Public Comments on Petitions for Rulemaking:
Calculated Maximum Fuel Element Cladding Temperature

PRM-50-93 and PRM-50-95
75 FR 3876 (January 25, 2010)
75 FR 66007 (October 27, 2010)
NRC-2009-0554

Submission ID	Name	Affiliation	ADAMS Accession Number
1	Anonymous		ML100610119
2	Mark Leyse		ML100820229
3	Aladar Stolmar		ML100830501
4	Robert Leyse		ML100850098
5	Kathryn Barnes	Don’t Waste Michigan	ML100890123
6	Jerry Martin		ML100890124
7	Bill and Jan Tache		ML100890125
8	Margo and Dennis Proksa		ML100890126
9	Debbie Highfill		ML100890127
10	Hugo Kobayashi		ML100890128
11	Betty Winholtz		ML100890130
12	Mary Johnston		ML100970080
13	Robert Leyse		ML101020563
14	Mark Leyse		ML101020564
15	Paul Gunter	Beyond Nuclear	ML101030142
16	John Butler	Nuclear Energy Institute	ML101040678
17	Robert Leyse		ML101040679
18	David Helker	Exelon	ML101130353
19	David Lochbaum	Union of Concerned Scientists	ML101180175
20	Mark Leyse		ML101230118
21	Mark Leyse		ML103340249
22	Mark Leyse		ML103340248
23	Aladar Stolmar		ML103340250
24	John Butler	Nuclear Energy Institute	ML103340251
25	Robert Leyse		ML103340252
26	Mark Leyse		ML110050023
27	Mark Leyse		ML111020046
28	Mark Leyse		ML11209C489
29	Mark Leyse		ML11213A211
30	Mark Leyse		ML12109A084
31	Mark Leyse		ML13031A698
32	Mark Leyse		ML14104B253
33	Mark Leyse		ML15288A506

Submission ID 1
Anonymous
ML100610119

PRM-50-93
(75FR03876)

DOCKETED
USNRC

March 1, 2010 (10:25am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

PUBLIC SUBMISSION

As of: March 01, 2010
Received: February 28, 2010
Status: Pending_Post
Tracking No. 80ab0a3a
Comments Due: April 12, 2010
Submission Type: Web

Docket: NRC-2009-0554
Mark Edward Leyse; Calculated Maximum Fuel Element Cladding Temperature

Comment On: NRC-2009-0554-0002
Mark Edward Leyse; Receipt of Petition for Rulemaking

Document: NRC-2009-0554-DRAFT-0002
Comment on FR Doc # 2010-01317

Submitter Information

Name: anonymous

General Comment

1-1

the petitioner seems to have ignored the following significant references, neither of which suggests anything is wrong with oxidation models at temperatures below 2300 F after an extensive review, although noting that Baker-Just should not be considered for best estimate calculations...

Nuclear Engineering and Design
Volume 232, Issue 1, July 2004, Pages 75-109
Advanced treatment of zircaloy cladding high-temperature oxidation in severe accident code calculations...

Schanz, G, 2003, "Recommendations and supporting information on the choice of Zirconium oxidation models in SA codes", FZKA 6827
[FZKA 6827. SAM-COLOSS-P043. Recommendations and Supporting....
bibliothek.fzk.de/zb/berichte/FZKA6827.pdf]

Template = SECY-067

DS 10

Rulemaking Comments

From: Gallagher, Carol
Sent: Monday, March 01, 2010 10:00 AM
To: Rulemaking Comments
Subject: Comment letter on PRM-50-93
Attachments: NRC-2009-0554-0002.pdf

Van,

Attached for docketing is a comment letter on PRM-50-93 that I received via the Regulations.gov website on February 28, 2010.

Thanks,
Carol

Received: from HQCLSTR01.nrc.gov ([148.184.44.79]) by OWMS01.nrc.gov
([148.184.100.43]) with mapi; Mon, 1 Mar 2010 10:00:54 -0500
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: "Gallagher, Carol" <Carol.Gallagher@nrc.gov>
To: Rulemaking Comments <Rulemaking.Comments@nrc.gov>
Date: Mon, 1 Mar 2010 10:00:17 -0500
Subject: Comment letter on PRM-50-93
Thread-Topic: Comment letter on PRM-50-93
Thread-Index: Acq5T+fXtB8NLureS4eUodtOeeDBHQ==
Message-ID:
<6F9E3C9DCAB9E448AAA49B8772A448C50C6C5ED77A@HQCLSTR01.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<6F9E3C9DCAB9E448AAA49B8772A448C50C6C5ED77A@HQCLSTR01.nrc.gov>
MIME-Version: 1.0

Submission ID 2
Mark Leyse
ML100820229

PRM-50-93
(75FR03876)

2

DOCKETED
USNRC

March 22, 2010 (10:50am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

March 15, 2010

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

Attention: Rulemakings and Adjudications Staff

Subject: Response to the 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, response to the NRC's notice of solicitation of public comments on PRM-50-93, published in the Federal Register, January 25, 2010.

Respectfully submitted,

Mark E Leyse

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

Template = SECY-067

DS 10

March 15, 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; NRC-2009-0554

TABLE OF CONTENTS

Appendix A Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation and Fig. 15. Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM

Appendix B Fig. 12. Temperatures during Test CORA-2 at [550] mm and 750 mm Elevation and Fig. 13. Temperatures Measured during Test CORA-3 at 450 mm and 550 mm Elevation

Appendix C Figure 15. Temperatures of Unheated Rods and Power History of CORA-5, Figure 16. Temperatures of Unheated Rods during CORA-12, Figure 17. Temperatures at Different Elevations during CORA-15, Figure 18. Temperatures of Unheated Rods during CORA-9, Figure 19. CORA-7; Temperatures at Elevations Given (750 mm), and Figure 20. Temperatures of Guide Tube and Absorber Rod during Test CORA-5

Appendix D Figure 37. Temperatures of the Heated Rods (CORA-13) and Figure 39. Temperatures of the Unheated Rods (CORA-13)

Appendix E Table 10. Zircaloy Oxidation, Energy Release, and Hydrogen Production during Various CORA Tests

Appendix F Figure A8.9. Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies and Figure A8.10. Analysis of Zr2K Thermal Response

March 15, 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; NRC-2009-0554

I. INTRODUCTION

On November 17, 2009, Mark Edward Leyse, Petitioner 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 ECCS evaluation calculations be based on data from multi-

---

[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 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.

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]

In these comments on PRM-50-93, Petitioner provides supplementary information to PRM-50-93. Petitioner provides supplementary information to the following sections of PRM-50-93: Section III.A.1., Section III.B., Section III.C.1.d., Section III.C.1.e., Section III.C.1.g., Section III.C.1.h., and Section III.D.4. Petitioner has also added a new section, at the end of these comments, titled "Examining the Autocatalytic Metal-Water Reaction that Occurred during the BWR FLECHT Zr2K Test."

II. BACKGROUND

Supplementary Information to PRM-50-93 Section III.A.1. Why "The Impression Left from Run 9573" Cannot be Separated from Zirconium-Water Reaction Models

2-1

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the Atomic Energy Commission ("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 Final Report"] when compared to the equivalent stainless steel tests."[6] 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."[7]

---

[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] 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.

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

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:

> The second 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.
>
> *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].[8]

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

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

> film. *The presence of this heat source should not affect the heat transfer coefficients, which depend on conditions in the coolant outside the rod.*[9]

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.

As stated in PRM-50-93, within the first 18.2 seconds of FLECHT run 9573,[10] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[11] *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 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 heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[12]

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.[13]

---

9 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 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.

10 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.

11 *Id.*, p. 3-98.

12 Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.

13 Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com/.

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."[14]

And, as quoted in PRM-50-93, regarding steam temperatures measured by the seven-foot steam probe, "PWR FLECHT 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).[15]

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 5...thermocouples"[16]—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.

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,"[17] the AEC Commissioners' conclusion that "the presence of...heat [generated from the exothermic zirconium-water reaction] should not

---

[14] 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.
[15] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[16] *Id.*, p. 3-98.
[17] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.

affect…heat transfer coefficients, which depend on conditions in the coolant outside the rod"[18] is erroneous. 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.

Supplementary Information to PRM-50-93 Section III.B. Reflood Rates

1. Reflood Rates and the AEC's ECCS Rulemaking Hearing

2-2

Reflood rates were a major subject in the AEC's ECCS rulemaking hearing: reflood rates are discussed to some extent on more than a half dozen pages of "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"[19] the concluding statement of Henry. W. Kendall and Daniel F. Ford, Union of Concerned Scientists ("UCS"), on behalf of Consolidated National Intervenors ("CNI"), in the AEC ECCS rulemaking hearing. "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" provides a concise summary of reactor safety issues, debated in the hearing, including some reactor safety issues that have not been resolved since 1973, when the hearing concluded.

Regarding an Advisory Committee on Reactor Safeguards ("ACRS") statement, regarding ECCS analysis, that was placed on the record in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> The ACRS explained that, in their view, ECCS analysis is proven to be conservative when it is fully confirmed by experimental evidence and supporting analytical studies. On this basis, the ACRS listed every major item of the present LOCA transient analysis methods that in their view had not been proven to be conservative.[20]

---

[18] 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."

[19] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," Concluding Statement—Safety Phase—Prepared by Union of Concerned Scientists on Behalf of Consolidated National Intervenors in the Matter of Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants, AEC Docket RM-50-l, April 1973, p. 5.20-5.23, 5.35, 5.48-5.49.

[20] *Id.*, pp. 4.42-4.43.

Among the items on ACRS's list were reflood rates and reflood heat transfer.[21]

It is significant that a Cal. Tech. paper written in 1975, recommended minimum reflood heat transfer rates or alternatively, minimum reflood rates; "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," states "[r]ecommendations are made for improvements in criteria conservatism, *especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates)*" [emphasis added].[22]

Regarding reflood rates and steam binding, the revised edition of *The Cult of the Atom: The Secret Papers of the Atomic Energy Commission* states:

> The industry had predicted that the E.C.C.S. in pressurized-water reactors would be able to deluge the core with water, quickly refilling the reactor and terminating the difficulties caused by the loss of normal cooling water. The industry's calculations showed that the "reflooding rate"—the speed at which the water level inside the reactor increased following the injection of E.C.C.S. water—would be several inches per second. Since the fuel rods in the core are twelve feet high, it would not take long to flood the core with cooling water once E.C.C.S. water [was injected].
>
> [George] Brocket and his associates, however, *reported that the reflooding rate might be only one and a half inches per second, or less.* The industry's analyses, they showed the A.E.C., had overlooked the fact that the steam pressure inside the reactor would drastically limit the rate at which emergency cooling water could rise up into the core. Because of "steam binding," they said, the current E.C.C.S. might have only a "marginal" capacity for preventing [a meltdown] [emphasis added].[23]

And explaining steam binding, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> As the cooling water reaches the hot core much of it would be converted to steam, and it is this steam together with entrained water droplets that would provide the initial cooling of the hotter regions of the core. For the reflood water to continue entering the core it must displace the steam, which would have to escape from the reactor vessel and find its way into the containment atmosphere. In the pressurized water reactors the steam

---

[21] *Id.*, p. 4.43.

[22] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Abstract, p. iii.

[23] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, pp. 100-101.

> would have to flow through the steam generator and pump to escape through a cold leg break; the reduction of [the] reflood rate by the relatively high resistance to flow of this path is called "steam binding." *Steam binding would severely limit the rate of reflooding the core, reducing it from an intended 6 to 11 inches per second to from 1.0 to 2.5 inches per second, depending on the reactor design.* The rule we announce considers all the evidence in the record on this important subject of steam binding and provides an acceptable overall assurance of ECCS effectiveness. The inquiry, however, should not end there. Thus the Commission urges the pressurized water reactor manufactures to seek out design changes that would overcome steam binding. This same point of view is reflected in the September 10, 1973, letter of the Advisory Committee on Reactor Safeguards [emphasis added].[24]

Discussing the testimony of Dr. Morris Rosen of the AEC, regarding George Brocket's statements about steam binding, in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> "My opinion of, let's say, the testimony of George Brockett is, I believe— I don't know his exact title but I think it is manager of development, nuclear safety development at [Aerojet Nuclear Company ("Aerojet")], *I think he came out strongly indicating that steam binding indeed was a problem.*
>
> "*I think he indicated perhaps that reductions in operating power levels were required.*
>
> "Personal observation about Mr. Brockett: I think in my opinion one would classify him as perhaps one of the leading experts in this country in emergency core cooling, in my opinion, if not the leading expert.
>
> "I think when that man comes out and says there is a problem, I take note of it" [emphasis added].[25]

Regarding reflood rates, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> CNI testimony set out the history of continuous and substantial decreases in predicted PWR core flooding rates that has occurred over recent years. It is now established that core flooding rates earlier considered as

---

[24] Ray, Larson, Doub, Kriegsman, Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1092.

[25] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 4.7-4.8.

> extremely degraded are now very close to the expected conditions for a double-ended PWR inlet line break. There is a widespread feeling in the community of reactor safety engineers that there is presently a relatively small and likely non-existent margin between cooling and non-cooling. ... [Robert] Colmar indicated that in his opinion reflood and refill were the areas of greatest uncertainty. [Rex] Shumway of [Aerojet] reported on several reflooding calculations he had performed for a Westinghouse ice condenser plant. The upper limit was 1.4 inches per second and for the lower limit, without the unbroken leg completely plugged and no water in the unbroken leg, found the computed reflood rate to be in the range from 0.45 to 0.55 inches per second. If these lower values prove to be correct CNI concludes that an accident in such a plant *cannot be controlled* [emphasis not added].[26]

Discussing the PWR FLECHT tests and reflood rates, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> A major difficulty in the program was that the flooding rate values selected for the test program were chosen when the low flooding rates now recognized as realistic were not [yet] identified. Essentially, as pointed out in CNI testimony, the base flooding rate was initially set for the tests at 12 inches per second. It soon was reduced to [six] inches per second and later lowered further as calculations indicated actual flooding rates [of] *around one inch per second.* The test program was modified in part to study this new region. A large bulk of the information, however, was taken for non-representative flood rates. Accordingly, a major portion of the program results are simply not applicable to the expected circumstances of a PWR LOCA.
>
> Zane of [Aerojet] testified that at the point when most of the FLECHT tests were completed, Westinghouse acknowledged the possibilities of lower flooding rates and the steam binding problem [emphasis not added].[27]

Additionally, as mentioned in PRM-50-93, it is significant that "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states, "Consolidated National Intervenors pointed out that most of [the Zircaloy] runs were made at unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding

---

[26] *Id.*, p. 5.21.
[27] *Id.*, p. 5.35.

2-3

rate was about one inch per second."[28] Furthermore, the "different result" that was obtained from run 9573, lead the Commissioners of the AEC to state "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[29]

2. Reflood Rates, Cladding Temperatures at the Onset of Reflood, and TRAC-M (TRACE)

It is significant that "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data" states:

> During a large-break LOCA, cladding temperature changes as follows:
>
> Cladding temperature increases during blowdown from normal operating conditions of approximately 325°C *to approximately 550-800°C (roughly 1000-1500°F)* [emphasis added].[30]

If indeed, the Zircaloy fuel cladding were to have temperatures between approximately 1000°F and 1500°F, especially between approximately 1200°F and 1500°F, at the onset of reflood, and there were a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower), with high probability, cladding temperatures would exceed the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F.

As discussed in PRM-50-93, 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.

---

[28] 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."

[29] *Id.*

[30] NRC, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data," NUREG-1744, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011520327, p. 3.

(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 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).
>
> 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.[31]

As discussed in PRM-50-93, in 1973, the Commissioners of the AEC stated, "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[32] Run 9573 was one of the four tests

[31] Mark Edward Leyse, PRM-50-93, November 17, 2009, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML093290250, p. 18.

[32] 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 Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.

Regarding "more than 50 tests [that] were conducted [in the early 1980s,] to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length 32-rod nuclear bundle during the heatup, reflood, and quench phases of a large-break LOCA," the NRC stated:

> The petitioner [Robert H. Leyse] states that more experiments with Zircaloy cladding have not been conducted on the scale necessary to overcome the impression left from run 9573. The NRC disagrees. In fact, additional Zircaloy tests have been performed. In the early 1980s, the NRC contracted with National Research Universal (NRU) at Chalk River, Ontario, Canada to run a series of LOCA tests in the NRU reactor. More than 50 tests were conducted to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length 32-rod nuclear bundle during the heatup, reflood, and quench phases of a large-break LOCA. The NRC is reviewing the data from this program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE).[33]

So, in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test] program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[34] It is clear that the NRC has failed to analyze the data from the NRU thermal-hydraulic and mechanical deformation tests that indicates that, in the event a LOCA, a constant core reflood rate of approximately 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, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

Furthermore, when the NRC's document, "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction," from 2002, states that "good core quenching rates are achieved even for flooding rates of one inch per second," it is important to

[33] 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. 18-19.
[34] *Id.*, p. 19.

remember that the NRC's claim is based on the results of tests conducted with stainless steel cladding.

In more detail, "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction" states:

> During the reflood phase of most reactor designs, the emergency core coolant is injected so that it passes through the downcomer and lower plenum and then up into the core. "Bottom reflood" of the core is the predominant mode of core recovery, and many experiments have been conducted to investigate the processes important in bottom reflooding. ...
>
> Tests conducted at less than one inch per second as part of the FLECHT and FLECHT-SEASET programs confirmed high rates of carryover from the bundle. ... These, along with other tests demonstrated the flowing:
>
> 1. 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.*
>
> 2. During reflood, the flow regime, cladding temperature rise and quench behavior is strongly dependant on the flooding rate [emphasis added].[35]

Regarding a FLECHT-SEASET test conducted with stainless steel cladding, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer" states:

> The FLECHT-SEASET test 31504 is commonly included as a benchmark test in the validation matrix of several computer codes. Run 31504 is a forced reflood test with 2.5 cm./sec. [(~1.0 in./sec.)] flooding rate. ... In the experiment the reflood is initiated when the PCT reaches 1144 K (1600°F). Subcooled liquid at 323 K is injected at the bottom of the test section at 2.5 cm./sec. The pressure (272 kPa) is set at the outlet of the bundle.[36]

The report, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," Volume 2, states that in the FLECHT-SEASET test 31504, the PCT at the onset of reflood was 1585°F, that the rod peak power was 0.7 kw/ft, and

---

[35] "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.

[36] Cesare Frepoli, John H. Mahaffy, and Lawrence E. Hochreiter, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer," Nuclear Engineering and Design, 224, 2003, pp. 139, 140.

that the PCT during reflood, remained under the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F; it states that the PCT was approximately 2100°F.[37]

(It is noteworthy that "A Moving Subgrid Model for Simulation of Reflood Heat Transfer" states that the original COBRA-TF and the new COBRA-TF/FHMG codes are used to simulate the FLECHT-SEASET test 31504 and that code predictions are compared with test data.[38])

Regarding FLECHT-SEASET tests 31504 and 32753, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data" states:

> This report presents the results of an assessment of the capabilities of the TRAC-M(F90), Version 3.580, and TRAC-M(F77), Version 5.5.2A, codes to calculate reflood and steam cooling phenomena for pressurized-water reactors (PWRs). The reflood assessment was performed using test data from FLECHT-SEASET Run 31504, while the steam cooling assessment was performed using test data from FLECHT-SEASET Run 32753. These tests simulate unblocked bundle forced reflood and steam cooling conditions in PWRs.[39]

And, regarding the assessment of the capabilities of the TRAC-M(F90), Version 3.580, and TRAC-M(F77), Version 5.5.2A, codes to calculate reflood and steam cooling phenomena, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data" states:

> The assessment shows that predictions of the reflood phenomena derived using both codes are inaccurate; however, it is judged that *they can conservatively predict peak clad temperatures in heated rods* since the code model expels more water from the test section than measured. The predictions of steam cooling in single-phase flow conditions are acceptable [emphasis added].[40]

It is significant that the FLECHT-SEASET test 31504 was conducted with a stainless steel bundle, not with a Zircaloy bundle. Therefore, the TRAC-M codes

---

[37] M. J. Loftus, *et al.*, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," Volume 2, NUREG/CR-1532, June 1980, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070740185, pp. 31504-1, 31504-2.
[38] Cesare Frepoli, John H. Mahaffy, and Lawrence E. Hochreiter, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer," Nuclear Engineering and Design, 224, 2003, p. 139.
[39] NRC, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data," NUREG-1744, p. 1.
[40] *Id.*, p. iii.

conservatively predict PCTs for heated stainless steel rods; however, the TRAC-M codes do not conservatively predict PCTs for the Zircaloy fuel rods that are used in PWRs.

In other words, if the FLECHT-SEASET test 31504 had been conducted with a Zircaloy bundle instead of a stainless steel bundle, the test results would have been different: with high probability, the Zircaloy bundle would have had a PCT that exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F and it would have incurred autocatalytic oxidation, like FLECHT run 9573.

As quoted in PRM-50-93, on page 68, 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:

> The rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400 K [(2060°F)]. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[41] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425 K [(2106°F)]. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain* [emphasis added].[42]

And regarding FLECHT stainless steel runs 6553 and 9278, and FLECHT Zircaloy run 9573, PRM-50-93 states:

> FLECHT stainless steel runs 6553 and 9278 (with the same peak power levels as Zircaloy run 9573), at the hot rod midplane elevation, at the onset of flood, had cladding temperatures of 2012°F and 2028°F, respectively, flood rates of 1 in./sec., and peak cladding temperatures of 2290°F and 2286°F, respectively.[43] In contrast to Zircaloy run 9573—with a slightly lower clad temperature at the onset of flood and a slightly higher flood rate—runs 6553 and 9278 did not incur autocatalytic oxidation reactions. In fact, runs 6553 and 9278 were conducted with the same stainless steel assembly, and after run 9278 was conducted, the assembly was reused for more tests, because it remained intact.[44]

[41] 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.
[42] *Id.*, p. 4.4.
[43] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-6.
[44] Mark Edward Leyse, PRM-50-93, November 17, 2009, pp. 68-69.

In PRM-50-93, on pages 59-71, Petitioner argued that stainless steel cladding heat transfer coefficients are not *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

2-4

3. Downcomer Boiling and Reflood Downcomer Bypass

It significant that reflood rates could be affected by downcomer boiling and reflood downcomer bypass.

Regarding downcomer boiling and reflood downcomer bypass, "Appendix K Non-Conservatisms" states:

> Downcomer hydraulics refers to two processes that were not anticipated in the original 1973 Rulemaking, nor recognized at the time of the 1988 Appendix K revision. The first process is downcomer boiling, which are the processes of subcooled and saturated boiling that may occur as fluid in the downcomer is brought to saturation by heat released by the core barrel, reactor vessel walls, and lower plenum metal. The second process is reflood downcomer bypass, which refers to the entrainment and carry-over of downcomer fluid to the break by steam that flows circumferentially around the downcomer from the intact cold legs. ... Both of these processes are relatively "new." That is, that neither process was recognized as potential non-conservatisms until the early 1990s. Their effects can be observed in experimental data as well as in recent calculations with realistic thermal-hydraulic codes.[45]

And regarding downcomer boiling, "Downcomer Boiling Phenomena during the Reflood Phase of a Large-Break LOCA for the APR1400" states:

> Downcomer boiling phenomena in a conventional pressurized water reactor has an important effect on the transient behavior of a postulated large-break LOCA..., because it can degrade the hydraulic head of the coolant in the downcomer and consequently affect the reflood flow rate for a core cooling and finally result in a failure of the nuclear fuel rods.[46]

---

[45] "Appendix K Non-Conservatisms," Attachment 4 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 3; Attachment 4 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720716; the letter's Accession Number: ML021720690.

[46] B. J. Yun, D. J. Euh, C. H. Song, "Downcomer Boiling Phenomena during the Reflood Phase of a Large-Break LOCA for the APR1400," Nuclear Engineering and Design, 238, 2008, p. 2064.

And regarding reflood downcomer bypass, "Appendix K Non-Conservatisms" states:

> The entrainment of downcomer water reduces the driving head for core reflood, similar to the downcomer boiling effect. The effect of reflood downcomer bypass was concluded to be non-conservative in ["Summary of Results form the UPTF Downcomer Separate Effects Tests, Comparison to Previous Scaled Tests, and Application to U.S. Pressurized Water Reactors"], although the impact on PCT was not expected to be large. In a later study, ["Evaluation of Proposed Changes to 10 CFR 50 Appendix K"], however, it was concluded that the UPTF and CCTF experimental tests under predicted the effect in a PWR, and thus a larger increase in PCT due to reflood downcomer bypass was possible. Therefore reflood downcomer bypass is considered a non-conservatism not appropriately accounted for in Appendix K.[47]

According to "Effect of Proposed Revisions on Evaluation Model Results," estimated increases in the PCT from downcomer boiling are:

+400°F (Westinghouse estimate from Best Estimate EM calculations for a W 4-loop PWR); +810°F (NRC contractor calculations using RELAP5 for a CE system 80+ (3800 MWt) unit; and +63°F (For downcomer boiling and reflood bypass. Estimate based on WCOBRA/TRAC calculations for an uprated CE System 80+ unit. Both downcomer boiling and ECC bypass during reflood were found to be important and contributed to increases in PCT.)[48]

2-5

Supplementary Information to PRM-50-93 Section III.C.1.d. The LOFT LP-FP-2 Experiment

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

[47] "Appendix K Non-Conservatisms," Attachment 4 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," p. 4.

[48] "Effect of Proposed Revisions on Evaluation Model Results," Attachment 5 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 4; Attachment 4 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720740; the letter's Accession Number: ML021720690.

(LPIS) of a four-loop PWR as described in "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2."[49] 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.[50]

And regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

From the analyses of core temperature measurements in [the LOFT] LP-FP-2 [experiment], the rapid increase in temperature shown in fig 14.[51] 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,"[52] 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.[53, 54]

---

[49] 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.

[50] 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. 133.

[51] See Appendix A Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation.

[52] 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.

[53] See Appendix A Fig. 15 Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM.

Additionally, regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> More phenomena were detected from the analyses of the recorded behavior of the 60 thermocouples in the CFM together with other thermocouples and measuring systems in the LOFT nuclear reactor.
>
> After the first indication of [the] metal-water reaction at 1430 [seconds] several instruments indicated a common event at 1500 [seconds]. These instruments included gross gamma monitor, momentum flux meter in the downcomer, upper tie plate and guide tube thermocouples. [According to "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2,"[55] t]his event is believed to be the rupture of the control rod cladding.[56]

And regarding the durability of pressure sensors, thermocouples, and radiation monitors in the LOFT-LP-FP-2 experiment and TMI-2 accident, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> Both in TMI-2 and [LOFT] LP-FP-2 only [a] few types of sensors were able to withstand the consequences of severe accidents and were able to deliver information for post-accident analysis. These were pressure sensors, thermocouples, and radiation monitors. Advanced instrumentation technology have proven to be able to utilize these three types of sensors in redundant and diverse instrumentation of Light Water Reactors (LWR) to manage severe accidents.[57]

It is significant that "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states that in the LOFT LP-FP-2 experiment "the rapid increase in temperature...was a result of the oxidation of zircaloy which became rapid at temperatures in excess of 1400 K." This would mean, as discussed in PRM-50-93 (pages 38-43), that during the LOFT LP-FP-2 experiment the onset of an autocatalytic

---

[54] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 135.

[55] 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.

[56] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 136.

[57] *Id.*, p. 147.

2-6

oxidation reaction of Zircaloy cladding occurred at approximately 1400 K (2060°F)—well below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

Supplementary Information to PRM-50-93 Section III.C.1.e. The CORA Experiments

1. Three Papers on the CORA Experiments

It is significant that the CORA-2 and CORA-3 experiments, initiated with a temperature ramp rate of 1 K/sec, had temperature excursions, due to the exothermal Zircaloy-steam reaction, that commenced at approximately 1000°C (1832°F),[58] leading the CORA-2 and CORA-3 bundles to maximum temperatures of 2000°C and 2400°C, respectively.[59]

Discussing the exothermal Zircaloy-steam reaction that occurred in these 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 Tests B and C)],[60] 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].[61]

As discussed in PRM-50-93, on pages 26-27, 38-43-45, 51-55, "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the

---

[58] See Appendix B Fig. 12. Temperatures during Test CORA-2 at [550] mm and 750 mm Elevation and Fig. 13. Temperatures Measured during Test CORA-3 at 450 mm and 550 mm Elevation.

[59] 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.

[60] 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.

[61] 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.

exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation."[62]

Regarding the CORA-2 and CORA-3 experiments, the abstract of "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:

> In the CORA experiments test bundles of usually 16 electrically heated fuel rod simulators and nine unheated rods are subjected to temperature transients of a slow heatup rate in a steam environment. Thus an accident sequence is simulated, which may develop from a small-break loss-of-coolant accident of an LWR.
>
> CORA-2 and CORA-3 were the first "Severe Fuel Damage" experiments of the program with UO_2 pellet material. The transient tests were performed on August 6, 1987, and on December 3, 1987, respectively. Both test bundles did not contain absorber rods. Therefore, CORA-2 and CORA-3 can serve as reference experiments for the future tests, in which the influence of absorber rods will be considered. An aim of CORA-2, as a first test of its kind, was also to gain experience in the test conduct and posttest handling of UO_2 specimens. CORA-3 was performed as a high-temperature test. With this test the limits of the electric power supply unit could be defined
>
> The transient phases of CORA-2 and CORA-3 were initiated with a temperature ramp rate of 1 K/sec. The temperature escalation due to the exothermal [Zircaloy]-steam reaction started at about 1000°C, leading the bundles to maximum temperatures of 2000°C and 2400°C for tests CORA-2 and CORA-3, respectively.[63]

And discussing video and still cameras that recorded 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:

> The high-temperature shield is located within the pressure tube. Through a number of holes in the shield, the test bundle is being inspected during

[62] 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.
[63] 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, Abstract.

the test by several video and still cameras. The holes are also used for temperature measurements by two-color pyrometers complementing the thermocouple readings at elevated temperatures.[64]

And discussing the interpretation of the CORA-2 and CORA-3 experiments results, "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:

> The tests CORA-2 and CORA-3 have been successfully conducted, accompanied by measurements and visual observations and evaluated by micro-structural and compositional analyses. On the basis of this information and the expertise from separate-effects investigations the following interpretation of the sequence of mechanisms during the degradation of the bundles is given.
>
> As already observed in previous tests [(CORA Tests B and C)],[65] 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. An effectively moderated escalation would be observed for smaller initial heatup rates, because the growth of protective scale during steam exposure counteracts by decreasing the oxidation rate of the material.
>
> This explains the observation that the temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward. It is reasonable to assume, that the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred.
>
> A first melting process starts already at about 1250°C at the central grid spacer of Inconel, due to diffusive interaction in contact with Zry cladding material, by which the melting temperatures of the interaction partners (ca. 1760°C for Zry, ca. 1450°C for Inconel) are dramatically lowered towards the eutectic temperature, where a range of molten mixtures solidifies.

---

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

[65] 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.

> (This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[66]

Two additional papers on the CORA experiments also provide information on cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding that occurred below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.[67]

First, regarding this phenomenon, the abstract of "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> The transient phases of the tests were initiated with a temperature ramp rate of 1 K/sec. *The temperature escalation due to the exothermal zircaloy (Zry)-steam reaction started at about 1100°C*, leading the bundles to maximum temperatures of approximately 2000°C [emphasis added].[68]

And regarding this phenomenon, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" also states:

> The transient of a SFD-type accident is initiated by a slow temperature rise in the order of 0.5 [to] 1.0 K/sec., followed by a *rapid temperature escalation (several tens of degrees Kelvin per second)* due to the exothermal heat produced by the cladding oxidation in steam environment [emphasis added].[69]

---

[66] 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.

[67] See Appendix C Figure 15. Temperatures of Unheated Rods and Power History of CORA-5, Figure 16. Temperatures of Unheated Rods during CORA-12, Figure 17. Temperatures at Different Elevations during CORA-15, Figure 18. Temperatures of Unheated Rods during CORA-9, Figure 19 CORA-7; Temperatures at Elevations Given (750 mm), and Figure 20 Temperatures of Guide Tube and Absorber Rod during Test CORA-5, which depict temperature excursions during various CORA tests; see also Appendix D Figure 37. Temperatures of the Heated Rods (CORA-13) and Figure 39. Temperatures of the Unheated Rods (CORA-13).

[68] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7448, 2008, Abstract, p. I.

[69] *Id.*, p. 1.

Second, regarding this phenomenon the abstract of "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" states:

> In the CORA experiments two different bundle configurations are tested: PWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor) bundles. The PWR-type assemblies usually consist of 25 rods with 16 electrically heated fuel rod simulators and nine unheated rods (full-pellet and absorber rods). Bundle CORA-13, a PWR-type assembly, contained two Ag/In/Cd-steel absorber rods. The test bundle was subjected to temperature transients of a slow heatup rate in a steam environment; *i.e.*, the transient phase of the test was initiated with a temperature ramp rate of 1 K/sec. *The temperature escalation due to the exothermal zircaloy(Zry}-steam reaction started at about 1100°C at an elevation of 850 mm (1000 sec. after [the] onset of the transient), leading to a temperature plateau of 1850°C* and after initiation of quenching to maximum temperatures of approximately 2000°C to 2300°C. CORA-13 was terminated by quenching with water from the bottom with a flooding rate of 1 cm/sec.
>
> Rod destruction started with the failure of the absorber rod cladding at about 1200°C; *i. e.*, about 250 K below the melting regime of steel. Penetration of the steel cladding was presumably caused by a eutectic interaction between steel and the zircaloy guide tube. As a consequence, the absorber-steel-zircaloy melt relocated radially outward and axially downward. Besides this melt relocation the test bundle experienced severe oxidation and partial melting of the cladding, fuel dissolution by Zry/UO_2 interaction, complete Inconel grid spacer destruction, and relocation of melts and fragments to lower elevations in the bundle. An extended flow blockage has formed at the axial midplane.
>
> Quenching of the hot test bundle by water resulted, besides additional fragmentation of fuel rods and shroud, in an additional temperature increase in the upper bundle region. Coinciding with the temperature response an additional hydrogen buildup was detected. During the flooding phase 48% of the total hydrogen [was] generated [emphasis added].[70]

And regarding this phenomenon "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" also states:

> The temperature rise shows the same general features already found in earlier tests. With the increase of the electrical power input, first the temperature rises proportional to the power. *Having reached about 1000°C, the exothermal Zry/steam reaction adds an increasing*

[70] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," Kernforschungszentrum Karlsruhe, KfK 5054, 1993, Abstract, p. v.

> *contribution to the energy input, resulting in a temperature escalation.* The escalation starts at [the] 950 mm and 750 mm elevation. For the outer fuel rod simulator [number] 3.7 the escalation is delayed at 750 mm by about 150 sec. A possible reason for this delay could be the heat losses due to the window at 790 mm adjacent to this rod. The escalation at the 550 mm elevation follows 200 sec. later. The escalation at 1150 mm develops before that at the 350 mm elevation [emphasis added].[71]

So "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" and "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" both state that temperature escalations due to the exothermic Zircaloy-steam reaction began at approximately 1100°C (2012°F). "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" also states that "having reached about 1000°C [(1832°F)], the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation." [72] Additionally, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states that the "rapid temperature escalation[s were] several tens of degrees Kelvin per second...due to the exothermal heat produced by the cladding oxidation in [a] steam environment."[73]

As stated above data from the CORA experiments indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

It is also significant that, regarding the percentage of additional energy from the exothermic zirconium-steam reaction during the escalation phase of the CORA tests, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> In the escalation phase; *i.e.*, starting from about 1100°C the slow temperature rise is followed by a rapid increase caused by the increased electric power input *and the additional energy from the exothermal*

[71] *Id.*, p. 12.
[72] *Id.*
[73] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, p. 1.

zirconium-steam reaction. The contribution of this exothermal heat to the total energy input is generally between 30 and 40% [emphasis added].[74]

And elsewhere, regarding this phenomenon, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> Based on the accumulated H_2 productions of tests CORA-15, CORA-9, and CORA-7 the oxidation energy is determined. Its percentage amounts to 30 - 45% of the total energy input (electric supply plus exothermal energy)...[75]

So the percentage of oxidation energy from the exothermic zirconium-steam reaction was generally between 30 and 40%, and in some cases was as high as 45%, of the total energy input during the escalation phase of the CORA tests (see Appendix E Table 10. Zircaloy Oxidation, Energy Release, and Hydrogen Production during Various CORA Tests).

2. The 1990 CORA Workshop at Kernforschungszentrum Karlsruhe

It is significant that in the 1990 CORA Workshop at Kernforschungszentrum Karlsruhe ("KfK") GmbH, Karlsruhe, FRG, October 1-4, 1990, problems with SCDAP/RELAP5's modeling of Zircaloy oxidation kinetics, in the 900-1200°C temperature range, were discussed.

The document, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," is partly a report on the 1990 CORA Workshop at KfK GmbH, Karlsruhe, FRG, October 1-4, 1990.[76]

Regarding temperature excursions during the CORA experiments and SCDAP/RELAP5's late prediction of the temperature excursion for the CORA-12

[74] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, p. 5.
[75] *Id.*, p. 7.
[76] 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, Cover Page.

experiment, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> Temperature escalation starts at ~1200°C and continues even after shutoff of the electric power as long as metallic Zircaloy and steam are available.
>
> . . .
>
> [Dr. T. J. Haste, United Kingdom Atomic Energy Agency,] *did note the late prediction (via SCDAP/RELAP5) for the oxidation excursion in CORA-12*... [emphasis added][77]

And regarding "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[78] "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) oxidation to be underpredicted.*
>
> ... 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.[79]

And regarding heatup rates, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> H. Plank (Siemens/KWU) made an interesting argument for the reduction of heatup rates in future CORA tests based on accident probabilities in German LWRs. Historically, the CORA structural heatup rate has been ~1 K/sec., which reflects the most probable German severe accident core heatup rates. However, backfits to German BWRs will make the long term sequences (4-10 hr. or >10 hr.) more likely and these sequences exhibit heatup rates of ~1/3 K/sec. There was some concern that this low rate could lead to complete oxidation of the Zircaloy with little or no metallic melting and relocation. (This has been predicted in previous studies for U.S. BWRs for long-term accident sequences with a small injection rate.) Low heatup rates will be considered as a future CORA test parameter as will bundle preoxidation. G. Shantz (KfK) presented the results of a study that focused on the temperature and duration for Zircaloy

[77] *Id.*, pp. 2, 3.
[78] *Id.*, p. 3.
[79] *Id.*

preoxidation with a recommendation of a 2 hr. pretest at 800°C maximum temperature.[80]

Supplementary Information to PRM-50-93 Section III.C.1.g. The QUENCH-04 Test

Since submitting PRM-50-93, it has come to Petitioner's attention that there is an explanation for the temperature excursions that were measured, commencing at temperatures between approximately 750°C and 800°C, in the unheated region at the top of the shroud, in the QUENCH experiments, other than the exothermic hydriding reaction of Zircaloy in the shroud: the thermocouple readings were erroneous.

2-7

In PRM-50-93, on page 47, Petitioner quoted "Degraded Core Quench: Summary of Progress 1996-1999," to provide information regarding such low temperature excursions:

> A notable feature of the experiments was the occurrence of temperature excursions starting in the unheated region at the top of the shroud, from temperatures of 750-800°C, which is more than 300°K lower than excursion temperatures associated with [the] runaway oxidation [of Zircaloy] by steam. FZKA have postulated that these excursions are driven by the exothermic hydriding reaction of Zircaloy in the shroud. ...[81]

It was latter concluded that the thermocouple readings at the top of the shroud in the QUENCH experiments were erroneous, because of cable routing through hot zones of the QUENCH bundles. Regarding this issue, "Results of the QUENCH-09 Experiment with a B_4C Control Rod" states:

> To verify the influence of [thermocouple] routing on the temperature reading, [thermocouple] pairs were mounted at three axial levels in the QUENCH-09 bundle. One pair was mounted on the rod surface (TFS-type thermocouple) at level 12, the other two pairs on the shroud surface (TSH-type thermocouple) at levels 15 and 16. The TSH-type thermocouple pair consisted of one [thermocouple] passing through the hot zone (direction to bundle top) and one [thermocouple] not passing the hot zone (direction bundle bottom). The cables of the TSH thermocouples were routed to the bundle bottom. The cables of the two "colder" shroud thermocouples were insulated by the ZrO_2 fiber insulation. ...

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

[81] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," Executive Summary, February 2000, p. 9.

2-8

> It is concluded that thermocouples, passing [through] the hot zone, show…higher values, than thermocouples, whose cable [is] located in [the] region with lower temperatures, than temperature at the [thermocouple] junction. Therefore, hot-zone errors can be avoided by routing the thermocouple cables out of the hot zone…and by insulating the shroud [thermocouple] cable… This will be done in future tests.
>
> The qualification of questionable thermocouple readings was done for earlier QUENCH tests…[82]

So the thermocouple readings at the top of the shroud in the QUENCH experiments were erroneous; however, the passage above, from "Degraded Core Quench: Summary of Progress 1996-1999," is still highly significant, because it states that "excursion temperatures associated with [the] runaway oxidation [of Zircaloy] by steam" are higher than 1050°C to 1100°C (1922°F to 2012°F).[83]

Supplementary Information to PRM-50-93 Section III.C.1.h. Examining the Autocatalytic Metal-Water Reaction that Occurred during FLECHT RUN 9573

As mentioned in PRM-50-93, there is no metallurgical data from the locations of run 9573 that incurred runaway (autocatalytic) oxidation, because Westinghouse did not obtain such data. When Westinghouse performed the metallurgical analyses for the assembly of FLECHT run 9573, Westinghouse measured oxide thicknesses in the locations of the assembly that did not incur autocatalytic oxidation.

It is significant that, regarding local steam starvation conditions postulated to have 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:

> [T]he temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward. It is reasonable to assume, that *the violent oxidation essentially consumed the available*

---

[82] M. Steinbrück, A. Miassoedov, G. Schanz, L. Sepold, U. Stegmaier, H. Steiner, J. Stuckert, "Results of the QUENCH-09 Experiment with a B_4C Control Rod," Appendix 2, Forschungszentrum Karlsruhe, FZKA 6829, 2004, pp. 181-182.

[83] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," p. 9.

steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred [emphasis added].[84]

It would also be reasonable to assume that, during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

As quoted in PRM-50-93, discussing the extensive oxidation of the assembly of FLECHT run 9573, in its comments regarding PRM-50-76, Westinghouse states:

> Despite 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. ...
>
> Westinghouse notes that the metallurgical analyses performed for FLECHT Run 9573 indicated that the measured oxide thickness was still within the expected range for specimens heated as high as 2500°F.[85]

(When Westinghouse performed the metallurgical analyses for the assemblies from the four FLECHT Zircaloy tests, it compared the measured oxide layer thicknesses to Baker-Just correlation predictions[86]—"the expected range.")

And as also quoted in PRM-50-93, in "Denial of Petition for Rulemaking (PRM-50-76)," discussing the metallurgical analyses performed for the Zircaloy FLECHT tests, the NRC states:

> The petitioner did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. The petitioner also ignored the Westinghouse application of the Baker-Just correlation to these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the

[84] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[85] 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, pp. 3-4.

[86] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 17, 21.

correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...

The NRC applied the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients.[87]

So, as stated in PRM-50-93, neither Westinghouse nor the NRC applied the Baker-Just correlation to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation; furthermore, the NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation. And, as stated above, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments—during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

2-9

Supplementary Information to PRM-50-93 Section III.D.4. A Comparison of the High Temperature Oxidation Behavior of Zircaloy and Stainless Steel Assemblies

Discussing criticisms Consolidated National Intervenors ("CNI") made in the AEC's ECCS rulemaking hearing of the PWR FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

Criticisms were made by the CNI concerning a number of problems [with the PWR FLECHT program]. The experimental design was faulted (especially the use of [stainless steel] rods in 84 of the 88 tests [versus Zircaloy] rods in only [four] of the 88).[88]

---

[87] *Id.*, pp. 21-22.
[88] 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, p. A8-28.

2-10

Discussing the durability of stainless steel heater-rod assemblies in the FLECHT program, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[s]tainless steel was used instead of Zircaloy as the cladding material for nearly all of the FLECHT tests because it is more durable under the test conditions."[89]

And also discussing the durability 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].[90]

Examining the Autocatalytic Metal-Water Reaction that Occurred during the BWR FLECHT Zr2K Test

It is significant that during the AEC's ECCS rulemaking hearing, conducted in the early '70s, that Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[91] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[92] conducted with a Zircaloy assembly. Among other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [metal-water] reactions, in spite of the

---

[89] 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. 1123. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

[90] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-6.

[91] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists.

[92] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-17; this paper cites Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

'failed' heater rods. They compared test results for SS2N [(conducted with a stainless steel assembly)] with Zr2K, showing satisfactory correlation during approximately the first five minutes of the test with substantial deviations (Zr2K temperatures greater than SS2N) during the subsequent periods of substantial heater failures."[93]

Discussing criticisms of the BWR-FLECHT tests, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The first complaint [of 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].[94]

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].[95]

General Electric ("GE") argued that the exothermic metal-water reactions were insignificant in the thermal response of the Zircaloy heater rods. Regarding this issue, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> Attempts by GE to show that [metal-water] reactions were insignificant in the thermal response of the rods were not overly convincing since they did

[93] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-18.
[94] *Id.*, pp. A8-2, A8-6.
[95] *Id.*, p. A8-6.

> not evaluate actual dynamic heat rate inputs but depended instead upon arbitrarily time averaged heat inputs over arbitrary time intervals...[96] Gross estimates were made of the total energy contributed to the thermal transient through the [metal-water] reaction of 1/4 B/inch of cladding length (based upon the maximum observed depth of ZrO_2 penetration for the Zr2K experiment of 1.8 mils). This was compared with a design total delivered decay power to the center of the maximum peaked rod over the 24 minute spray cooling transient of 29.7 B/inch (14.5 B/inch over the first 10 minutes). Thus, GE inferred the total [metal-water] reaction to be 5-10 percent of the decay energy depending upon which of the two time periods was used in the estimation. They acknowledge that the rate of [metal-water reaction] energy addition is more significant than the comparisons with [the] total energy shown above, but state that rate information cannot be obtained from the Zr2K data. Irrespective of the validity of this observation, it seems that comparisons with rod input energy increments taken over 10 to 24 minute intervals are too insensitive to be adequate indications of the significance of the [metal-water reaction] energy contribution. No feeling of confidence is gained that [metal-water] reactions were unimportant as a result of this GE analysis. However, the case for [metal-water reaction] induced thermal runaway in the Zr2K test is equally weak.[97]

First, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies, it is clear that GE's claim that the metal-water reactions were insignificant during the Zr2K test is erroneous. For example, the CORA experiments were conducted with electrically heated bundles of Zircaloy fuel rod simulators—like the Zr2K test—and, as a result of the exothermic Zircaloy-water reaction, "in the CORA test facility, [cladding] temperature escalation start[ed] between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec."[98] Furthermore, during the escalation phase of the CORA experiments, the percentage of oxidation energy from the exothermic Zircaloy-water reaction was

---

[96] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, Appendix A.

[97] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-18, A8-19.

[98] 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," p. 83.

generally between 30 and 40%, and in some cases was as high as 45%,[99] of the total energy input.[100]

So during the Zr2K test it is highly probable that—like the CORA experiments—the energy from the exothermic Zircaloy-water reaction was between 30 and 40% of the total energy input,[101] not between 5 and 10% as GE estimated. (It is noteworthy that GE "acknowledge[d] that the rate of [metal-water reaction] energy addition [was] more significant than the[ir] comparisons with [the] total energy…but state[d] that rate information [could not] be obtained from the Zr2K data."[102])

Second, when taking into account data from the CORA experiments and other severe fuel damage experiments, it is highly probable that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI…implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[103, 104] It is significant that "in the CORA test facility, [cladding] temperature escalation start[ed] between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec:"[105] "a rapid [cladding]

---

[99] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, 2008, p. 7.

[100] *Id.*, p. 5.

[101] See Appendix E Table 10. Zircaloy Oxidation, Energy Release, and Hydrogen Production during Various CORA Tests, which depicts percentages of oxidation energy during various CORA tests.

[102] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-19.

[103] Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, p. 5.63.

[104] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-24.

[105] 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," p. 83.

temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[106]

Furthermore, the graphs of "Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies"[107] and "Analysis of Zr2K Thermal Response"[108] depict thermocouple measurements taken during the Zr2K test that resemble thermocouple measurements taken during severe fuel damage experiments: the graphs depict temperature excursions that began when cladding temperatures reached between approximately 2100 and 2200°F. The graphs depict cladding-temperature values at separate points in approximately 20-second intervals; in some cases the temperature increases by several hundred degrees Fahrenheit within approximately 20 seconds, indicating the onset of temperature excursions, at rates greater than 10°K/sec (see Appendix F Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies and Figure A8.10 Analysis of Zr2K Thermal Response).

It is significant that GE concluded that the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test were not valid. GE stated "that the 'erratic thermocouple outputs do not represent actual cladding temperatures, but are the result of equipment malfunctions'[109] associated with the Zr2K test."[110] However, when taking into account data from the CORA experiments and other severe fuel damage

---

[106] F. E. Panisko, N. J. Lombardo, "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," p. 282.

[107] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-25; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13197, June 1971, Figures A-11 and A-12, as the source of this information.

[108] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Figure 12, as the source of this information.

[109] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Appendix D, p. 107.

[110] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-24, A8-27.

experiments conducted with Zircaloy assemblies it is highly probable that GE's claim that the thermocouple measurements did not represent actual cladding temperatures is erroneous; after all, the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test resemble thermocouple measurements of cladding-temperature excursions taken during severe fuel damage experiments.

In its analysis of the cladding temperature excursion that occurred during the Zr2K test, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> One of the more difficult aspects of evaluation of Zr2K test results is associated with the fundamental data for the tests, the recorded thermocouple...responses. *GE has been very liberal with their accreditation of observed [thermocouple] responses as erratic.* However, several proffered examples of erratic response seem to show well defined inter-rod correlations. Under such circumstances, "unexplained" might be a better description for the observed [thermocouple] behavior than "erratic" [emphasis added].[111]

Discussing the "well defined inter-rod correlations"[112] that occurred during "the extreme temperature excursion,"[113] "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> A rigorously thorough analysis of the Zr2K thermal response measurements is beyond the scope of this report. It should be noted, however, that the recorded temperatures of rod 16, which developed the first electrical anomaly after the official start of the test, were almost identical to those of rod 24, which was given credit for the maximum temperature measurement. The intra- and inter-rod temperature measurements for rod 16 and its neighbors show consistent correlations over the first two minutes of the transient, in spite of the current anomaly being experienced by the rod (which started essentially at the beginning of the thermal transient test period and lasted for nearly six minutes). Between 2 and 3 minutes after transient initiation, however, thermocouples...on rod 16 indicate an apparent sharp temperature rise. Because of the anomalous electrical activity of rod 16 at this time, experimental analysts have been inclined to discount this [thermocouple] response as anomalous also. *However, it is interesting to note that the extreme temperature excursion... (adjacent to rod 16) occurred at the same time the rod 16 [thermocouple] excursion occurred and is matched*

[111] *Id.*, p. A8-19.
[112] *Id.*
[113] *Id.*, p. A8-21.

> *by [the] nearly identical temperature excursion in rod 9, the other rod diametrically adjacent to rod 16. Moreover, it seems entirely too coincidental that temperature turnaround should be achieved in rod 24 at essentially the same time that the actual failure (rod current going to zero) for both rods 16 and 24 occurred.* Under those circumstances, it does not seem surprising that rod 17, still being driven by "normal" electric current and in direct view of the three hottest rods in the test (rods 16, 23. and 24) should then become the highest temperature rod for most of [the] remaining significant portion of the temperature transient. During this period, rods 17 and 23 both underwent electrical anomalies in which excessive currents were delivered to them. It was not until the current to both of these rods actually went to zero, approximately 12 minutes after the thermal transient began, that rod 17 relinquished its role as the highest temperature rod for the test.
>
> The relationships described above seem to indicate a systematic correlation between the electrical anomalies of the "failed" rods and temperature extremes for the bundle [emphasis added].[114]

So, as "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states, the observed thermocouple measurements were not erratic. And, as stated above, the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test resemble thermocouple measurements of cladding-temperature excursions taken during severe fuel damage experiments.

In the conclusion of its analysis of the cladding temperature excursion that occurred during the Zr2K test, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> Based upon analysis of the material presented, it appears unquestionable that the [thermocouple] response was badly affected by short circuits and equipment malfunction. The net result is that it is not possible to certify that [metal-water] reactions were insignificant in the measured thermal transient, but the case for near "thermal runaway" proposed by the CNI is also unconvincing. It is probable that most of the dramatic [thermocouple] slope changes, as well as several of the other [thermocouple] aberrations associated with the test, were short-circuit induced rather than [metal-water] reactions. *However, more results seem to be systematically correlatable between rods [than] the GE test analysis is willing to concede. This leads to uncertainty over the proper interpretation of [the] results. A more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable* [emphasis added].[115]

---

[114] *Id.*, pp. A8-21, A8-23.
[115] *Id.*, p. A8-27.

Indeed, "a more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable."[116] However, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies more than a decade after the Zr2K test, it is clear that GE's claim that the metal-water reactions were insignificant during the Zr2K test is erroneous and that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI...implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[117, 118]

Of course, in the event of an actual LOCA, the energy from decay heating would not suddenly terminate if cladding temperatures were to reach the same temperatures that caused the heaters to fail during the Zr2K test. And during the Zr2K test it is highly probable that—like the CORA experiments—the energy from the exothermic Zircaloy-water reaction was between 30 and 40% of the total energy input, not between 5 and 10% as GE estimated. Additionally, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies more than a decade after the Zr2K test, it is clear that the Zr2K test—which had cladding-temperature increases of several hundred degrees Fahrenheit within approximately 20 seconds, at some locations of its assembly, after cladding temperatures reached between approximately 2100 and 2200°F—incurred an autocatalytic oxidation reaction.

Furthermore, it is significant that in the AEC's ECCS rulemaking hearing, Dr. Roger Griebe, the Aerojet project engineer for BWR-FLECHT, testified that "there is *no*

[116] *Id.*
[117] Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, p. 5.63.
[118] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-24.

convincing proof available from [Zr2K] test data to demonstrate that [a] near-thermal runaway [condition] definitely did not exist [in the Zr2K test] [emphasis not added].[119]

(In "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," the BWR-FLECHT Zr2K test is termed "Test ZR-2;" therefore, in the passages below the BWR-FLECHT Zr2K test will be termed "Test ZR-2.")

Regarding Dr. Roger Griebe's testimony, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> CNI's direct testimony concluded that a near thermal runaway condition existed in Test ZR-2.[120] It is of compelling importance that Roger Griebe, the [Aerojet] project engineer for BWR-FLECHT, stated a similar interpretation of this test, which they submitted to [General Electric ("GE")], and Griebe testified, there is *no* convincing proof available from ZR-2 test data to demonstrate that this near-thermal runaway definitely did not exist [emphasis not added].[121, 122]

And regarding Aerojet internal memoranda that provide commentary on the BWR-FLECHT program consistent with that presented by CNI, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] internal memoranda provide commentary on the BWR-FLECHT program quite consistent with that presented by CNI. Thus, for example, 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

---

[119] 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.
[120] Daniel F. Ford and Henry. W. Kendall, Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Volume I, Direct Testimony prepared in behalf of the Consolidated National Intervenors, USAEC Docket RM-50-l, 23 March 1972, p. 5.63.
[121] Official Transcript of the AEC's Emergency Core Cooling Systems Rulemaking Hearing, pp. 7138-7139.
[122] 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.

> 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]."[123]

Additionally, regarding Dr. Griebe's review of the data presented by GE regarding the maximum cladding history of ZR-2, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> It is important to note that GE's interpretation of Test ZR-2 is based on a bundle maximum cladding temperature curve that CNI contended in its direct testimony constituted false reporting of the test data. The basis that GE asserts for the correctness of its reported maximum temperature curve are the thermocouple data available from Sanborn strip recorders that were used by GE. It is important to note that the GE report published on Test ZR-2 (Exhibit 133) does not present any reporting of the strip data. Moreover, the Board turned down CNI's request for discovery that the data be made available. Finally, Dr. Roger Griebe, who had the Sanborn tapes available, was addressed an interrogatory by CNI concerning what the test data established to be the true maximum cladding temperature curve for Test ZR-2. Dr. Griebe's answer, which presented detailed documentation from the Sanborn strip data, completely confirmed CNI's position that the maximum cladding temperature curve used in GE analysis of ZR-2 is false and that the much more severe temperature history from Exhibit 125 is, in fact, the correct data for Test ZR-2, as CNI had asserted.
>
> Dr. Griebe's review of the data presented by GE regarding the maximum cladding history of ZR-2 provides quite precise technical support for his testimony earlier that GE "tremendously slanted" BWR-FLECHT data "towards the lower temperatures and towards the interpretation GE obviously presented in their report" (Tr. 7127). ...
>
> CNI's interpretation of both the correct maximum cladding temperature curve and their more reasonable assessment of the test was concurred in by Dr. Griebe. Yet the Regulatory Staff provides *no commentary whatsoever on either the issue of the correct temperature curve for ZR-2 or the issue of the existence of a near thermal runaway condition* [emphasis added].[124]

[123] *Id.*
[124] *Id.*, pp. 5.12, 5.14.

Indeed, it is unfortunate that the AEC Regulatory Staff did not provide commentary "on either the issue of the correct temperature curve for ZR-2 or the issue of the existence of a near thermal runaway condition [in the ZR-2 test]."[125]

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 of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[126]

Petitioner, would add that such a new BWR-FLECHT program would have to be conduced with Zircaloy fuel assemblies. It would also be necessary that the PCTs of such tests exceeded those of the PWR Thermal-Hydraulic Experiment 1 ("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[127]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

---

[125] *Id.*

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

[127] 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. 3-3.

III. CONCLUSION

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: March 15, 2010

Appendix A Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation and Fig. 15 Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM[1]

[1] 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, pp. 143-144.

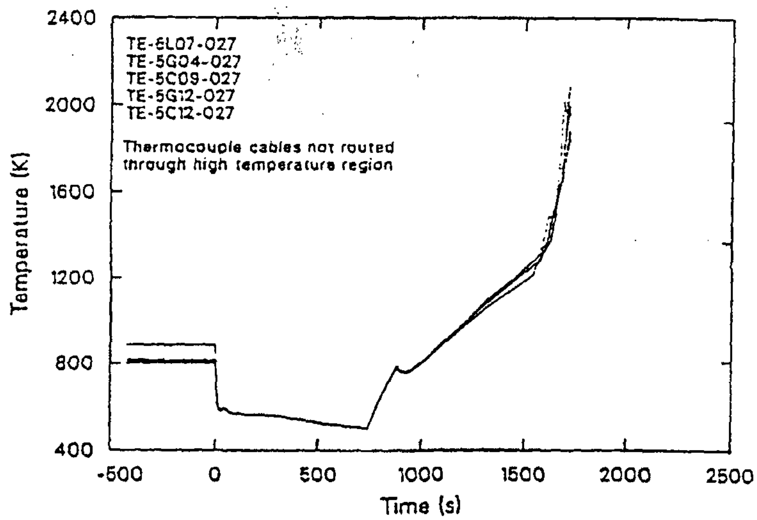


Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

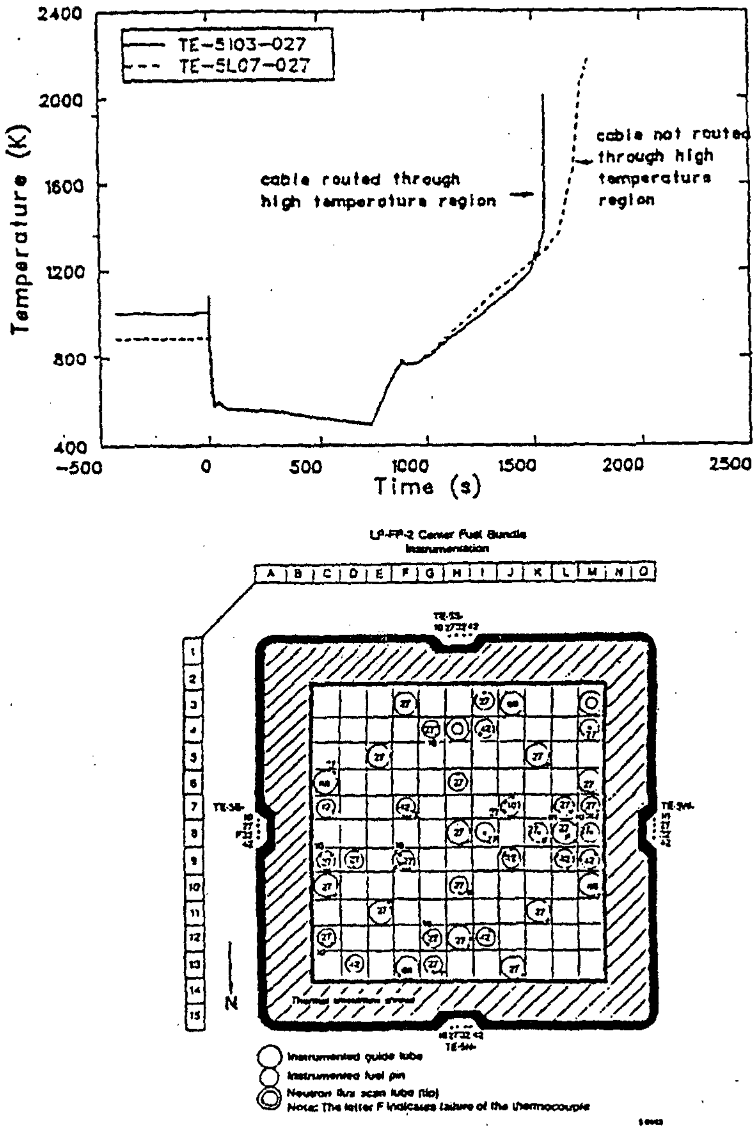
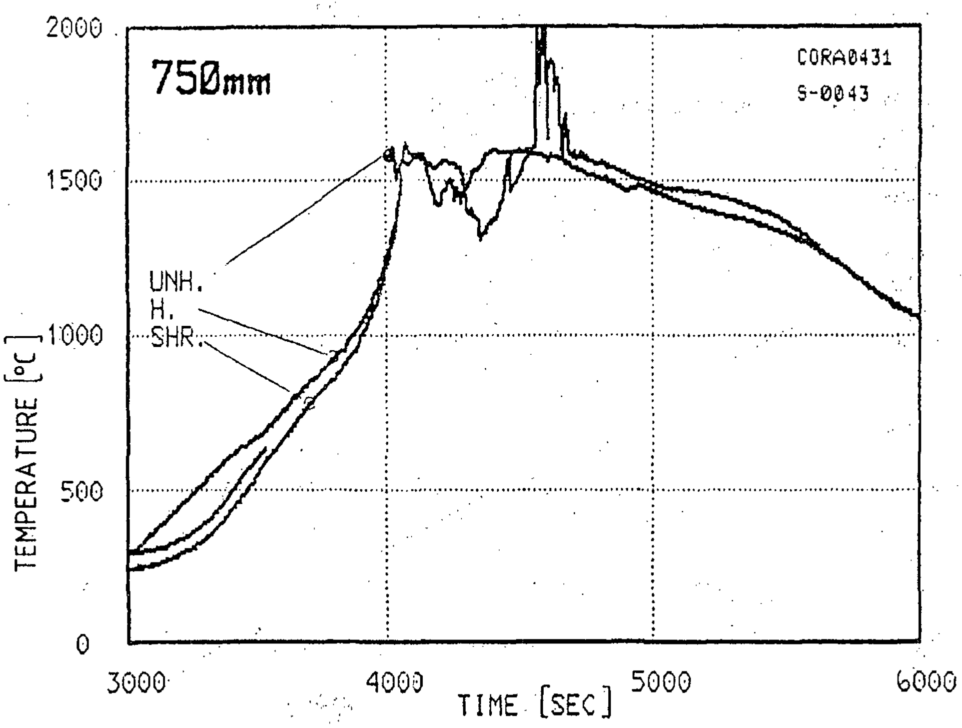


Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

Appendix B Fig. 12. Temperatures during Test CORA-2 at [550] mm and 750 mm Elevation and Fig. 13. Temperatures Measured during Test CORA-3 at 450 mm and 550 mm Elevation[2]

[2] 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, pp. 79-80.



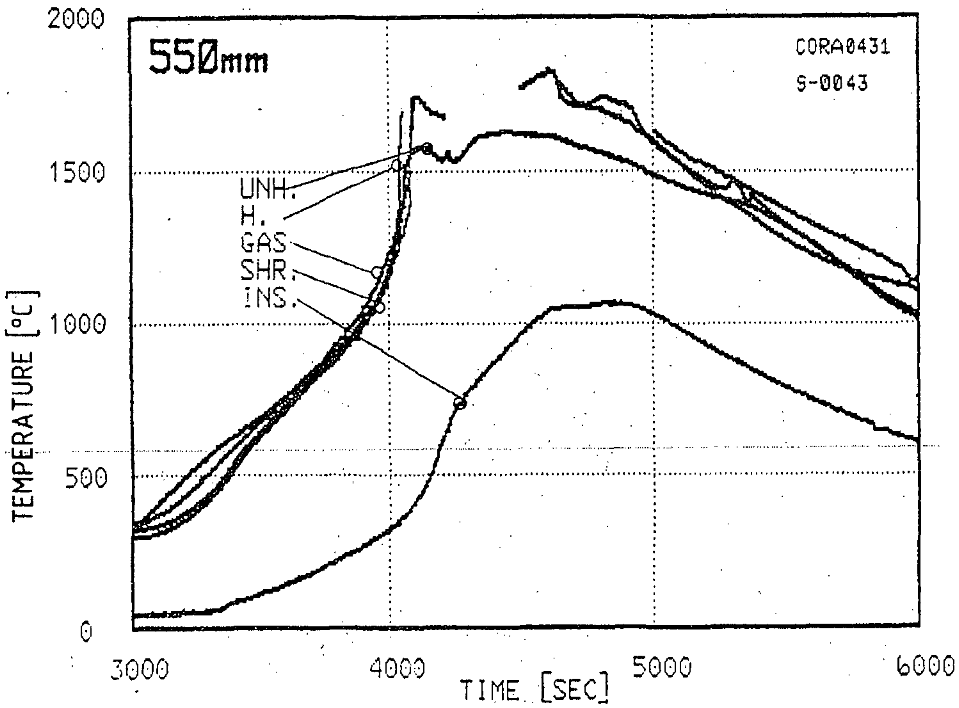


Fig. 12. Temperatures during test CORA-2 at 500 mm and 750 mm elevation. Temperatures of heated (H) and unheated rod (UNH), atmosphere (gas), shroud (SHR), and outer surface of shroud insulation (INS)

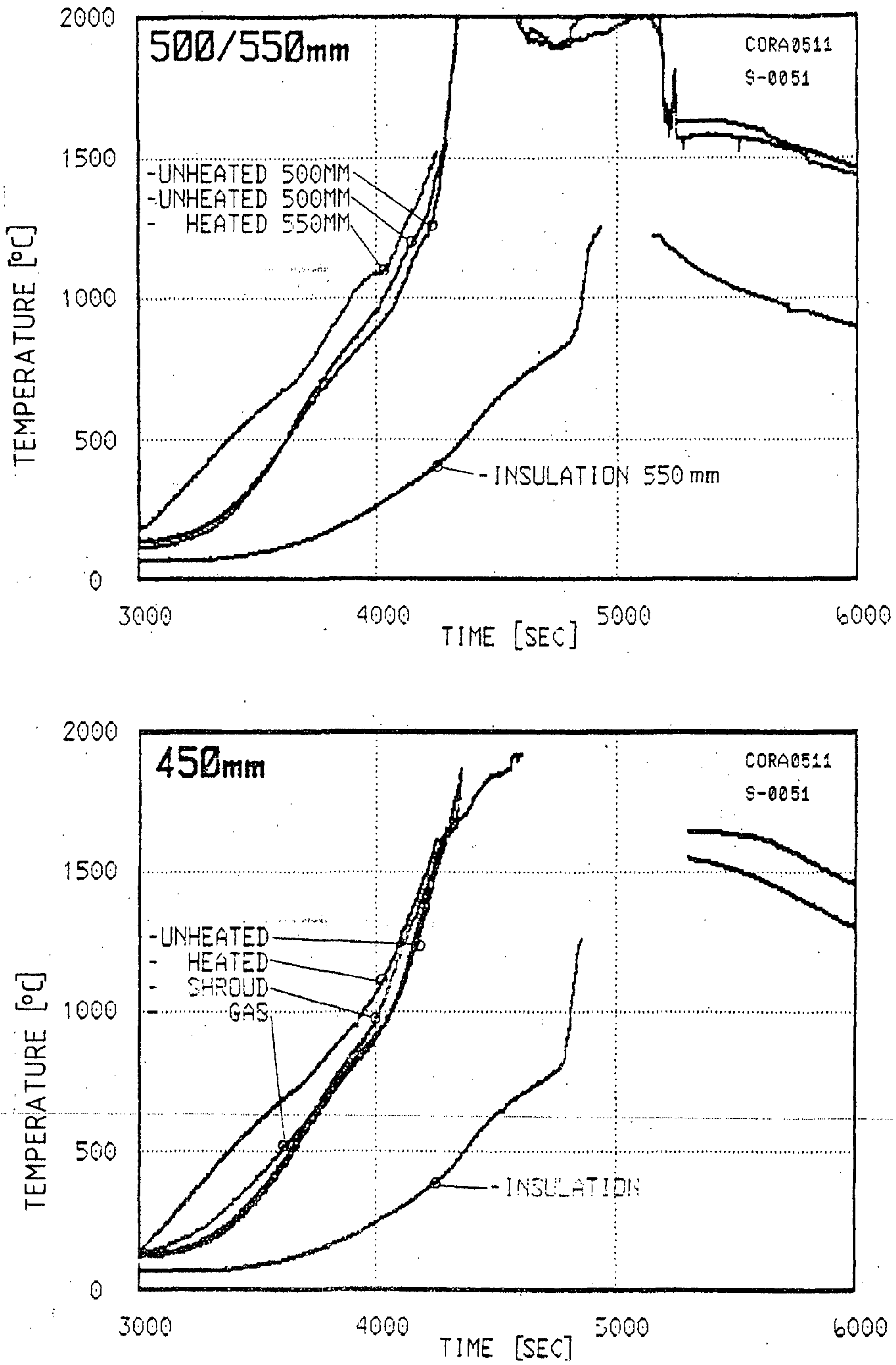


Fig. 13. Temperatures measured during test CORA-3 at 450 mm and 550 mm elevation

Appendix C Figure 15. Temperatures of Unheated Rods and Power History of CORA-5, Figure 16. Temperatures of Unheated Rods during CORA-12, Figure 17. Temperatures at Different Elevations during CORA-15, Figure 18. Temperatures of Unheated Rods during CORA-9, Figure 19. CORA-7; Temperatures at Elevations Given (750 mm), and Figure 20. Temperatures of Guide Tube and Absorber Rod during Test CORA-5[3]

[3] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, pp. 75-80.

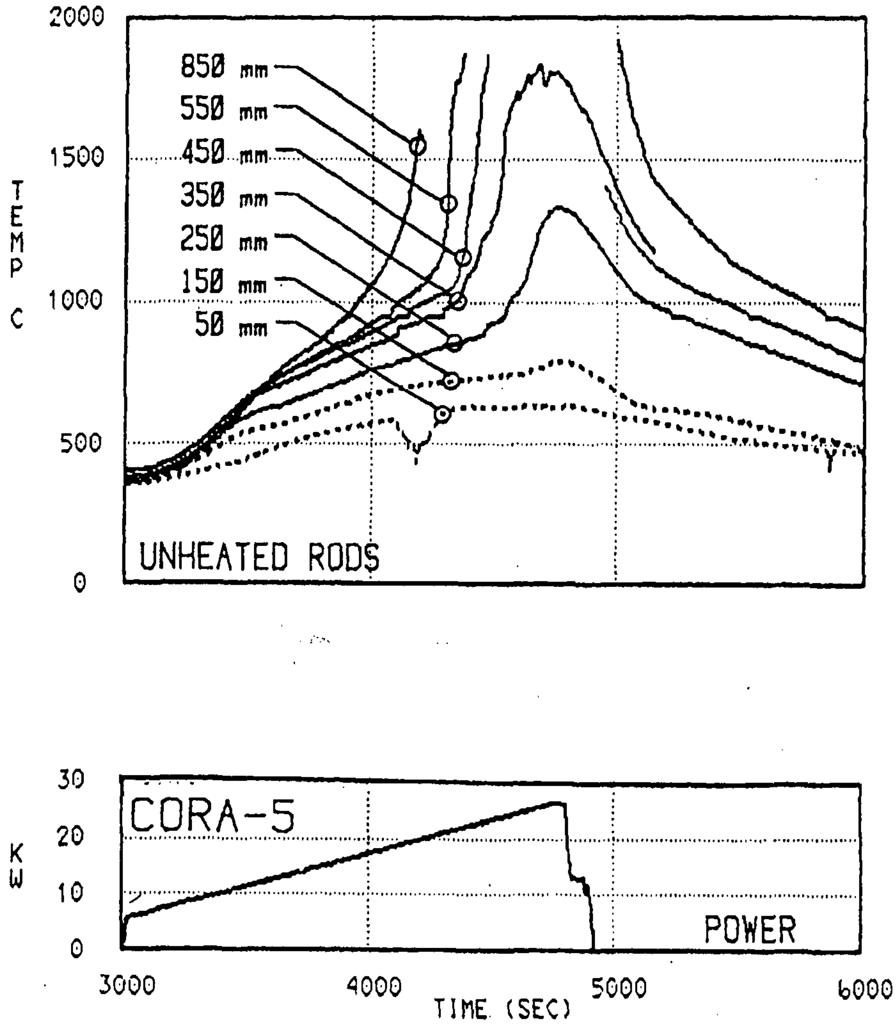


Fig.15: Temperatures of unheated rods and power history of CORA-5

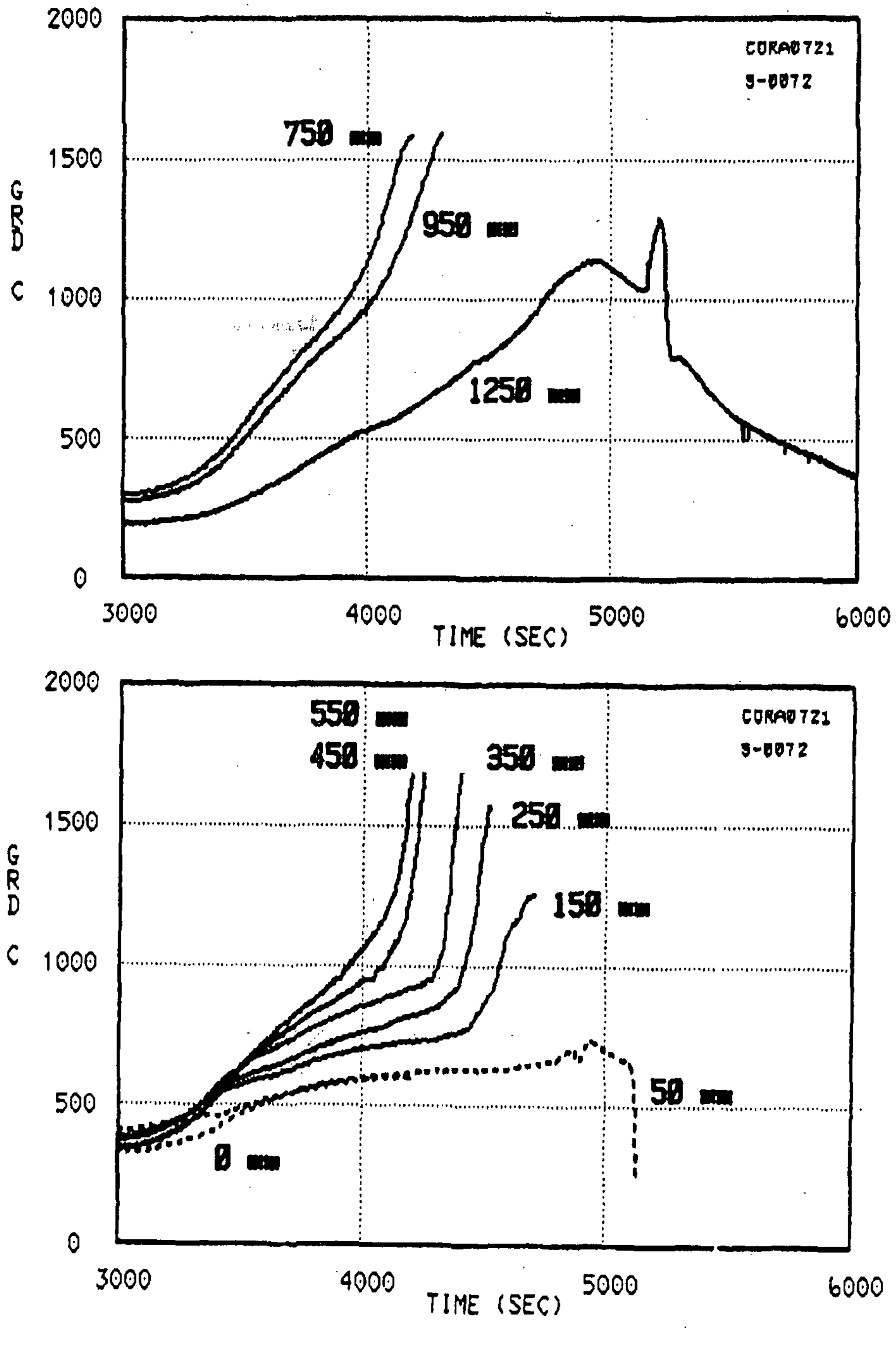


Fig.16: Temperatures of unheated rods during CORA-12

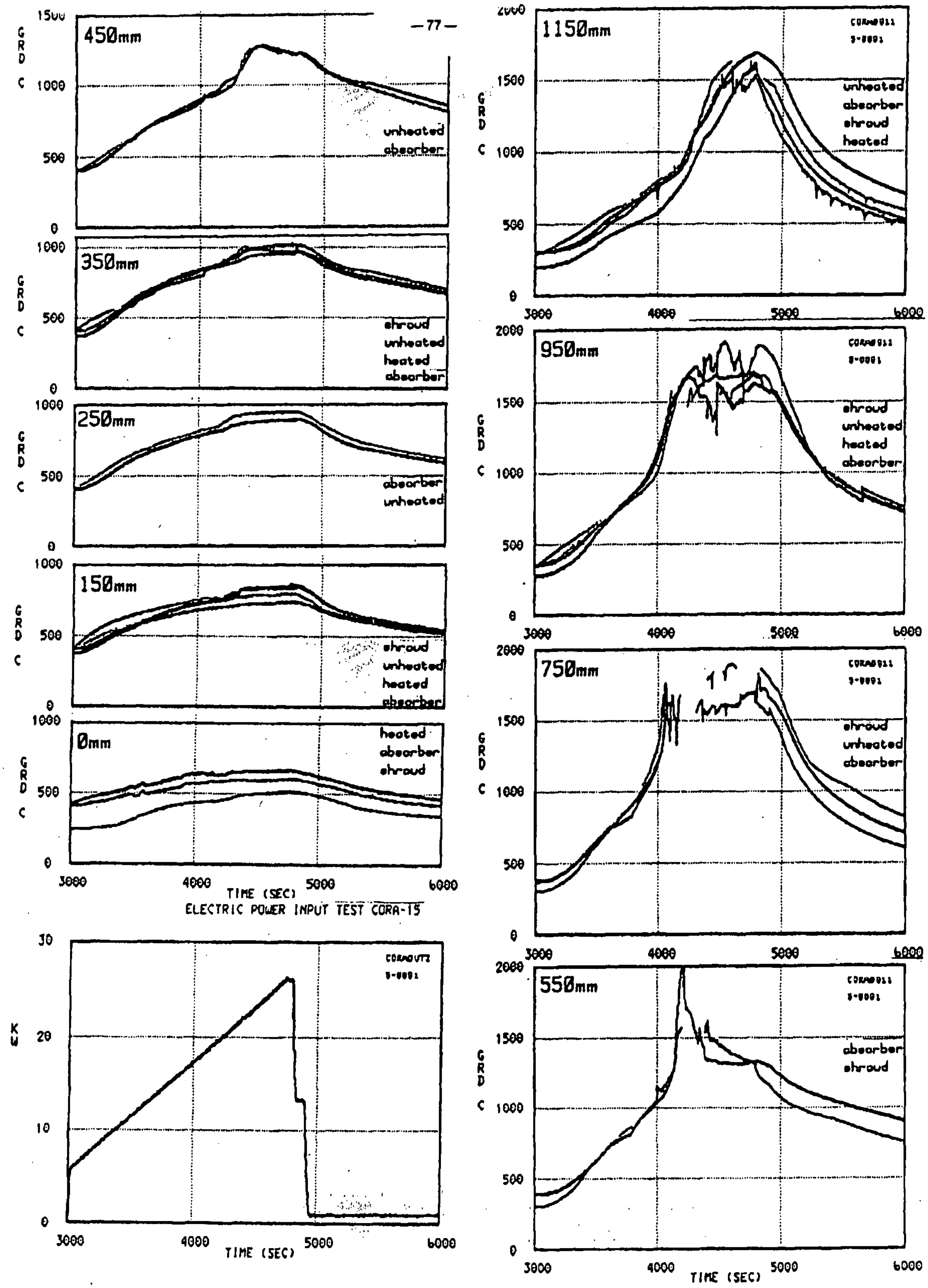


Fig.17: Temperatures at different elevations during CORA-15

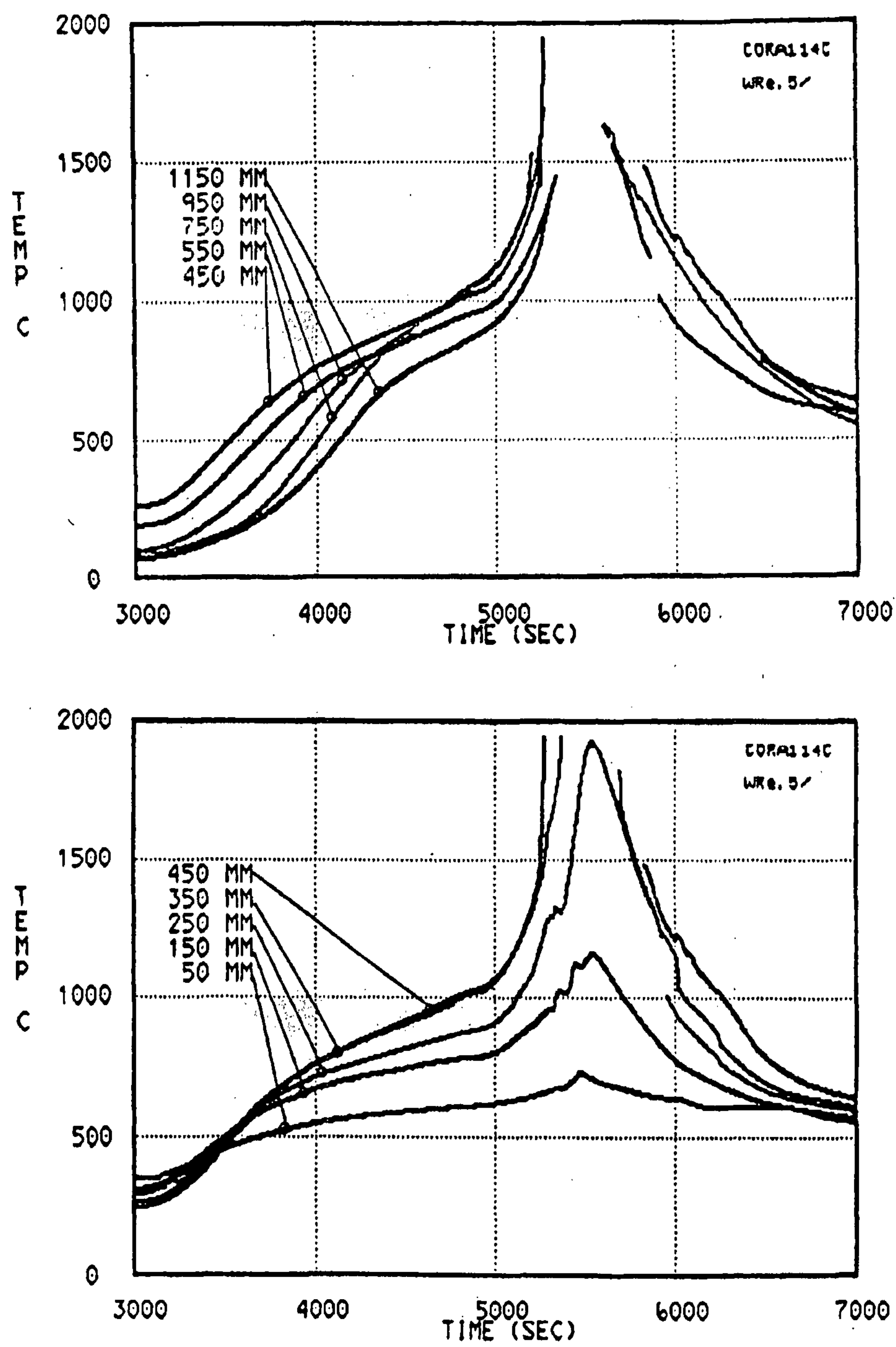
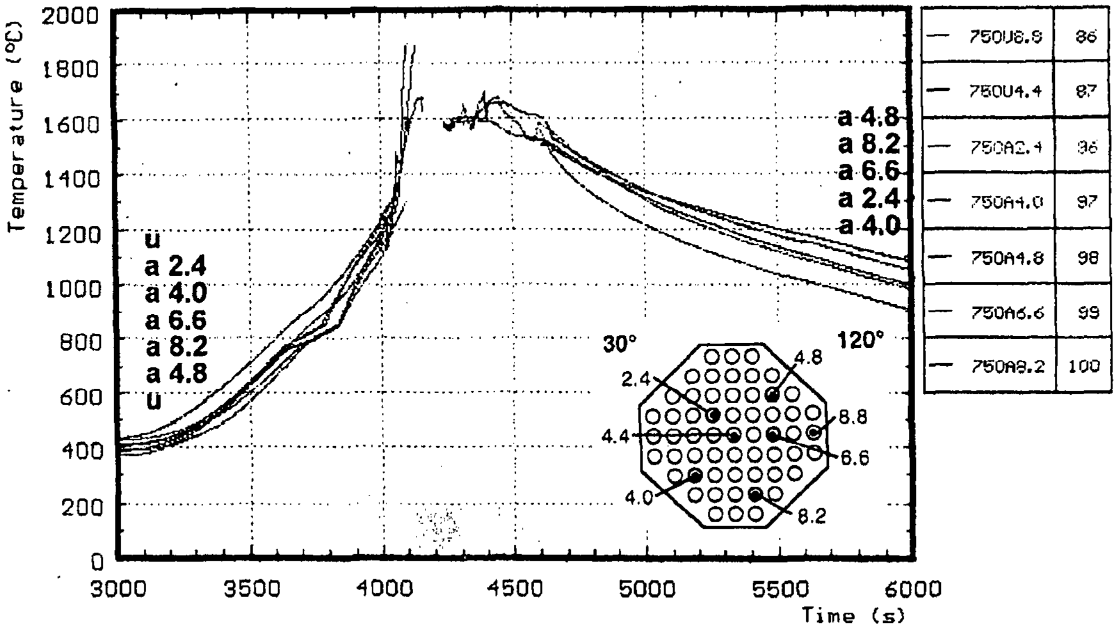
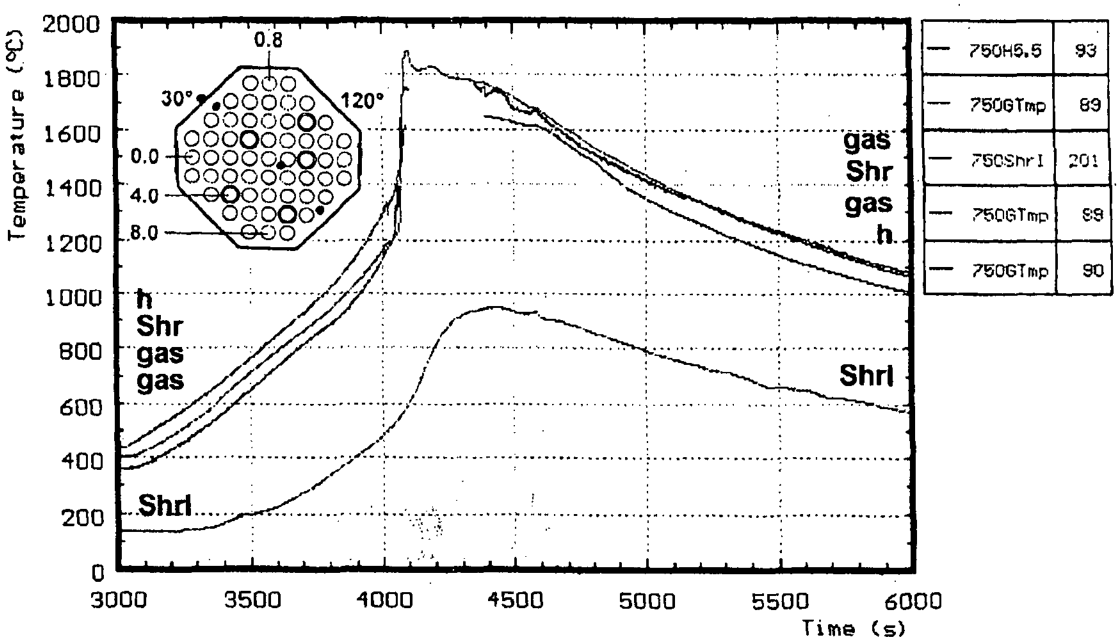


Fig.18: Temperatures of unheated rods during CORA-9





h : heated rods
u : unheated rods
a : in absorber

shr : outer side of shroud
shrl : on shroud insulation
gas : gas temperature

Fig. 19: CORA-7; Temperatures at elevations given (750 mm)

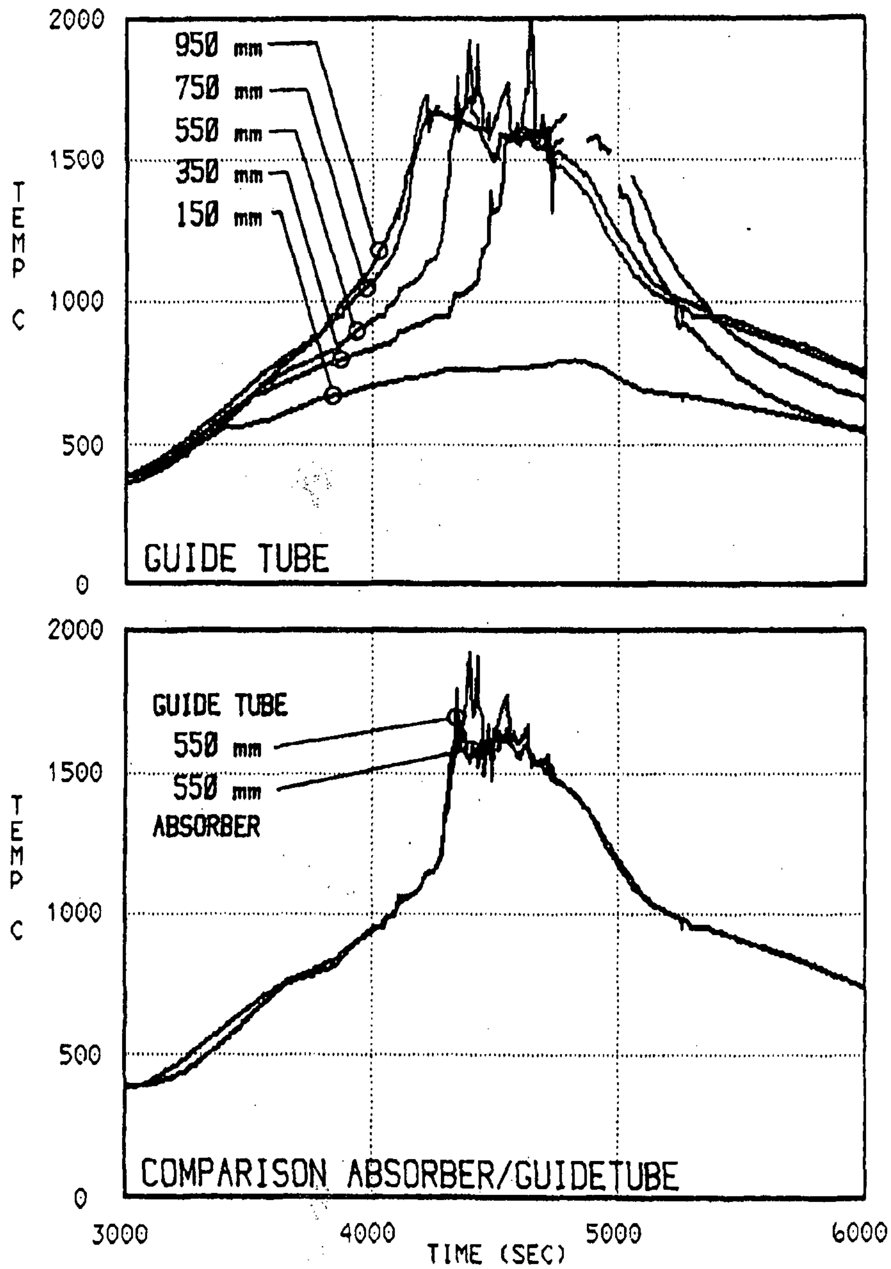


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

Appendix D Figure 37. Temperatures of the Heated Rods (CORA-13) and Figure 39. Temperatures of the Unheated Rods (CORA-13)[4]

---

[4] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, Kernforschungszentrum Karlsruhe, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," 1993, pp. 76, 78.

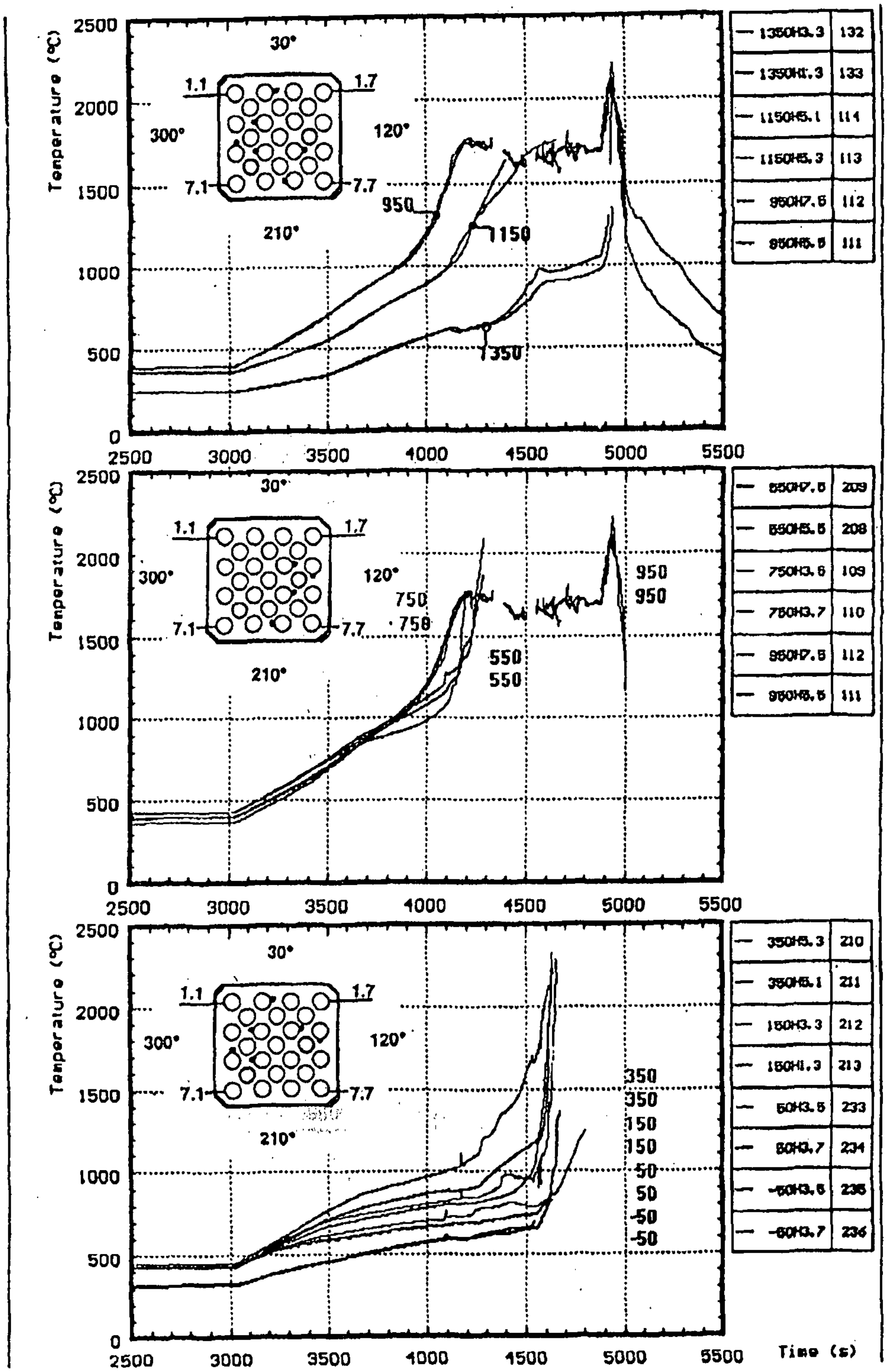


Fig. 37: Temperatures of the heated rods (CORA-13)

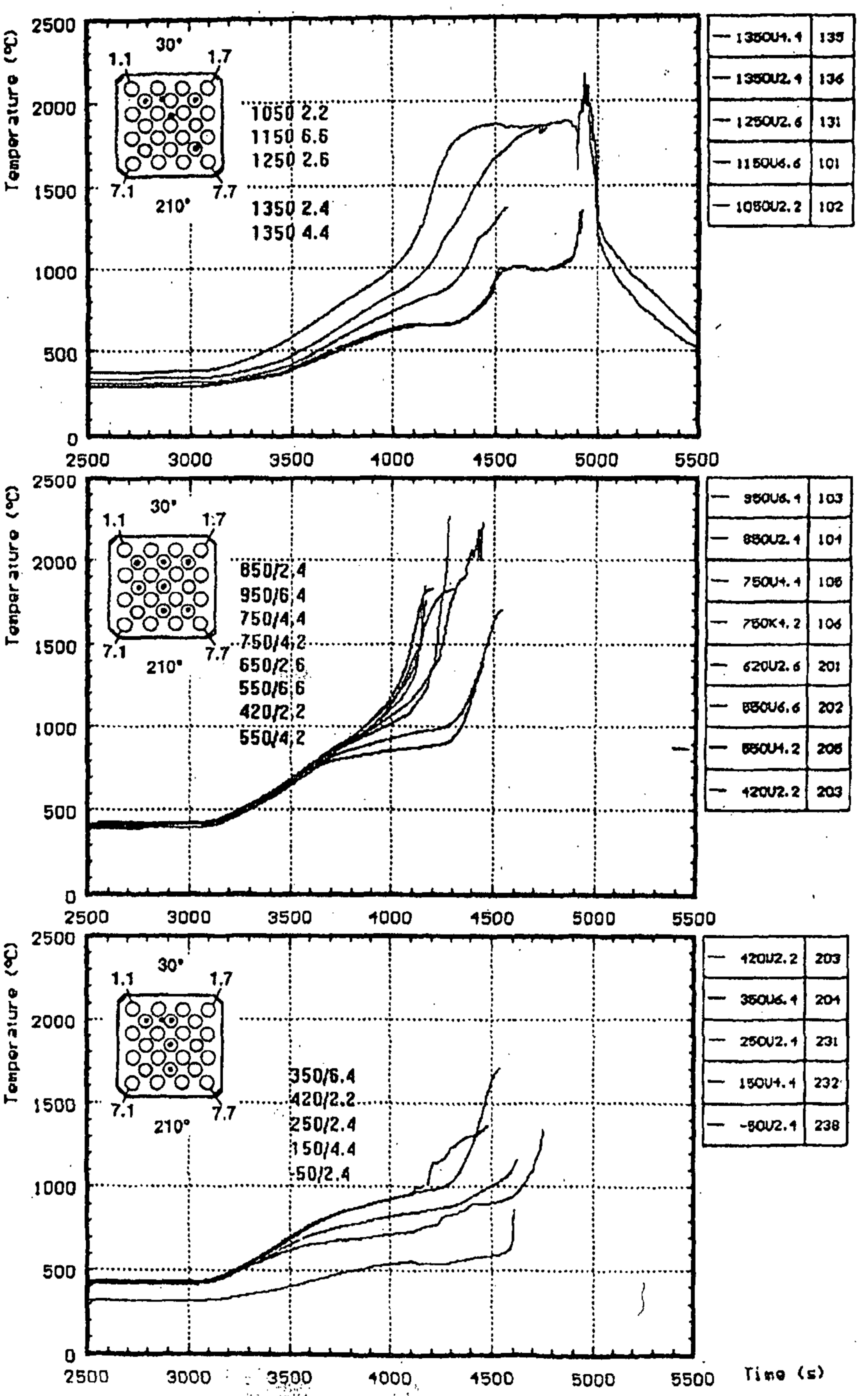


Fig. 39: Temperatures of the unheated rods (CORA-13)

Appendix E Table 10. Zircaloy Oxidation, Energy Release, and Hydrogen Production during Various CORA Tests[5]

[5] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, p. 38.

Table 10: Zircaloy oxidation, energy release, and hydrogen production during various CORA tests

Test	Steam flow [g/s]	Total H_2 production [g]	Oxidation energy [MJ]	Percentage of oxidation energy [a] [%]	Total Zr oxidation [b] [%]	Test time at T › 1400°C [s]	Fraction of H_2O consumed [%]
CORA-15	6	180	27.4	45	74	~ 1000	27
CORA-9	6	159	24.2	30	48	~ 800	30
CORA-7	12	114	17.3	34	28	~ 500	17

[a] Percentage of total energy, i.e. chemical reaction power and electric power input

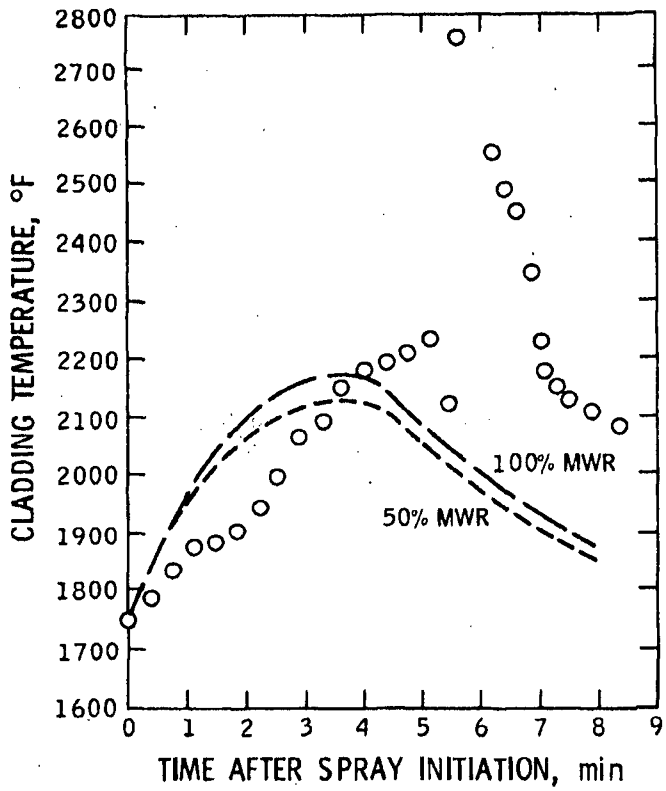
[b] Percentage referred to bundle length of 1.2 m;

Appendix F Figure A8.9. Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[6] and Figure A8.10. Analysis of Zr2K Thermal Response[7]

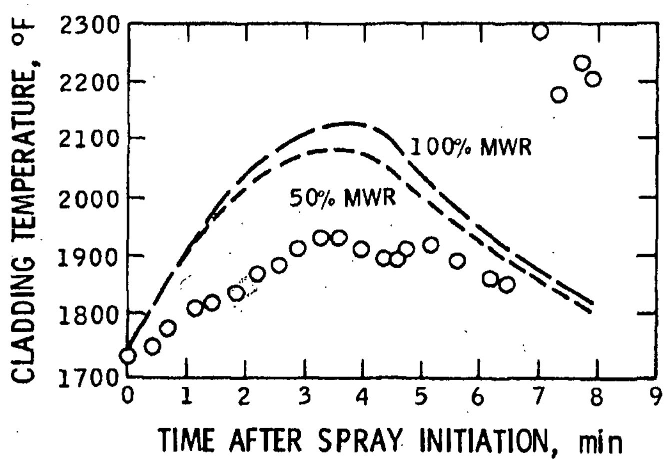
---

[6] 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, p. A8-25; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13197, June 1971, Figures A-11 and A-12, as the source of this information.

[7] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, Figure 12, as the source of this information.



Bundle Zr2K Rod 24 Midplane Thermal Response Prediction

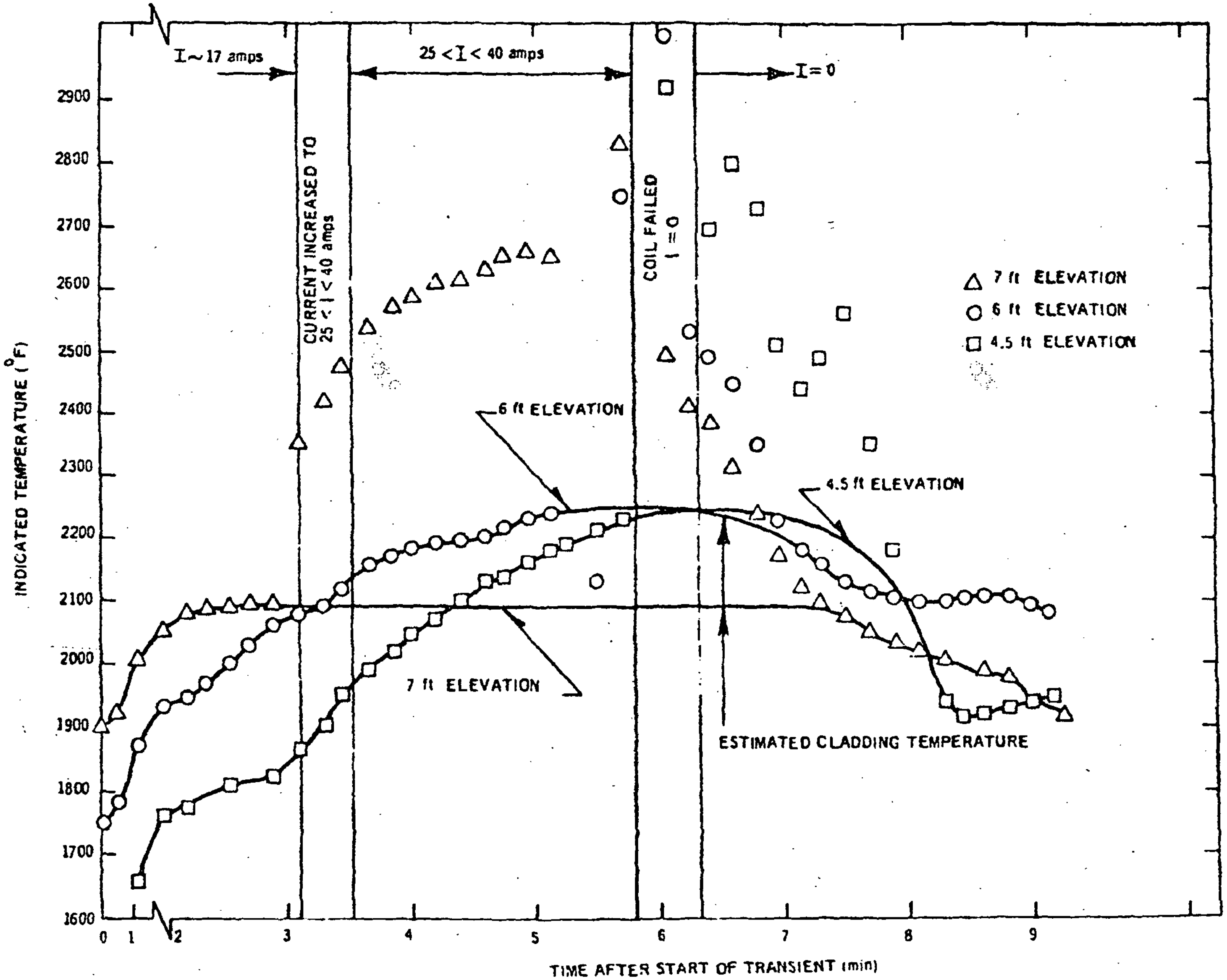


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies (After Figures A-11 and A-12 from 52 by permission.)

Figure A8.10

Analysis of Zr2K Thermal Response



(After Figure 12, 54, by permission.)

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Sunday, March 21, 2010 9:27 PM
To: Rulemaking Comments
Subject: NRC-2009-0554
Attachments: Petition Comments 2010.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is a cover letter and my response, dated March 15, 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.

Sincerely,

Mark Leyse

Received: from mail1.nrc.gov (148.184.176.41) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Sun, 21 Mar 2010
21:27:01 -0400
X-Ironport-ID: mail1
X-SBRS: 4.5
X-MID: 12506254
X-fn: Petition Comments 2010.pdf
X-IronPort-AV: E=Sophos;i="4.51,284,1267419600";
d="pdf'?scan'208";a="12506254"
Received: from mail-pw0-f41.google.com ([209.85.160.41]) by mail1.nrc.gov
with ESMTP; 21 Mar 2010 21:26:53 -0400
Received: by pwi4 with SMTP id 4so3327122pwi.14 for
<rulemaking.comments@nrc.gov>; Sun, 21 Mar 2010 18:26:52 -0700 (PDT)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:date:message-id:subject
:from:to:content-type;
bh=FgA2Et+toJPpEzWhRkL9rglg+I1CqfsaIPLJmNC5TH4=;
b=unjpCpdm4P6tdZtLEJBlXfCzuVELkoALZOv+4wSF2O/VgXPdSJDTwiTCOtArqeWh5L

jWbZMkEY7mt3XtmVLzWZJl1tMC3ZOZg1dQDgyhnnI/YoLmOdGRzJhdAJNUFYhXtM2dGu
qkpLEwEKD06Zg1tGQDaGmG0WYpfiLXFq0yd8o=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:content-type;
b=M9vhNbccqOyP7qwRvpx919eYCFsPUFz+2yD7hKcjLMf/V13zF52rOa0jtKG2t2lgXm
OLGFf7csb5n52v23Yr8ipseSTIdG03aQPFjtURwD+Q8sBpbuZf1XUNjt1ys9M4W1ibpu
Y49IPwIctpGirzeRHBuZhzW1GEkLqYOMeWnSE=
MIME-Version: 1.0
Received: by 10.143.194.5 with SMTP id w5mr475348wfp.155.1269221212576; Sun,
21 Mar 2010 18:26:52 -0700 (PDT)
Date: Sun, 21 Mar 2010 21:26:52 -0400
Message-ID: <edacd5761003211826o31829f2fh475ce9a8a0b0481d@mail.gmail.com>
Subject: NRC-2009-0554
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>
Content-Type: multipart/mixed; boundary="000e0cd605108468050482599766"
Return-Path: markleyse@gmail.com

Submission ID 3
Aladar Stolmar
ML100830501

From: Aladar Stolmar [astolmar@gmail.com]
Sent: Wednesday, March 24, 2010 2:59 AM
To: Rulemaking Comments
Subject: Docket ID NRC-2009-0554
Attachments: Docket ID NRC.pdf

Docket ID NRC-2009-0554

Comments from Aladár Stolmár, HU-3021 Lőrinci, Szabadság tér 3, Hungary

Phone: +36-20-404-2713, email: astolmar@gmail.com

Background: I received my nuclear engineering degree (BS) at the Moscow Power (Engineering) Institute in 1973 and worked on the Nuclear Power Plant Paks project in Hungary. In 1985, I immigrated to the USA and worked as a consultant for Westinghouse on the Chernobyl-4 accident investigation, assigned to A. David Rossin, Assistant Secretary of Energy for Nuclear Energy. I also worked for Westinghouse on the AP-600 design and as a senior engineer in the Probabilistic Risk Assessment group investigating processes in accident progression for several NPP worldwide. Currently, I'm back in Hungary and involved in the expansion of NPP Paks. Already, at Westinghouse, I raised a Safety Concern over the misrepresentation of cladding heat-up and ignition in the predecessor codes of Relap, which I still have not seen corrected.

In fact, the correct representation of the cladding condition (locations without any oxide) and the correct representation of the temperature distribution in the steam cooling regime results in an ignition at a much lower temperature than it is predicted in the Relap-5 computer model. I mean, the **prediction** for a steam cooled environment temperature by the code could be as low as 1000 K and the **real, factual** local temperature could already exceed the ignition condition for the Zirconium fire in the steam. And, in fact, the ignition of the Zirconium fire will result-in a non-extinguishable firestorm in the core, as occurred in the TMI-2 core, the Chernobyl-4 core and the Paks-2 refueling pond fuel bundle washing vessel, and had been indicated by the experiments cited by Mark Leyse and others I cite below as well. Until we have a much more detailed experimental investigation of real conditions, I suggest to mandate the prevention of steam bubble formation in the core and in nuclear fuel containing vessels, and I suggest to regulate the containment to consider the maximum possible Hydrogen generation, which is equal to the reaction of the entire Zirconium inventory. (The more detailed investigation also may turn out to require the same strict, conservative limitations.)

I agree with Mark Leyse that the current 10 CFR 50 regulation series is not conservative, because it does not require the demonstration of the prevention of steam bubble formation in the core, leading to a Zirconium fire in the steam; and the prevention of the destruction of the reactor core as it happened in the TMI-2 and Chernobyl-4 severe reactor accidents, nor the prevention of the destruction of nuclear reactor fuel as it happened in the Paks-2 refueling pond. It is due to the fact that the very rapid development of the ignition condition after the bubble

Template - SECY-067

DS 10

formation in the core is misrepresented, shown by the required codes to be much slower than it is in reality. http://aladar-mychernobyl.blogspot.com/

Furthermore, the current 10 CFR 50 series of regulation is not conservative, because it does not require the demonstration of preservation of the containment surrounding the reactor in the event of the detonation of a Hydrogen-air mixture, calculated from the generated amount of Hydrogen from the Zirconium-steam fire, consuming the entire inventory of Zirconium in the core in a single firestorm event.

http://www.osti.gov/energycitations/servlets/purl/10188341-UMoU6M/native/

FULL-LENGTH HIGH-TEMPERATURE SEVERE FUEL DAMAGE TEST #5

D. D. Lanning at al.

April 1988 – Completion Date

September 1993 – Publication Date

Prepared for

U. S. Nuclear Regulatory Commission

Under U.S. Department of Energy

Contract DE-AC06-76RLO1830

Pacific Northwest Laboratory

Richland, Washington 99352

Reports on page 6:

"TEST RESULTS

Following the uncovering and dryout during the coolant boilaway, the rods heated at a rate of 2 to 5 K/s until peak cladding temperatures of 1700° K were attained, at which time the autocatalytic oxidation reaction resulted in a temperature excursion (at a rate of 10 to 50° K/s) and hydrogen generation. Peak local cladding temperatures are estimated to have exceeded 2600° K, based on information from thermocouples on the outside of the bundle liner.

The high-temperature oxidation reaction began at the 2.4- to 3.04-m elevation and formed a localized burn front that moved quickly downward as far as the 1.2-m elevation and then steadily upward. The burn front reached the top end caps (3.80m) and ceased 15 min. before the end of the test. The oxidation reaction consumed 75% of the total zircaloy or almost 100% of the zircaloy in the path of the burn front. The remaining 25% of the zircaloy was always below or near the bundle water level. The amount of hydrogen generated was 300±30 g, close to the total conversion of the 1.26-g/s make-up coolant flow within the 45-min. high-temperature period. The hydrogen flow fluctuated during the 45-min. high-temperature period in response to similar fluctuations (10% to 20% relative) in the bundle coolant flow. The peak hydrogen flow was 190 mg/s, which corresponded to an oxidation power of 28 kW."

This description is a very clear presentation of ignition and fire of Zirconium in the steam in a steam-starved environment.

http://itu.jrc.ec.europa.eu/uploads/media/Activity_Report_2004.pdf

page 42 "Bundle tomography revealed that a large central cavity was apparent above the corium pool at approximately one-third bundle height. At the top there were remnants of distorted, degraded fuel rods, whereas below the corium pool there was small streams of melt material and debris evident."

page 43. "The differences between the degraded bundle geometries of FPT1 and FPT2 can be explained by the fact that under steam-starved conditions (FPT2) Zircaloy metal melts and relocates at a lower temperature, whereas under oxidising conditions (FPT1) the Zircaloy cladding oxidises to a refractory oxide (ZrO_2) and remains in place until very high temperatures are reached later in the accident."

Similar destruction and relocation of nuclear reactor fuel was observed in the TMI-2 and Chernobyl-4 severe reactor accidents and in the Paks-2 refueling pond reactor fuel washing accident.

The similarities in these tests and accidents are the formation of gaseous (steam) bubbles in the upper regions of fuel bundles, the ignition of Zirconium in the steam and generation of Hydrogen and zirconia (ZrO_2) reaction products in a very intense fire, essentially in a firestorm. Therefore, the conservative regulation shall mandate that the owners and operators of Nuclear Reactors and Reactor Fuel Handling Facilities shall demonstrate that **there will be no dry-out of the fuel bundles in any circumstances**.

Also, in order to prevent the exposure of the public to the harmful consequences of an accident in a reactor, the housing of the reactor **(containment) shall withstand the detonation of the air-Hydrogen mixture with the amount of Hydrogen** calculated from the consumption of the **entire inventory of Zircaloy in the reactor core** or in the entire enclosed in a vessel volume, where such bubble formation is possible.

- There are several reports presenting the same issue as Mark Leyse. The cladding of nuclear fuel made of **Zirconium alloy ignites and burns in the steam**. The same process can be recognized (and should be recognized) as **the common cause** of the TMI-2 and Chernobyl-4 reactor severe accidents and the Paks-2 refueling pond accident. And the regulations in 10 CFR 50 series shall mandate to deal with the real issues and real processes.

Docket ID NRC-2009-0554

Comments from Aladár Stolmár, HU-3021 Lőrinci, Szabadság tér 3, Hungary
Phone: +36-20-404-2713, email: astolmar@gmail.com

Background: I received my nuclear engineering degree (BS) at the Moscow Power (Engineering) Institute in 1973 and worked on the Nuclear Power Plant Paks project in Hungary. In 1985, I immigrated to the USA and worked as a consultant for Westinghouse on the Chernobyl-4 accident investigation, assigned to A. David Rossin, Assistant Secretary of Energy for Nuclear Energy. I also worked for Westinghouse on the AP-600 design and as a senior engineer in the Probabilistic Risk Assessment group investigating processes in accident progression for several NPP worldwide. Currently, I'm back in Hungary and involved in the expansion of NPP Paks. Already, at Westinghouse, I raised a Safety Concern over the misrepresentation of cladding heat-up and ignition in the predecessor codes of Relap, which I still have not seen corrected.

3-1

In fact, the correct representation of the cladding condition (locations without any oxide) and the correct representation of the temperature distribution in the steam cooling regime results in an ignition at a much lower temperature than it is predicted in the Relap-5 computer model. I mean, the **prediction** for a steam cooled environment temperature by the code could be as low as 1000 K and the **real, factual** local temperature could already exceed the ignition condition for the Zirconium fire in the steam. And, in fact, the ignition of the Zirconium fire will result-in a non-extinguishable firestorm in the core, as occurred in the TMI-2 core, the Chernobyl-4 core and the Paks-2 refueling pond fuel bundle washing vessel, and had been indicated by the experiments cited by Mark Leyse and others I cite below as well. Until we have a much more detailed experimental investigation of real conditions, I suggest to mandate the prevention of steam bubble formation in the core and in nuclear fuel containing vessels, and I suggest to regulate the containment to consider the maximum possible Hydrogen generation, which is equal to the reaction of the entire Zirconium inventory. (The more detailed investigation also may turn out to require the same strict, conservative limitations.)

I agree with Mark Leyse that the current 10 CFR 50 regulation series is not conservative, because it does not require the demonstration of the prevention of steam bubble formation in the core, leading to a Zirconium fire in the steam; and the prevention of the destruction of the reactor core as it happened in the TMI-2 and Chernobyl-4 severe reactor accidents, nor the prevention of the destruction of nuclear reactor fuel as it happened in the Paks-2 refueling pond. It is due to the fact that the very rapid development of the ignition condition after the bubble formation in the core is misrepresented, shown by the required codes to be much slower than it is in reality. http://aladar-mychernobyl.blogspot.com/

Furthermore, the current 10 CFR 50 series of regulation is not conservative, because it does not require the demonstration of preservation of the containment surrounding the reactor in the event of the detonation of a Hydrogen-air mixture, calculated from the generated amount of Hydrogen from the Zirconium-steam fire, consuming the entire inventory of Zirconium in the core in a single firestorm event.

http://www.osti.gov/energycitations/servlets/purl/10188341-UMoU6M/native/
FULL-LENGTH HIGH-TEMPERATURE SEVERE FUEL DAMAGE TEST #5
D. D. Lanning at al.

April 1988 – Completion Date
September 1993 – Publication Date
Prepared for
U. S. Nuclear Regulatory Commission
Under U.S. Department of Energy
Contract DE-ACO6-76RLO1830
Pacific Northwest Laboratory
Richland, Washington 99352

Reports on page 6:

"TEST RESULTS
Following the uncovering and dryout during the coolant boilaway, the rods heated at a rate of 2 to 5 K/s until peak cladding temperatures of 1700° K were attained, at which time the autocatalytic oxidation reaction resulted in a temperature excursion (at a rate of 10 to 50° K/s) and hydrogen generation. Peak local cladding temperatures are estimated to have exceeded 2600° K, based on information from thermocouples on the outside of the bundle liner.
The high-temperature oxidation reaction began at the 2.4- to 3.04-m elevation and formed a localized burn front that moved quickly downward as far as the 1.2-m elevation and then steadily upward. The burn front reached the top end caps (3.80m) and ceased 15 min. before the end of the test. The oxidation reaction consumed 75% of the total zircaloy or almost 100% of the zircaloy in the path of the burn front. The remaining 25% of the zircaloy was always below or near the bundle water level. The amount of hydrogen generated was 300±30 g, close to the total conversion of the 1.26-g/s make-up coolant flow within the 45-min. high-temperature period. The hydrogen flow fluctuated during the 45-min. high-temperature period in response to similar fluctuations (10% to 20% relative) in the bundle coolant flow. The peak hydrogen flow was 190 mg/s, which corresponded to an oxidation power of 28 kW."

TEST RESULTS

Following the uncovering and dryout during the coolant boilaway, the rods heated at a rate of 2 to 5 K/s until peak cladding temperatures of 1700 K were attained, at which time the autocatalytic oxidation reaction resulted in a temperature excursion (at a rate of 10 to 50 K/s) and hydrogen generation. Peak local cladding temperatures are estimated to have exceeded ~2600 K, based on information from thermocouples on the outside of the bundle liner.

The high-temperature oxidation reaction began at the 2.4- to 3.04-m elevation and formed a localized burn front that moved quickly downward as far as the 1.2-m elevation and then steadily upward. The burn front reached the top end caps (3.80 m) and ceased 15 min before the end of the test. The oxidation reaction consumed 75% of the total Zircaloy or almost 100% of the Zircaloy in the path of the burn front. The remaining 25% of the Zircaloy was always below or near the bundle water level. The amount of hydrogen generated was 300±30 g, close to the total conversion of the 1.26-g/s makeup coolant flow within the 45-min high-temperature period. The hydrogen flow fluctuated during the 45-min high-temperature period in response to similar fluctuations (10% to 20% relative) in the bundle coolant flow. The peak hydrogen flow was 190 mg/s, which corresponded to an oxidation power of 28 kW.

vi

This description is a very clear presentation of ignition and fire of Zirconium in the steam in a steam-starved environment.

http://itu.jrc.ec.europa.eu/uploads/media/Activity_Report_2004.pdf
page 42 "Bundle tomography revealed that a large central cavity was apparent above the corium pool at approximately one-third bundle height. At the top there were remnants of distorted, degraded fuel rods, whereas below the corium pool there was small streams of melt material and debris evident."

page 43. "The differences between the degraded bundle geometries of FPT1 and FPT2 can be explained by the fact that under steam-starved conditions (FPT2) Zircaloy metal melts and relocates at a lower temperature, whereas under oxidising conditions (FPT1) the Zircaloy cladding oxidises to a refractory oxide (ZrO_2) and remains in place until very high temperatures are reached later in the accident."

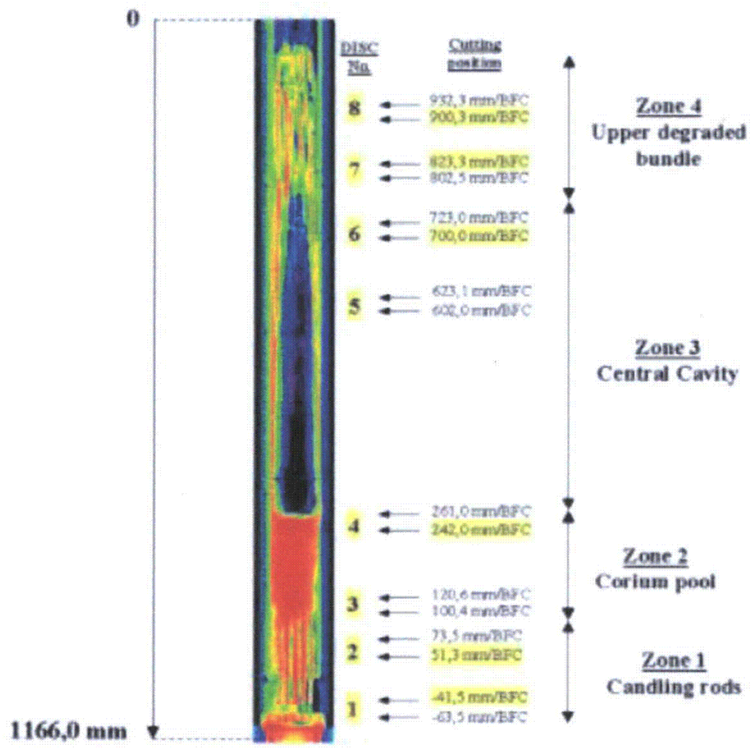


Fig. 6: False colour tomography of FPT2 bundle after testing showing the central cavity in black (least density) and the corium pool of melted material beneath in red (highest density)

Similar destruction and relocation of nuclear reactor fuel was observed in the TMI-2 and Chernobyl-4 severe reactor accidents and in the Paks-2 refueling pond reactor fuel washing accident.

The similarities in these tests and accidents are the formation of gaseous (steam) bubbles in the upper regions of fuel bundles, the ignition of Zirconium in the steam and generation of Hydrogen and zirconia (ZrO_2) reaction products in a very intense fire, essentially in a firestorm. Therefore, the conservative regulation shall mandate that the owners and operators of Nuclear Reactors and Reactor Fuel Handling Facilities shall demonstrate that **there will be no dry-out of the fuel bundles in any circumstances.**

Also, in order to prevent the exposure of the public to the harmful consequences of an accident in a reactor, the housing of the reactor **(containment) shall withstand the detonation of the air-Hydrogen mixture with the amount of Hydrogen** calculated from the consumption of the **entire inventory of Zircaloy in the reactor core** or in the entire enclosed in a vessel volume, where such bubble formation is possible.

There are several reports presenting the same issue as Mark Leyse. The cladding of nuclear fuel made of **Zirconium alloy ignites and burns in the steam**. The same process can be recognized (and should be recognized) as **the common cause** of the TMI-2 and Chernobyl-4 reactor severe accidents and the Paks-2 refueling pond accident. And the regulations in 10 CFR 50 series shall mandate to deal with the real issues and real processes.

Received: from mail1.nrc.gov (148.184.176.41) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Wed, 24 Mar 2010
02:59:01 -0400
X-Ironport-ID: mail1
X-SBRS: 5.2
X-MID: 12583603
X-fn: Docket ID NRC.pdf
X-IronPort-AV: E=Sophos;i="4.51,300,1267419600";
d="pdf'?scan'208";a="12583603"
Received: from mail-ww0-f41.google.com ([74.125.82.41]) by mail1.nrc.gov with
ESMTP; 24 Mar 2010 02:58:59 -0400
Received: by wwb39 with SMTP id 39so2214190wwb.14 for
<Rulemaking.Comments@nrc.gov>; Tue, 23 Mar 2010 23:58:59 -0700 (PDT)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:date:message-id:subject
:from:to:content-type;
bh=z14vkGtLiDMnPfnJgv/xvQGhJgm0jPA90Boij2dwsAg=;
b=Uw+jVhB7S/5J8oLL6278wYl+4epCqX5PE2m+/sw0F04qqYa3TnocAHu0Z+SYtJ8wfl
Si0PxeZ98sRTLAAHSrtq8lWWsbLJdDBPCHwBx/50sUoVrQPaQavDG9FVqnPy86iCoe2V
iyYwWYiky/KKz9cwhxdPLyHouDEucXBDX6aYE=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:content-type;
b=mG3RKKuik5kNlwqXzi7QNDi+zlEU8VRl+dn4/SS6pmtp6lrNlqeIER6vnl9c5Q1w72
Mr2dmJeYlnS7xrnA43XMsf5k2lZUsPpWJJJF+MWxr5H05cn1Xn8KVebKXRC7ZHgjrC1k
tn0ofGKr+jl0s77hkZqqyvTXWtekcPgPcdnzg=
MIME-Version: 1.0
Received: by 10.216.162.147 with SMTP id y19mr850000wek.0.1269413938096; Tue,
23 Mar 2010 23:58:58 -0700 (PDT)
Date: Wed, 24 Mar 2010 07:58:57 +0100
Message-ID: <ac89c8501003232358h37b67561m2cc9e125f5567fe1@mail.gmail.com>
Subject: Docket ID NRC-2009-0554
From: Aladar Stolmar <astolmar@gmail.com>
To: Rulemaking.Comments@nrc.gov
Content-Type: multipart/mixed; boundary="0016367fae49da6d720482867685"
Return-Path: astolmar@gmail.com

Submission ID 4
Robert Leyse
ML100850098

PRM-50-93
(75FR03876)

DOCKETED
USNRC

March 25, 2010 (10:15am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

4

Comments on PRM-50-93

As an introduction to these comments, the following is copied from PRM-50-93:

> Petitioner is submitting this petition, because Petitioner is aware that data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. Data from such experiments also 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.

Investigations by P. Hofmann and V. Noak at Forschungszentrum Karlsruhe further confirm the Petitioner's assertion that the Baker-Just equation is non-conservative for calculating the temperature at which runaway oxidation will occur in a LOCA. Their report is, Physico-Chemical Behavior of Zircaloy Fuel Rod Cladding Tubes During LWR Severe Accident Reflood, Part I: Experimental results of single rod quench experiments, FZKA 5846, Institut für Materialforschung, Projekt Nukleare Sicherheitsforschung, Mai 1997.

4-1

In FZKA 5846, Hofmann and Noack report:

*A series of separate-effects tests is being carried out on Zircaloy PWR fuel rod cladding to study the enhanced oxidation which can occur on quenching. In these tests, performed in the QUENCH rig, **single tube specimens are heated by induction** to a high temperature and then quenched by water or rapidly cooled down by steam injection.*

No significant temperature excursion during quenching occurred such as had been observed for example in the quenched (flooded) CORA-bundle tests /4, 5/. This absence of any temperature escalation is believed to be due to the high radiative heat losses in the QUENCH rig.

The Baker-Just report, ANL-6548, is predominantly based on work by Bostrom, and Lemmon:

W. A. Bostrom, The High Temperature Oxidation of Zircaloy in Water, WAPD-104 (March 1954).

A. W. Lemmon, Jr., Studies Relating to the Reaction between Zirconium and Water at High Temperatures, BMI-1154 (jan 1957)

Bostrom and Lemmon each used induction heating of single specimens. In ANL-6548, Baker and Just did not recognize the high radiative heat losses in the Bostrom and Lemmon work. In contrast, in the CORA bundle tests runaway

Template=SECY-067

DS10

oxidation began in the range of 1100 and 1200 °C. (2012 to 2192°F) and this runaway is described as follows in PRM-50-93, page 26:

There is also experimental data from multi-rod severe accident tests that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. 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.

[134] 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. 77.

And from PRM-50-93, page 44:

It is significant that in the CORA Experiments, at cladding temperatures between 1100°C and 1200°C (2012°F to 2192°F), that the cladding began to rapidly oxidize and cladding temperatures started increasing at a maximum rate of 15°C/sec. (27°F/sec.), because the Baker-Just and Cathcart-Pawel equations calculate that autocatalytic oxidation occurs at approximately 2600°F and approximately 2700°F, respectively;[137] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."'[138] Data from the CORA Experiments also indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. It is also significant that the CORA experiments demonstrated that "[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."[139]

[137] 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."
[138] 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," p. 282.
[139] 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," p. 83.

Comment submitted by

Robert H. Leyse*
Chemical Engineer and Nuclear Engineer
P. O. Box 2850
Sun Valley, ID 83353

***Experience:**

Career to date: Commenter's ongoing career spans several decades: General Electric at Hanford Works (1950), Argonne, DuPont Savannah River Plant, General Electric Vallecitos, Westinghouse Pittsburgh, Scandpower Norway, Consulting with Westinghouse at TMI-2, EPRI Nuclear Safety Analysis Center, EPRI Exploratory Research, and now self employed (2010).

Selected Experience pertinent to this comment on PRM-50-93:

PWR FLECHT: Test design, discoveries and reporting as referenced in PRM-50-93.

Presentation at 2003 RELAP5 International Users Seminar, West Yellowstone, Montana
Unmet Challenges for SCDAP/RELAP5-3D. Analysis of Severe Accidents for Light Water Nuclear Reactors with Heavily Fouled Cores. Robert H. Leyse,
www.inl.gov/relap5/rius/yellowstone/**leyse**.pdf

Comment NEI PETITION FOR RULEMAKING: PRM-50-78 (Cladding Materials) September 9, 2002
The petition should be denied because the evaluations of cladding materials do not account for the realities of plant operation under so-called normal conditions as well as the LOCA environment.

PETITION FOR RULEMAKING: PRM-50-76 May 8,2002
Petitioner is aware of deficiencies in Appendix K. **1.** A. 5. The Baker-Just equation does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures. Likewise, petitioner is aware of deficiencies in Regulatory Guide 1.157, BESTESTIMATE CALCULATIONS OF ECCS PERFORMANCE, Paragraph 3.2.5.1. The report NUREG-17 does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures.

PETITION FOR RULEMAKING: PRM-50-73 September 04, 2001
The specific issue is that 50.46 and Appendix K do not address the impact of crud on coolability during a fast moving (large break) LOCA.

PETITION FOR RULEMAKING: PRM-50-78 September 9, 2002
Regulations are needed to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants.

Current field is microscale heat transfer at ultra-high heat fluxes to pressurized water.

Microscale Heat Transfer to Subcooled Water
LEYSE: MICROSCALE HEAT TRANSFER
doi.wiley.com/10.1111/j.1749-6632.2002.tb05912.x
Or go to: *http://www3.interscience.wiley.com/journal/118947467/abstract*

MICROSCALE PHASE CHANGE HEAT TRANSFER AT HIGH HEAT FLUX. Robert H. Leyse. Inz, Inc. Phani K. Meduri, Gopinath R. Warrier and Vijay K. Dhir ...
boiling.seas.ucla.edu/Publications/Conf_LMWD2003

Rulemaking Comments

From: Gallagher, Carol
Sent: Wednesday, March 24, 2010 4:19 PM
To: Rulemaking Comments
Subject: Comment on PRM-50-93
Attachments: NRC-2009-0554-DRAFT-0003.1[1].pdf

Van,

Attached for docketing is a comment from Robert H. Leyse on PRM-50-93 that I received via the regulations.gov website on 3/24/10.

Thanks,
Carol

Received: from HQCLSTR01.nrc.gov ([148.184.44.79]) by TWMS01.nrc.gov
([148.184.200.145]) with mapi; Wed, 24 Mar 2010 16:18:41 -0400
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: "Gallagher, Carol" <Carol.Gallagher@nrc.gov>
To: Rulemaking Comments <Rulemaking.Comments@nrc.gov>
Date: Wed, 24 Mar 2010 16:18:32 -0400
Subject: Comment on PRM-50-93
Thread-Topic: Comment on PRM-50-93
Thread-Index: AcrLjyypl2i7zSlKTt+ZTkxjFcvrGg==
Message-ID:
<6F9E3C9DCAB9E448AAA49B8772A448C50CEAA9118A@HQCLSTR01.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<6F9E3C9DCAB9E448AAA49B8772A448C50CEAA9118A@HQCLSTR01.nrc.gov>
MIME-Version: 1.0

Submission ID 5
Kathryn Barnes, Don’t Waste Michigan
ML100890123

PRM-50-93
(75FR03876)

Rulemaking Comments

DOCKETED
USNRC

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

5

From: Kathryn Barnes [greenwoodsart@msn.com]
Sent: Thursday, March 25, 2010 5:25 PM
To: Rulemaking Comments
Subject: Nuclear waste issue

5-1

NRC:

I support the Mark Leyse petition (PRM- 50-93) because it will help prevent a nuclear accident such as what happened at Chernobyl. The nuclear industry should not have liaise faire to operate unsafely.

Sincerely,

Kathryn Barnes

Don't Waste Michigan

Sherwood Chapter

Michigan

Ps. Thank you for wisely stopping the Yucca MT. depository project. Nuclear waste liability should not be shoved onto the taxpayers nor dumped on Native people's land.

Template = SECY-067

DS 10

Received: from mail1.nrc.gov (148.184.176.41) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Thu, 25 Mar 2010
17:25:27 -0400
X-Ironport-ID: mail1
X-SBRS: 2.9
X-MID: 12647040
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AiIDAD9yq0tBNyLUgWdsb2JhbACBRY9JihsVAQELCQoHFQMfvzCEfQQ
X-IronPort-AV: E=Sophos;i="4.51,309,1267419600";
d="scan'208,217";a="12647040"
Received: from col0-omc4-s10.col0.hotmail.com ([65.55.34.212]) by
mail1.nrc.gov with ESMTP; 25 Mar 2010 17:25:18 -0400
Received: from COL115-DS18 ([65.55.34.201]) by col0-omc4-s10.col0.hotmail.com
with Microsoft SMTPSVC(6.0.3790.3959); Thu, 25 Mar 2010 14:25:15 -0700
X-Originating-IP: [67.142.162.22]
X-Originating-Email: [greenwoodsart@msn.com]
Message-ID: <COL115-DS18A58BBF5982AF1E91C545B9240@phx.gbl>
Return-Path: greenwoodsart@msn.com
From: Kathryn Barnes <greenwoodsart@msn.com>
To: <Rulemaking.Comments@nrc.gov>
Subject: Nuclear waste issue
Date: Thu, 25 Mar 2010 17:25:16 -0400
MIME-Version: 1.0
Content-Type: multipart/alternative;
boundary="----=_NextPart_000_0036_01CACC40.226C3310"
X-Priority: 3
X-MSMail-Priority: Normal
Importance: Normal
X-Mailer: Microsoft Windows Live Mail 14.0.8064.206
X-MimeOLE: Produced By Microsoft MimeOLE V14.0.8064.206
X-OriginalArrivalTime: 25 Mar 2010 21:25:15.0787 (UTC) FILETIME=[A94E85B0:01CACC61]

Submission ID 6
Jerry Martin
ML100890124

PRM-50-93
(75FR03876)

Rulemaking Comments

6

From: JERRY MARTIN [mrjerrymartin@gmail.com] on behalf of JERRY MARTIN [MUMARTIN@COX.NET]
Sent: Thursday, March 25, 2010 6:25 PM
To: Rulemaking Comments
Subject: "Power Uprate" procedures

6-1

Dear Secretary, United States Nuclear Commission,

It is critical that you support the Mark Leyse petition (PRM- 50-93) and proceed to rescind the "power uprate" procedures of the Nuclear Industry, Nation wide. My son and his family live next to Diablo Canyon Plant at Avila Bay, CA, which is built on an earthquake fault. Between the "power uprate" procedure and the dangerous fault, I feel it is only a matter of time before his and his families lives are endangered. Please take action as soon as possible.

Jerry Martin

DOCKETED
USNRC

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Template = SECY-067

DS 10

Received: from mail2.nrc.gov (148.184.176.43) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Thu, 25 Mar 2010
18:25:05 -0400
X-Ironport-ID: mail2
X-SBRS: 4.5
X-MID: 14245630
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
Al8CANd/q0vRVd61kWdsb2JhbACblAgVAQEBAQkLCgcTAx+vaoVDiHkBAQMFhHgEgx6BJod
x
X-IronPort-AV: E=Sophos;i="4.51,309,1267419600";
d="scan'208,217";a="14245630"
Received: from mail-pz0-f181.google.com ([209.85.222.181]) by mail2.nrc.gov
with ESMTP; 25 Mar 2010 18:25:05 -0400
Received: by pzk11 with SMTP id 11so5029854pzk.17 for
<Rulemaking.Comments@nrc.gov>; Thu, 25 Mar 2010 15:25:03 -0700 (PDT)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:received:received:sender:message-id:from:to
:content-type:mime-version:subject:date:x-mailer;
bh=ddAU71vBBV8zjCxpH90SSinZDEyCQ//LxoKWW/PBeRs=;

b=bLdkTRYszUdRn+6KPMOGN88Pv8XQbHUweOOu7RK2Do1fC72sM0jxEJMs7yH8AG3xVr
Bm8Bb/caiT0ml+mtvFwsW7ZUdspEL3bknmopaGTv+f1wxAABjdNuUPwBGqK1lJaPh2H6
5DibngZCfsMLhXgsdhll1tyyGkIDQeR4WsDwE=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=sender:message-id:from:to:content-type:mime-version:subject:date
:x-mailer;
b=g7j6wDj3oia6/3TECuDFxS/TMUBfyBYWb+il7shwgnnj863nh7dlgEMCNzy0PwhEO0
Cm2rfht4jwUtyhM8Fq4UYehFok8foAeNTlWG8dFi0Mdj+eh2C98glwmOFlTK+oKNks0j
MRq2Cv1QbbCjn9li5cUmElOOztmojgmZ6DOzs=
Received: by 10.141.213.27 with SMTP id p27mr3744485rvq.176.1269555903294;
Thu, 25 Mar 2010 15:25:03 -0700 (PDT)
Return-Path: <mrjerrymartin@gmail.com>
Received: from [10.0.1.3] (ip68-6-88-122.sb.sd.cox.net [68.6.88.122])
by mx.google.com with ESMTPS id 23sm168802iwn.2.2010.03.25.15.25.01
(version=TLSv1/SSLv3 cipher=RC4-MD5); Thu, 25 Mar 2010 15:25:02
-0700 (PDT)
Sender: JERRY MARTIN <mrjerrymartin@gmail.com>
Message-ID: <C7F229AE-A186-4722-952A-FE52E6F5C721@COX.NET>
From: JERRY MARTIN <MUMARTIN@COX.NET>
To: Rulemaking.Comments@nrc.gov
Content-Type: multipart/alternative; boundary="Apple-Mail-6--809076452"
MIME-Version: 1.0 (Apple Message framework v936)
Subject: "Power Uprate" procedures
Date: Thu, 25 Mar 2010 15:24:59 -0700
X-Mailer: Apple Mail (2.936)

Submission ID 7
Bill and Jan Tache
ML100890125

PRM-50-93
(75FR03876)

DOCKETED
USNRC

Rulemaking Comments

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

7

From: Tache [tache@together.net]
Sent: Friday, March 26, 2010 2:24 AM
To: Rulemaking Comments
Subject: PRM-50-93

7-1

Dear Sirs,

My wife and I strongly support PRM-50-93, the Mark Leyse petition.
There is real danger in the recommendations of the NRC and the concept of power uprate, we believe.

The entire nuclear energy idea is fraught with risk, certainly including the poisoning of the environment for millions and millions of years. We should not be using it at all. The down sides of the industry---possible accidents, disposal of waste materials are dangerous to the utmost. Makes us lose sleep at night.

Thank you.

Bill and Jan Tache
PO Box 279
Big Sur, CA 93920

Template=SECY-067

DS 10

Received: from mail2.nrc.gov (148.184.176.43) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Fri, 26 Mar 2010
02:24:50 -0400
X-Ironport-ID: mail2
X-SBRS: 4.8
X-MID: 14250427
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AtACAFbwq0vRVlIAlGdsb2JhbAA3mnMVAQEBAQkLCAkTAx+vaY4xBIR+gyI
X-IronPort-AV: E=Sophos;i="4.51,312,1267419600";
d="scan'208";a="14250427"
Received: from elasmtp-curtail.atl.sa.earthlink.net ([209.86.89.64]) by
mail2.nrc.gov with ESMTP; 26 Mar 2010 02:24:50 -0400
DomainKey-Signature: a=rsa-sha1; q=dns; c=nofws;
s=dk20050327; d=together.net;
b=huk18Sqdrxaf5BS5Rofdq+WVe3iUJv9Ck4dzzLWMvYZIgYKWxth6pllCEZxOQnM7;

h=Received:Message-Id:From:To:Content-Type:Content-Transfer-Encoding:Mime-Version:Subject:Date:X-Mailer:X-ELNK-Trace:X-Originating-IP;
Received: from [63.207.68.181] (helo=[192.168.2.100]) by
elasmtp-curtail.atl.sa.earthlink.net with esmtpa (Exim 4.67) (envelope-from
<tache@together.net>) id 1Nv2yA-00021q-Kv for
Rulemaking.Comments@nrc.gov;
Fri, 26 Mar 2010 02:24:34 -0400
Message-ID: <24A9862B-547E-4ACB-A59A-E47F348C8928@together.net>
From: Tache <tache@together.net>
To: Rulemaking.Comments@nrc.gov
Content-Type: text/plain; charset="US-ASCII"; format=flowed; delsp=yes
Content-Transfer-Encoding: 7bit
MIME-Version: 1.0 (Apple Message framework v936)
Subject: PRM-50-93
Date: Thu, 25 Mar 2010 23:24:08 -0700
X-Mailer: Apple Mail (2.936)
X-ELNK-Trace:
b2ee1e6bbd4f9f358f17f9b52bcb53cd456e50a2f7853fae9396751993769ca6ac30c6e5e32ee181350badd9bab72f9c350badd9bab72f9c350badd9bab72f9c
X-Originating-IP: 63.207.68.181
Return-Path: tache@together.net

Submission ID 8
Margo and Dennis Proksa
ML100890126

PRM-50-93
(75FR03876)

DOCKETED
USNRC

8

Rulemaking Comments

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

From: Dennis Proksa [blackrockforge@cableone.net]
Sent: Friday, March 26, 2010 1:16 PM
To: Rulemaking Comments
Subject: Our 2 Cents

8-1

RULE-MAKING AND ADJUDICATIONS STAFF,

We support the Mark Leyse petition PRM-50-93.

We understand that the NRC and the nuclear industry have selectively excluded multiple-rod fuel damage test experiments to arrive at their calculated "conservative" safety margins.

NOW is the time to RAISE the safety standards for ALL nuclear reactors.

Thank you for looking out for the citizens of this nation, rather than our nation's INDUSTRIAL giants and their Profit margin.

Sincerely,

Margo & Dennis Proksa

Template = SECY-067

DS 10

Received: from mail1.nrc.gov (148.184.176.41) by OWMS01.nrc.gov
 (148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Fri, 26 Mar 2010
 13:16:27 -0400
X-Ironport-ID: mail1
X-SBRS: 0.4
X-MID: 12674455
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
Ai0DACeJrEsYdADoaGdsb2JhbACBRZlhFQ0FBgoHEwMfwAiEfgSDHg
X-IronPort-AV: E=Sophos;i="4.51,315,1267419600";
 d="scan'208,217";a="12674455"
Received: from s6.cableone.net ([24.116.0.232]) by mail1.nrc.gov with ESMTP;
 26 Mar 2010 13:16:27 -0400
Received: from [10.0.1.2] (unverified [24.116.157.120]) by s6.cableone.net
 (CableOne SMTP Service s6) with ESMTP id 30185618-1872270 for
 <Rulemaking.Comments@nrc.gov>; Fri, 26 Mar 2010 10:16:25 -0700
Return-Path: <blackrockforge@cableone.net>
Message-ID: <7F492412-B5CF-410C-86A2-7D4648E5792E@cableone.net>
From: Dennis Proksa <blackrockforge@cableone.net>
To: Rulemaking.Comments@nrc.gov
Content-Type: multipart/alternative; boundary="Apple-Mail-9--741188931"
MIME-Version: 1.0 (Apple Message framework v936)
Subject: Our 2 Cents
Date: Fri, 26 Mar 2010 11:16:27 -0600
X-Mailer: Apple Mail (2.936)
X-Vpipe: Scanner said ok (av_avast)
X-IP-stats: Incoming Last 3, First 701, in=211, out=0, spam=0 ip=24.116.157.120
X-Originating-IP: 24.116.157.120
X-Abuse-Info: Send abuse complaints to abuse@cableone.net

Submission ID 9
Debbie Highfill
ML100890127

PRM-50-93
(75FR03876)

DOCKETED
USNRC

9

Rulemaking Comments

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

From: debbie highfill [debbiehighfill@yahoo.com]
Sent: Friday, March 26, 2010 1:55 PM
To: Rulemaking Comments
Subject: Mark Leyse petition (PRM- 50-93) - Nuclear Power Plant Safety

9-1

Dear NRC Commissioners,
I urge you to study and accept the Mark Leyse petition (PRM- 50-93). I have read the research he has cited and I feel strongly that it should considered carefully.
I live within the evacuation zone of Diablo Canyon Nuclear Power Plant.
Please act with caution in regard to protecting our families.

Sincerely,
A concerned citizen,
Debbie Highfill
Morro Bay, CA

Template=SECY-067

DS10

Received: from mail1.nrc.gov (148.184.176.41) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Fri, 26 Mar 2010
13:54:46 -0400
X-Ironport-ID: mail1
X-SBRS: 4.5
X-MID: 12676460
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AsoBAB+RrEtFk1x0kWdsb2JhbACPEYF5ijEBAQEBCQsKBxMDH69wgWmFOYh5AQQEAYR5
BIMe
X-IronPort-AV: E=Sophos;i="4.51,315,1267419600";
d="scan'208";a="12676460"
Received: from web80011.mail.sp1.yahoo.com ([69.147.92.116]) by mail1.nrc.gov
with SMTP; 26 Mar 2010 13:54:46 -0400
Received: (qmail 71523 invoked by uid 60001); 26 Mar 2010 17:54:45 -0000
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed; d=yahoo.com; s=s1024;
t=1269626085; bh=2varPwoNPJtWbwakjqw9IpdyF3ZfGW8HMuUcMNCdYAs=;
h=Message-ID:X-YMail-OSG:Received:X-Mailer:Date:From:Subject:To:MIME-Version:Content-Type;
b=jHYy4mZzIT0jWCVNDg0FBcvrAQCG1IB0P1JNjGwdUW9tIVsju59hORXP8Gpzg/n5KD9+acp
arLHQFazunPCJ8wiqDa8pNUGRJGkDxyiCbiiwyd3a5SyfVWuo7qEKyemJxGY13MzUFgBTRae
du9Prl4L9RLHChAYIkSGHb3X5zkM=
DomainKey-Signature: a=rsa-sha1; q=dns; c=nofws;
s=s1024; d=yahoo.com;

h=Message-ID:X-YMail-OSG:Received:X-Mailer:Date:From:Subject:To:MIME-Version:Content-Type;

b=1g3V6lNz1n6bF8L6S99xBRMsBC9LIaYVc2kIY9TYvhviR31I65cXf23muijVsOgoEO9jE/Wves
wK3VmSY54lNpPyBV2lRH6UZwFzQK4jkcXetAywl+s3BYp72Lw5pqbtELiLeomxQA+32ST3YdU
7GWVftokQrKgw6Rs8AX0Dp6k=;
Message-ID: <300297.71139.qm@web80011.mail.sp1.yahoo.com>
X-YMail-OSG: XK0wWuUVM1kiH0sxgUWWgrbIUyy.cdttpt8MhcDIbitfMkz
tbvCDkpt7WiZozsu02xe_rHmzBDA.kmeATBtw1zJAKgffcGbVcG_WD9GRQPW
I9OKHJFjSjoqxhrWAp1IawpPcRLF0Hiogvn5R59CoIY91_hOoPrgqwBfsBZ8
oVT9nswPeaqiVK.4wVzDScG.cAmJS9kD16BqrxunO1E3LE.XZ7ntB4bTI8HG
3t55eGb.o8TGpsgLwEzshwoYJMZdSaIQ_EqMiLr5tHgHTrJ4twYZouUC1Jjc
0zO7xnWwBKhBsVyZYzFdWgGugDl21BNzszw6.
Received: from [99.36.40.193] by web80011.mail.sp1.yahoo.com via HTTP; Fri, 26
Mar 2010 10:54:45 PDT
X-Mailer: YahooMailClassic/10.0.8 YahooMailWebService/0.8.100.260964
Date: Fri, 26 Mar 2010 10:54:45 -0700
From: debbie highfill <debbiehighfill@yahoo.com>
Subject: Mark Leyse petition (PRM- 50-93) - Nuclear Power Plant Safety
To: Rulemaking.Comments@nrc.gov
MIME-Version: 1.0
Content-Type: multipart/alternative;
boundary="0-203062002-1269626085=:71139"
Return-Path: debbiehighfill@yahoo.com

Submission ID 10
Hugo Kobayashi
ML100890128

PRM-50-93
(75FR03876)

DOCKETED
USNRC

Rulemaking Comments

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

From: Kobayashi, Hugo [Hugo.Kobayashi@morganstanley.com]
Sent: Friday, March 26, 2010 6:39 PM
To: Rulemaking Comments
Subject: NRC-2009-0554

10-1

10

I support Mark Leyse's petition (PRM-50-93) that I read about in the Beyond Nuclear Bulletin.

The Leyse petition raises serious concerns that the Nuclear Regulatory Commission and the nuclear industry have selectively excluded multiple-rod severe fuel rod damage test experiments to arrive at their calculated "conservative" safety margins. Leyse likens the NRC/industry action and result to "studying a burning match to predict what would occur in a forest fire." Not surprising. For decades, the nuclear power industry has prioritized raising the thermal energy and narrowing safety margins in its reactors to build more steam and more power by as much as 18% to 20% in a process called "power uprate." As such, the nuclear industry views the Leyse petition as a direct threat to these increased power levels at old reactors, higher projected power levels for new reactors and more profitable energy production margins.

Please take Mr. Leyse's petition seriously.

Hugo Kobayashi

Template = SECY-067

DS 10

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Fri, 26 Mar 2010
18:39:17 -0400
X-Ironport-ID: mail2
X-SBRS: 4.5
X-MID: 14297442
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
Am4AAObUrEvHWWdFkWdsb2JhbACBRZlmFQEBAQEJCwoHEwUdvy6CVoloBA
X-IronPort-AV: E=Sophos;i="4.51,316,1267419600";
d="scan'208,217";a="14297442"
Received: from pimtaint02.ms.com ([199.89.103.69]) by mail2.nrc.gov with
ESMTP; 26 Mar 2010 18:39:16 -0400
Received: from pimtaint02 (localhost.ms.com [127.0.0.1]) by pimtaint02.ms.com
(output Postfix) with ESMTP id CA2E0904C45 for <Rulemaking.Comments@nrc.gov>;
Fri, 26 Mar 2010 18:39:15 -0400 (EDT)
Received: from ny0019as01 (unknown [144.203.194.205]) by pimtaint02.ms.com
(internal Postfix) with ESMTP id A4A5A92C038 for
<Rulemaking.Comments@nrc.gov>; Fri, 26 Mar 2010 18:39:15 -0400 (EDT)
Received: from ny0019as01 (localhost [127.0.0.1]) by ny0019as01 (msa-out
Postfix) with ESMTP id 9535D3DC131 for <Rulemaking.Comments@nrc.gov>; Fri, 26
Mar 2010 18:39:15 -0400 (EDT)
Received: from HNWEXGOB02.msad.ms.com (hn212c1n1 [10.184.121.167]) by
ny0019as01 (mta-in Postfix) with ESMTP id 9294242C045 for
<Rulemaking.Comments@nrc.gov>; Fri, 26 Mar 2010 18:39:15 -0400 (EDT)
Received: from npwexhub01.msad.ms.com (10.164.54.2) by HNWEXGOB02.msad.ms.com
(10.184.121.167) with Microsoft SMTP Server (TLS) id 8.2.176.0; Fri, 26 Mar
2010 18:39:14 -0400
Received: from NYWEXMBX2128.msad.ms.com ([10.184.95.6]) by
npwexhub01.msad.ms.com ([10.164.54.2]) with mapi; Fri, 26 Mar 2010 18:39:14
-0400
From: "Kobayashi, Hugo" <Hugo.Kobayashi@morganstanley.com>
To: <Rulemaking.Comments@nrc.gov>
Date: Fri, 26 Mar 2010 18:39:12 -0400
Subject: NRC-2009-0554
Content-Transfer-Encoding: 7bit
Thread-Topic: NRC-2009-0554
thread-index: AcrNNSedxeoa3rkwR6yMvlhoVAew9A==
Message-ID:
<CDF156C5E5933D498E413205B13AB68C028987008B@NYWEXMBX2128.msad.ms.com>
Accept-Language: en-US
Content-Language: en-US
Content-Class: urn:content-classes:message
Importance: normal
Priority: normal
X-MimeOLE: Produced By Microsoft MimeOLE V6.00.3790.4325
X-MS-Has-Attach:
X-MS-TNEF-Correlator:
acceptlanguage: en-US
Content-Type: multipart/alternative;

boundary="_000_CDF156C5E5933D498E413205B13AB68C028987008BNYWEXMBX2128m_
"
MIME-Version: 1.0
X-Anti-Virus: Kaspersky Anti-Virus for MailServers 5.5.35/RELEASE, bases: 26032010
#3644506, status: clean
Return-Path: Hugo.Kobayashi@morganstanley.com

Submission ID 11
Betty Winholtz
ML100890130

PRM-50-93
(75FR03876)

11

DOCKETED
USNRC

Rulemaking Comments

March 29, 2010 (10:45am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

From: betty winholtz [winholtz@sbcglobal.net]
Sent: Monday, March 29, 2010 12:27 AM
To: Rulemaking Comments
Subject: Mark Leyse petition (PRM- 50-93) - Nuclear Power Plant Safety

11-1

Dear NRC Commissioners,
I hear from a friend in town about this. I live in the same town within the evacuation zone, so please study and accept the Mark Leyse petition (PRM- 50-93). I have read the research he has cited and I feel strongly that it should considered carefully.
As I stated, I, too, live within the evacuation zone of Diablo Canyon Nuclear Power Plant on the Central Coast of California. Please act with caution in regard to protecting our families.
Betty Winholtz

Template = SECY-067

DS 10

Received: from mail2.nrc.gov (148.184.176.43) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Mon, 29 Mar 2010
00:26:55 -0400
X-Ironport-ID: mail2
X-SBRS: 4.8
X-MID: 14317876
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AvgCAPPIr0tEjslTIGdsb2JhbACPQ4t8AQEBAQkLCAkRAx+wIwEFjT4BBIUBgyI
X-IronPort-AV: E=Sophos;i="4.51,326,1267419600";
d="scan'208,217";a="14317876"
Received: from web82702.mail.mud.yahoo.com ([68.142.201.83]) by mail2.nrc.gov
with SMTP; 29 Mar 2010 00:26:55 -0400
Received: (qmail 36040 invoked by uid 60001); 29 Mar 2010 04:26:54 -0000
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed; d=sbcglobal.net; s=s1024;
t=1269836814; bh=2L9ZT0Y/HX6v/Fo9oDN1Lzs8RW+k6kACFSNCN98g2wo=;
h=Message-ID:X-YMail-OSG:Received:X-Mailer:Date:From:Subject:To:MIME-Version:Content-Type;
b=REPcc3m5vbHJsWSHpS2EtBo+sUiFNDJvMs8+8PZbBlDshHbEwjPj4GDbBjtCiCLqQ0XpNW
7i8nzr6aOzURaqWMfLFmK67vxd0A7NZCDXELmRKhq3CjyHOO5mzuOuF6N8wtYuO/fdpjhbL
Du980GsOPfebMdUn6V/16ek9MzTRtQ=
DomainKey-Signature: a=rsa-sha1; q=dns; c=nofws;
s=s1024; d=sbcglobal.net;

h=Message-ID:X-YMail-OSG:Received:X-Mailer:Date:From:Subject:To:MIME-Version:Content-Type;

b=A3ahGQmcZRVfFxp0B8TSWyRnB1te8a2dIDFRzr6iG01OvoWeJ9pLG37YOaoKutsmhxVMS
WABYtzMEgNeOo+z9LRSvKJe7ysAr52GNpHwSm+bXalRo57g5fNNQnwxM0IoCOb1uPxU9XD
HFZiig3qbjR76DX+gGh0ESlrTaCXJpCo=;
Message-ID: <293308.35532.qm@web82702.mail.mud.yahoo.com>
X-YMail-OSG: VHMIKCYVM1lYNDgzIJ6.3Q8bO9OZ1RKUzRGx8P8xji.NVBZ
EhuUVLFpxABqM0U6bR5p3VQ5vppOvJicSJVMjXWT6CaMMTSJf9OrXndNDcqJ
kNSw1igHOcVg4toovbjZkln1aY04pmBbLNud4OasuAgjMuIV3L8dQ8PY1wzu
EkU1dtBxiHYd4X5cqTdfLjMNEwwemNRuR1udmOYvOC06VQ0.vBaP8P1p4CiV
gwvjD7CR_EfRH2fSdpylVLR85IJ6AQfIjlw3q6HJCZGjqd4iT2EwrtTDw5go
u
Received: from [75.15.159.32] by web82702.mail.mud.yahoo.com via HTTP; Sun, 28
Mar 2010 21:26:54 PDT
X-Mailer: YahooMailClassic/10.0.8 YahooMailWebService/0.8.100.260964
Date: Sun, 28 Mar 2010 21:26:54 -0700
From: betty winholtz <winholtz@sbcglobal.net>
Subject: Mark Leyse petition (PRM- 50-93) - Nuclear Power Plant Safety
To: Rulemaking.Comments@nrc.gov
MIME-Version: 1.0
Content-Type: multipart/alternative;
boundary="0-1999587480-1269836814=:35532"
Return-Path: winholtz@sbcglobal.net

Submission ID 12
Mary Johnston
ML100970080

PRM-50-93
(75FR03876)

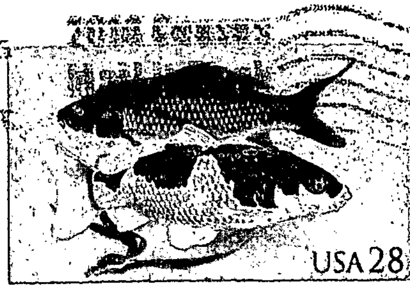
12

DOCKETED
USNRC

April 5, 2010 (4:23pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Mrs. Dean Johnston
124 Friendship Cir
Joplin, MO 64801



Secretary, U.S. Nuclear Commission
Washington, D.C. 20555-001

Attn.: Rule-
making and
Adjudications
Staff

12-1

I would like to see greater safety margins to protect fuel cladding in nuclear reactors. NRC conducted experiments in 1985 which generally showed that 1800° F, as a peak temperature would prevent runaway oxidation of the zircaloy cladding much more safely than the 2200° F. limit would. I support the Mark Leyse petition (PRM-50-93).

Thank you, Mary E Johnston 29 March, 2010

Template = SECY-067

DS 10

Submission ID 13
Robert Leyse
ML101020563

DOCKETED
USNRC

April 12, 2010 (9:10am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

13

PRM-50-93 is based on sound science

Of course, the Commissioners of the NRC were not presented with the very substantial amount of the available and applicable documentation when they approved the denial of PRM-50-76 that was posted in the Federal Register on Tuesday, September 6, 2005. The NRC very strongly asserted: "NRC's technical safety analysis demonstrates that current procedures for evaluating ECCS performance are based on sound science and that no amendments to the NRC's regulations and guidance documents are necessary." In PRM-50-93, the very thorough analysis and documentation by the Petitioner, Mark Edward Leyse, unambiguously demonstrates that the NRC's current procedures for evaluating ECCS performance are most certainly not based on sound science.

13-1

In its posted denial of PRM-50-76, the NRC states, "The remaining data from Bostrum (``The High Temperature Oxidation of Zircaloy in Water," W. A. Bostrum, WAPD-104 March 1954) and Lemmon (``Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," A. W. Lemmon, Jr., BMI-1154, January 1957), at more relevant zirconium cladding conditions, were used by Baker and Just in the derivation of their equation." However, it is unlikely that the authors of NRC's technical safety analysis, ML041210109, ever looked at either WAPD-104 or BMI-1154. It is more likely that those authors merely lifted the description of those references from the Baker-Just report, ML050550198. Thus the authors of ML041210109 were not aware that Bostrom and Lemmon each used induction heating in their investigations. Furthermore, those authors also were likely unaware of FZKA 5846, the Hofmann and Noack report from which this Commenter has already quoted as follows in his Comment 4 to PRM-50-93 on 04/05/10.

A series of separate-effects tests is being carried out on Zircaloy PWR fuel rod cladding to study the enhanced oxidation which can occur on quenching. In these tests, performed in the QUENCH rig, ***single tube specimens are heated by induction*** *to a high temperature and then quenched by water or rapidly cooled down by steam injection.*

No significant temperature excursion during quenching occurred such as had been observed for example in the quenched (flooded) CORA-bundle tests /4, 5/. This absence of any temperature escalation is believed to be due to the high radiative heat losses in the QUENCH rig.

The opening statement in PMR-50-93 is founded on sound science, "Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") revise 10 C.F.R. § 50.46(b)(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."

Finally, in view of its importance, I want to assure the public that I have a well-documented e-mail trail in support of the following assertion that I have copied from the foregoing: *However, it is unlikely that the authors of NRC's technical safety analysis,*

Template = SECY-067

DS 10

ML041210109, ever looked at either WAPD-104 or BMI-1154. It is more likely that those authors merely lifted the description of those references from the Baker-Just report, ML050550198. Thus on 2/7/2010 I e-mailed chairman@nrc.gov , "Please make these two reports, WAPD-104 and BMI-1154, available to the public." Receiving no reply, I forwarded that e-mail to stephen.burns@nrc.gov on 2/17/2010. On 3/4/2010 I received an e-mail from Richard.Dudley@nrc.gov informing me that BMI-1154 was now in ADAMS (ML100570218). On 3/31/2010, Richard.Dudley@nrc.gov e-mailed a pdf version of WAPD-104. These are the facts regarding the timing of NRC's access to WAPD-104 and BMI-1154.

Comment submitted by

Robert H. Leyse*
Chemical Engineer and Nuclear Engineer
P. O. Box 2850
Sun Valley, ID 83353

***Experience:**

Career to date: Commenter's ongoing career spans several decades: General Electric at Hanford Works (1950), Argonne, DuPont Savannah River Plant, General Electric Vallecitos, Westinghouse Pittsburgh, Scandpower Norway, Consulting with Westinghouse at TMI-2, EPRI Nuclear Safety Analysis Center, EPRI Exploratory Research, and now self employed (2010).

Selected Experience pertinent to this comment on PRM-50-93:

PWR FLECHT: Test design, discoveries and reporting as referenced in PRM-50-93.

Presentation at 2003 RELAP5 International Users Seminar, West Yellowstone, Montana
Unmet Challenges for SCDAP/RELAP5-3D. Analysis of Severe Accidents for Light Water Nuclear Reactors with Heavily Fouled Cores. Robert H. Leyse,
www.inl.gov/relap5/rius/yellowstone/**leyse**.pdf

Comment NEI PETITION FOR RULEMAKING: PRM-50-78 (Cladding Materials) September 9, 2002
The petition should be denied because the evaluations of cladding materials do not account for the realities of plant operation under so-called normal conditions as well as the LOCA environment.

PETITION FOR RULEMAKING: PRM-50-76 May 8, 2002
Petitioner is aware of deficiencies in Appendix K. 1. A. 5. The Baker-Just equation does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures. Likewise, petitioner is aware of deficiencies in Regulatory Guide 1.157, BESTESTIMATE CALCULATIONS OF ECCS PERFORMANCE, Paragraph 3.2.5.1. The report NUREG-17 does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures.

PETITION FOR RULEMAKING: PRM-50-73 September 04, 2001
The specific issue is that 50.46 and Appendix K do not address the impact of crud on coolability during a fast moving (large break) LOCA.

PETITION FOR RULEMAKING: PRM-50-78 September 9, 2002
Regulations are needed to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants.

Current field is microscale heat transfer to pressurized water at ultra-high heat fluxes.

Microscale Heat Transfer to Subcooled Water
LEYSE: MICROSCALE HEAT TRANSFER
doi.wiley.com/10.1111/j.1749-6632.2002.tb05912.x
Or go to: *http://www3.interscience.wiley.com/journal/118947467/abstract*

MICROSCALE PHASE CHANGE HEAT TRANSFER AT HIGH HEAT FLUX. Robert H. Leyse. Inz, Inc.
Phani K. Meduri, Gopinath R. Warrier and Vijay K. Dhir ...
boiling.seas.ucla.edu/Publications/Conf_LMWD2003

Rulemaking Comments

From: Gallagher, Carol
Sent: Friday, April 09, 2010 10:24 AM
To: Rulemaking Comments
Subject: Comment on PRM-50-93
Attachments: NRC-2009-0554-DRAFT-0013.1[1].doc

Van,

Attached for docketing is a comment from Robert H. Leyse on PRM-50-93 that I received via the regulations.gov website on April 8, 2010.

Thanks,
Carol

Received: from HQCLSTR01.nrc.gov ([148.184.44.79]) by OWMS01.nrc.gov
([148.184.100.43]) with mapi; Fri, 9 Apr 2010 10:24:17 -0400
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: "Gallagher, Carol" <Carol.Gallagher@nrc.gov>
To: Rulemaking Comments <Rulemaking.Comments@nrc.gov>
Date: Fri, 9 Apr 2010 10:23:49 -0400
Subject: Comment on PRM-50-93
Thread-Topic: Comment on PRM-50-93
Thread-Index: AcrX8EWL3MAwAY6jRHOrAWIT938GrQ==
Message-ID:
<6F9E3C9DCAB9E448AAA49B8772A448C50CEAD4FBA1@HQCLSTR01.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<6F9E3C9DCAB9E448AAA49B8772A448C50CEAD4FBA1@HQCLSTR01.nrc.gov>
MIME-Version: 1.0

Submission ID 14
Mark Leyse
ML101020564

PRM-50-93
(75FR03876)

DOCKETED
USNRC

14

April 12, 2010 (9:10am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 12, 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, second response to the NRC's notice of solicitation of public comments on PRM-50-93, published in the Federal Register, January 25, 2010.

Respectfully submitted,

Mark E Leyse

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

Template = SECY-067

DS10

April 12, 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; NRC-2009-0554

TABLE OF CONTENTS

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 ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] These same requirements also need to

---

[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.

[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.

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.

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 provides supplementary information to section III.C.2. of PRM-50-93.

---

[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.

II. Supplementary Information to PRM-50-93 Section III.C.2. The Fact that the Baker-Just and Cathcart-Pawel Equations were Not Developed to Consider how Heat Transfer would Affect Zirconium-Water Reaction Kinetics in the Event of a LOCA

> 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.[8]—Dr. Ralph Meyer

> ...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?[9]—Dr. Dana A. Powers

14-1

1. A Recommendation for a Set of Correlations for Severe Fuel Damage Codes for the High Temperature Range and the 1990 CORA Workshop

Regarding a recommendation for severe fuel damage ("SFD") codes, "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys" states:

> For the high temperature range, SFD codes of integral type rely generally on a simplified treatment of the steam oxidation, neglecting limited system geometry and anomalies: In those the oxidation is described using reaction rate functions of parabolic time and Arrhenius temperature dependence. *Recently a set of correlations was recommended, based on the critical review of experimental data*,[10] their statistical evaluation within the diffusion system concept, *and their verification against rod bundle experiments* [emphasis added].[11, 12]

---

[8] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Transcript, 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.

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

[10] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Forschungszentrum Karlsruhe, FZKA 6827, 2003.

[11] A. Volchek et al., "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations, Part I. Experimental Database and Basic Modeling, Part II. Best-Fitted Parabolic Correlations, Part III. Verification Against Representative Transient Tests," Nuclear Engineering and Design, 232, 2004, pp. 75-109.

[12] G. Schanz, "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys," Forschungszentrum Karlsruhe, FZKA 7329, 2007, p. 2.

In the passage above, Schanz, the author, is referring to zirconium-steam reaction kinetics in "the high temperature range," at temperatures far greater than the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F. However, the recommendation to have "correlations...based on the critical review of experimental data...and their verification against rod bundle experiments,"[13] should also be applied to zirconium-steam reaction kinetics, at temperatures lower than "the high temperature range," including temperatures lower than 2200°F.

(In "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Schanz, refers to temperatures that are greater than 1900 K (2961°F) as "the high-temperature range."[14])

It is significant that in the 1990 CORA Workshop at Kernforschungszentrum Karlsruhe ("KfK") GmbH, Karlsruhe, FRG, October 1-4, 1990, problems with SCDAP/RELAP5's modeling of Zircaloy oxidation kinetics, in the 900-1200°C temperature range, were discussed.

The document, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," is partly a report on the 1990 CORA Workshop at KfK GmbH, Karlsruhe, FRG, October 1-4, 1990.[15]

Regarding temperature excursions during the CORA experiments and SCDAP/RELAP5's late prediction of the temperature excursion for the CORA-12 experiment, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> Temperature escalation starts at ~1200°C and continues even after shutoff of the electric power as long as metallic Zircaloy and steam are available.
>
> ...
>
> [Dr. T. J. Haste, United Kingdom Atomic Energy Agency,] did note *the late prediction (via SCDAP/RELAP5) for the oxidation excursion in CORA-12*... [emphasis added][16]

---

[13] *Id.*

[14] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," refers to temperatures that are lower than 1800 K (2781°F) as the low-temperature range, p. 9; and refers to temperatures that are greater than 1900 K (2961°F) as the high-temperature range, p.10.

[15] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," October 16, 1990, Cover Page.

[16] *Id.*, pp. 2, 3.

And regarding "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[17] "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) oxidation to be underpredicted.*
>
> ... 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].[18]

So in the ORNL SCDAP/RELAP5 calculations performed for the CORA-16 experiment, a rod bundle experiment, "[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) oxidation to be underpredicted."[19] This indicates that in the early '90s there were deficiencies in SCDAP/RELAP5 calculations of Zircaloy oxidation kinetics in the 900-1200°C temperature range. And such deficiencies in ECCS evaluation calculations for LOCAs continue to this day (2010).

As quoted above, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> [Dr. T. J. Haste, United Kingdom Atomic Energy Agency,] did note *the late prediction (via SCDAP/RELAP5) for the oxidation excursion in CORA-12*... [emphasis added][20]

And regarding the same problem in an ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," published in 2010, states:

> The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17),[21] *but the onset of temperature escalation in the upper part of the CFM was delayed* [emphasis added].[22]

---

[17] *Id.*, p. 3.
[18] *Id.*
[19] *Id.*
[20] *Id.*, pp. 2, 3.

In "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17, the graph of the cladding-temperature values depicts that the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment has the onset of the temperature escalation (at the 1.067 m. elevation) occurring at a temperature greater than 1700 K (2600°F); figure 17 also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m. elevation) occurred at a temperature well below 1500 K (2240°F). So the difference between the calculated and actual experimental value for the onset of the temperature escalation, at the 1.067 m. elevation is greater than 200 K (360°F)—a significant difference.

(It is noteworthy that 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), at the 0.69 m. elevation.)

The reason for the deficiencies in ECCS evaluation calculations of Zircaloy oxidation kinetics in the LOFT LP-FP-2 and other experiments, is that ECCS evaluation calculations use the Baker-Just and Cathcart-Pawel equations to calculate metal-water reaction rates.

In PRM-50-93, regarding RELAP5/Mod3 calculations using the Baker-Just and Cathcart-Pawel equations, Petitioner quoted, "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;" the document states:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F... Thus, with regard to runaway temperature escalation, the peak

[21] See Appendix A Fig. 17. LOFT LP-FP-2 CFM Clad Temperature at Elevation 1.067 m.
[22] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," Progress in Nuclear Energy, 52, 2010, p. 155.

cladding temperature could be raised to 2300°F without affecting this sensitivity and without reducing the margin that the Commission would have perceived in 1973.

To explore this sensitivity further, we performed more than 50 LOCA calculations with RELAP5/Mod3. In about half of the cases, the Baker-Just equation was used for the metal-water heat generation rate, and in the other half, the Cathcart-Pawel equation was used. Reactor power just prior to the LOCA was varied parametrically to simulate incremental variations in decay heat. The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F. Each series of calculations done with the two metal-water models always showed peak cladding temperatures without runaway to be at least 100°F higher with Cathcart-Pawel, which is consistent with the temperature difference in the rate equations. Thus in these calculations, the margin between 2300°F and the calculational instability using Cathcart-Pawel was always equal to or greater than the margin between 2200°F and the calculational instability using Baker-Just.[23]

It is significant that the Baker-Just and Cathcart-Pawel equations calculated autocatalytic (runaway) oxidation to occur when cladding temperatures increased above approximately 2600°F and above approximately 2700°F, respectively, in the NRC's more than 50 LOCA calculations with RELAP5/Mod3, because data from severe fuel damage experiments indicates that autocatalytic oxidation of Zircaloy cladding occurs at far lower temperatures. Furthermore, such experiments indicate that the Baker-Just equation is not substantially conservative at 2200°F.

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

---

[23] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, pp. 3-4; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.

> 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.[24]

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."[25] So at the point 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.

So the recommendation to have "correlations...based on the critical review of experimental data...and their verification against rod bundle experiments,"[26] should certainly be applied to zirconium-steam reaction kinetics at temperatures below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(It is noteworthy that Schanz is one of the coauthors of "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," which states that "[w]ith 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."[27])

---

[24] 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.

[25] 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.

[26] G. Schanz, "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys," FZKA 7329, 2007, p. 2.

[27] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "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, p. 83.

14-2

2. Inductive Heating and Furnace Experiments

a. The Two Inductive Specimen Heating Experiments that the Baker-Just Equation is Almost Entirely Based On

Regarding the experiments that the Baker-Just equation is based on, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes" states:

> The Baker-Just correlation itself is based on results of their own experiments for just the melting temperature of zirconium, (in which fine [zirconium] wires were directly heated in water and the hydrogen evolution from the resulting molten droplets was measured to calculate the reaction rate), together with literature results from Lemmon and Bostrom, who used inductive specimen heating and a hydrogen evolution measurement for evaluation.[28]

And regarding Bostrom and Lemmon's experiments with inductive zirconium specimen heating, "High-Temperature Oxidation of Zircaloy-2 and Zircaloy-4 in Steam" states:

> Bostrom inductively heated specimens of Zircaloy-2 in water (with a steam bubble enveloping the specimen) under isothermal conditions and determined K_p in the temperature range 1300-1860°C by the hydrogen evolution method. Lemmon measured the rates of reaction between Zircaloy-2 and steam in the temperature range 1000-1700°C by inductively heating specimens in steam at 50 psia and measuring the rate of hydrogen evolution.[29]

Describing Lemmon's experiments in more detail, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures" states:

> The reaction between solid Zircaloy 2 and steam at 50 psia was measured over the temperature range 1000 to 1690°C. … The Zircaloy 2 specimens were heated by electrical induction and reacted with flowing steam at a pressure of 50 psia. … The [Zircaloy 2] specimen was supported on a thermocouple protection tube and enclosed inside a Vycor tube; it was inductively heated to the reaction temperature by power applied through the induction coil.[30]

---

[28] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, 2003, p. 2.
[29] V. F. Urbanic and T. R. Heidrick, "High-Temperature Oxidation of Zircaloy-2 and Zircaloy-4 in Steam," Journal of Nuclear Materials 75, 1978, p. 252.
[30] Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, located at: www.nrc.gov,

Regarding radiative heat losses experienced in Lemmon's experiments, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures" states:

> The passage of steam through the reactor [unit] greatly increased the heat losses from the samples; and a large increase in power to the induction coil was required. Sample temperatures dropped as much as 100 or 200°C below the desired temperature before the power adjustment was effective. This sometimes took as long as [five] min.[31]

It is significant that both Bostrom and Lemmon conducted their experiments with inductive zirconium specimen heating, because the zirconium specimens would have had radiative heat losses. And such radiative heat losses would have had an effect on oxidation kinetics.

Regarding how radiative heat losses in inductive specimen heating experiments affect oxidation kinetics, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes" states:

> [Ocken] stated that [the] advantage [of experiments with inductive (Urbanic and Heidrick) and direct electrical heating (Biederman, *et al.*) of a specimen in a cool environment[32]] would be the temperature gradient from heated specimen to cool surrounding[s], leading to temperature gradients in the cladding wall in the same sense as in a reactor. In total disagreement with the argument of Ocken, the author of this paper stresses the advantage of a constant cladding wall temperature and thus of a better defined specimen temperature, as provided in furnace experiments! ... This argument was already used by Sawatzky, *et al.*, who performed similar studies with inductive specimen heating as Urbanic and Heidrick. *Sawatzky reached an important improvement of the specimen temperature homogeneity by only optimizing the geometry of the specimen and registered considerably increased reaction rates* [emphasis added].[33]

Electronic Reading Room, ADAMS Documents, Accession Number: ML100570218, pp. C-1, C-2, C3.

[31] *Id.*, p. C-7.

[32] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, 2003, pp. 4-5.

[33] *Id.*

And regarding how radiative heat losses in induction heating experiments prevented Zircaloy cladding tubes from having significant temperature excursions in the single rod quench experiments, "Experimental Results of Single Rod Quench Experiments" states:

> In these tests, performed in the QUENCH rig, single tube specimens are heated by induction to a high temperature and then quenched by water or rapidly cooled down by steam injection. ...
>
> *Because of the high radiative heat loses in the QUENCH rig, none of the tests conducted have resulted in significant temperature excursion* occurring during quenching such as had been observed for example in the quenched (flooded) CORA-bundle tests[34] [emphasis added].[35]

So the radiative heat losses of the zirconium specimens in Bostrom and Lemmon's induction heating experiments would have affected the oxidation kinetics that Bostrom and Lemmon measured. Bostrom and Lemmon's experiments certainly did not replicate the oxidation kinetics that would occur in a nuclear power plant's core, in the event of a LOCA. Yet the Baker-Just equation—required by Appendix K to Part 50 I(A)(5)—is almost entirely based on the results of Bostrom and Lemmon's experiments.

It is no wonder that the Baker-Just equation calculated autocatalytic (runaway) oxidation to occur when cladding temperatures increased above approximately 2600°F in the NRC's more than 50 LOCA calculations with RELAP5/Mod3,[36] while the LOFT LP-FP-2 experiment demonstrated that autocatalytic oxidation commences at cladding temperatures as low as approximately 1400°K (2060°F)[37] or 1500°K (2240°F).[38]

---

[34] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," Kernforschungszentrum Karlsruhe, KfK 5054, 1993 and S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," Forschungszentrum Karlsruhe, FZKA 5679, 1996, are cited as the source of this information.

[35] P. Hofmann and V. Noack, "Experimental Results of Single Rod Quench Experiments," Part I of "Physico-Chemical Behavior of Zircaloy Fuel Rod Cladding Tubes During LWR Severe Accident Reflood," Forschungszentrum Karlsruhe, FZKA 5846, 1997, Summary page, pp. 2-3.

[36] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, pp. 3-4; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.

[37] 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:

14-3

b. The Hobson/Rittenhouse Furnace Experiments that the Criteria of 10 C.F.R. § 50.46(b)(1) and (2) are Primarily Based On

It is significant that "The History of LOCA Embrittlement Criteria" states that "the 17%-ECR[39] and 1204°C [PCT] criteria [of 10 C.F.R. § 50.46(b)] were primarily based on the results of post-quench ductility tests conducted by Hobson."[40, 41]

And regarding the 1204°C PCT criterion, "The History of LOCA Embrittlement Criteria" states:

> The 2200°F (1204°C) peak cladding temperature (PCT) criterion was selected on the basis of Hobson's slow-ring-compression tests that were performed at 25-150°C. *Samples oxidized at 2400°F (1315°C) were far more brittle than samples oxidized at <2200°F (<1204°C) in spite of comparable level of total oxidation.* This is because oxygen solid-solution hardening of the prior-beta phase is excessive at oxygen concentrations >0.7 wt%.
>
> The selection of the 1204°C criterion was subsequently justified by the observations from the ANL 0.3-J impact tests and the handling failure of rods tested in the Power Burst Facility. These results also take into account of the effect of large hydrogen uptake that occurred near the burst opening. *Consideration of potential for runaway oxidation alone would have [led] to a PCT limit somewhat higher than 2200°F (1204°C).* In conjunction with the 17% oxidation criterion, the primary objective of the PCT criterion is to ensure adequate margin of protection against post-quench failure that may occur under hydraulic, impact, handling, and seismic loading [emphasis added].[42]

---

www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML062840091, pp. 30, 33.

[38] 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.

[39] "ECR" is the acronym for "equivalent cladding reacted."

[40] The experimental data that 50.46(b)(1) and 50.46(b)(2) are primarily based on is reported on in Hobson, D. O. and 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," Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, 26 March, 1973.

[41] G. Hache and H. M. Chung, "The History of LOCA Embrittlement Criteria," Proc. 28th Water Reactor Safety Information Meeting, Bethesda, USA, October 23-25, 2000, p. 10.

[42] *Id.*, pp. 27-28.

Describing the Hobson/Rittenhouse furnace experiments, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report" states:

> In early 1970s, cladding tube oxidation tests seem to have been regarded as fairly simple tests and in many cases only very sketchy descriptions are given. Hobson and Rittenhouse[43] describe oxidation of 0.45 m long cladding specimens in a ceramic muffle tube inserted in a furnace. Steam was supplied from below in amounts so that the reaction was not steam limited. Exposure temperatures were from 926 to 1370°C with exposure times from 2 to 60 minutes.
>
> A very important aspect of the early experiments of Hobson and Rittenhouse[44] and Hobson[45] in [the] early 1970s is that apparently specimen temperature was not measured but was assumed to be the same as the measured furnace temperature. This assumption may be reasonably accurate for low temperatures; *e.g.*, for <800°C. *However, for high temperatures; e.g., >1100°C, self-heat generation from large exothermic heat of Zr oxidation is significant*, and true specimen temperature must have been measured directly, *e.g.*, by use of spot-welded thermocouples. *Their papers do not mention this, and only describe [a] temperature variation of 6°C over a distance of 7.5 cm at the center of the furnace heat zone.*
>
> In view of this and similar lack of direct measurement of specimen temperatures in the oxidation experiment of Baker-Just... [emphasis added][46]

And 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,[47] specimen temperatures were not

---

[43] 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.

[44] *Id.*

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

[46] Nuclear Energy Agency, OECD, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," NEA No. 6846, 2009, p. 103.

[47] 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

> measured directly; *e.g.*, by using spot-welded thermocouples. Likewise, specimen temperatures in the experiment of Baker-Just[48] 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,[49] 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].[50]

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.

(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,"[51] 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.)

Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972; and Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," ANS Topical Meeting on Water Reactor Safety, 1973, Salt Lake City, pp. 274-288.

[48] 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.

[49] 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," ANS Topical Meeting on Water Reactor Safety, 1973, Salt Lake City, pp. 274-288.

[50] 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.

[51] *Id.*

14-4

c. The Cathcart/Pawel Furnace Experiments that the Cathcart-Pawel Equation is Based On

Regarding Zircaloy specimen temperature "overshoots," when the exothermic heat of reaction caused the specimen temperature to exceed that of its environment, in the Cathcart/Pawel furnace experiments with the MaxiZWOK, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies" states:

> For reaction at 1000°C (1832°F), the exothermic heat of reaction is sufficient to drive the specimen temperature above that of its environment, creating the "overshoot" that was typical of MaxiZWOK experiments in this temperature range. In [one] particular case...an overshoot of about 18°C (32°F) was observed before the specimen temperature began to return to its steady-state value, and several minutes were required for the effects of specimen self-heating to be dissipated.[52]

And regarding the same phenomenon in the MaxiZWOK experiments, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies" also states:

> Three mixed-temperature experiments were conducted with the steam temperatures varying from 894 to 994°C (1641-1821°F) and furnace temperatures varying from 1040 to 1110°C (1904-2030°F). Except for the degree of overshoot, the specimen temperature response in all three runs was similar... [In one run] the steam temperature was controlled at 994°C (1821°F) while the furnace was maintained at 1110°C (2030°F). In this environment the Zircaloy-4 PWR tube specimen experienced a 43°C (77°F) temperature overshoot before its temperature decreased to an equilibrium value of 1057°C (1935°F). Thus, even at this comparatively low temperature, it is evident that appreciable specimen self-heating can occur. It would be anticipated that for similar heat transfer characteristics, the extent of self-heating would increase substantially with increasing temperature.[53]

Regarding the isothermal rate of oxidation of Zircaloy-4 in the Cathcart/Pawel furnace experiments, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies" states:

> Neither steam flow rate (above levels leading to steam starvation), steam temperature, the presence in the steam of reasonable concentrations of

---

[52] 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. 79.

[53] *Id.*, pp. 102-103.

oxygen, nitrogen, or hydrogen; nor small variations in alloy composition significantly influence the isothermal rate of oxidation of Zircaloy-4. Obviously, however, both steam temperature and flow rate are important parameters in heat transfer calculations, and *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* [emphasis added].[54]

(It is noteworthy that PRM-50-76 states "it is not possible to achieve an isothermal rate of oxidation of Zircaloy-4 if the Zircaloy-4 is exposed to LOCA fluid conditions at elevated temperatures."[55])

And regarding temperature control in the Cathcart/Pawel MaxiZWOK and MiniZWOK experiments, "Denial of Petition for Rulemaking (PRM-50-76)" states:

> Controlling sample temperature by adjusting heater power (MiniZWOK) was much more successful than adjusting steam flow (MaxiZWOK). As the petitioner notes, *temperature overshoot was a problem with MaxiZWOK and at high temperatures could have led to temperature runaway*. As noted previously, temperature control is absolutely necessary in reaction kinetics experiments such as these [emphasis added].[56]

(It is also noteworthy that in the MaxiZWOK, steam flow was at least an order of magnitude greater than it was in the MiniZWOK;[57] and that "[t]he bulk of the reaction rate experiments [conducted by Cathcart and Pawel] were performed in the MiniZWOK apparatus."[58])

The NRC states that "temperature control is absolutely necessary in reaction kinetics experiments such as [those conducted with the MaxiZWOK and MiniZWOK]."[59] 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

[54] *Id.*, pp. 118-119.
[55] Robert H. Leyse, PRM-50-76, May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 5.
[56] 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. 14.
[57] *Id.*, p. 15.
[58] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, p. 14.
[59] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 14.

protocol of isothermal conditions, because the energy from the exothermal Zircaloy-steam oxidation would cause a temperature excursion.

In the MaxiZWOK experiment, at 1000°C (1832°F), the Zircaloy specimen was able to return to its steady-state value and the specimen self-heating was able to dissipate; however, in a Zircaloy bundle experiment between 1000°C and 1200°C there would be 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 capability is lost*, it determines a feedback between temperature increase and cladding oxidation [emphasis added].[60]

Furthermore, Figure 1[61] 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),[62] and Figure 13[63] 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.[64]

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;"[65] 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."[66]

---

60 Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 195.
61 See Appendix B Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.
62 Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 196.
63 See Appendix B Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core.
64 Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 205.
65 *Id.*, p. 195.
66 J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, p. 119.

So, as argued in PRM-50-76, the experiments that the Cathcart-Pawel equation is based on "did not include any consideration of the complex thermal hydraulic conditions [that would occur] during [a] LOCA."[67] And this would include the fact that the Cathcart-Pawel equation was not developed to consider the heat transfer of multi-rod bundles, or of multi-bundles, that would occur in the event of a LOCA.

It is significant that in the Advisory Committee on Reactor Safeguards, 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.[68]

And, as stated in PRM-50-93, the Cathcart-Pawel equation is non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA, precisely because it was not developed to consider how heat transfer would affect zirconium-water reaction kinetics.

Clearly, the Cathcart-Pawel equation is an equation for predicting the oxidation kinetics of Zircaloy specimens in furnaces; it is not adequate for predicting the oxidation kinetics of Zircaloy bundles in a nuclear power plant core in the event of a LOCA.

14-5

3. Multi-Rod Bundle (Assembly) Experiments

> ...I have a basic distrust of very elaborate calculations of complex situations, especially where the calculations have not been checked by full-scale experiments. As you know, much of our trust in the ECCS depends on the reliability of complex codes. It seems to me—when the consequences of failure are serious—then the ability of the codes to arrive at a conservative prediction must be verified in experiments of complexity and scale approaching those of the system being calculated. I therefore believe that serious consideration should be given first to cross-checking different codes and then to verifying ECCS computations by experiments on large scale and, if necessary, on full scale. This is expensive, but there

[67] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 1.
[68] Dr. Dana A. Powers, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Transcript, September 29, 2003, pp. 211-212.

is precedent for such experimentation—for example, in the full-scale tests on COMET and on nuclear weapons.[69]—Alvin Weinberg

Clearly, temperature controlled inductive heating and furnace experiments with Zircaloy specimens do not replicate the oxidation kinetics that would occur in a nuclear power plant's core, in the event of a LOCA. So, as discussed above, the recommendation to have "correlations...based on the critical review of experimental data...and their verification against rod bundle experiments,"[70] should be applied to zirconium-steam reaction kinetics at temperatures lower than 1900 K (2961°F), including temperatures below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

It is significant that, discussing the 2200°F PCT limit and autocatalytic (runaway) oxidation, as well as a method for assessing the conservatism of the PCT limit, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> One of the bases for selecting 2200°F (1204°C) as the PCT [limit] was that it provided a safe margin, or conservatism, away from an area of zircaloy oxidation behavior known as the autocatalytic regime. *The autocatalytic condition occurs when the heat released by the exothermic zircaloy-steam reaction (6.45 megajoules per kg zircaloy reacted) is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant.* This reaction heat then further raises the zircaloy temperature, which in turn increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.
>
> 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... [E]ven 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].[71]

---

[69] From a letter, dated February 9, 1972, from Oak Ridge National Laboratory Director Alvin Weinberg to AEC Chairman James Schlesinger; in Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 4.28-4.29.

[70] G. Schanz, "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys," Forschungszentrum Karlsruhe, FZKA 7329, 2007, p. 2.

[71] 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.

(It is noteworthy that, as discussed in PRM-50-93, according to some reports, experiments like the LOFT LP-FP-2 experiment demonstrated that autocatalytic oxidation commences at cladding temperatures as low as approximately 1400°K (2060°F);[72] therefore, a margin above 2200°F does *not* exist.)

It is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" states that "[t]he autocatalytic condition *occurs when the heat released by the exothermic zircaloy-steam reaction…is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant*;" and discusses "*thresholds for temperature excursion…when design basis heat transfer and decay heat are considered*" [emphasis added].[73]

Clearly, as argued in PRM-50-76, the experiments that the Baker-Just and Cathcart-Pawel equations are based on, "did not include any consideration of the complex thermal hydraulic conditions [that would occur] during [a] LOCA."[74] So, as stated in PRM-50-93, 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, precisely because they were not developed to consider how heat transfer would affect zirconium-water reaction kinetics.

Discussing how the oxidation rate of Zircaloy increases with increasing temperature in the conditions of excellent thermal insulation that multi-rod bundle tests provide, "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 Tests B and C)],[75] 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

---

[72] 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, pp. 30, 33.
[73] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, p. 8-2.
[74] Robert H. Leyse, "PRM-50-76," p. 1.
[75] 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.

temperature, *together with the excellent thermal insulation of the bundles* [emphasis added].[76]

(It is noteworthy that Schanz is one of the coauthors of "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).")

As stated above, 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 exothermal Zircaloy-steam oxidation would cause a temperature excursion.

a. The "Spreading Zircaloy Fires" that occurred in the Power Burst Facility Severe Fuel Damage Scoping Test and CORA-2 and CORA-3 Experiments

Regarding the rapid oxidation and "spreading zircaloy fire" that occurred in the Severe Fuel Damage Scoping Test conducted at the Power Burst Facility ("PBF") in 1982, at an Advisory Committee on Reactor Safeguards meeting, Philip MacDonald of Idaho National Engineering Laboratory stated:

14-6

> We observed rapid oxidation of the lower portion of the bundle. It wasn't expected. It cannot be calculated with existing models. It is a flame-front phenomenon which is not addressed in existing models. It will probably be addressed in coming months or years. Think of a sparkler. That kind of phenomenon. One problem with the existing models, all the axial loadings are extremely coarse. They just do not deal with a spreading zircaloy fire.[77]

The same phenomenon of "a spreading zircaloy fire" occurred and was observed by video and still cameras in the CORA-2 and CORA-3 experiments.[78] Discussing

[76] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[77] Philip MacDonald, NRC, Advisory Committee on Reactor Safeguards ("ACRS"), "Transcript of ACRS Subcommittees on Class 9 Accidents and Reactor Radiological Effects," February 22, 1983, Washington D.C., located in ADAMS Public Legacy, Accession Number: 8302240211.

[78] 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. 2.

observations of 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 Tests B and C)],[79] 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. ...
>
> This explains the observation that the temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. *From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward.* It is reasonable to assume, that the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred [emphasis added].[80]

Clearly, the temperature controlled inductive heating and furnace tests with Zircaloy specimens that the Baker-Just and Cathcart-Pawel equations are based on, did not replicate the oxidation kinetics of the "spreading zircaloy fire" in the PBF Severe Fuel Damage Scoping Test, or the "slowly moving fronts of bright light, which illuminated the bundle[s]...indicating the spreading of the temperature escalation upward and downward," that commenced at approximately 1000°C, in the CORA-2 and CORA-3 experiments.

14-7

b. The LOFT LP-FP-2 Experiment and the "Cold" Guide Tube

It is significant that in the LOFT LP-FP-2 experiment "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about...1400 K on a guide tube."

---

[79] 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.

[80] 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.

Regarding how the metal-water reaction propagates away from the initiation point, "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. … A cladding thermocouple at the same elevation…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).[81]

So, in the LOFT LP-FP-2 experiment, the cooler environment and "cold" surfaces surrounding the rapidly oxidizing fuel assembly did not prevent autocatalytic oxidation from commencing at a cladding temperature well below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

c. The Autocatalytic Metal-Water Reaction that Occurred during PWR FLECHT RUN 9573 with the Fuel Bundle Housing that "Constituted a 700°F Cold Spot"

14-8

Regarding criticisms that the fuel bundle housing in the PWR FLECHT tests "constituted a 700°F cold spot," "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> The record contains an enormous body of criticism of the PWR-FLECHT tests: in addition to the views set forth by [Consolidated National Intervenors] in its testimony, numerous critical remarks were made by experts from [Aerojet], ORNL, BMI, and others.
>
> There were substantial criticisms of the fuel bundle housing. It represented an inadequate simulation of the extended rod array found in a

[81] 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, pp. 30, 33.

> reactor core. *In the tests it constituted a 700°F cold spot.* [Rex] Shumway remarked that the temperature history of the housing was not representative of a PWR (Tr. 6781). Robert Colmar expressed his criticisms of the housing both in [his written testimony][82] and Tr. 11399-11419 [emphasis added].[83]

So, in the PWR-FLECHT tests, there were radiative heat losses from the multi-rod bundles surrounded by fuel bundle housing that "constituted a 700°F cold spot." Yet, nonetheless, as discussed in PRM-50-93, in FLECHT Run 9573, an autocatalytic oxidation reaction commenced at a temperature lower than what both the Baker-Just and Cathcart-Pawel equations would predict.

14-9

4. An Argument Against Schanz's Claim that the Baker-Just Equation is Conservative for Calculating Oxidation Kinetics for Temperatures Below 2200°F

In "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Schanz claims that the Baker-Just equation is conservative for calculating oxidation kinetics for temperatures below 2200°F.

Regarding the Baker-Just equation, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes" states:

> The Baker-Just correlation will retain its importance for comparison and licensing purposes (conservative approach). However, it should not be considered for application in best-estimate calculations. At high temperature, near the melting point of Zry, the correlation is less conservative.[84]

So despite having criticized induction heating experiments with high radiative heat losses that affect oxidation kinetics and having coauthored "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," which states

---

[82] Exhibit 1044: Testimony of Robert J. Colmar, Division of Reactor Licensing, ECCS Hearing, March 23, 1972.

[83] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," Concluding Statement—Safety Phase—Prepared by Union of Concerned Scientists on Behalf of Consolidated National Intervenors in the Matter of Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants, AEC Docket RM-50-1, April 1973, p. 5.31.

[84] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, p. 8.

that "[w]ith 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,"[85] Schanz claims that "the Baker-Just Correlation will retain its importance for comparison and licensing purposes."[86]

Indeed, it would be quite easy to disprove Schanz's claim that "the Baker-Just Correlation will retain its importance for comparison and licensing purposes," by citing experimental data from numerous papers that Schanz has coauthored (two of which are quoted from above). In making his claim, Schanz seems to have forgotten about the experimental data from many of the CORA experiments he reported on, in which there were autocatalytic oxidation reactions and temperature excursions that commenced below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

5. An Argument Against Schanz's Claim that the Cathcart-Pawel Equation is of High Reliability for Calculating Oxidation Kinetics for Temperatures Below 2200°F

As quoted above, in "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Schanz states:

14-10

> The Baker-Just correlation will retain its importance for comparison and licensing purposes (conservative approach). However, it should not be considered for application in best-estimate calculations.[87]

In the passage above, "best-estimate calculations" for licensing purposes, refers to calculations using the Cathcart-Pawel equation.

(Regulatory Guide 1.157 states that "[t]he rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam

---

[85] 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.
[86] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, p. 8.
[87] *Id.*

should be calculated in a best-estimate manner;"[88] *i.e.*, with the Cathcart-Pawel equation.[89])

Regarding the Cathcart-Pawel equation (and Leistikow correlations), in "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Schanz states:

> The Cathcart-Pawel correlations and the Leistikow correlations are judged to be of equal and high reliability. This standard is understood to result from strong efforts towards precise temperature measurement and control, the volume of the data bases and adequate and consistent evaluation procedures.[90]

(It is noteworthy that, regarding the Cathcart/Pawel furnace experiments (ZMWOK Program), "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies" states:

> The [ZMWOK] Program has yielded a set of isothermal reaction rate data for the oxidation of Zircaloy-4 in steam between 900 and 1500°C (1652-2732°F). ...
>
> The ZMWOK data set provides a basis for quantifying the degree of conservatism of the Baker-Just correlation for the oxidation rate of Zircaloy. Under the conditions used for our experiments, the Baker-Just relationship predicts oxidation rate constants 32, 78, and 147% higher than [the Cathcart-Pawel correlation] at temperatures of 1000, 1200, and 1500°C (1832, 2192, and 2732°F), respectively.[91]

The passage above, states that the Baker-Just correlation is more conservative than the Cathcart-Pawel correlation. However, this means that, in fact, the Cathcart-Pawel correlation is more non-conservative than the Baker-Just correlation.)

---

[88] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, p. 6.
[89] NRC, Regulatory Guide 1.157, p. 6, states that "[t]he data of ["Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies"] are considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F;" J. V. Cathcart et al., "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977.
[90] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, p. 8.
[91] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, pp. 117, 118.

So despite having coauthored "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," which states that "[w]ith 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,"[92] Schanz claims that the Cathcart-Pawel equation is of high reliability for calculating oxidation kinetics for temperatures below 2200°F.[93]

Indeed, it would be quite easy to disprove Schanz's claim that the Cathcart-Pawel equation is of high reliability for calculating oxidation kinetics for temperatures below 2200°F, by citing experimental data from numerous papers that Schanz has coauthored (two of which are quoted from above). In making his claim, Schanz seems to have forgotten about the experimental data from many of the CORA experiments he reported on, in which there were autocatalytic oxidation reactions and temperature excursions that commenced below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

14-11

6. The "Verification" of a Set of Correlations for SFD Codes for the High Temperature Range

Regarding recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F), "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys" states:

> Recently a set of correlations was recommended, based on the critical review of experimental data,[94] their statistical evaluation within the diffusion system concept, and their verification against rod bundle experiments.[95, 96]

---

[92] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "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, p. 83.

[93] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, p. 8.

[94] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," Forschungszentrum Karlsruhe, FZKA 6827, 2003.

[95] A. Volchek, *et al.*, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations, Part I. Experimental Database and Basic Modeling, Part II. Best-Fitted Parabolic Correlations, Part III. Verification Against Representative Transient Tests," Nuclear Engineering and Design, 232, 2004, pp. 75-109.

[96] G. Schanz, "Semi-Mechanistic Approach for the Kinetic Evaluation of Experiments on the Oxidation of Zirconium Alloys," FZKA 7329, p. 2.

According to "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations: Part III. Verification Against Representative Transient Tests," the rod bundle experiments that "verified" the set of recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F) were the QUENCH-06 and PHEBUS B9+ experiments.

It is significant that "[t]he bundle integral experiments QUENCH-06 and PHEBUS B9+ did not lead to extremely large temperature excursions."[97] So the set of recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F) were not verified by the results of rod bundle experiments that had significant temperature excursions—like the CORA-2 and CORA-3 experiments—that commenced below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. Nor were the set of recommended correlations verified by the results of the LOFT LP-FP-2 experiment.

The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[98]

And regarding the importance 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:

> The [LOFT LP-FP-2] experiment...provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> 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.[99]

And 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

---

[97] F. Fichot, *et al.*, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations: Part III. Verification Against Representative Transient Tests," Nuclear Engineering and Design, 232, 2004, p. 97.

[98] 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.

[99] 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.

14-12

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.

(It is noteworthy that in PRM-50-93, on page 40, Petitioner erroneously states that "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states that the peak measured cladding temperature reached 2100°K (3320°F) within approximately 35 seconds and that after the onset of rapid oxidation, cladding temperatures increased at an average rate of approximately 20°K/sec. (36°F/sec.); according to the paper average rate was approximately 10°K/sec. (18°F/sec.). However, according to other reports the heat up rate was between 10°K/sec and 20°K/sec.)

Clearly, the set of recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F) would not be able to be validated by rod bundle experiments, like the LOFT LP-FP-2 experiment, that had temperature excursions that commenced below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

7. The LOFT LP-FP-2 Experiment and the Validation of the ICARE/CATHARE and ASTEC Codes

a. The Treatment of Zirconium Oxidation Kinetics in Severe Accident Codes and the ICARE/CATHARE Code

Regarding high-temperature correlations for SFD codes, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations: Part III. Verification Against Representative Transient Tests" states:

> The treatment of zirconium oxidation kinetics in severe accident (SA) codes has been the subject of many discussions and controversies in recent years. The main problem was the existence of several correlations which could lead to large differences in the calculated results. It appeared clearly that there was a need to converge towards a common understanding of the physical processes that must be modeled (oxygen diffusion, blanketing effect, etc.) and an agreed database among code developers and users. It would help reducing an important source of uncertainties in SA calculations.

> The kinetic correlation database, obtained as a result of examination of complementary experimental data in Parts I[100] and II,[101] is applied here to analyze a few high-temperature separate-effects tests and bundle experiments where Zry oxidation reaction played a dominant role. The ICARE/CATHARE computer code developed by [Institut de Radioprotection et de Sûreté Nucléaire], is used to check the validity of the high-temperature correlations derived in Parts I and II.[102]

So it was the ICARE/CATHARE code that "verified" the set of recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F), with the results of the QUENCH-06 and PHEBUS B9+ experiments.

According to a JSRI Projects report from 2001, the ICARE/CATHARE V1 code had a validation program, which included validation with the LOFT LP-FP-2 experiment. Unfortunately, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations: Part III. Verification Against Representative Transient Tests," did not discuss the ICARE/CATHARE analysis of the LOFT LP-FP-2 experiment.

b. The ASTEC Code

Regarding the Accident Source Term Evaluation Code ("ASTEC"), "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> ASTEC is an integral code jointly developed by IRSN (France) and GRS (Germany) to assess the whole sequence of a severe accident in nuclear power plants (NPP), from the initiating event up to fission product (FP) release and behavior in the containment, and finally radioactive release out of the containment. The code consists of several coupled modules, each one of them dealing with different severe accident phenomena or NPP zones. Among them, the CESAR module, which computes the two-phase thermal-hydraulics in primary and secondary systems, is coupled to the

---

[100] G. Schanz, *et al.*, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations, Part I. Experimental Database and Basic Modeling," Nuclear Engineering and Design, 232, 2004, pp. 75-84.

[101] A. Volchek, *et al.*, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations, Part II. Best-Fitted Parabolic Correlations," Nuclear Engineering and Design, 232, 2004, pp. 85-96.

[102] F. Fichot, *et al.*, "Advanced Treatment of Zircaloy Cladding High-Temperature Oxidation in Severe Accident Code Calculations: Part III. Verification Against Representative Transient Tests," Nuclear Engineering and Design, 232, 2004, p. 97.

DIVA module able to calculate core degradation, corium relocation and behavior in the lower head up to vessel failure. Most DIVA models are issued from the ICARE2 IRSN mechanistic code for core degradation (Chatelard, et al., 2006), except some fast-running models that were specifically developed for ASTEC (2D core gas thermal-hydraulics and corium behavior in the lower plenum).

Many partners of the SARNET network of excellence (in the 6th Framework Programme of the European Commission) were involved in ASTEC V1 code validation against experiments. This paper summarizes the main results of the validation performed on the CESAR and DIVA modules of the successive code versions up to ASTEC V1.3rev2 delivered in December 2007. Table 1 presents the selected experiments that include several International Standard Problems (ISP) of OECD/CSNI (Committee on the Safety of Nuclear Installations).[103]

The LOFT LP-FP-2 experiment is listed among the experiments in Table 1. "CESAR and DIVA Module Validation Tasks." The LOFT LP-FP-2 experiment is the only experiment listed to validate both the CESAR and DIVA modules, for their simulations of reactor cooling system thermal-hydraulics and core degradation, respectively. The TMI-2 accident is also listed to validate phenomena modeled by both the CESAR and DIVA modules.

8. The LOFT LP-FP-2 Experiment Simulated by the ASTEC V1 and ICARE/CATHARE Codes

14-13

It is significant that, regarding an ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17),[104] *but the onset of temperature escalation in the upper part of the CFM was delayed* [emphasis added].[105]

In "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17,[106] the graph of the cladding-temperature values depicts

[103] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," Progress in Nuclear Energy, 52, 2010, pp. 148-149.
[104] See Appendix A Fig. 17. LOFT LP-FP-2 CFM Clad Temperature at Elevation 1.067 m.
[105] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," p. 155.

that the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment has the onset of the temperature escalation (at the 1.067 m. elevation) occurring at a temperature greater than 1700 K (2600°F); figure 17 also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m. elevation) occurred at a temperature well below 1500 K (2240°F). So the difference between the calculated and actual experimental value for the onset of the temperature escalation, at the 1.067 m. elevation is greater than 200 K (360°F)—a significant difference.

(It is noteworthy that 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), at the 0.69 m. elevation.)

Again, as stated above, the set of recommended correlations for SFD codes for temperatures greater than 1900 K (2961°F) would not be able to be validated by rod bundle experiments, like the LOFT LP-FP-2 experiment, that had temperature excursions that commenced below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(It is noteworthy that, regarding the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment during reflood, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflooding were not reproduced by ASTEC due to lack of adequate modeling.[107])

It is significant that, regarding an ASTEC V1 analysis the LOFT LP-FP-2 experiment that was compared with the results of an ICARE/CATHARE code analysis, ENEA's "2006 Progress Report" states:

> LOFT LP-FP-2 experiment analysis. The LOFT LP-FP-2 [experiment], performed in the Loss-of-Fluid Test (LOFT) facility at the Idaho National Engineering Laboratory (INEL) USA to provide information on fuel rod behavior, hydrogen generation, and fission-product release during a loss-of-coolant accident scenario in a pressurized water reactor (PWR) up to core reflood, was analyzed with ASTEC V1 to assess the ability of the

[106] See Appendix A Fig. 17. LOFT LP-FP-2 CFM Clad Temperature at Elevation 1.067 m.
[107] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," p. 155.

code to stimulate thermal-hydraulic conditions and core degradation phenomena. The ASTEC results were then compared with the results of the ICARE/CATHARE code.

ASTEC simulates reasonably well the transient phase of the experiment before the reflood phase, that is, reactor system thermal-hydraulics, core uncovery and heatup, hydrogen generation and fission-product release. The total hydrogen release is in good agreement with test measurements. Instead the code needs some improvement in order to investigate the reflood phase *because temperature excursions and consequent heavy degradation of the fuel rods, hydrogen release and primary pressure increase are not reproduced by ASTEC because of the inadequate modeling.*

In general, the ICARE/CATHARE results confirm the validity of the ASTEC results [emphasis added].[108]

Unfortunately, the passage above, does not discuss the results of the ICARE/CATHARE analysis of the LOFT LP-FP-2 experiment. However, it is clear that there are serious problems with ASTEC's prediction of the onset of the temperature escalation that occurred in the LOFT LP-FP-2 experiment: the difference between the calculated and actual experimental value for the onset of the temperature escalation, at the 1.067 m. elevation is greater than 200 K (360°F).

14-14

9. It is Highly Problematic that Data from the LOFT LP-FP-2 Experiment and Other SFD Experiments has Not been Considered Relevant to ECCS Evaluation Calculations for Design Basis Accidents

As quoted above, regarding an ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17),[109] *but the onset of temperature escalation in the upper part of the CFM was delayed* [emphasis added].[110]

---

[108] ENEA, Nuclear Fusion and Fission, and Related Technologies Department, "2006 Progress Report," pp. 109-110.
[109] See Appendix A Fig. 17. LOFT LP-FP-2 CFM Clad Temperature at Elevation 1.067 m.
[110] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," p. 155.

And as discussed above, in "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17, the graph of the cladding-temperature values depicts that the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment has the onset of the temperature escalation (at the 1.067 m. elevation) occurring at a temperature greater than 1700 K (2600°F); figure 17 also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m. elevation) occurred at a temperature well below 1500 K (2240°F). So the difference between the calculated and actual experimental value for the onset of the temperature escalation, at the 1.067 m. elevation is greater than 200 K (360°F)—a significant difference.

It is clear that there are serious problems with ASTEC's prediction of the onset of the temperature escalation that occurred in the LOFT LP-FP-2 experiment. Yet "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" concludes that "[g]ood results have been obtained for early-phase models of core heat-up [and] oxidation...for all calculated experiments;"[111] the LOFT LP-FP-2 experiment is listed among the calculated experiments.

Clearly, "good results" were *not* obtained for early-phase models of core heat-up and oxidation for the LOFT LP-FP-2 experiment: the difference between the calculated and actual experimental value for the onset of the temperature escalation, at the 1.067 m. elevation is greater than 200 K (360°F).

(It is noteworthy that 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), at the 0.69 m. elevation.)

Furthermore, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" is a European paper but it does not raise any concerns over the fact some reports state that the temperature excursion in the LOFT LP-FP-2 experiment commenced at a cladding temperature below the European PCT limit.

(It is noteworthy that Petitioner has not found any papers raising any concerns over the fact some reports state that the temperature excursion in the LOFT LP-FP-2

[111] *Id.*, p. 156.

experiment commenced at a cladding temperature below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.)

It is significant that "European Validation of the Integral Code ASTEC (EVITA)" states:

> Severe accident management (SAM) measures are currently being developed and implemented at nuclear power plants (NPP) worldwide in order to prevent or mitigate severe accidents. This needs a deep understanding of processes leading to severe accidents and of phenomena related to them. As greater account of severe accident measures is taken in the regulation of plants, there will be the need to show a greater degree of validation of codes and a better understanding of uncertainties and their impact on plant evaluations.[112]

Clearly, it would help to prevent severe accidents at nuclear power plants worldwide by first acknowledging that the temperature excursion in the LOFT LP-FP-2 experiment commenced at a cladding temperature of approximately 1400 K. Then it would help to correct the current deficiencies of ECCS evaluation models for design basis accidents; *i.e.*, their problems calculating metal-water reaction rates. The rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations must be based on data from multi-rod (assembly) severe fuel damage experiments.

And it would help to lower PCT limits worldwide to values that provide necessary margins of safety. In the United States, the NRC must lower the 10 C.F.R. § 50.46(b)(1) PCT limit to a value that provides a necessary margin of safety.

III. Conclusion

Unfortunately, experiment conductors and reviewers and regulators have not acknowledged that the rapid oxidation and temperature excursions that occurred at "low" temperatures in multi-rod (assembly) severe fuel damage experiments like the LOFT LP-FP-2 experiment are pertinent to ECCS evaluation models for design basis accidents. Experimental data that indicates the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-

[112] H. J. Allelein, *et al.*, "European Validation of the Integral Code ASTEC (EVITA)," Nuclear Engineering and Design, 221, 2003, p. 96.

conservative has not been considered pertinent for predicting the phenomena that would occur in the event of a LOCA.

(Such experimental data also 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 these equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.)

For example, as quoted above, "The History of LOCA Embrittlement Criteria," presented in October 2000, by G. Hache of Institut de Protection et de Sûreté Nucléaire, Cadarache, France and H. M. Chung of Argonne National Laboratory, Argonne, Illinois, USA states:

> The 2200°F (1204°C) peak cladding temperature (PCT) criterion was selected on the basis of Hobson's slow-ring-compression tests that were performed at 25-150°C. Samples oxidized at 2400°F (1315°C) were far more brittle than samples oxidized at <2200°F (<1204°C) in spite of comparable level of total oxidation. ... *Consideration of potential for runaway oxidation alone would have [led] to a PCT limit somewhat higher than 2200°F (1204°C)* [emphasis added].[113]

So, clearly, the Institut de Protection et de Sûreté Nucléaire and Argonne National Laboratory and their various counterparts, still have not acknowledged that in multi-rod bundle experiments, like the LOFT LP-FP-2 experiment, the onset of runaway oxidation commenced at temperatures below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F (1204°C).

Furthermore, experiment conductors and reviewers and regulators continue to believe that the data from temperature controlled, isothermal reaction kinetics experiments with Zircaloy tube specimens such as those conducted by Cathcart and Pawel are pertinent to ECCS evaluation models for design basis accidents.

In the MaxiZWOK experiment, at 1000°C (1832°F), the Zircaloy specimen was able to return to its steady-state value and the specimen self-heating was able to dissipate;

[113] G. Hache and H. M. Chung, "The History of LOCA Embrittlement Criteria," Proc. 28th Water Reactor Safety Information Meeting, Bethesda, USA, October 23-25, 2000, pp. 27-28.

however, in a Zircaloy bundle experiment between 1000°C and 1200°C there would be a significant temperature excursion.

(It is noteworthy that "[t]he bulk of the reaction rate experiments [conducted by Cathcart and Pawel] were performed in the MiniZWOK apparatus."[114])

Experiment conductors and reviewers and regulators have not acknowledged that 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 exothermal Zircaloy-steam oxidation would cause a temperature excursion.

Clearly, the Cathcart-Pawel equation is an equation for predicting the oxidation kinetics of Zircaloy specimens in furnaces; it is not adequate for predicting the oxidation kinetics of Zircaloy bundles in a nuclear power plant core in the event of a LOCA.

And deficient ECCS evaluation models that use the Cathcart-Pawel and Baker-Just equations cannot realistically model the phenomena that would occur in the event of a LOCA. Deficient ECCS evaluation models are also potentially dangerous because they provide erroneous simulations of the phenomena that would occur in the event of a LOCA.

For example, the ECCS evaluation calculations that helped qualify Indian Point Unit 2's ("IP-2") 2004 stretch power uprate, calculated IP-2's PCT at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies during a postulated large break ("LB") LOCA.[115] This is highly problematic because, with high probability, if there were a LB LOCA at IP-2, there would be a partial or complete meltdown.

This is demonstrated by examining data from multi-rod (assembly) severe fuel damage experiments. During the LOFT LP-FP-2 experiment, when peak cladding

---

[114] 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. 14.

[115] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," October 27, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042960007, Enclosure 2, p. 18.

temperatures reached between approximately 2060°F[116] and 2240°F,[117] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec.;[118] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[119]

And the CORA experiments demonstrated that with good fuel assembly insulation—like what the core of a nuclear power plant has—that cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, starts when the cladding reaches between 2012°F and 2192°F; cladding temperatures then start increasing at a maximum rate of 27°F/sec.[120]

So, in the event of a LOCA at IP-2, if peak cladding temperatures exceeded approximately 2060°F, with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec or greater. Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[121] the melting point of Zircaloy is approximately 3308°F.[122]

Furthermore, there are other deficiencies in the NRC's and nuclear industry's ECCS evaluation models, discussed in PRM-50-93. Such deficiencies must be corrected.

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

---

[116] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.

[117] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.

[118] *Id.*

[119] F. E. Panisko, N. J. Lombardo, "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," p. 282.

[120] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "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, p. 83.

[121] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p 23.

[122] 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.

Respectfully submitted,

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

Dated: April 12, 2010

Appendix A Fig. 17. LOFT LP-FP-2 CFM Clad Temperature at Elevation 1.067 m.[1]

[1] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," Progress in Nuclear Energy, 52, 2010, p. 155.

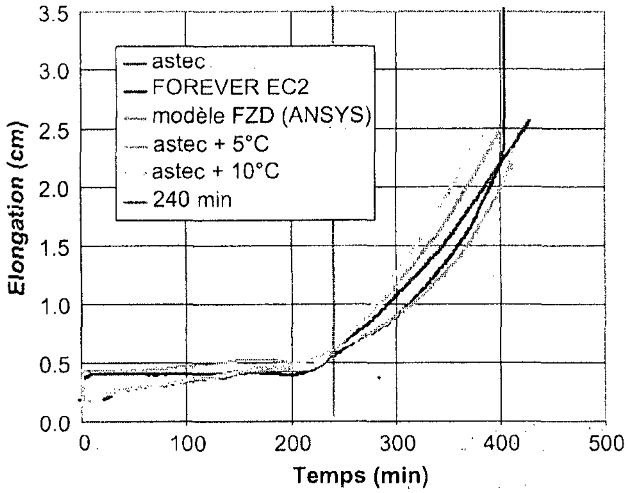


Fig. 16. FOREVER EC2 lower head displacement.

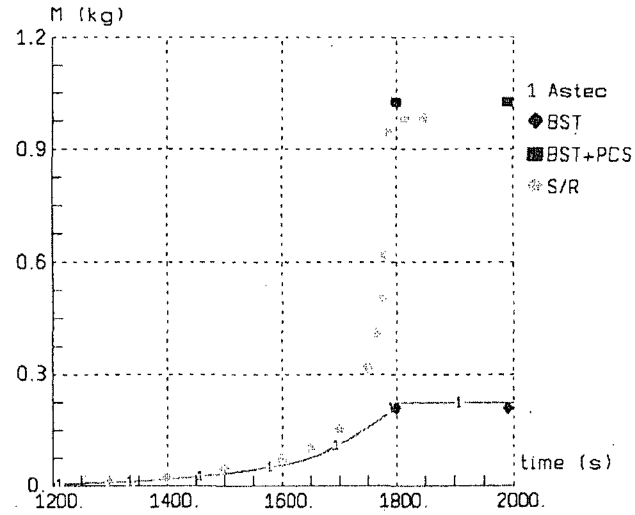


Fig. 18. LOFT-LP-FP-2 total mass of hydrogen produced.

4. ASTEC validation for coupling of circuit thermal-hydraulics and core degradation

The main results of CESAR and DIVA coupling validation work on the LOFT-LP-FP-2 experiment and on the TMI-2 accident are presented below.

4.1. LOFT-LP-FP-2 experiment

LP-FP-2 was the second experiment performed in the Loss-of Fluid Test (LOFT) facility at INEL (USA) under the sponsorship of OECD. The objectives of the test were to provide information on fuel rod behaviour, hydrogen generation, and FP release and transport during a LOCA accident scenario that resulted in severe core damage. The initial conditions of the experiment represented typical commercial PWR operating conditions. The simulated accident scenario was a pipe break in the low pressure injection system line, which represents a potential pathway for the release of primary coolant from the reactor vessel to the containment. The transient was terminated by core reflooding.

Although the hydraulic separation between the Central Fuel Module (CFM) and the surrounding driver core zone could not be simulated in a simple way with ASTEC V1.3, the thermal-hydraulic behaviour of the circuits was reasonably well predicted by the code. Primary system pressure was underestimated at the end of the transient during the bundle degradation: it is likely due to under prediction of heat transfer to primary fluid from hot vessel lower plenum structures. The onset of core uncovery and heat-up was very well reproduced by ASTEC (Fig. 17), but the onset of temperature escalation in the upper part of the CFM was delayed. The total mass of hydrogen produced before reflooding was very well predicted by the code (Fig. 18). In spite of the high CFM temperatures reached, the FP release fractions calculated by ASTEC before reflooding were lower than expected but, however, in reasonable agreement with the test estimations. High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflooding were not reproduced by ASTEC due to lack of adequate modelling.

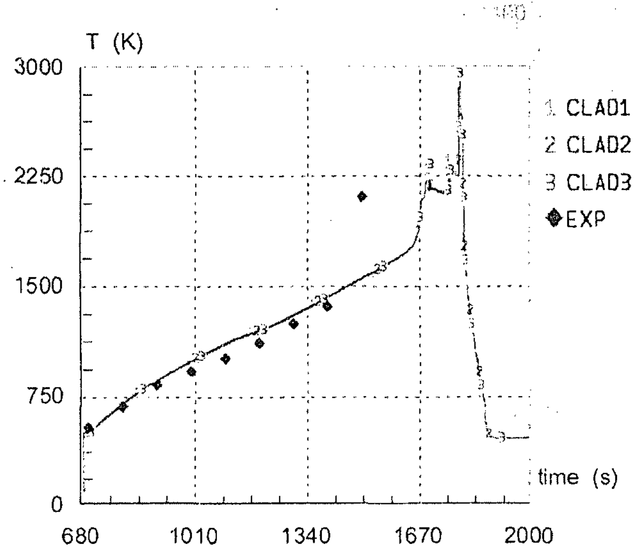


Fig. 17. LOFT-LP-FP-2 CFM clad temperature at elevation 1.067 m.

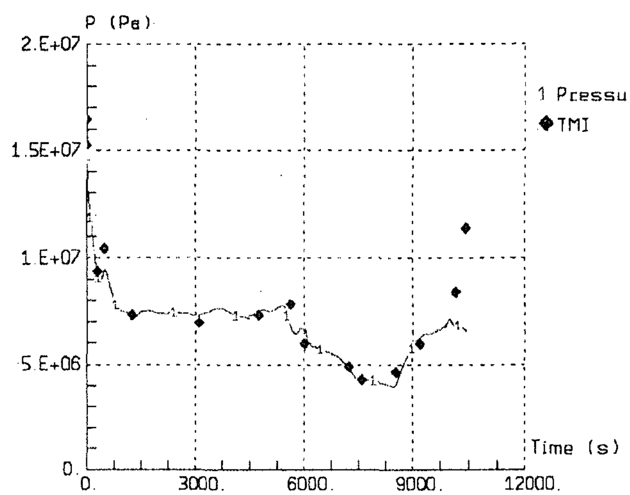


Fig. 19. TMI-2 primary system pressure.

Appendix B Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases and Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core[2]

[2] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, pp. 196, 205.

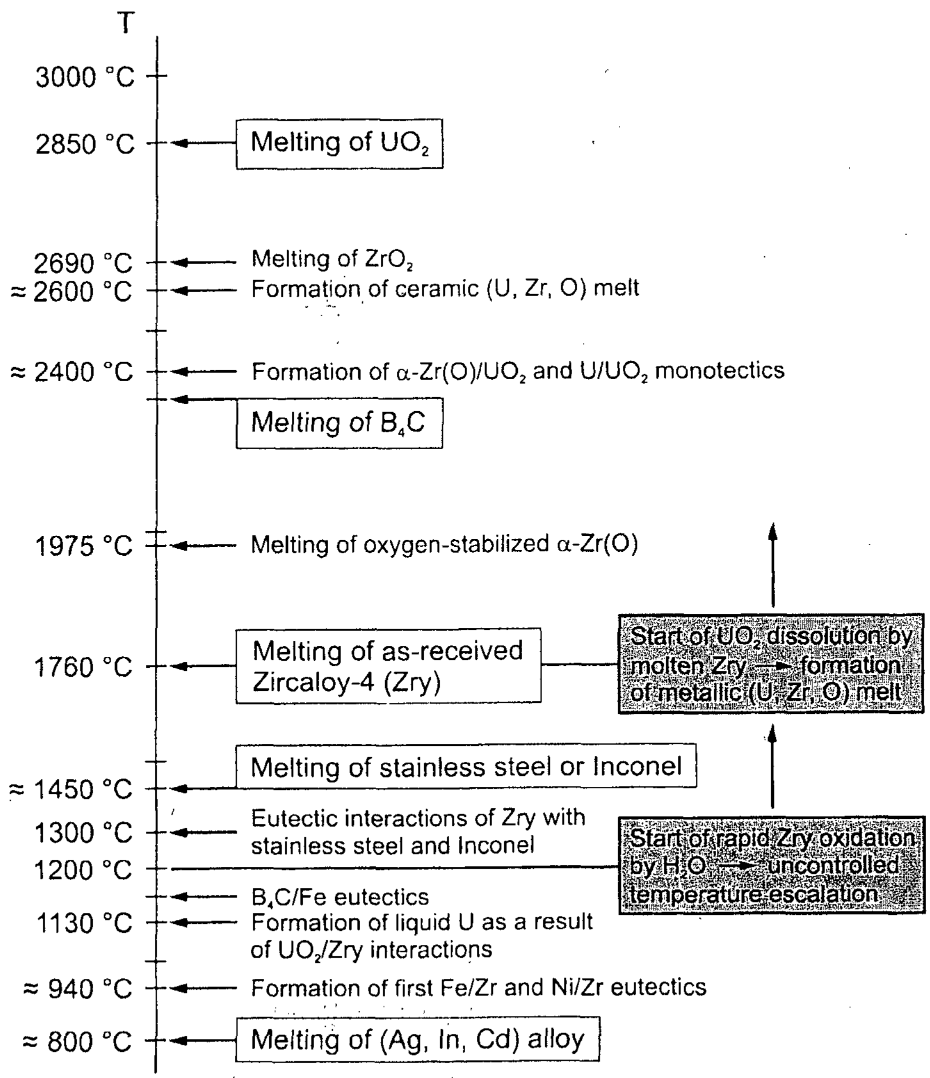


Fig. 1. LWR severe accident-relevant melting and chemical interaction temperatures which result in the formation of liquid phases.

- eutectic and monotectic reactions between α-Zr(O) and UO_2,
- melting of ZrO_2 and UO_2 forming a ceramic Zr–U–O melt,
- formation of immiscible metallic and ceramic melts in different parts of the reactor core,
- relocation of the solid and liquid materials into the lower reactor pressure vessel (RPV) head, and
- thermal, mechanical and chemical attack of the RPV wall.

At temperatures above 1200°C the rapid oxidation of Zircaloy and of stainless steel by steam results in local uncontrolled temperature escalations within the core with peak temperatures >2000°C. As soon as the Zircaloy cladding starts to melt (>1760°C), the solid UO_2 fuel may be chemically dissolved and thus liquefied about 1000 K below its melting point. As a result, liquefied fuel relocations can already take place at about 2000°C.

Many of these physical and chemical processes have been identified in separate-effects tests, out-of-pile and in-pile integral severe fuel damage (SFD) experiments, and Three Mile Island Unit 2 (TMI-2) core material examinations [5–10,33]. All of these interactions are of concern in a severe accident, because relocation and/or solidification of the resulting fragments or melts may result in local cooling channel blockages of different sizes and may cause further heatup of these core regions steam starvation. At high heat-up rates >5 K/s, the ZrO_2 layer will probably be too thin to hold the metallic melt in place and relocation will occur after mechanical and/or chemical breach of the ZrO_2 shell (Fig. 13).

It is evident from the foregoing discussion that the in-vessel melt progression process is very complex. It can only be understood by a combination of experiments and computer modeling and careful verification and validation of such codes. This requires detailed and thorough analysis of the out-of-pile and in-pile tests, the large-sized LOFT LP-FP2 experiment, and the TMI-2 accident. Both TMI-2 and LOFT LP-FP2 can be linked to smaller scale separate-effects tests to look at particular phenomena. The computer models, when validated against these smaller scale experiments, must allow application to reactor plant conditions where scaling effects become important.

5.3. Material distribution in integral experiments

The materials redistribution within the various types of fuel elements examined in the integral test program

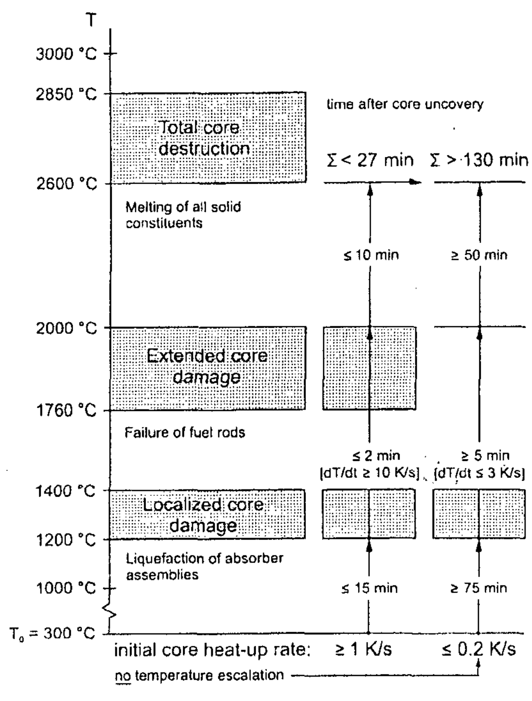


Fig. 13. Dependence of the temperature regimes on liquid phase formation on the initial heat-up rate of the core. Small heat-up rates drastically reduce the amount of molten Zircaloy (1800–2000°C) and give more time for possible accident management measures.

CORA showed interesting results [26]. The absorber materials initiate melt formation and melt relocation and shift the temperature escalation as a result of the zirconium–steam reaction to the lower end of the bundle by the relocation, i.e., by movement of molten (hot) material. The relocation of melts occurs by rivulet and droplet flow. The various melts solidify on cool-down at different temperatures, i.e., at different axial locations. The viscosity of the molten material has an impact on the relocation behavior and has to be considered in modeling of these phenomena [37]. Material relocations induce a temperature escalation at about 1200°C. The release of chemical energy results in renewed melt formation and relocation. Therefore, the processes are closely coupled. Pre-oxidation of the cladding results in reduced melt formation and shifts the onset of temperature escalation to higher temperatures. Inconel and stainless steel spacers relocate above 1250°C as a result of chemical interactions and do not act as materials catchers. Pre-oxidized Zircaloy spacers still exist at temperatures >1700°C and therefore have a significant impact on the relocation processes at lower temperatures [26].

The CORA-10 test simulated the behavior of a rod bundle with additional cooling at its lower end (TMI-2 conditions) [34]. Fig. 14 depicts the axial bundle temperature profile at different times and the material relocation. One can recognize the influence of the higher heat losses at the lower end (30 cm) of the bundle in the axial temperature profiles. Two steep axial temperature gradients form at 4400 s, one at 45 cm and one at the 30 cm bundle elevation. Corresponding to the steep axial temperature gradients, the main blockage formed at the 40 cm bundle elevation. The absorber rods cannot be found in the cross sections as a result of liquefaction and relocation. A part of the UO_2 was dissolved by molten Zircaloy and relocated [26].

The axial material distributions of CORA-W1 [35] and CORA-W2 [36] are compared in Fig. 15, together with the boundary conditions of the experiments. The two tests were performed with fuel-element components typical of Russian type VVER-1000 reactors, Zr 1% Nb fuel rod cladding, and B_4C absorber material in stainless steel cladding. Fig. 15 underlines the extraordinary influence of the low-temperature eutectic interaction between B_4C and stainless steel on melt relocation, damage progression, and blockage formation. The absorber material interactions initiate the formation of liquid phases. Relocating melts transport heat to lower bundle positions and initiate the exothermic zirconium–steam reaction, which leads to a renewed temperature increase, melt formation, and relocation. Compared with the CORA-W1 bundle, the axial region of fuel rod damage in the CORA-W2 bundle extended to the very lowest end of the bundle, despite the fact that the input of electrical energy was smaller [26].

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Sunday, April 11, 2010 8:56 PM
To: Rulemaking Comments
Subject: NRC-2009-0554
Attachments: Comment on PRM-50-93.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is a cover letter and my response, dated April 12, 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.

Sincerely,

Mark Leyse

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Sun, 11 Apr 2010
20:56:09 -0400
X-Ironport-ID: mail2
X-SBRS: 3.8
X-MID: 14825005
X-fn: Comment on PRM-50-93.pdf
X-IronPort-AV: E=Sophos;i="4.52,186,1270440000";
d="pdf'?scan'208";a="14825005"
Received: from mail-pw0-f41.google.com ([209.85.160.41]) by mail2.nrc.gov
with ESMTP; 11 Apr 2010 20:56:02 -0400
Received: by pwi2 with SMTP id 2so3982266pwi.14 for
<rulemaking.comments@nrc.gov>; Sun, 11 Apr 2010 17:55:59 -0700 (PDT)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:date:received:message-id
:subject:from:to:content-type;
bh=6SvZJBoxykcJw6pL52Q4EuUiotZjdRYY734JeGCD7a8=;

b=MaZPBvdCMR1SkPlRLe8WCCGOmyLj1RNdyOJxCWBtX2B/cAmFNVXDfte6csUyfdfLOy
Ty4HDt1E+tOpQF4z2SJdBQ/5tgLEtQ36rRRdNMgB6xiK8xByU+7Hk1YbowlDvvAwbmu/
e8nwoc62JM60c1w9H+Byz6LbrlfZ2W/EuiaSA=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:content-type;
b=pC44mF9/T37AVxyk9EYuJyk6QKIzR+grzfeWWqRPkyjrU3ZgG+pCM8Vhd/2lczhJjF

47WZbGD1WpmFUBrNx5D+Xb8UTz9KYAeSBY0bu/cyKXfQkT2FgZP9Lv3yZcs09algWcV1
EUvlmpUawyQwxs9drZ6+bE8vfxnPF3+FJjDYM=
MIME-Version: 1.0
Received: by 10.142.44.15 with HTTP; Sun, 11 Apr 2010 17:55:57 -0700 (PDT)
Date: Sun, 11 Apr 2010 20:55:57 -0400
Received: by 10.142.119.33 with SMTP id r33mr1286530wfc.213.1271033757792;
Sun, 11 Apr 2010 17:55:57 -0700 (PDT)
Message-ID: <y2xedacd5761004111755nf28539aar626df27a2d25c7f9@mail.gmail.com>
Subject: NRC-2009-0554
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>
Content-Type: multipart/mixed; boundary="001636e8ff91a186890483ff9bda"
Return-Path: markleyse@gmail.com

Submission ID 15
Paul Gunter, Beyond Nuclear
ML101030142

Beyond Nuclear
6930 Carroll Avenue, Suite 400,
Takoma Park, MD 20912
Tel: 301.270.2209 Fax: 301.270.4000
Email: paul@beyondnuclear.org
Web: www.beyondnuclear.org

PRM-50-93
(75FR03876)

15

DOCKETED
USNRC

April 12, 2010 (3:10pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 12, 2010

The Secretary
United States Nuclear Commission
Washington, DC 20555-0001
Attention: Rulemaking and Adjudications Staff
By email to Rulemaking.Comments@nrc.gov
FAX to 301-415-1677

Comments of Beyond Nuclear in Support of Petition for Rulemaking of Mark Leyse (PRM 50-93)

Ms. Secretary,

15-1

On behalf of Beyond Nuclear, I am submitting supporting comments for PRM 50-93. The NRC should adopt the petition for rulemaking submitted by Mark Leyse in the best interest of public safety.

On March 28, 1979, the United States of America experienced what was thought to be an inconceivable event when the Three Mile Island Unit 2 reactor near Harrisburg, Pennsylvania had a nuclear meltdown. The nuclear industry and its apologists still insist that there have been no human health consequences from the accident despite convincing evidence to the contrary.

Template=SECY-067

DS10

Two such compelling commemorative presentations on the TMI accident and its consequences can be viewed at http://www.tmia.com/march26 .

The Leyse petition raises the concern that safety margins that determine Nuclear Regulatory Commission (NRC) regulation and current industry practice are based on selectively cherry picking through experimental test data that otherwise points to less than adequate safety margins for maintaining the first protective boundary against another such accident and a catastrophic radiation release, the nuclear fuel rod cladding.

Mr. Leyse has filed the petition seeking to raise technical safety margins for reactor systems. The petition raises the concern that the nuclear power industry does not now have adequate safety margins against the consequences of a Loss of Coolant Accident (LOCA) and unduly risks another core melt accident and potentially large radioactive releases worse than what occurred at TMI.

Mr. Leyse focuses on two critical and credible technical issues that regard such a LOCA at a US reactor: 1) the temperature at which these nuclear fuel rods must be maintained by emergency core cooling systems to prevent another meltdown and; 2) the rate at which emergency cooling water is introduced to re-flood the reactor vessel to cover the reactor core following a significant loss of reactor coolant. Current NRC regulations require that following a loss of coolant accident fuel cladding temperatures be maintained by emergency cooling systems to remain below 2200^{O} Fahrenheit (F). This temperature is calculated by NRC and industry as an adequate safety margin against a core meltdown or "runaway oxidization." Mr. Leyse persuasively argues that the NRC regulations need to be revised to lower the fuel cladding temperature (Peak Cladding Temperature) to at least 1800^{O} F to maintain an adequate safety margin. Mr. Leyse has based his argument for the revised regulation of the fuel cladding temperature margin on extensive documentation of actual mock-up experiments including those sponsored by the NRC in 1985 that demonstrated that such a runaway oxidization of Zircaloy fuel cladding can occur at 2060^{O} F, well below the current legal safety margin limit of 2200^{O} F. The experiment demonstrated that once the fuel cladding temperature exceeds 2060^{O} F runaway oxidation can occur and within less than 60 seconds increase to 3300^{O} F, the melting point of the cladding material.

The Leyse petition for rulemaking further argues that NRC must shorten the re-flood delay time and increase the re-flood rate within the reactor vessel to recover the core with water before a runaway fuel melt accident can initiate.

The nuclear industry uses Zircaloy, an alloy of zirconium, as the cladding material for its uranium fuel rod assemblies. If ignited in a nuclear accident, the Zircaloy fuel cladding will burn in an intensely hot flare-like reaction and in a water/steam rich environment generate explosive hydrogen gas that can detonate and endanger a nuclear reactor containment structure and downwind communities. In fact, this is what happened during the Three Mile Island accident as expertly explained by in the above mentioned and hyperlinked presentation by Arnie Gunderson at the 30th commemoration of the Three Mile Island accident in Harrisburg, Pennsylvania.

The Leyse petition raises serious concerns that the Nuclear Regulatory Commission and the nuclear industry have selectively excluded multiple-rod severe fuel rod damage test experiments to arrive at their calculated "conservative" safety margins. Leyse has likened the current NRC/industry Zircoloy cladding margins as being based on "studying a burning match to predict what would occur in a forest fire."

For decades now, the nuclear power industry has prioritized raising the thermal energy and narrowing safety margins in its reactors to build more steam and more power by as much as 18% to 20% in a process called "power uprate." The Leyse petition raises particularly legitimate issues for the adequacy of existing technical specifications and safety margins at these uprated operating reactors and cause for concern of current public safety.

The issues raised by the Leyse petition need to be addressed with the agency's priority set on raising the bar for public safety and not an industry production agenda. Incidents where management and regulator have collaborated to subordinate safety to production, ignoring obvious warning signs such as surfaced at the Davis-Besse nuclear power station in 2002 will only serve to undermine public confidence that the agency is true to its mandate to promote public health and safety first.

By adopting the petition, the agency can build this public confidence and demonstrate that its priorities are indeed focused first on public safety.

Sincerely,

----/s/------

Paul Gunter, Director
Reactor Oversight Project
Beyond Nuclear

Rulemaking Comments

From: Paul Gunter [paul@beyondnuclear.org]
Sent: Monday, April 12, 2010 2:49 PM
To: Rulemaking Comments
Subject: Comments of Beyond Nuclear in support of PRM 50-93
Attachments: beyond_nuclear_04122010_comment_prm50-93.pdf

To whom it may concern:

Attached please find the comments of Beyond Nuclear in support of PRM 50-93.

Thank you,

Paul Gunter, Director
Reactor Oversight Project
Beyond Nuclear
6930 Carroll Avenue Suite 400
Takoma Park, MD 20912
Tel. 301 270 2209
www.beyondnuclear.org

Received: from mail2.nrc.gov (148.184.176.43) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Mon, 12 Apr 2010
14:49:11 -0400
X-Ironport-ID: mail2
X-SBRS: 4.5
X-MID: 14854740
X-fn: beyond_nuclear_04122010_comment_prm50-93.pdf
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
Ak4DAP0Hw0vRVdnckWdsb2JhbACPXwEBi1IIFQEBAQEJCwoHEQMfvQCCc4IZBA
X-IronPort-AV: E=Sophos;i="4.52,192,1270440000";
d="pdf'?scan'208";a="14854740"
Received: from mail-gx0-f220.google.com ([209.85.217.220]) by mail2.nrc.gov
with ESMTP; 12 Apr 2010 14:49:10 -0400
Received: by gxk20 with SMTP id 20so973348gxk.12 for
<Rulemaking.Comments@nrc.gov>; Mon, 12 Apr 2010 11:49:09 -0700 (PDT)
MIME-Version: 1.0
Received: by 10.150.137.1 with HTTP; Mon, 12 Apr 2010 11:49:09 -0700 (PDT)
Date: Mon, 12 Apr 2010 14:49:09 -0400
Received: by 10.150.80.6 with SMTP id d6mr2434011ybb.26.1271098149291; Mon, 12
Apr 2010 11:49:09 -0700 (PDT)
Message-ID: <r2s7eefda751004121149qb48d77fenea308f1f363a4ade@mail.gmail.com>
Subject: Comments of Beyond Nuclear in support of PRM 50-93
From: Paul Gunter <paul@beyondnuclear.org>
To: Rulemaking.Comments@nrc.gov
Content-Type: multipart/mixed; boundary="000e0cd5c500a9c8b704840e99d0"
Return-Path: paul@beyondnuclear.org

Submission ID 16
John Butler, Nuclear Energy Institute
ML101040678

PRM-50-93
(75FR03876)

NUCLEAR ENERGY INSTITUTE

16

John C. Butler
DIRECTOR
ENGINEERING AND OPERATIONS SUPPORT
NUCLEAR GENERATION DIVISION

DOCKETED
USNRC

April 13, 2010 (4:00pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 12, 2010

Ms. Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attn: Rulemakings and Adjudications Staff

Subject: Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554

Project Number: 689

Dear Ms. Vietti-Cook:

The attachment to this letter provides comments from the Nuclear Energy Institute (NEI)[1] on behalf of the nuclear energy industry on the Petition for Rulemaking (PRM-50-93), in response to the *Federal Register* notice of January 25, 2010. This petition, dated November 17, 2009, requests that the NRC amend its regulations regarding the domestic licensing of production and utilization facilities.

Specifically, the petitioner requests that the NRC amend its regulations based on data from multi-rod (assembly) severe fuel damage experiments and promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a loss-of-coolant accident.

In support of this request, the petitioner cites results from two out of many tests performed over 25 years ago. The first of these tests was performed under non-prototypic conditions well beyond the envelope for current plant designs. Results from the second test were discounted by the original experimenters because of instrumentation problems. Neither one of these tests, whether reviewed in isolation or in combination with the other tests, support the changes to the regulations sought by

[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

Template = SECY-067

DS 10

the petitioner. The petitioner's request that the NRC amend regulations regarding the domestic licensing of production and utilization facilities should be denied.

If you have any questions regarding this matter, please contact Gordon Clefton (gac@nei.org; 202-739-8086) or me.

Sincerely,

John C. Butler

Attachment

NEI Comments on Petition for Rulemaking (PRM-50-93)

Petitioner's Request

The petitioner requests that the Nuclear Regulatory Commission (NRC) revise its regulations based on data from multirod (fuel assembly) severe fuel damage experiments. The petitioner also requests that the NRC promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a loss-of-coolant accident (LOCA).

Specifically, the petitioner states that his interpretation of data from select multirod severe fuel damage experiments indicates that the current regulations at 10CFR Part 50 are non-conservative in their peak cladding temperature limit of 2200 °F (1204°C) and that the Baker-Just and Cathcart-Pawel equations are also non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. The petitioner requests that the NRC revise its regulations at 10CFR50.46(b)(1) and Appendix K to 10CFR Part 50 based on this interpretation. The petitioner also requests that the NRC promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a LOCA.

Background

The petitioner uses data from select multirod tests in an attempt to demonstrate that the peak fuel cladding temperature as stated by 10CFR50.46(b)(1) is not adequate to protect the cladding from reaching the autocatalytic zirconium-water regime. In addition, the petitioner questions the adequacy of the correlations used in 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 that time, the NRC concluded that Appendix K of 10CFR 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.

Zirconium-Water Reaction

16-1

One of the key premises of the petitioner's request for rule change is that the Baker-Just and Cathcart-Pawel correlations are non-conservative for calculating the metal-water reaction rates. It is hypothesized that neither correlation predicts the autocatalytic temperature of the zirconium-water reaction. The effects of the exothermic zirconium-water reaction are considered in the ECCS design

16-2

because of their potential influence on the thermal and mechanical behavior of the system. A review of related literature concludes that the zirconium-water reaction is relatively slow and corrosion-like under most conditions; however, at very high temperatures a self-sustained reaction can occur.

The Baker-Just correlation is specified in Appendix K of 10CFR Part 50 for the calculation of the energy release rate due to oxidation, hydrogen generation, and equivalent cladding reacted (ECR). The 2200°F cladding temperature limit for LOCA was implemented in order to limit oxygen induced embrittlement. The temperature limit in combination with the 17% ECR limit calculated by Baker-Just was specified to ensure the cladding remains ductile following a LOCA. The Baker-Just correlation, using the current range of parameter inputs, has been shown to be conservative and adequate to assess Appendix K ECCS performance. Data published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F. 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 correlation over-predicts the zirconium-water reaction by as much as 30% at the limiting temperature (2200°F).

NRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance" allows the use of a best-estimate correlation to calculate the zirconium-water reaction for temperatures greater than 1900°F and recommends the use of the Cathcart-Pawel correlation (NUREG-17, "Zirconium Metal - Water Oxidation Kinetics: IV Reaction Rate Studies"). The NRC, foreign organizations such as JAEA in Japan and CEA in France, and the United States nuclear Industry are currently conducting and evaluating experimental and analytical programs on fuel cladding behavior under LOCA conditions. These programs include both well-characterized isothermal high temperature oxidation tests and integral rodlet tests conducted at temperatures up to 2200°F that have confirmed the predictive capability of the Cathcart-Pawel correlation.

Multirod Severe Fuel Tests

The petitioner relies heavily on the results of two assembly tests with fuel damage, FLECHT Run 9573 and LOFT LP-FP-2. FLECHT Run 9573 refers to one of four Zircaloy clad FLECHT experiments performed in 1969 and reported in WCAP-7665. Westinghouse responded to similar claims in PRM-50-76 in LTR-NRC-02-52 Rev. 1. The petitioner claims that this test 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. The FLECHT Run 9573 was based on extremely severe conditions. Reflood was initiated when the cladding temperature reached 1970°F. This temperature in combination with a low flooding rate (1.1 in/sec) allowed a temperature excursion leading to failures of the heating elements at about 18 seconds into the transient. At that time, the measured cladding temperature reached 2300°F and the steam temperature was in excess of 2500°F. The high steam temperature was a result of the exothermic reaction between the zirconium and the steam. This reaction occurred at hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe. Metallurgical analyses were performed on specimens extracted from heater rods. The heater rods were exposed to temperatures as high as 2500°F. The measured oxide thickness was within the predicted range calculated using Baker-Just.

From a LOCA perspective, the test conditions of FLECHT Run 9573 were extremely severe and well beyond those conditions which the design of the plants would allow to occur. 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 were used in the test are possible only after significant cooling is established within the core.

The LP-FP-2 experiment was the second fission product release and transport test performed in the Loss of Fluid Test (LOFT) facility at Idaho National Engineering Laboratory (INEL) under the sponsorship of the Organisation for Economic Co-operation and Development (OECD). The objective of the test was to provide information on fuel rod behavior, hydrogen generation, and fission product release and transport during a LOCA scenario that resulted in severe core damage. 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. Additionally, 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 this experiment. The reported temperature at the initiation of rapid oxidation is not an accurate depiction of the cladding temperature without some form of interpretation. The petitioner supplies no analytical evaluation of the data to support the claim that the rate of oxidation became excessive below 2200°F.

Reflood Rates

16-3

The petitioner bases the claim for a fixed minimum reflood rate on FLECHT Run 9573. As was pointed out above, the test conditions of FLECHT Run 9573 were extremely severe and can be considered beyond those that would be allowed by US plant design. It is also important to understand the past and current role of rod bundle reflood heat transfer tests. In the late 1960s, a mechanistic understanding of reflood heat transfer did not exist. To develop heat transfer models as expeditiously as possible, the Atomic Energy Commission (AEC) and Industry cooperatively developed the Pressurized Water Reactor (PWR) FLECHT program which was run by Westinghouse. The principal objective was to determine reflood heat transfer coefficients as a function of key initial and boundary conditions, rod elevation, and time after the beginning of reflood. Additionally, the program developed empirical correlations based on that dependency.

There is no reason to establish a new minimum allowable core reflood rate in the LOCA evaluation models as the petitioner is proposing. In the 10CFR50.46 Appendix K Section I.D.5.b, a restriction to use steam cooling for the convective portion of the reflood heat transfer at flooding rates less than one inch per second is already included. In best-estimate models it is indicated (RG 1.157 Section 3.12.4) that "heat transfer calculations that account for two phase conditions in the core during refilling of the reactor vessel should be justified through comparison with experimental data. Best-estimate models will be considered acceptable provided their technical basis is demonstrated through comparison with appropriate data and analyses". Regulatory Guide 1.157 includes the FLECHT experiments as appropriate data for comparison; therefore, the results from the FLECHT experiments have already influenced best-estimate LOCA evaluation models and their allowable core reflood rates.

Conclusions

The petitioner expressed concerns leading to his request that the NRC revise its regulations at 10CFR 50.46(b)(1) and Appendix K to 10CFR Part 50. The petitioner also requested that the NRC promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a LOCA. The petitioner relies heavily on the results of two assembly tests with fuel damage, FLECHT Run 9573 and LOFT LP-FP-2.

16-4

A significant amount of LOCA testing has been conducted, since the completion of these early test programs. Experimental programs have been conducted by numerous organizations on both isothermal oxidation conditions and integral test conditions. The results from these programs to date confirm that Baker-Just is a conservative correlation for the prediction of metal-water reactions and that Cathcart-Pawel provides a best-estimate prediction of oxidation kinetics. These later tests conducted at 2200°F have shown no evidence of rapid oxidation Thus, the petitioner's claim that the autocatalytic runaway regime begins below 2200°F and that the current 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). In regard to defining a minimum reflood rate, the conditions of FLECHT Run 9573 were extremely severe and from a LOCA stand point should be considered beyond those possible with current ECCS designs.

Based on these considerations, the lack of scientific evaluation results to the contrary of the referenced experiments, and the counter indications associated with analysis, testing, and evaluation conducted over the last thirty years, it is concluded that the proposed revisions to 10CFR50.46(b)(1) and Appendix K to 10CFR Part 50 are unwarranted.

Rulemaking Comments

From:	REED, Joseph [jsr@nei.org] on behalf of BUTLER, John [jcb@nei.org]
Sent:	Tuesday, April 13, 2010 10:48 AM
Subject:	Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554
Attachments:	04-12-10_NRC_Industry Comments on PRM-50-93.pdf; 04-12-10_NRC_Industry Comments on PRM-50-93_Attachment.pdf

April 12, 2010

Ms. Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attn: Rulemakings and Adjudications Staff

Subject: Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC–2009–0554

Project Number: 689

Dear Ms. Vietti-Cook:

The attachment to this letter provides comments from the Nuclear Energy Institute (NEI) on behalf of the nuclear energy industry on the Petition for Rulemaking (PRM-50-93), in response to the *Federal Register* notice of January 25, 2010. This petition, dated November 17, 2009, requests that the NRC amend its regulations regarding the domestic licensing of production and utilization facilities.

Specifically, the petitioner requests that the NRC amend its regulations based on data from multi-rod (assembly) severe fuel damage experiments and promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a loss-of-coolant accident.

In support of this request, the petitioner cites results from two out of many tests performed over 25 years ago. The first of these tests was performed under non-prototypic conditions well beyond the envelope for current plant designs. Results from the second test were discounted by the original experimenters because of instrumentation problems. Neither one of these tests, whether reviewed in isolation or in combination with the other tests, support the changes to the regulations sought by the petitioner. The petitioner's request that the NRC amend regulations regarding the domestic licensing of production and utilization facilities should be denied.

If you have any questions regarding this matter, please contact Gordon Clefton (gac@nei.org; 202-739-8086) or me.

Sincerely,

John C. Butler
Director, Engineering & Operations Support

Nuclear Energy Institute
1776 I Street NW, Suite 400
Washington, DC 20006
www.nei.org

P: 202-739-8108
F: 202-533-0113
E: jcb@nei.org

nuclear. clean air energy.

Received: from mail1.nrc.gov (148.184.176.41) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Tue, 13 Apr 2010
10:50:52 -0400
X-Ironport-ID: mail1
X-SBRS: 4.4
X-MID: 13138299
X-fn: 04-12-10_NRC_Industry Comments on PRM-50-93.pdf,
04-12-10_NRC_Industry Comments on PRM-50-93_Attachment.pdf
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AqoBAHYhxEvYILQPkGdsb2JhbACBP5k/YQEBAQEJCQwHEQQepBiZE4J0ghkE
X-IronPort-AV: E=Sophos;i="4.52,197,1270440000";
d="pdf'?scan'208,217";a="13138299"
Received: from va3ehsobe005.messaging.microsoft.com (HELO
VA3EHSOBE006.bigfish.com) ([216.32.180.15]) by mail1.nrc.gov with ESMTP; 13
Apr 2010 10:50:51 -0400
Received: from mail11-va3-R.bigfish.com (10.7.14.236) by
VA3EHSOBE006.bigfish.com (10.7.40.26) with Microsoft SMTP Server id
8.1.240.5; Tue, 13 Apr 2010 14:50:51 +0000
Received: from mail11-va3 (localhost.localdomain [127.0.0.1]) by
mail11-va3-R.bigfish.com (Postfix) with ESMTP id 197FC12288CE; Tue, 13 Apr
2010 14:50:51 +0000 (UTC)
X-SpamScore: -7
X-BigFish:
VPS-7(z4614mza0dJ18c1J179cMzz1202hzz186Mz31j467h27ah2a8h6bh34h43h61h)
X-Spam-TCS-SCL: 0:0
X-FB-SS: 5,5,5,5,
Received: from mail11-va3 (localhost.localdomain [127.0.0.1]) by mail11-va3
(MessageSwitch) id 1271170248581598_5500; Tue, 13 Apr 2010 14:50:48 +0000
(UTC)
Received: from VA3EHSMHS026.bigfish.com (unknown [10.7.14.250]) by
mail11-va3.bigfish.com (Postfix) with ESMTP id 89E27F90052; Tue, 13 Apr 2010
14:50:48 +0000 (UTC)
Received: from NEIEMAIL01.nei.org (208.116.169.4) by VA3EHSMHS026.bigfish.com
(10.7.99.36) with Microsoft SMTP Server (TLS) id 14.0.482.44; Tue, 13 Apr
2010 14:50:44 +0000
Received: from NEIEMAIL01.nei.org ([10.2.100.2]) by NEIEMAIL01 ([10.2.100.2])
with mapi; Tue, 13 Apr 2010 10:48:19 -0400
From: "BUTLER, John" <jcb@nei.org>
Sender: "REED, Joseph" <jsr@nei.org>
Date: Tue, 13 Apr 2010 10:48:18 -0400
Subject: Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod
(Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554
Thread-Topic: Industry Comments on Petition for Rulemaking (PRM-50-93);
Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554
Thread-Index: Acrae+qDiqrJQVwcSfiybOdSXMN4FwAlyRfQAAE6jOA=
Message-ID: <1E82BFD961F25B49B78B156D195B54EB01FBE2E162@NEIEMAIL01>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes

X-MS-TNEF-Correlator:
acceptlanguage: en-US
Content-Type: multipart/mixed;

boundary="_005_1E82BFD961F25B49B78B156D195B54EB01FBE2E162NEIEMAIL01_"
MIME-Version: 1.0
To: Undisclosed recipients:;
X-Bypass-Agent: EF-1;
X-Reverse-DNS: unknown
Return-Path: jsr@nei.org

Submission ID 17
Robert Leyse
ML101040679

PRM-50-93
(75FR03876)

17

DOCKETED
USNRC

April 13, 2010 (4:00pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

PRM-50-93 is a Wake-up Call

There are two parts to this comment.

1. NRC has applied Baker-Just in recent actions.

2. Commissioners must wake-up.

1. NRC applied Baker-Just in recent actions.

In its posted denial of PRM-50-76, the NRC states, "The remaining data from Bostrum (``The High Temperature Oxidation of Zircaloy in Water," W. A. Bostrum, WAPD-104 March 1954) and Lemmon (``Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," A. W. Lemmon, Jr., BMI-1154, January 1957), at more relevant zirconium cladding conditions, were used by Baker and Just in the derivation of their equation." However, it is unlikely that the authors of NRC's technical safety analysis, ML041210109, ever looked at either WAPD-104 or BMI-1154. It is more likely that those authors merely lifted the description of those references from the Baker-Just report, ML050550198. Thus the authors of ML041210109 were not aware that Bostrom and Lemmon each used induction heating in their investigations. Furthermore, those authors also were likely unaware of FZKA 5846, the Hofmann and Noack report. Recently NRC has placed reports in ADAMS: WAPD-104 is ML100900446 and BMI-1154 is ML100570218.

17-1

NRC reviewers have not been aware that single rod tests with induction heating do not yield conservative values for the temperature at which runaway oxidation proceeds. Thus several nuclear power plant licensees have been allowed to install lead assemblies with alloys such as M-5. One typical example is:

1.	(80)	San Onofre, Units 2 and 3 - Temporary Exemption from the Requirement of 10 CFR Part 50, Section 50.46 and Appendix K for Lead Fuel Assemblies.	ML090860429	2009-12-17	7

In the above we read:

Metal-water reaction tests performed on M5 alloy material by AREVA NP (as discussed in topical report BAW-10227P-A) demonstrate conservative reaction rates relative to the Baker-Just equation.

Here is the reference to the safety evaluation of topical report BAW-10227P-A:

5.	(80)	REVISED SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR TOPICAL REPORT BAW-10227P, "EVALUATION OF ADVANCED CLADDING AND STRUCTURAL MATERIAL (M5) IN PWR REACTOR FUEL."	ML003681490	2000-02-04	34

The safety evaluation tells us:

Template = SECY-067

DS 10

Evaluation -FCF uses approved LOCA evaluation models along with the Baker-Just correlation, as required by 10 CFR Part 50 Appendix K, for demonstrating compliance with the 2200 OF PCT and 17 percent oxidation criteria for the fuel cladding during a LOCA. FCF has performed high-temperature oxidation tests for M5 cladding (Appendix D of Reference 1) to confirm that the Baker-Just oxidation correlation remains conservative in relation to M5 high-temperature oxidation. ***The FCF high temperature oxidation tests were performed in super heated flowing steam where the sample (both M5 and Zr-4) was inductively heated to temperatures of 1050, 1150, and 1250°C for various times.*** *The measured oxidation rates for the M5 samples were significantly lower than those for the Zr-4 samples at 1050°C; however, at 1150 and 1250°C the oxidation rates were nearly identical. A comparison of M5 measured values to Baker-Just predictions demonstrated that the Baker-Just correlation remained conservative for temperatures typically calculated for LOCA. The staff asked FCF (Reference 4) to provide Arrehenius plots of the high-temperature oxidation data in order to provide a measure of bias and uncertainty in the data. FCF provided these plots (Reference 6) which demonstrated only small uncertainties and essentially no biases in the data.* ***The FCF data demonstrates that high-temperature oxidation of the M5 alloy is bounded by the Baker-Just correlation and that the Appendix K requirement for the use of Baker-Just remains conservative in relation to the use of M5.***

Oxidation tests where single rod or tubing specimens are inductively heated do not yield conservative data for the temperature at which runaway is initiated. Clearly, the NRC staff's review and approval of topical report BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," dated February 4, 2000 (ADAMS Accession Nos. ML003681479 and ML003681490), has not been based on sound science.

2. Commissioners must wake-up.

17-2

The Commissioners should not blindly follow the staff recommendations and findings as they did in the case of PRM-50-76. The Commissioners must have an awareness of what is going on as the review proceeds instead of casually reviewing a completed set of recommendations. Furthermore, the Commissioners should not tolerate undue delays in the review.

For example, the Commissioners did not do their homework in their unanimous approval of the staff recommendation to deny PRM-50-76. They had no awareness of the staff activities as the staff review proceeded from the date of docketing, May 8, 2002, until they received the following on June 29, 2005; a duration of almost 38 months.

RULEMAKING ISSUE
NOTATION VOTE

SECY-05-0113

June 29, 2005

FOR: The Commissioners

FROM: Luis A. Reyes
Executive Director for Operations

SUBJECT: DENIAL OF A PETITION FOR RULEMAKING TO REVISE APPENDIX K TO 10 CFR PART 50 AND ASSOCIATED GUIDANCE DOCUMENTS (PRM-50-76)

The proposed federal register document was Attachment 1 to the above.

Authorized Fed Register Attachment to above Attachment 1

[7590-01-P]
NUCLEAR REGULATORY COMMISSION
10 CFR Part 50
[Docket No. PRM-50-76]
Robert H. Leyse; Denial of Petition for Rulemaking
AGENCY: Nuclear Regulatory Commission
ACTION: Petition for Rulemaking; Denial

This proposed federal register document included a list of references including the two keystone references, WAPD-104 and BMI-1154 that are cited in Baker –Just;

"The High Temperature Oxidation of Zircaloy in Water," W. A. Bostrum, WAPD-104 (March 1954)

"Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," A. W. Lemmon, Jr., BMI-1154, (January 1957)

The document that appeared in the Federal Register on September 6, 2005, **ACTION:** Petition for rulemaking; denial, had a list of references; however the vital reports, WAPD-104 and BMI-1154 were not listed.

On August 5, 2005, the voting record was released:

UNITED STATES NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
August 5, 2005
SECRETARY
COMMISSION VOTING RECORD
DECISION ITEM: SECY-05-0113
TITLE: DENIAL OF A PETITION FOR RULEMAKING TO REVISE APPENDIX K TO 10 CFR PART 50 AND ASSOCIATED GUIDANCE DOCUMENTS (PRM-50-76)

The voting record includes a signed note from each of the four Commissioners. In each case the opening sentence is: "I approve …"

Commissioner Lyons wrote:

I approve the staffs recommendation to deny the petition for rulemaking and concur with the comments of Commissioner Merrifield. The staff needs to address the following comments in both the Federal Register Notice and the denial letter to the petitioner:

1. The following sentence contained on page 2, lines 4 and 5 of the letter to the petitioner and page 21, lines 8 and 9 of the Federal Register Notice needs to be modified to clarify how these experiments relate to the denial of the petition. 'The NRC funded more than 50 Zircaloy clad bundle reflood experiments at the National Research Universal (NRU) reactor."

2. The following sentence contained on page 2, lines 5 through 10 of the letter to the petitioner and page 21, lines 10 through 13 of the Federal Register Notice needs to be modified to clarify how these programs relate to the denial of the petition. 'The NRC is currently conducting and evaluating experimental and analytical programs on fuel cladding behavior ..... to evaluate the adequacy of current 50.46 oxidation-related criteria and models."

3. The following paragraph on page 2 of the letter to the petitioner and page 22 of the Federal Register Notice needs to be modified to clarify how this information relates to the denial of the petition.

"The NRC applied the Cathcart-Pawel oxygen uptake and ZRO2 thickness equations to the four FLECT Zircaloy experiments .... [start new paragraph] The NRC applied the Cathcart-Pawel oxide thickness equation to 15 of their transient temperature experiments ..... This result is consistent with the application of the Cathcart-Pawel equations, which are intended for use in best-estimate LOCA calculations in accordance with RG 1.157."

However, the denial was effected and published in the Federal Register on September 6, 2005, and the requirements listed by Commissioner Lyons were never met. Those requirements are met among the well documented teachings of PRM-50-93.

Comment submitted by

Robert H. Leyse*
Chemical Engineer and Nuclear Engineer
P. O. Box 2850
Sun Valley, ID 83353

***Experience:**

Career to date: Commenter's ongoing career spans several decades: General Electric at Hanford Works (1950), Argonne, DuPont Savannah River Plant, General Electric Vallecitos, Westinghouse Pittsburgh, Scandpower Norway, Consulting with Westinghouse at TMI-2, EPRI Nuclear Safety Analysis Center, EPRI Exploratory Research, and now self employed (2010).

Selected Experience pertinent to this comment on PRM-50-93:

PWR FLECHT: Test design, discoveries and reporting as referenced in PRM-50-93.

Presentation at 2003 RELAP5 International Users Seminar, West Yellowstone, Montana
Unmet Challenges for SCDAP/RELAP5-3D. Analysis of Severe Accidents for Light Water Nuclear Reactors with Heavily Fouled Cores. Robert H. Leyse,
www.inl.gov/relap5/rius/yellowstone/**leyse**.pdf

Comment NEI PETITION FOR RULEMAKING: PRM-50-78 (Cladding Materials) September 9, 2002
The petition should be denied because the evaluations of cladding materials do not account for the realities of plant operation under so-called normal conditions as well as the LOCA environment.

PETITION FOR RULEMAKING: PRM-50-76 May 8, 2002
Petitioner is aware of deficiencies in Appendix K. 1. A. 5. The Baker-Just equation does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures. Likewise, petitioner is aware of deficiencies in Regulatory Guide 1.157, BESTESTIMATE CALCULATIONS OF ECCS PERFORMANCE, Paragraph 3.2.5.1. The report NUREG-17 does not include any consideration of the complex thermal hydraulic conditions during LOCA including the potential for very high fluid temperatures.

PETITION FOR RULEMAKING: PRM-50-73 September 04, 2001
The specific issue is that 50.46 and Appendix K do not address the impact of crud on coolability during a fast moving (large break) LOCA.

PETITION FOR RULEMAKING: PRM-50-78 September 9, 2002
Regulations are needed to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants.

Current field is microscale heat transfer to pressurized water at ultra-high heat fluxes.

Microscale Heat Transfer to Subcooled Water
LEYSE: MICROSCALE HEAT TRANSFER
doi.wiley.com/10.1111/j.1749-6632.2002.tb05912.x
Or go to: *http://www3.interscience.wiley.com/journal/118947467/abstract*

MICROSCALE PHASE CHANGE HEAT TRANSFER AT HIGH HEAT FLUX. Robert H. Leyse. Inz, Inc.
Phani K. Meduri, Gopinath R. Warrier and Vijay K. Dhir ...
boiling.seas.ucla.edu/Publications/Conf_LMWD2003

Rulemaking Comments

From: NRCREP Resource
Sent: Tuesday, April 13, 2010 12:32 PM
To: Rulemaking Comments
Subject: Comment on PRM-50-93
Attachments: NRC-2009-0554-DRAFT-0014.1[1].doc

Van,

Attached for docketing is a comment from Robert H. Leyse on PRM-50-93 that I received via the regulations.gov website on April 12, 2010.

Thanks,
Carol

Received: from HQCLSTR02.nrc.gov ([148.184.44.80]) by TWMS01.nrc.gov
([148.184.200.145]) with mapi; Tue, 13 Apr 2010 12:32:05 -0400
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: NRCREP Resource <NRCREP.Resource@nrc.gov>
To: Rulemaking Comments <Rulemaking.Comments@nrc.gov>
Date: Tue, 13 Apr 2010 12:32:06 -0400
Subject: Comment on PRM-50-93
Thread-Topic: Comment on PRM-50-93
Thread-Index: AcrbJlfU7OSoqruLTyGwihUBjdTDpg==
Message-ID:
<4792243393EAA04CBAD2CFFB94765C0D0F9D3AC0AE@HQCLSTR02.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<4792243393EAA04CBAD2CFFB94765C0D0F9D3AC0AE@HQCLSTR02.nrc.gov>
MIME-Version: 1.0

Submission ID 18
David Helker, Exelon
ML101130353

Exelon Nuclear
200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

Exelon. Nuclear

PRM-50-93
(75FR3876)

18

DOCKETED
USNRC

April 22, 2010 (4:30pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 12, 2010

Secretary
U.S. Nuclear Regulatory Commission
Attn: Rulemakings and Adjudications Staff
Washington, DC 20555–0001

Subject: Response to Request for Comments Concerning 10 CFR 50 Petition for Rulemaking Filed by Mark Edward Leyse (Federal Register Notice 75 FR 3876, dated January 25, 2010)

Exelon Generation Company, LLC (Exelon) is submitting this letter in response to the U.S. Nuclear Regulatory Commission's (NRC's) request for comments concerning a 10 CFR 50 Petition for Rulemaking filed by Mark Edward Leyse, which was published in the Federal Register (i.e., 75FR3876, dated January 25, 2010).

The petitioner requests that the NRC revise its regulations based on data from multi-rod (assembly) severe fuel damage experiments. The petitioner also requests that the NRC promulgate a regulation that will stipulate minimum allowable core reflood rates in the event of a Loss-of-Coolant Accident (LOCA).

18-1

Exelon appreciates the opportunity to comment on this Petition for Rulemaking. Exelon does not consider the proposed revisions to 10 CFR 50 necessary and we fully support the comments submitted by the Nuclear Energy Institute on behalf of the industry related to this Petition for Rulemaking.

If you have any questions or require additional information, please do not hesitate to contact me at 610-765-5525.

Respectfully,

D. P. Helker

David P. Helker
Manager - Licensing

Template-SECY-067

DS 10

Submission ID 19
David Lochbaum, Union of Concerned Scientists
ML101180175

PRM-50-93
(75FR03876)

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

DOCKETED
USNRC

April 27, 2010 (12:05pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 27, 2010

19

Comments submitted by the Union of Concerned Scientists on the petition for rulmaking submitted by Mark Edward Leyse (Docket No. PRM-50-93; NRC-200909554)

In response to the notice published January 25, 2010, in the *Federal Register* (Vol. 75, No 5, pp. 3876-3877), I submit the following comments on behalf of the Union of Concerned Scientists.

It is readily apparent from the materials submitted by Mr. Leyse with his petition for rulemaking that considerable effort went into its research and preparation. UCS recognizes and appreciates the unselfish commitment to public health and safety this petition represents.

In our opinion, Mr. Leyse's petition addresses a genuine safety problem. We believe the NRC should embark on a rulemaking process based on this petition. We are confident that this process would culminate in revised regulations – perhaps not precisely the ones proposed by Mr. Leyse but ones that would adequately resolve the issues he has meticulously identified – that would better ensure safety in event of a loss of coolant accident.

19-1

To date, there has only been one loss of coolant accident of significant consequence at a nuclear power reactor licensed by the NRC – the March 1979 accident at Three Mile Island Unit 2. What deeply concerns UCS about safety analyses for postulated loss of coolant accidents is the inability to accurately explain what happened when during this past loss of coolant accident. Many talented, capable researchers have attempted to explain what happened when the TMI-2 overheated. Their results vary widely, even when examining the same aspects of this past event. For example, J. O. Henrie and A. K. Postma from Rockwell Hanford Operations authored "*Lessons Learned from Hydrogen Generation and Burning During the TMI-2 Event,*" (GEND-061, May 1987) reported results from their own and other researchers efforts to specify how much hydrogen was generated when during the accident. Figures 4-7 and 4-8 from their report are presented below. The results may all be in the same ballpark, but it is very clearly and undeniably a very large ballpark.

www.ucsusa.org
Two Brattle Square · Cambridge, MA 02238-9105 · TEL: 617.547.5552 · FAX: 617.864.9405
1825 K Street NW · Suite 800 · Washington, DC 20006-1232 · TEL: 202.223.6133 · FAX: 202.223.6162
2397 Shattuck Avenue · Suite 203 · Berkeley, CA 94704-1567 · TEL: 510.843.1872 · FAX: 510.843.3785
One North LaSalle Street · Suite 1904 · Chicago, IL 60602-4064 · TEL: 312.578.1750 · FAX: 312.578.1751

Template = SECY-067

DS 10

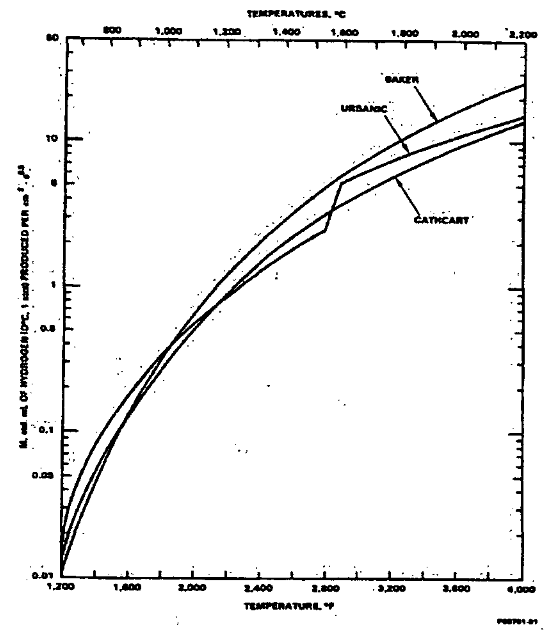


Figure 4-7. Comparison of Zirconium-Water Reaction Correlations.

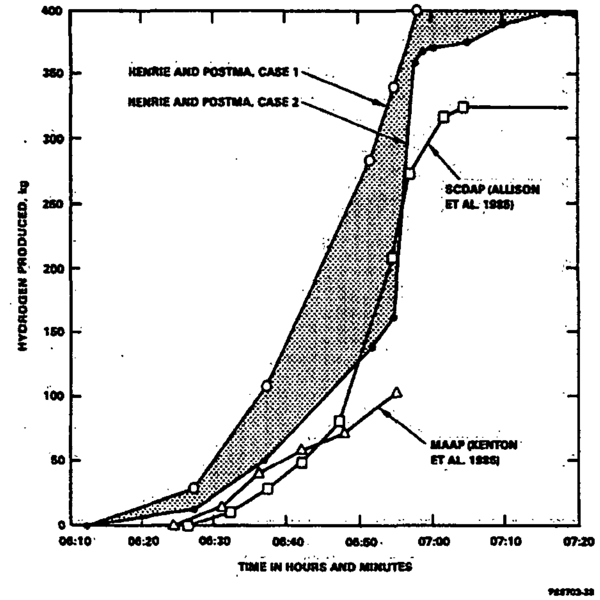


Figure 4-8. Calculated TMI-2 Hydrogen Production from the Zirconium-Water Reaction.

The inability to explain a historical event with precision raises reasonable doubts about the ability to forecast future events with precision – in essence, the heart of Mr. Leyse's petition.

The urgency behind the need to resolve the issues raised in Mr. Leyse's petition is best demonstrated by narrowing margins to the magical 2,200°F peak cladding temperature.

Plant Name	Predicted Peak Clad Temperature	Source
Braidwood	Unit 1 – 2,161°F Unit 2 – 2,168°F	ML090990492 April 9, 2009
Byron	Unit 1 – 2,161°F Unit 2 – 2,168°F	ML090990492 April 9, 2009
Catawba	Unit 1 – 2,145°F Unit 2 – 2,145°F	ML092180407 August 3, 2009
Cook	Unit 1 – 2,128°F Unit 2 – 2,139°F	ML092520238 August 28, 2009
McGuire	Unit 1 – 2,145°F Unit 2 – 2,145°F	ML092180407 August 3, 2009
North Anna	Unit 1 – 2,131°F Unit 2 – 2,131°F	ML091820272 June 30, 2009
Palo Verde	Unit 1 – 2,152°F Unit 2 – 2,148°F Unit 3 – 2,148°F	ML091810703 December 22, 2009

In his petition, Mr. Leyse raised valid questions about the models used to show "margin" during postulated loss of coolant accidents. Prudent protection of public health and safety warrants that his questions be answered. The NRC should answer these vital questions by pursuing the rulemaking that Mr. Leyse has proposed.

Questions were raised about the ability of o-rings to function at low temperatures before the space shuttle Challenger was launched in January 1986. About seventy seconds later, those questions were answered in about the hardest way possible.

Questions were raised about the impact of foam on the surfaces of the space shuttle Columbia during its launch. A computer model developed during the Apollo program was used and it indicated that the foam would fail the integrity of the tiles. Those undesired answers were set aside, attributed to an "out-dated" model. The Columbia's subsequent landing in Texas and Louisiana tragically re-demonstrated the need to get the right answers to all the right questions.

Questions were raised about the safety of cracked nozzles through the reactor vessel head at Davis-Besse. An order was drafted by the NRC requiring that Davis-Besse be shut down to obtain the answers to these safety questions. That order was not issued and Davis-Besse was allowed to continue to operate – the closest near-miss since the TMI-2 accident according to the NRC's accident sequence precursor program.

It was wrong that these past safety questions were not properly answered. It would be equally wrong now not to properly answer the safety questions posed by Mr. Leyse in his petition.

Sincerely,

David A. Lochbaum

David Lochbaum
Director, Nuclear Safety Project
PO Box 15316
Chattanooga, TN 37415
(423) 468-9272, office

Rulemaking Comments

From: Dave Lochbaum [dlochbaum@ucsusa.org]
Sent: Tuesday, April 27, 2010 7:58 AM
To: Rulemaking Comments
Subject: Docket ID NRC-2009-0554
Attachments: 20100427-ucs-nrc-comments-leyse-petition.pdf

Good Day:

Attached are belated comments on the subject petition for rulemaking.

Thanks,

David Lochbaum
Director, Nuclear Safety Project
Union of Concerned Scientists
PO Box 15316
Chattanooga, TN 37415
(423) 468-9272 office
(423) 488-8318 cell
dlochbaum@ucsusa.org

Received: from mail1.nrc.gov (148.184.176.41) by OWMS01.nrc.gov
 (148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Tue, 27 Apr 2010
 07:59:21 -0400
X-Ironport-ID: mail1
X-SBRS: 5.3
X-MID: 13555072
X-fn: 20100427-ucs-nrc-comments-leyse-petition.pdf
X-IronPort-AV: E=Sophos;i="4.52,280,1270440000";
  d="pdf'?scan'208";a="13555072"
Received: from mail.ucsusa.org ([208.50.113.51]) by mail1.nrc.gov with ESMTP;
 27 Apr 2010 07:59:18 -0400
Received: from UCSUSA-MTA by mail.ucsusa.org with Novell_GroupWise; Tue, 27
 Apr 2010 07:58:58 -0400
Message-ID: <4BD69925020000A100012B01@mail.ucsusa.org>
X-Mailer: Novell GroupWise Internet Agent 7.0.1
Date: Tue, 27 Apr 2010 07:58:28 -0400
From: Dave Lochbaum <dlochbaum@ucsusa.org>
To: <Rulemaking.Comments@nrc.gov>
Subject: Docket ID NRC-2009-0554
MIME-Version: 1.0
Content-Type: multipart/mixed; boundary="=__Part507AC074.1__="
Return-Path: dlochbaum@ucsusa.org

Submission ID 20
Mark Leyse
ML101230118

PRM-50-93
(75FR03876)

DOCKETED
USNRC

20

April 30, 2010 (2:22pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
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,

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

Template = SECY-067

DS10

April 28, 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; NRC-2009-0554

TABLE OF CONTENTS

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

20-1

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

[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.

20-2

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).

20-3

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.)

20-4

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.)

20-5

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.

20-6

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.

---

[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.)

20-7

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]

---

[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.

---

[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.

---

[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]

---

[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].

---

[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

---

[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.

---

[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.

20-8

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]

---

[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.

---

[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.)

---

[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

[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*

[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]

---

[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.)

20-9 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.)

20-10 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

20-11

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

20-12

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]

---

[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

[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."

20-13

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.

---

[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).

[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.

20-14

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.

---

[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.

20-15

20-16

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]

---

[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.

20-17

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.

---

[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.

20-18 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

---

[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

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Thu, 29 Apr 2010
16:01:08 -0400
X-Ironport-ID: mail2
X-SBRS: 4.4
X-MID: 15618433
X-fn: Response to NEI Comments on PRM-50-93.pdf
X-IronPort-AV: E=Sophos;i="4.52,297,1270440000";
d="pdf'?scan'208";a="15618433"
Received: from mail-pz0-f195.google.com ([209.85.222.195]) by mail2.nrc.gov
with ESMTP; 29 Apr 2010 16:01:05 -0400
Received: by pzk33 with SMTP id 33so11060647pzk.17 for <multiple
recipients>; Thu, 29 Apr 2010 13:01:02 -0700 (PDT)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:received:date:message-id
:subject:from:to:content-type;
bh=H/rsKubkl6XVwPHX8Dq1m6h703HNNRW4hU8jUIl9HUU=;
b=kCh4kEODKnfL7Zo9XBAuihd9m83KuqzZ/Qg+mo8g1SK/4lzDO4A+A0PipZlqNZILBD
D0eMlgxwa391x4GBa2Hg7okoe0XXdJH2rzWaltQETAx0GlSheleW4s7snz0Bb2zxksw5
2wiB//xZlcNs79Age/Jr2kc1nUPIJ17LzEDlw=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:content-type;
b=w8cFVYifsnZospzWf829lVrXWMEZr6lVrMwq/7Dh6TE7ARl6kBjKRxdhzbyRkhJOd9
PAPXCH1z0BjkpuEvpHmEuWMsvB9ZrtZ8deXxYD5pcjDkmjs7Fq7hsZijQs/tiJorRtVy
ZhZfWFMzJN2yYVUtnkGXNJvuPlcEDN/qQ4GMw=
MIME-Version: 1.0
Received: by 10.142.249.3 with SMTP id w3mr5985875wfh.68.1272571262044; Thu,
29 Apr 2010 13:01:02 -0700 (PDT)
Received: by 10.142.178.1 with HTTP; Thu, 29 Apr 2010 13:01:01 -0700 (PDT)
Date: Thu, 29 Apr 2010 16:01:01 -0400
Message-ID: <x2jedacd5761004291301u9eeb9127o2e247e863638b44b@mail.gmail.com>
Subject: NRC-2009-0554
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>
Content-Type: multipart/mixed; boundary="00504502ce880692ae048565961f"
Return-Path: markleyse@gmail.com

Submission ID 21
Mark Leyse
ML103340249

PRM-50-95
(75FR66007)

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Tuesday, November 23, 2010 11:18 PM
To: Rulemaking Comments; PDR Resource
Cc: Dave Lochbaum; necnp@necnp.org; Raymond Shadis; Powers, Dana A
Subject: NRC-2009-0554 (First)
Attachments: Comments November 2010.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is my first response, dated November 23, 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.

Sincerely,

Mark Leyse

DOCKETED
USNRC

November 24, 2010 (9:15am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Template = SECY-067

DS 10

November 23, 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

TABLE OF CONTENTS

November 23, 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

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

[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 an enforcement action 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 ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a 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.

21-1

II. Supplementary Information to PRM-50-93 and PRM-50-95

> Daniel Ford: I am concerned with one of the many gaps in the Interim Policy Statement and the computer code. I am concerned with a variety of chemical-metal-water reactions that are not considered at all in these codes, metal-water reactions which various recent experimental data indicate can prove [to] very significantly [impact] local temperature

---

[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 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.

> during an accident, and [cause] extensive cladding damage. The specific metal-water reaction I am concerned with at the moment is the reaction between the Zircaloy-Inconel eutectic and steam, I am concerned to find out how the Applicant's analysis contained in the computer code, which does not consider this, how it would be different if it did.
>
> Leonard M. Trosten: I thank you for the explanation. I recognize this as being one of the principal points of concern in the critique by the Union of Concerned Scientists...[7]—IP-2 licensing hearing, November 1971

Experimental data discussed in PRM-50-93 (partly), PRM-50-95, and in these comments on PRM-50-93 and PRM-50-95—among other things—indicates that "low temperature" eutectic reactions could affect the progression of damage during a LOCA. For example, Inconel grid spacers would effect the progression of damage in a reactor core during a LOCA if their temperatures were to reach approximately 2012°F;[8] and experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.[9]

A. Some Parts of PRM-50-95 can be Interpreted as a Commentary on the Safety Issues Raised in PRM-50-93

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.

Some parts of PRM-50-95 can be interpreted as a commentary on the safety issues raised in PRM-50-93.

---

[7] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2520-2522.

[8] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 202.

[9] 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, p. 427.

In particular, when PRM-50-95 states that experimental data indicates that VYNPS's LBPCT of 1960°F[10] does not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA, and that such data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety,"[11] it means that the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit needs to be decreased to a temperature lower than 1832°F.

So NRC needs to determine how far below 1832°F the 10 C.F.R. § 50.46(b)(1) PCT limit needs to be decreased in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

(In PRM-50-93, Petitioner requests that NRC revise 10 C.F.R. § 50.46(b)(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.)

B. Clarifications of Some of Petitioner's Statements in PRM-50-93 and Comments on PRM-50-93

1. Clarifications of Statements on Reflood Rates

Petitioner wants to clarify that Petitioner was referring to experimental data from Thermal-Hydraulic Experiment 1 ("TH-1"), conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, when Petitioner stated in PRM-50-93 that:

21-2

> 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, 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.

---

[10] Entergy, "VYNPS 10 C.F.R. § 50.46(a)(3)(ii) Annual Report for 2009," January 14, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100260386, p. 2.

[11] Mark Edward Leyse, PRM-50-95, June 7, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101610121, pp. 5-6.

Additionally, in TH-1, a total of 28 tests were conducted to simulate large break ("LB") LOCAs, so Petitioner was referring to LB LOCAs, when Petitioner stated "[i]t 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, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F."

As discussed in PRM-50-93, 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).

2. Clarifications of Statements on the Baker-Just and Cathcart-Pawel Correlations

In PRM-50-93, Petitioner's phrasing was imprecise when Petitioner stated that "[d]ata from multi-rod...severe fuel damage experiments...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." Petitioner should have more clearly stated that "data from multi-rod...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."

21-3

Also, in PRM-50-93, Petitioner's phrasing was imprecise when Petitioner stated that "[t]his...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." Petitioner should have more clearly stated that "[t]his...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."

Additionally, in other sentences in PRM-50-93, Petitioner stated imprecise phrases similar to "the Baker-Just and Cathcart-Pawel equations are both non-conservative *for calculating*." Petitioner should have more clearly stated that "the Baker-Just and Cathcart-Pawel correlations are both non-conservative *for use in analyses that would predict*."

Petitioner will now provide a brief explanation of Petitioner's claim that:

> 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 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.

First, it was reported in the early 1990s that for the CORA-16 experiment (a multi-rod severe fuel damage experiment), "[c]ladding oxidation was not accurately predicted by available correlations"[12] 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."[13] 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.

Second, it is significant that in AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above

[12] 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.

[13] 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.

> 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[14]

Clearly, the Zircaloy-steam reaction is not negligible below 1900°F, as experimental data from multi-rod experiments demonstrates. For example, a Karlsruhe paper states:

> As already observed in previous tests, the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above [1832°F]. 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.[15]

It is clear that ECCS evaluation calculations using the Baker-Just correlation under-predict the Zircaloy-steam reaction that would occur in a LOCA environment; this also applies to ECCS evaluation calculations using the Cathcart-Pawel correlation.

3. Additional Clarifications of Statements on the Baker-Just and Cathcart-Pawel Correlations

In PRM-50-93, in a number of places, petitioner stated that "the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F."[16]

First (as stated above for a similar phrase), Petitioner's phrasing was imprecise when Petitioner stated in PRM-50-93 that that "the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F." Petitioner

---

[14] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

[15] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[16] 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."

should have more clearly stated that "*analyses that use* the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and *analyses that use* the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA."

Second, in PRM-50-93, when Petitioner stated that (now rephrased) "analyses that use the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and analyses that use the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA," Petitioner qualified such statements with footnotes that stated:

> 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.[17]

So, it is important to clarify that Petitioner's statement that "analyses that use the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and analyses that use the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA," was qualified in PRM-50-93, as specifically referring to the 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.

21-4

C. A Postulation that Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at 1000°C in the Three Mile Island Unit 2 Accident

First, Petitioner does not intend to present Dr. Robert E. Henry's postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C (1832°F) in the Three Mile Island Unit 2 ("TMI-2") accident as evidence that an autocatalytic reaction did in fact commence at 1000°C in the TMI-2 accident: there is no thermocouple data from the hot spots of the fuel assemblies to confirm if Dr. Henry is correct.

---

[17] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, pp. 3-4; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.

(It is acknowledged that runaway oxidation occurred in the TMI-2 accident; Petitioner's point, is to draw attention to the fact that Dr. Henry of Fauske & Associates postulated runaway oxidation commenced at 1832°F—368°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. 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 and MAAP Users Group.)

Second, Petitioner does not intend to use Dr. Henry's postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident to support Petitioner's argument that the 10 C.F.R. § 50.46(b)(1) PCT limit should be decreased to a temperature lower than 1832°F.

Third, Petitioner is discussing what Dr. Henry postulated, because Petitioner finds it compelling that Dr. Henry postulated that an autocatalytic reaction commenced at 1000°C in the TMI-2 accident. In Dr. Henry's presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007,[18] Dr. Henry states that "[a]t about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal runaway state."[19]

Fourth, information presented in "TMI-2: A Textbook in Severe Accident Management," regarding the Zircaloy-steam reaction and core damage phenomena, does pertain to PRM-50-93 and PRM-50-95.

Fifth, it is significant that in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in PRM-50-93 and PRM-50-95—including the CORA experiments and LOFT LP-FP-2 experiment.

---

[18] 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.

[19] *Id.*

(It is significant that Dr. Robert E. Henry is clearly very knowledgeable about severe accident phenomena. It is also significant that, in the acknowledgements for "TMI-2: A Textbook in Severe Accident Management," one of the presentation slides states that Dr. Dana Powers sent Dr. Henry the slides Dr. Powers had used in lectures on the TMI-2 accident and that Hans Fauske, D.Sc., reviewed all of the slides presented in "TMI-2: A Textbook in Severe Accident Management": Dr. Powers and Fauske, D.Sc., are also clearly very knowledgeable about severe accident phenomena.)

It is compelling that one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

> Fuel Cladding Oxidation
>
> - As the boil-off of the water in the core continued, the uncovered region continued to heatup with the highest cladding/fuel temperatures being at about the 3/4-core height location.
>
> - Increasing temperatures caused the Zircaloy oxidation rate to increase which was accompanied by an increased release rate of chemical energy.
> - *At about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal-runaway state.* During this interval the Zircaloy reaction was limited by the rate of steam generated in the covered part of the core which decreased as the water level decreased [emphasis added].[20]

So Dr. Henry postulated that runaway oxidation commenced at approximately 1000°C. And another one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," 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."[21]

It is significant that one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

> Fuel Cladding Oxidation
>
> - The Zr in the Zircaloy cladding will oxidize in a high temperature steam environment: hydrogen and energy (heat) are released by this reaction:
>
> $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \Delta H_R$

[20] *Id.*
[21] *Id.*

- The heat of reaction, ΔH_R, is about 6.5 MJ/kg.

- At about 1000°C, the rate of chemical energy release approximately equals the decay power.

- The oxidation rate increases with increasing temperature, which leads to an escalating core heatup rate.

- Therefore, the core damage was generally caused by the cladding oxidation.[22]

It is also significant that another one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

Example: Core Heatup Rate Escalation Due to Cladding Oxidation

- Important Tests:

- Out-of-Reactor: CORA

- In-Reactor: [PBF] SFD, FLHT, LOFT LP-FP-2, and PHEBUS[23]

So in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in PRM-50-93 and PRM-50-95—including the CORA experiments and LOFT LP-FP-2 experiment. And it is compelling that Dr. Henry postulated that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident—368°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

D. National Research Universal Thermal-Hydraulic Experiment 1

21-5

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

[22] *Id.*
[23] *Id.*

TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[24]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[25] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[26] "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."[27])

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.[28]

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 cladding temperature was 2040°F.[29] 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.

---

[24] 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.
[25] *Id.*, p. 10.
[26] 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.
[27] *Id.*
[28] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.
[29] *Id.*

(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,[30] 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 that, in the event of a LOCA, at a nuclear power plant, 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, with high probability, 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.[31]

(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.)

21-6

E. The Damage PWR Fuel Assembly Components would Incur at "Low Temperatures"

"Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents" states that "[e]xperiments were performed at several laboratories to investigate the behavior of (Ag,In,Cd) control rods during severe reactor accidents"[32, 33] and that the

[30] *Id.*, p. 10.
[31] 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.
[32] D.A. Powers, "Behavior of Control Rods during Core Degradation," NUREG/CR-4401, SAND85-0469, 1985; B.R. Bowsher, R.A. Jenkins, A.L. Nichols, N.A. Rowe, J.A.H. Simpson, "Silver-Indium-Cadmium Control Rod Behavior during a Severe Reactor Accident," AEEWR-R 1991, 1986; David A. Petti, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," NUREG/CR-4876, EG + E-2501, 1987; and F. Nagase, H. Uetsuka, "Some Topics from the Basic Experiments on High-Temperature Core Materials Behavior at JAERI," JAERI, Tokai Research Establishment, Japan.

essential results of the experiments are summarized in a paper titled "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents."[34]

Regarding the results of the experiments, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents." states:

> *The (Ag,In,Cd) alloy melts at about 800°C, but will not affect core degradation as long as the molten material is contained within the stainless steel cladding.* As the temperature increases, some of the control rod constituents will vaporize within the cladding until failure occurs, either from internal pressurization or from melting of the cladding. At low pressures of the primary system and when no Zircaloy is present, the control rod fails between 1350 and 1450°C. *Failure of the control rods with the Zircaloy guide tubes occurs at about 1200°C as a result of thermal expansion, physical contact, and eutectic chemical interactions between the stainless steel cladding and the Zircaloy guide tube.* The high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the cladding fails. The ejected material results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy. Due to the high vapor pressure of Cd it vaporizes. Liquid control rod material continues to vaporize if it remains at high temperatures. The control rod material which will flow out of the hot regions of the core freezes and may inhibit steam and/or water flow. At high system pressure, over-pressurization of the rod does not occur. Instead, upon failure, the alloy flows to cooler regions of the reactor core. In all cases the resulting reaction products melt at low temperatures and enhance by this the degradation of the reactor core [emphasis added].[35, 36]

So "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," states that "[f]ailure of the control rods with the Zircaloy guide tubes occurs at about 1200°C"[37] and that "[t]he high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the

---

[33] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," Kernforschungszentrum Karlsruhe, KfK 4670, 1989, p. 1.
[34] David A. Petti, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," NUREG/CR-4876, EG + E-2501, 1987.
[35] *Id.*
[36] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, pp. 1-2.
[37] *Id.*, p. 1.

cladding fails"[38] that "results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy."[39]

And regarding eutectic interactions of the absorber rod's steel cladding tube and the Zircaloy guide tube that can cause liquefaction to occur locally at approximately 1200°C, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> The absorber rod should fail; *i.e.*, melt down, upon attainment of the melting point of the steel cladding tube (~1400°C) at the latest. *On account of eutectic interactions of the steel cladding tube and the zircaloy guide tube, liquefaction can take place locally as early as from 1200°C on.* The (Ag,In,Cd) absorber melt contributes essentially to the propagation of damage in the bundle which is an unambiguous finding of chemical-analytical studies of the reaction products by means of the scanning electron microscope [emphasis added].[40]

It is significant that "when no Zircaloy is present, the control rod fails between 1350°C and 1450°C"[41] or that the control rod fails at ~1400°C, at the latest.[42] So when Zircaloy is present, the control rod fails at a temperature—approximately 1200°C—that is between 150°C and 250°C lower—a substantial temperature difference.

Describing the damage PWR fuel assembly components would incur at relatively low temperatures, in more detail, the conclusion of "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents" states:

> - *The (Ag,In,Cd) absorber alloy starts to melt at about 800°C, but this will not affect core degradation as long as the molten material is contained within the stainless steel (AISI 316) cladding.* The chemical interaction between the absorber alloy and stainless steel is negligible.
>
> - Failure of the stainless steel absorber rod cladding takes place as a result of either internal pressurization (high Cd vapor pressure) or eutectic interactions with the Zircaloy guide tube (bowing of the rods at high

[38] *Id.*
[39] *Id.*, pp. 1-2.
[40] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7448, 2008, p. 14.
[41] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, p. 1.
[42] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, p. 14.

temperatures). The released (Ag,In,Cd) melt can then interact with the Zircaloy guide tube.

- The Zircaloy will be chemically dissolved by the absorber alloy. The dissolution of the Zircaloy can be described by a parabolic rate law. *The dissolution rate is very fast; at 1200°C, it takes only about 50 [seconds] to dissolve 1 mm Zircaloy and about 4 minutes to destroy the entire 2.25 mm thick Zircaloy crucible wall.*

- As soon as solid state contact occurs between the stainless steel cladding and the Zircaloy guide tube, eutectic interactions take place which can be described by parabolic rate laws. *Liquid phases form at around 1000°C, and a fast and complete liquefaction of both components takes place above 1250°C. Only small amounts of stainless steel are necessary to dissolve great amounts of Zircaloy, and it takes only a little more than 2 minutes to destroy the 2.25 mm thick Zircaloy crucible wall at 1200°C.*

- Thin ZrO_2 layers (~10 μm) on the Zircaloy surface delay the chemical interactions of Zircaloy with the (Ag,In,Cd) alloy or the stainless steel, but cannot prevent them. The ZrO_2 layer must be dissolved by the Zry before chemical interactions can take place. The required incubation period depends on temperature and time. Dissolved oxygen in the Zircaloy, forming oxygen-stabilized α-Zr(O), reduces the reaction rates and shifts the liquefaction temperature to slightly higher levels.

- With respect to the chemical behavior of (Ag,In,Cd) absorber rods during severe reactor accidents, meltdown and relocation must be assumed to occur at temperatures around 1250°C. The resulting melt destroys the Zircaloy cladding of the fuel rods and dissolves a part of the UO_2, contributing substantially to fuel element degradation. *Since UO_2 fuel can be liquefied at temperatures as low as 1250°C, this process has a strong impact on the release of volatile fission products.*

- The premature low-temperature failure of the PWR absorber rods and the localized relocation of (Ag,In,Cd) alloy within the reactor core may cause criticality problems during flooding of the destroyed core [emphasis added].[43]

[43] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, pp. 13-14.

And describing chemical interactions between the (Ag, In, Cd) absorber rod alloy and Zircaloy, in detail, "Current Knowledge on Core Degradation Phenomena, a Review" states:

> The absorber rod alloy (80 wt% silver, 15% indium, 5% cadmium) is thermodynamically stable with its stainless steel cladding, even in the liquid state (>800°C). *However, the absorber rod guide tube is made from Zircaloy, which will chemically interact with the stainless steel cladding of the absorber rod. During a severe reactor accident, localized contact between stainless steel and Zircaloy exists at many places. This solid-state contact results in chemical interactions with the formation of liquid phases around 1150°C.* After failure of the absorber rod cladding, the molten Ag-In-Cd alloy (melting point ~800°C) comes into contact with the Zircaloy guide tube and chemically destroys it. Then, the molten Ag-In-Cd can even attack and chemically dissolve the Zircaloy cladding of the fuel rods well below the melting point of Zircaloy (~1760°C). The relocating Ag-In-Cd alloy is therefore able to propagate and accelerate the core-melt progression at rather low temperatures.
>
> The chemical interactions between Ag-In-Cd and Zircaloy were studied in separate-effects tests which are described in [reference 19].[44] The reaction zone growth rate (decrease in Zircaloy wall thickness) is plotted in an Arrhenius diagram against the reciprocal temperature in Fig. 10. *At temperatures >1200°C, the chemical interactions result in a sudden and complete liquefaction of the compatibility specimens.* As a consequence, the Zircaloy cladding can be chemically dissolved ~600 K below its melting point and may even result in a low-temperature UO_2 fuel dissolution. For phase considerations of melting reactions, the quaternary U-Zr-Fe-O system may be regarded as a model system for the complicated multi-component system of a beginning core melt; iron represents the stainless steel. A detailed description of the phase relations is given in [reference 4].[45]
>
> The chemical interaction between the Ag-In-Cd alloy and Zircaloy is theoretically described by a model under conditions of convective mixing in the (Zr, Ag, In) liquid phase in [reference 20].[46] Homogeneous bulk saturation of the liquid phase with Zr takes place in the course of the Zircaloy dissolution by the absorber melt resulting in a gradual decrease of the interaction process. Two main parameters of the model are calculated: Zr concentration in the saturated melt and convective mass transfer coefficient in the liquid phase[47] [emphasis added].[48]

---

[44] P. Hofmann, M. Markiewicz, Journal of Nuclear Materials, 209, 1994, p. 92.
[45] P. Hofmann, *et al.*, Nuclear Technology 87, 1989, p. 146.
[46] M.S. Veshchunov, P. Hofmann, Journal of Nuclear Materials, 228, 1996, p. 318.
[47] *Id.*

21-7

And regarding the fact that control rod material (Ag-In-Cd) may influence the chemical reaction between Inconel grid spacers and Zircaloy fuel cladding, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core" states:

> The CORA-7 test[49] indicated that the reaction between [Inconel] grid spacer and [Zircaloy] cladding was not symmetrical and that control rod material (Ag-In-Cd) may influence the interaction between grid spacer and cladding.[50]

So clearly, in the event of a LOCA, PWR core component damage could commence at relatively low temperatures.

F. Chemical Interactions Between Zircaloy and Inconel and Between Zircaloy and Stainless Steel 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."[51]

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[52] and Japan,[53] in-pile experiments at the PBF facility in Idaho,[54]

---

48 P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, pp. 201-202.

49 P. Hofmann, *et al.*, "Material Behavior in the Large PWR Bundle Experiment CORA-7," International CORA Workshop 1991, September 23-26, 1991, Karlsruhe, Germany.

50 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, p. 436.

51 L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," Abstract, p. 427.

52 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.

53 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.

> and by examinations of the damaged Three Mile Island (TMI-2) core.[55] *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 with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers* [emphasis added].[56]

It is significant that Inconel grid spacers would effect the progression of damage in a reactor core during a LOCA if their temperatures were to reach approximately 2012°F;[57] and significant that experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.

And discussing the fact that a first melting process started at approximately 1250°C at the central Inconel grid spacer 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:

> A first melting process starts already at about 1250°C at the central grid spacer of Inconel, due to diffusive interaction in contact with Zry cladding material, by which the melting temperatures of the interaction partners (ca. 1760°C for Zry, ca. 1450°C for Inconel) are dramatically lowered towards the eutectic temperature, where a range of molten mixtures solidifies. (This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[58]

---

[54] 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.
[55] E.L. Tolman, *et al.*, "TMI-2 Accident Scenario Update," EGG-TMI-7489, EG&G Idaho, Inc., December 1986.
[56] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.
[57] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[58] 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.

It is also significant that "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:

> Only small amounts of Inconel are necessary to dissolve great amounts of [Zircaloy].[59]

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.*[60]
>
> 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).*[61]

[59] *Id.*, p. 40.
[60] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[61] 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.[62] *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*[63] [emphasis added].[64]

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"[65] and that "[l]ocalized liquefaction of these components started around 1200°C [(2192°F)]."[66] It was also observed in the CORA tests that "[l]iquid phases form at temperatures <1000°C [(1832°F)]" and that "the reaction kinetics become significant only above 1100°C [(2012°F)]."[67]

21-8

1. Additional Information—from the 1970s—on the Chemical Interaction Between Zircaloy and Inconel at "Low Temperatures"

The chemical interaction between Zircaloy and Inconel was a subject of the IP-2 licensing hearing, in 1971, as the selection below from the transcript of the IP-2 licensing hearing demonstrates:

> Daniel Ford: I am concerned with one of the many gaps in the Interim Policy Statement and the computer code. I am concerned with a variety of chemical-metal-water reactions that are not considered at all in these codes, metal-water reactions which various recent experimental data indicate can prove [to] very significantly [impact] local temperature during an accident, and [cause] extensive cladding damage. The specific

[62] P. Hofmann, M. Markiewicz, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Inconel 718 at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 4729, 1994.
[63] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[64] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[65] *Id.*
[66] *Id.*
[67] *Id.*

> metal-water reaction I am concerned with at the moment is the reaction between the Zircaloy-Inconel eutectic and steam, I am concerned to find out how the Applicant's analysis contained in the computer code, which does not consider this, how it would be different if it did.
>
> Leonard M. Trosten: I thank you for the explanation. I recognize this as being one of the principal points of concern in the critique by the Union of Concerned Scientists...[68]

So in 1971, Daniel Ford of UCS stated that he was concerned with one of the many gaps in the Interim Policy Statement: the Zircaloy-Inconel eutectic reaction. Unfortunately, to this day, nearly 40 years latter, NRC's regulations still do not consider the Zircaloy-Inconel eutectic reaction that would, with high probability, occur at temperatures lower than 2200°F.

Discussing chemical interactions between Zircaloy and Inconel X-750, a paper published in 1975, "Incompatibility between Zircaloy-2 and Inconel X-750 during Temperature Transients," states:

> All current designs of water reactors contain various components made from high-nickel and/or high-iron content alloys. In certain specific cases these components are either in contact with, or in close proximity to, other components constructed from high-zirconium content alloys. Typical examples of such high-nickel or high-iron content alloy components are, without mentioning the reactor type specifically, plenum springs, fuel element spacer grids, control rod claddings and slides, and wear pads.
>
> All of these components could potentially react with the adjacent zirconium alloy component, since, although the alloys themselves have relatively high melting points (*e.g.*, Inconel X-750 melts at 1395°C), their major components nickel and iron form eutectics with zirconium at the much lower temperatures of 960°C [(1760°F)] and 940°C [(1724°F)], respectively.[69]
>
> The problems which could arise because of the existence of these low melting point eutectics have been demonstrated during heat-treatments, carried out as part of the BWR-FLECHT test under emergency core

[68] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2520-2522.

[69] M. Hansen, "Construction of Binary Alloys" McGraw-Hill, 1958.

> cooling conditions,[70] in which Inconel springs became inseparably fused to zircaloy tubing because of melting at the points of contact. Similar results have been observed occasionally during vacuum brazing of Inconel in which zirconium has been present as a getter.[71] Also, it has long been recognized that high temperature mechanical testing of zirconium and its alloys cannot be accomplished using nickel or iron based alloys for the gripping devices.[72, 73]

And providing additional information on views from the early 1970s of the chemical interaction between Zircaloy and Inconel, the selection below from the transcript of the IP-2 licensing hearing states:

> James S. Moore: Well, we had the Inconel grid in a reactor contacting a Zircaloy rod as it does in a reactor at a local point with a spring, and then we heated up the total assemblage of the rod and the grid and to the point of temperatures, I forget the exact numbers, but they were up well into the zirc-water reaction temperatures approaching 2300 degrees [Fahrenheit], and then observed what happened at the local contact point between the Inconel and the zirc rod, and as I indicated and as other people have observed there is a eutectic formed between the Zircaloy and the nickel of the Inconel grid, which has a melting point of about 1760 degrees [Fahrenheit] and we did get in the test very local melting at this contact point. But this did not create any difficulty in that the heat was carried away from that local point sufficiently so that there was no blockage, any additional blockage, or effects on the Zircaloy rod itself. So that this is based on holding this rod in the Inconel grid at about 2300 degrees [Fahrenheit] for several minutes, which is well beyond what you would expect in a loss of coolant situation.
>
> Anthony Z. Roisman: Let me see if I understand this. The eutectic is formed at about 1760 degrees [Fahrenheit], is that right?
>
> James S. Moore: It melts at 1760 [degrees Fahrenheit].
>
> Anthony Z. Roisman: You are saying that no reaction occurs with the water until it reaches almost 2300 degrees Fahrenheit?

---

[70] M.J. Grater, W.F. Zelenzny, R.E. Schmunk, Idaho Nuclear Corp., IN-1453, Idaho Falls, March 1971.

[71] J. Christensen: Danish Atomic Energy Commission, Research Establishment Risø, private communication, 1974.

[72] B. Weiler Madsen: Danish Atomic Energy Commission, Research Establishment Risø, private communication, 1975.

[73] M.R.Warren, K. Rørbo, E. Adolf, Danish Atomic Energy Commission, Research Establishment Risø, "Incompatibility between Zircaloy-2 and Inconel X-750 during Temperature Transients," Journal of Nuclear Materials, 58, 1975, p. 185.

> James S. Moore: No.
>
> Anthony Z. Roisman: Exactly how does the melting occur? Are droplets formed that would tend to drop down between the rods?
>
> James S. Moore: Yes. There was very local melting, but we didn't observe any droplets or any sputtering which you might postulate under those conditions. It was a very localized effect and that is what gave us the assurance that this was not a problem for us.[74]

Clearly, James S. Moore's claim that the Inconel-Zircaloy reaction would not be a problem at 2300°F (1260°C) is erroneous.

Regarding the Inconel-Zircaloy reaction at 2282°F (1250°C), "Current Knowledge on Core Degradation Phenomena, a Review" states:

> During heat-up of the...Inconel/Zircaloy reaction systems, a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C [(2282°F)]. This may be the reason that melt progression in a fuel rod bundle initiates at...Inconel spacer grid/Zircaloy fuel rod contact locations.[75, 76]

Furthermore, "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[77] and Japan,[78] in-pile experiments at the PBF facility in Idaho,[79]

---

[74] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, pp. 2170-2171.

[75] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.

[76] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.

[77] 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.

[78] 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.

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

> and by examinations of the damaged Three Mile Island (TMI-2) core.[80] 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 with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers* [emphasis added].[81]

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by erroneous notions of the Inconel-Zircaloy reaction.

21-9

G. The Zircaloy-Steam Reaction could be Affected by the Boron Carbide (B_4C) Absorber

It is significant that in PRM-50-95, Petitioner discussed the fact that in a BWR LOCA, the Zircaloy-steam reaction could be affected by the boron carbide (B_4C) absorber.

In PRM-50-95, Petitioner quotes "Degraded Core Quench: A Status Report," which compares the BWR CORA-17 experiment with the PWR CORA-12 and CORA-13 experiments (which used typical PWR bundles and Ag-In-Cd absorber).

Regarding this issue, "Degraded Core Quench: A Status Report" states:

> The earlier starting and stronger reaction in the [CORA-17] BWR test can be interpreted as being due to the additional influence of the boron carbide [(B_4C)] absorber. *This material has an exothermic reaction rate three times larger than that of Zircaloy* and produces [four] to [eight] times more hydrogen [emphasis added].[82]

So according to "Degraded Core Quench: A Status Report," boron carbide (B_4C) has an exothermic reaction rate approximately three times greater than that of Zircaloy.

---

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

[81] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.

[82] 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," OCDE/GD(97)5, August 1996, p. 16.

Additionally, comparing the BWR CORA-17 experiment with the PWR CORA-12 and CORA-13 experiments "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17" states:

> Immediately after quenching BWR test bundle CORA-17 experiences a modest increase for 20 sec. and changed then in a steep increase resulting in the highest temperature and hydrogen peaks of the three tests [(CORA-12, CORA-13, CORA-17)]. CORA-17 also showed a temperature increase in the lower part of the bundle... We interpret this earlier starting and stronger reaction [as being] due to the influence of the boron carbide, the absorber material of the BWR test.
>
> *B_4C has an exothermic reaction energy [four] to [five] times larger than Zry* and produces about [six] times more hydrogen. Probably the hot remained columns of B_4C (seen in the non-quench test CORA-16) react early in the quench process with the increased upcoming steam. *The bundle temperature, raised by this reaction increases the reaction rate of the remained metallic Zry* (exponential dependence) [emphasis added].[83]

And according to "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," boron carbide (B_4C) has an exothermic reaction rate approximately four to five times greater than that of Zircaloy. Furthermore, the increased bundle temperature—a consequence of the B_4C exothermic reaction energy—in turn, increases the reaction rate of the remaining Zircaloy.

H. A Portion of the IP-2 Licensing Hearing Transcript: Superheated Steam in a LOCA Environment

It is significant that in 1971, in the IP-2 licensing hearing, Daniel Ford of UCS was concerned about the role that superheated steam would play in a LOCA environment.

Regarding this issue, a portion of IP-2 licensing hearing transcript states:

> Daniel Ford: Mr. Moore, is it correct that in the [PWR] FLECHT tests[84] negative heat transfer coefficients [calculated as a result of heat transfer

---

[83] 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, Abstract, pp. ii.

[84] The transcript states "BWR FLECHT tests, not "PWR FLECHT tests";" however, it is most likely that Daniel Ford was actually asking a question about the PWR FLECHT tests. First, James S. Moore was an employee of Westinghouse Electric, which conducted the PWR FLECHT tests. Second, negative heat transfer coefficients—calculated as a result of heat transfer from the coolant to the fuel cladding—occurred in the PWR FLECHT tests. Third, results from the PWR

from the coolant to the fuel cladding] were observed at axial levels in a number of different instances?

James S. Moore [of Westinghouse Electric]: They were recorded as negative heat transfer coefficients. What they actually indicate is reverse heat transfer from the coolant to the [fuel] cladding.

Daniel Ford: For the purpose of this discussion and since they are plotted as heat transfer coefficients, would you just accept the definition of terms, that is a negative heat transfer coefficient?

James S. Moore: I guess I'd prefer reverse heat transfer, which is more descriptive.

Daniel Ford: I see. It is correct, though, that the reverse heat transfer coefficients are represented in your data as negative heat transfer coefficients, is that correct?

James S. Moore: Yes, yes.

Daniel Ford: Thank you.

Do you agree that if you passed a saturated vapor, saturated steam through a furnace that you'd create superheated steam?

James S. Moore: If I pass saturated steam through a furnace I create superheated steam?

Daniel Ford: Yes.

James S. Moore: Yes.

Daniel Ford: Do the codes that you use for analyzing the loss-of-coolant accidents explicitly consider the formation of superheated steam or do they regard the coolant at different axial levels being simply liquid entrained in steam, period?

James S. Moore: It depends on which calculations you are talking about.

Daniel Ford: In the calculations that you have used for Indian Point 2 to calculate the maximum [fuel] clad temperature, have you separately considered the role of superheated steam in precipitating or yield[ing] the maximum clad temperature?

FLECHT tests would have been used for ECCS evaluation calculations for IP-2, because IP-2 is a PWR.

James S. Moore: In terms of reflooding, yes.

Daniel Ford: In terms of the code analysis that you have done, do you use negative heat transfer coefficients under any assumptions of flooding rate or pressure?

James S. Moore: If they would exist, yes. For the hot spot calculation, such a condition never does exist.

Daniel Ford: I see. In terms of the negative heat transfer coefficients that were observed, can you tell me at what axial levels these were observed?

James S. Moore: They were well above the hot spot. That is specifically the point. They were where the temperature was quite low [on] the [fuel] cladding.

Daniel Ford: Have you done any calculations which guarantee that the superheated steam, a negative heat transfer coefficient would always occur above the mid point?

James S. Moore: Yes.

Daniel Ford: Where are those calculations presented?

James S. Moore: Any one of these core cooling analyses were computed with the hot spot temperature. You can see the temperature itself is much greater than any saturated or even superheated condition that could exist.

Daniel Ford: Those are the calculations that you have presented. What I am asking is whether you have performed parametric calculations that indicate under no circumstances, that is, under no combination of parameters, which you get superheated steam at lower than the ten-foot elevations that it was observed at in the FLECHT test?

James S. Moore: Yes.[85]

It is rather odd that James S. Moore would answer, "yes," to Daniel Ford's last question in the portion of the IP-2 licensing hearing transcript quoted above, given the results of PWR FLECHT run 9573. Ford had asked Moore, "whether [Westinghouse had] performed parametric calculations that indicate under no circumstances, that is,

[85] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 8, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350639, pp. 2921-2924.

under no combination of parameters, [would there be] superheated steam at lower than the ten-foot elevations[, where] it was observed at in the FLECHT test[s],"[86] in the event of a LOCA?

Additionally, it is rather odd that James S. Moore would claim that "[f]or the hot spot calculation, [negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding would] never...exist."[87] And that "[negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding occurred] well above the hot spot. ... [That t]hey were where the temperature was quite low [on] the [fuel] cladding,"[88] given the results of PWR FLECHT run 9573.

It is significant that in FLECHT run 9573—a test conducted with a Zircaloy bundle—negative heat transfer coefficients were observed *at the bundle midplane* for 5 of 14 thermocouples and that steam temperatures exceeded 2500°F *at the seven-foot steam probe*, during a portion of the test.

This was reported in a Westinghouse document, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," in April 1971, months before James S. Moore's testimony.

Regarding the superheated steam, which exceeded 2500°F, and negative heat transfer coefficients observed at the bundle midplane in FLECHT run 9573, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states:

> At the time of the initial [heater element: fuel-cladding simulator] failures [in FLECHT run 9573], 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). ...anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. These may have been related to the high steam probe temperatures measured at the 7 ft elevation.[89]

[86] *Id.*, p. 2924.
[87] *Id.*, p. 2923.
[88] *Id.*
[89] 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, pp. 3-97, 3-98.

It is also significant that, in 2002, regarding superheated steam being located at the hot spots of the fuel rod simulators in FLECHT run 9573, Westinghouse stated:

> The high fluid [superheated steam] temperature [that occurred in 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."[90]

And discussing, in more detail, the superheated steam that was observed one foot above the midplane in FLECHT run 9573, a Westinghouse memorandum, written by Robert H. Leyse states:

> The final FLECHT test (Bundle Z-10) was completed on December 11, 1970. The test was run with flooding of 1 in./sec. beginning at 2000°F. Several heaters failed approximately 18 seconds after flooding when the peak indicated midplane temperature was 2325°F. Heater failure at this temperature is unlikely, particularly under conditions of decay heat and increasing temperature. *The steam probe thermocouple located one foot above midplane in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec.) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding.* It appears likely that *ignition of the Zircaloy grids led to high rates of heat input* at the elevation one foot above (and below) midplane* and this caused over-temperature and failure of the heaters. Test results are currently being studied.
>
> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.
>
> *The ratio of surface area to heat capacity for a Zircaloy grid is approximately 15 times that of a heater rod; hence, Zircaloy-steam reactions can lead [to] steeper temperature ramps in the vicinity of a Zircaloy grid [emphasis added].[91]

---

[90] 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.

[91] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.

It is significant that "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states that the negative heat transfer coefficients that were observed at the bundle midplane in FLECHT run 9573 were "anomalous."[92] Perhaps that is why James S. Moore claimed that negative heat transfer coefficients would never occur at the hot spot of the fuel cladding, in the event of a LOCA. However, Westinghouse's conclusion that the negative heat transfer coefficients observed in FLECHT run 9573 were anomalous had no scientific basis. For example, Westinghouse did not conduct any subsequent tests with Zircaloy bundles after FLECHT run 9573 (with similar test parameters) to confirm that the negative heat transfer coefficients were in fact anomalous. This is unfortunate, given the importance of the safety issues involved.

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by unconfirmed notions that negative heat transfer coefficients—calculated as a result of heat transfer from the coolant to the fuel cladding—would never occur at the hot spot of the fuel cladding, in the event of a LOCA.

I. A Portion of the IP-2 Licensing Hearing Transcript: Integral Experiments Versus Separate Effects Experiments

It is significant that in 1971, in the IP-2 licensing hearing, UCS was concerned that the metal-water reaction rates predicted to occur in IP-2's core, in the event of a LOCA, had not been confirmed by data from large-scale integral experiments. In the portion of the IP-2 licensing hearing transcript quoted below, Daniel Ford of UCS questions the validity of the Baker-Just correlation for use in ECCS evaluation calculations and points out that the Baker-Just correlation was "derived from experimental data that is completely outside of the context of nuclear systems;" *i.e.*, from single-rod separate-effects experiments.

The 1971 IP-2 licensing hearing transcript states:

> James S. Moore: No. There are no large-scale tests for the core. You are talking about a very complex chain of events. You are ending up with a zirc-water reaction. And you have to start with the loss of coolant and go through the blowdown, the reflood, the heat-up, the time and temperature, and then the zirc-water reaction.

---

92 F. F. Cadek, D. P. Dominicis, R. H. Leyse, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.

Daniel Ford: Right. Now in terms of simply your experimental philosophy do you see the necessity, since there are as you note so many complicated factors behind any independent phenomenon, do you see the necessity, the experimental necessity for the kind of integral test that I am talking about or do you think that you can just test individual small components of the problem, you know, assuming all the input from other phenomena?

James S. Moore: I believe it is my opinion that we can properly bound the calculation without a total completely integrated test.

...

Daniel Ford: I am talking about the water reactor safety program which has a variety of experiments on different... Using a variety of different equipment to simulate loss-of-coolant accident[s]. And I am talking about some of the large-scale experiments that are planned to take place [in] 1975 or so in which we will actually have a live reactor and have it subjected to loss-of-coolant transients and see what happens. I am talking about whether or not that is necessary in Mr. Moore's opinion, whether that would make a substantive contribution to the confirmation of these results on metal-water reactions inasmuch as they depend on all the other phenomena of the transient. I am asking him whether that is necessary or whether you can simply take Baker-Just's correlation, which is derived from experimental data that is completely outside of the context of nuclear systems? I am asking him whether we should have these kinds of integral experiments or whether we can just take empirical correlations and just use them with no hesitation...

James S. Moore: I count at least four or five questions in it. Do I think it necessary, do I think it would contribute?

Daniel Ford: I am purposely trying to find out what your philosophy is, what you regard as convincing experimental confirmation of, in this particular case, the metal-water reaction rates that you compute.

...

James S. Moore: In my opinion the totally integrated test is not necessarily a prerequisite to describe a physical phenomenon and in the case of the loss of coolant I don't think it is a requirement. I think you can get very good indications of what phenomena do occur with these separate effects kinds of experiments that have been performed. With respect to zirc-water reaction I would point out that we have come very close to simulating this through the FLECHT test[s].

> Daniel Ford: Now in terms of the water reactor safety research program would you tend not to think that the integral tests were ever really worth their expenditure?
>
> James S. Moore: I didn't say that. Are you asking that question?
>
> Daniel Ford: Yes.
>
> James S. Moore: It's my opinion we will get useful information out of that test, yes.
>
> Daniel Ford: Are there any specific uncertainties that in relation to which the output of these tests will provide useful information?
>
> James S. Moore: None specifically that I am aware of.
>
> Daniel Ford: In terms of the experiments pertaining to accumulator water, are there any that have confirmed in any kind of integral way your own metal-water [reaction] prediction for Indian Point 2?
>
> James S. Moore: I am again having trouble relating between [the] metal-water reaction and accumulators. Could we repeat the question again? That's a long train, from the accumulator to the metal-water reaction.
>
> Daniel Ford: I see. Well your prediction of metal-water reactions as a function of accumulator water, the total reaction rate, has that prediction of yours been confirmed by any experiments?
>
> James S. Moore: No specific experiment, complete integrated experiment.[93]

21-10

Unfortunately, to this day, nearly 40 years after the original IP-2 licensing hearing, the metal-water reaction rates predicted to occur in the event of a LOCA at IP-2 still have *not* been confirmed by data from large-scale integral experiments. In fact, now in 2010, there is a preponderance of metal-water-reaction-rate data from multi-rod severe fuel damage experiments like the LOFT LP-FP-2 experiment; nevertheless, IP-2's ECCS evaluation calculations still use metal-water reaction rate correlations that were derived from the data of single-rod experiments.

[93] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2550-2553.

The LOFT LP-FP-2 experiment, conducted in 1985, is considered "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."[94] In the LOFT LP-FP-2 experiment, "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about...[2060°F]"[95]—approximately 140°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

21-11

J. The First Transient Experiment of a Zircaloy Fuel Rod Cluster in TREAT and the Baker-Just Correlation

1. Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT

In this section Petitioner discusses "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, from March 1971 and the Baker-Just correlation. The First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") was conducted in the Transient Reactor Test Facility ("TREAT").

Describing the FRF-1 experiment, the abstract of "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states:

> The first fuel rod failure experiment in Transient Reactor Test Facility ("TREAT") was performed with a seven-rod bundle of 27 in. long Zircaloy-clad UO_2 fuel rods in flowing steam atmosphere. A water reactor loss-of-coolant accident was simulated by operating TREAT reactor at constant power for 20 sec so that fission heat in the UO_2 pellets caused the Zircaloy cladding temperature to rise 72°F/sec to a maximum of approximately 1800°F. The fuel rods were initially pressurized with helium between 115 and 215 psia (77°F) to simulate accumulated fission gas.

[94] 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.

[95] 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. 30.

> The Zircaloy cladding swelled and ruptured resulting in 48% blockage of the bundle coolant channel area at the location of maximum swelling. The average rod maximum circumferential swelling was 36%. Calculations related the hoop stress and ultimate strength at the onset of rapid expansion. The ideal gas law was used to calculate the rate of cladding expansion from measured rod temperature and internal pressure. Metallographic examination revealed ductile ruptures and significant oxygen pickup. Zirconium-steam reaction was 0.2%.[96]

And describing the Zircaloy-steam reaction that occurred in the FRF-1 experiment, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states:

> [W]e estimate the volume of hydrogen generated by metal-water reaction to be 1.2 ± 0.6 liters (STP). This is equivalent to about 0.2% metal-water reaction based on total cladding volume.[97]

It is significant that "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states that the maximum Zircaloy cladding temperature was approximately 1800°F[98] and that the volume of hydrogen generated by the metal-water reaction was estimated to be 1.2 ± 0.6 liters, which, in turn, was estimated to have been caused by approximately a 0.2% metal-water reaction of the total cladding volume. Because the volume of hydrogen generated by the metal-water reaction was estimated to be 1.2 ± 0.6 liters, it would have made sense for "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" to have also reported that the metal-water reaction was estimated to be 0.2% ± 0.1%.

2. Discussions of the Results of FRF-1 and the Baker-Just Correlation in the 1971 Indian Point Unit 2 Licensing Hearing

In 1971, in the IP-2 licensing hearing, the validity of the Baker-Just correlation for use in LOCA analyses was called into question, because data from the FRF-1 experiment indicated that in the experiment, at approximately cladding temperatures of 1800°F, the metal-water reaction had generated approximately 1.2 ± 0.6 liters of

---

[96] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, Abstract.

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

[98] See Appendix A Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1.

hydrogen. In the IP-2 licensing hearing, the Baker-Just correlation was criticized, because AEC had stated that at 1800°F, LOCA analyses using the Baker-Just correlation predicted that the metal-water reaction is "negligible."[99]

In AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[100]

And in two selections from the transcript of the IP-2 licensing hearing, from November 1, 1971, below, Anthony Z. Roisman—on behalf of Citizens' Committee for the Protection of the Environment and Environmental Defense Fund—addresses the fact that LOCA analyses using the Baker-Just correlation predict that the metal-water reaction is negligible at 1800°F and that that result does not agree with the results of the FRF-1 experiment (reported in ORNL-4635).

On this topic, the transcript states:

> Anthony Z. Roisman: In ORNL-4635, the 0.2 per cent was determined to be the amount of metal-water reaction that had occurred in rods at the 1800 degree Fahrenheit level. You said that the report pointed out there could be an error of plus or minus fifty per cent. In short, [the metal-water reaction] could have been 0.3 per cent or 0.1 per cent.
>
> You also mentioned that two per cent of the cladding in the analysis that Westinghouse does is assumed to reach 1800 degrees Fahrenheit or more temperature.
>
> What percentage of metal-water reaction do you predict will occur for that, for those rods at 1800 degrees Fahrenheit?
>
> James S. Moore [of Westinghouse Electric]: We would predict zero.
>
> Anthony Z. Roisman: You would predict no metal-water reaction at 1800 degrees Fahrenheit?
>
> James S. Moore: Yes.

---

[99] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.
[100] *Id.*

Anthony Z. Roisman: What about at 1900 degrees Fahrenheit?

James S. Moore: Well, now it is a function of how long you are at that temperature.[101]

Then, continuing on the same topic, the transcript states:

Anthony Z. Roisman: ... I'd like to go back to the metal-water reaction. I guess I am still a little unclear about this, ORNL-4635 had predicted 0.2 percent of metal-water reaction for rods at 1800 degrees Fahrenheit.

You said that you predict for this plant no metal-water reaction at 1800 degrees Fahrenheit.

James S. Moore: I don't think the Oak Ridge report predicted that. They assumed they measured it.

Anthony Z. Roisman: Can you tell me, is your basis for not predicting 0.2 percent metal-water reaction at 1800 degrees Fahrenheit based upon some experiments which Westinghouse has run or with which you are familiar, that demonstrate that there won't be any metal-water reaction at that temperature?

James S. Moore: These are based on experiments that have been performed by others. I'm not aware of any specific Westinghouse experiments in this area. But these are the experiments which were added to and summarized in the reference by Baker and Just that I believe you have from Argonne. That's the basis for the parabolic rate assumption.

Anthony Z. Roisman: Can you explain in a little more detail that the most recent Oak Ridge National Laboratory report doesn't require modification of that?

James S. Moore: I'm not an expert on Zirc-water reaction per se. But just looking at the report, it seemed to me, number one, this is one data point only. Also, it looked like it was pretty susceptible to interpretation and measurement in the way they derived the amount of hydrogen, and related this back to Zirc-water reaction.

So I think I really don't know the validity of that or any conclusions with respect to Zirc-water. It was not an experiment, as I understood, to specifically work on Zirc-water reaction aspects.

---

[101] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, pp. 2152-2153.

> Anthony Z. Roisman: Mr. Wiesemann, is this your area, the [Zirc-]water reaction? Mr. Moore testified that he is not really expert in that. Are you?
>
> Mr. Wiesemann: No. However, I am knowledgeable in general. I don't think my expertise in Zirc-water reaction area is any greater than Mr. Moore's. With regard to the question he just answered, about tests performed by Westinghouse, I am personally aware of some exploratory type tests which were done a long time ago when we first went into the use of Zircaloy cladding in reactors, exploring just exactly the thing you were discussing. That is the temperature range of 1800 to 2000 degrees [Fahrenheit], and the effect of this type of condition on Zircaloy rods to confirm for ourselves that there was no significant metal-water reaction in that range of temperatures in order to confirm for ourselves. The results of this, as far as I know, were never published. In order to get further details on that, we would probably have to consult some of the people who actually performed those tests. I observed those tests but I was not actually performing the test myself.[102]

And in a selection from the transcript of the IP-2 licensing hearing, from November 2, 1971, below, Leonard M. Trosten—on behalf of Consolidated Edison Company of New York, Inc.—also addresses the Baker-Just correlation, FRF-1 experiment, and the Zircaloy-water reaction—in response to Anthony Z. Roisman's questions from the previous day, on the same subjects.

On this topic, the transcript states:

> Leonard M. Trosten: ... Now with respect to the question which appears on the transcript page 1720 [a different page number than the current transcript], relating to the zirconium-water reaction, are you familiar with the question that was raised by Mr. Roisman yesterday concerning that matter?
>
> Dr. Jack Roll [of Westinghouse Electric]: Yes, sir. I reviewed the transcript.
>
> Leonard M. Trosten: Would you please comment with regard to the question raised by Mr. Roisman.
>
> Dr. Jack Roll: I believe the context of the question was that based upon the results reported in the reference ORNL document [ORNL-4635] did we have any reason to re-evaluate our application, I believe, of the Baker-Just equation to a computation of degree of zirc-water reaction, and I believe that Mr. Moore provided essentially the answer that I would have

[102] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, pp. 2166-2168.

provided, that that is no we could not use that single data point to re-evaluate or reapply the Baker-Just equations.

As pointed out by Mr. Moore in yesterday's proceedings, the measurement of the extent of zirc-water reaction was in fact by an inferred route, and there were no direct measurements taken. There was a large uncertainty in the measurement of total hydrogen evolution during the experiment.

The subtraction of other known effects resulted in a fifty per cent uncertainty in the amount of hydrogen which can be associated or applied with the zirc-water reaction, and from this they inferred the two-tenths per cent raw metal-water reaction. This was then compared, presumably by Mr. Roisman, to indicate that perhaps there was more zirc-water reaction here than one would expect based on reported temperatures.

But however, I pointed out in the Oak Ridge report there was not a direct measurement of temperature and they point out that the effects of thermocouple effects themselves and the power distribution with the bundle it enters result in an uncertainty in the temperatures of the fuel during the experiment..

Therefore, one cannot make a direct inference on reported temperatures and lead yourself to the conclusion that the extent of zirc-water reaction was higher or much higher than would have been predicted by Baker-Just.

I'd like to add further that we have, as a part of our work, in particular under the FLECHT program, reviewed the extent of zirc-water reaction, under what we considered to be much more representative conditions, that is zircaloy clad fuel rods with our particular time and temperature histories and our particular coolant content, that is our particular water conditions, and I believe as reported in the documentation summarized in the FLECHT reports we find very good agreement with the Baker-Just equation, and so we believe in summary that the Oak Ridge report presents a single data point to germaneness to our specific application must be questioned inasmuch as the data point was not, the test was not run to substantiate the Baker-Just equation.

And secondly, in summary, the work that we have done under the FLECHT program and reported in the FLECHT reports we believe reaffirms our use of the Baker-Just equations in evaluating zirc-water reaction under our conditions of loss-of-coolant accident.[103]

[103] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350642, pp. 2297-2299.

In the transcript of the IP-2 licensing hearing, from November 2, 1971, right after, Leonard M. Trosten questioned Dr. Jack Roll, Daniel Ford stated "[t]he authors of that Oak Ridge report, ORNL-4635,[104] contend that [the FRF-1 experiment] is the most realistic simulation of loss-of-coolant accident conditions to date;" then Daniel Ford asks Dr. Roll, "[d]o you dispute that claim?"[105]

It is significant that, discussing the FRF-1 experiment, in *A Distant Light: Scientists and Public Policy*, Henry W. Kendall states, "[h]ydrogen generated by zirconium-steam reactions was identified. In the words of the report:

> The Zircaloy cladding swelled and ruptured resulting in 48% blockage of the bundle coolant channel area at the location of maximum swelling. ...examination revealed ductile ruptures and significant oxygen pickup.[106]

The relevance of these results derives from the fact that the test 'was conducted under the most realistic loss-of-coolant accident conditions of any experiment to date.' "[107]

(It is noteworthy that "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," states that "[t]he transient test program [for Zircaloy-clad fuel rod clusters in the TREAT facility] is presently inactive because funding is not available"[108] and that "[s]upport of ORNL work on fuel rod failure is now scheduled to be terminated at the end of FY-71."[109]

So the experimental program that conducted "the most realistic loss-of-coolant accident conditions of any experiment to date"[110]—up to 1971—was not provided with funding so investigators could continue researching important safety issues.)

---

[104] The authors of ORNL-4635 are R. A. Lorenz, D. O. Hobson, and G. W. Parker.
[105] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2300.
[106] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, Abstract.
[107] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer , New York, 2000, p. 43.
[108] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.
[109] *Id.*, p. ix.
[110] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43:

It is also significant that, in one of the selections from the transcript of the IP-2 licensing hearing above, from November 2, 1971, Dr. Roll opines that the FLECHT program provided a more realistic representation of the Zircaloy-steam reaction in a LOCA environment, than the FRF-1 experiment; and that the FLECHT results were in "very good agreement with the Baker-Just equation."[111]

And on this topic—to repeat a section of the IP-2 licensing hearing transcript quoted above, from November 2, 1971—Dr. Jack Roll of Westinghouse Electric states:

> I'd like to add further that [Westinghouse Electric has], as a part of our work, in particular under the FLECHT program, reviewed the extent of zirc-water reaction, under what we considered to be much more representative conditions, that is zircaloy clad fuel rods with our particular time and temperature histories and our particular coolant content, that is our particular water conditions, and I believe as reported in the documentation summarized in the FLECHT reports we find very good agreement with the Baker-Just equation, and so we believe in summary that the Oak Ridge report [ORNL-4635] presents a single data point [that at cladding temperatures of approximately 1800°F, the metal-water reaction generated approximately 1.2 ± 0.6 liters of hydrogen and that the metal-water reaction was estimated to be 0.2% ± 0.1%] to germaneness to our specific application must be questioned inasmuch as the data point was not, the test was not run to substantiate the Baker-Just equation.
>
> And secondly, in summary, the work that we have done under the FLECHT program and reported in the FLECHT reports we believe reaffirms our use of the Baker-Just equations in evaluating zirc-water reaction under our conditions of loss-of-coolant accident.[112]

Then, soon afterwards in the transcript, describing metalographic cross-sections that were taken from rods from the four Zircaloy PWR FLECHT tests, Dr. Jack Roll states:

> The measurement that [Westinghouse Electric] took in evaluating the result of our FLECHT test with regard to the extent of [the] zirc-water reaction were in fact metalographic cross-sections at various enlargements from which the experienced metalographers can infer [the] nature of the phases in the cross-section. That is they can determine the portion of the original Zircaloy which remains as original Zircaloy. That portion which is oxygen saturated, that portion which is in fact converted to zirconium

---

[111] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2299.
[112] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2299.

> oxide. With these direct measurements at a number of cross-sections, one can then calculate explicitly the quantity of zirconium which has been converted to zirconium dioxide and the quantity of zirconium which is oxygen saturated from which you can then determine the total quantity of zirconium which has in fact reacted in some way with the oxygen.
>
> ...
>
> I believe the technique of looking at zirconium and zirconium oxide is in itself a primary source of data and need not be substantiated somewhere else. The question is, how do we know what is the extent of zirconium and oxygen reaction. The answer is, you know this by looking at the quantity of zirconium which has been converted to zirconium oxide.[113]

So Dr. Jack Roll explains that it was through examinations of metalographic cross-sections that were taken from rods from the four Zircaloy PWR FLECHT tests that "the work that [Westinghouse Electric did] under the FLECHT program...reaffirms [the] use of the Baker-Just equations in evaluating [the] zirc-water reaction under [the] conditions of [a] loss-of-coolant accident.[114]

However, this is problematic—as Petitioner explained in PRM50-93 and in Petitioner's comments on PRM-50-93, dated March 15, 2010—because there is no metallurgical data from the locations of run 9573 that incurred runaway (autocatalytic) oxidation: Westinghouse did not obtain such data. To explain this problem more completely, in the next section, Petitioner will replicate the text from Petitioner's comments on PRM-50-93, dated March 15, 2010 in the section titled "Supplementary Information to PRM-50-93 Section III.C.1.h. Examining the Autocatalytic Metal-Water Reaction that Occurred during FLECHT RUN 9573."

21-12

3. There is No Metallurgical Data from the Locations of FLECHT Run 9573 that Incurred Runaway (Autocatalytic) Oxidation

As mentioned in PRM-50-93, there is no metallurgical data from the locations of run 9573 that incurred runaway (autocatalytic) oxidation, because Westinghouse did not obtain such data. When Westinghouse performed the metallurgical analyses for the assembly of FLECHT run 9573, Westinghouse measured oxide thicknesses in the locations of the assembly that did not incur autocatalytic oxidation.

[113] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, pp. 2302-2303.
[114] *Id.*, p. 2299.

It is significant that, regarding local steam starvation conditions postulated to have 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:

> [T]he temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward. It is reasonable to assume, that *the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred* [emphasis added].[115]

It would also be reasonable to assume that, during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

As quoted in PRM-50-93, discussing the extensive oxidation of the assembly of FLECHT run 9573, in its comments regarding PRM-50-76, Westinghouse states:

> Despite 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. ...
>
> Westinghouse notes that the metallurgical analyses performed for FLECHT Run 9573 indicated that the measured oxide thickness was still within the expected range for specimens heated as high as 2500°F.[116]

(When Westinghouse performed the metallurgical analyses for the assemblies from the four FLECHT Zircaloy tests, it compared the measured oxide layer thicknesses to Baker-Just correlation predictions[117]—"the expected range.")

---

[115] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[116] 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, pp. 3-4.

[117] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 17, 21.

And as also quoted in PRM-50-93, in "Denial of Petition for Rulemaking (PRM-50-76)," discussing the metallurgical analyses performed for the Zircaloy FLECHT tests, NRC states:

> The petitioner did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. The petitioner also ignored the Westinghouse application of the Baker-Just correlation to these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients.[118]

So, as stated in PRM-50-93, neither Westinghouse nor NRC applied the Baker-Just correlation to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation; furthermore, NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation. And, as stated above, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments—during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

Therefore, Dr. Jack Roll's conclusion that the metallurgical data from the four Zircaloy PWR FLECHT tests reaffirmed the use of the Baker-Just correlation for evaluating the Zircaloy-steam reaction in the conditions of a loss-of-coolant accident is incorrect.

---

[118] *Id.*, pp. 21-22.

21-13

4. It is Incorrect that the Zircaloy-Steam Reaction is Negligible below 1900°F, as Computer Codes Using the Baker-Just Correlation Predict

In AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[119]

Indeed, computer codes using the Baker-Just correlation may calculate that the Zircaloy-steam reaction is negligible below 1900°F; however, experimental data from multi-rod experiments demonstrates that the Zircaloy-steam reaction is very substantial below 1900°F.

For example, discussing the fact that the Zircaloy-steam reaction was very substantial below 1900°F 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 Tests B and C)],[120] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C [1832°F]. This temperature escalation [several tens of degrees Kelvin per second[121]] 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. An effectively moderated escalation would be observed for smaller initial heatup rates, because the growth of protective scale during steam exposure counteracts by decreasing the oxidation rate of the material.
>
> This explains the observation that the temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward

---

[119] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

[120] 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.

[121] 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. 1.

> and downward. It is reasonable to assume, that the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred.[122]

So "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 that autocatalytic oxidation commenced at 1832°F in the CORA-2 and CORA-3 experiments: peak cladding temperatures started increasing at several tens of degrees Fahrenheit per second.

Furthermore, it is also significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory" ("In-Vessel Phenomena—CORA"), presented in 1991, explicitly states that "[c]ladding oxidation [in the CORA-16 experiment] was not accurately predicted by available correlations."[123] (In 1991, the Baker-Just correlation was among the available correlations.)

Discussing "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[124] a 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.*
>
> ... 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].[125]

---

[122] *Id.*, p. 41.
[123] 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.
[124] 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.
[125] *Id.*

So, in the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations"[126] 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."[127] This indicates that available correlations—including the Baker-Just correlation—are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

Clearly, the Zircaloy-steam reaction is not negligible below 1900°F, as experimental data from multi-rod experiments demonstrates. And ECCS evaluation calculations using the Baker-Just correlation under-predict the Zircaloy-steam reaction that would occur in a LOCA environment.

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by non-conservative ECCS evaluation calculations that used the Baker-Just correlation.

(It is noteworthy that the current power levels at Indian Point Unit 2 were qualified by non-conservative ECCS evaluation calculations that used the Cathcart-Pawel correlation. In 1991, the Cathcart-Pawel correlation was among the available correlations that when used in computer codes failed to predict cladding oxidation in the CORA-16 experiment.)

K. The Atomic Energy Commission Emergency Core Cooling Systems Rulemaking Hearing

> A.E.C. lawyers, at a meeting for A.E.C. staff witnesses a few days before the start of the hearing, also addressed the question of what the staff would say during cross-examination. The A.E.C. witnesses were given a one-page instruction sheet entitled "Hints at Being a Witness." It contained fifteen numbered instructions. ...although they were appearing as expert witnesses and were testifying under oath, item number ten on the list

[126] 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.
[127] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.

> admonished them: "Never disagree with established policy."[128]—Daniel Ford

> Before the hearing Rittenhouse and other Oak Ridge staff members who would testify also spoke with Alvin Weinberg, the director of the Oak Ridge lab. Unlike the managers at Idaho, Weinberg told his researchers to "act responsibly and tell the truth."[129] —Daniel Ford

In this section, Petitioner discusses the Atomic Energy Commission's ("AEC") emergency core cooling system ("ECCS") rulemaking hearing. PRM-50-93 addresses issues that were debated in the ECCS rulemaking hearings: reflood rates, the Full Length Emergency Cooling Heat Transfer ("FLECHT") tests, the metal-water reaction, and what the PCT limit should be in the event of a LOCA. (Of these subjects, PRM-50-95 addresses the metal-water reaction and the PCT limit.)

In this section, Petitioner extensively quotes from "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"[130] the concluding statement of Henry. W. Kendall and Daniel F. Ford of Union of Concerned Scientists ("UCS"), on behalf of Consolidated National Intervenors ("CNI"), in the AEC ECCS rulemaking hearing. "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" provides a concise summary of reactor safety issues, debated in the AEC ECCS rulemaking hearing, including reactor safety issues that have not been resolved since 1973, when the hearing concluded.

Petitioner also quotes from "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," *Meltdown: The Secret Papers of the Atomic Energy Commission*, ORNL Review Vol. 25, Nos. 3 and 4, Chapter 6, "Responding to Social Needs," and *A Distant Light: Scientists and Public Policy.*

---

[128] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, Simon & Schuster, New York, 1986, p. 119.
[129] *Id.*, p. 123.
[130] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," Concluding Statement—Safety Phase—Prepared by Union of Concerned Scientists on Behalf of Consolidated National Intervenors in the Matter of Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants, AEC Docket RM-50-l, April 1973.

1. Some of the Reactor Safety Issues Debated in the AEC ECCS Rulemaking Hearing, have Not been Resolved to this Day, Nearly Forty Years Latter

It is unfortunate that—despite extensive ECCS research, conducted after the rulemaking hearing concluded in 1973—some of the reactor safety issues, debated in the AEC ECCS rulemaking hearing, have not been resolved to this day, nearly forty years latter.

(Discussing an estimate—in 1988 dollars—of the total amount of money spent on ECCS performance research between 1974 and 1988, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> In the years following the rulemaking [issued in January 1974], over $700 [million] has been spent by the NRC on research investigating ECCS performance. It is estimated that a similar amount has been spent by DOE (including AEC and ERDA), the U.S. industry, and foreign researchers, resulting in a total estimated expenditure of over $1.5 billion. The majority of this LOCA research is complete and has greatly improved the understanding of ECCS performance during a LOCA.[131]

Clearly, since 1988, substantial additional amounts of money have been spent on continuing LOCA research. So—in 2010 dollars—billions of dollars have been spent on LOCA research, yet NRC has ignored the data from LOCA research experiments that indicates that some of its regulations are not conservative enough to help ensure public safety.

21-14

For example, "Compendium of ECCS Research for Realistic LOCA Analysis," states that "[a]ssessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature;"[132] and that "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."[133] However, "Compendium of ECCS Research for Realistic LOCA Analysis" does not mention it is

[131] 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. 8-1.
[132] *Id.*, p. 8-2.
[133] *Id.*

reported that in the LOFT LP-FP-2 experiment, autocatalytic oxidation commenced at cladding temperatures of approximately 2060°F[134].

And 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 experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> 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.[135]

2. A Brief Summary of the AEC ECCS Rulemaking Hearing

Regarding the AEC ECCS rulemaking hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The rulemaking hearing on reactor safety began in January 1972 and took place over a period of almost two years, until December 1973. The hearings...generated a record of more than 22,000 pages of transcript [of oral testimony] and thousands of pages of written direct testimony and exhibits.[136]

---

[134] 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, pp. 30, 33.

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

[136] 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. 1086. 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.

Additionally, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The [AEC's] Hearing Board consisted of Nathaniel H. Goodrich, Esq., presiding, Dr. Lawrence H. Quarles, and Dr. John H. Buck[, AEC employees[137]]. ... The primary participants included the Commission Regulatory Staff, four reactor manufactures, a consolidated group of electric utility companies, and the Consolidated National Intervenors ("CNI"), a group of about 60 organizations and individuals [UCS "served as the technical arm of CNI."[138]]. In addition, three states, the Lloyd Harbor Study Group, and several individuals participated to a lesser degree.[139]

Regarding the principal changes to AEC regulations, as a result of the hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The principal changes to the AEC's regulations were to lower the maximum allowed fuel cladding temperature, in the event of a loss-of-coolant accident ("LOCA"), from 2300°F to 2200°F, and to add a 17% local cladding oxidation limit.[140]

(It is noteworthy that "The History of LOCA Embrittlement Criteria" states that "the 17%-ECR[141] and 1204°C [PCT] criteria [of 10 C.F.R. § 50.46(b)] were primarily based on the results of post-quench ductility tests conducted by Hobson."[142] Furthermore, the experimental data that 50.46(b)(1) and (2) are primarily based on, is reported on in "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," ORNL-4758,[143] and "Ductile-Brittle Behavior of Zircaloy Fuel Cladding."[144]

---

[137] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, Forward.
[138] *Id.*
[139] Dixy Lee Ray, *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1086.
[140] *Id.*, pp. 1130-1133.
[141] "ECR" is the initialism for "equivalent cladding reacted."
[142] G. Hache and H. M. Chung, "The History of LOCA Embrittlement Criteria," Proc. 28th Water Reactor Safety Information Meeting, Bethesda, USA, October 23-25, 2000, p. 10.
[143] D. O. Hobson and P. L. Rittenhouse, "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972.
[144] D. O. Hobson, "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, 26 March, 1973.

Additionally, it is noteworthy that "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," ORNL-4758, is currently (November 2010) "non-publicly available" in NRC's ADAMS Documents (Accession Number: ML082410413).[145] So one of the papers (from the early 1970s) that is one of the primary foundations of 50.46(b)(1) and (2) is non-publicly available in NRC's ADAMS Documents.)

And regarding AEC regulations that were not changed, as a result of the hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The other three criteria [of the AEC's regulations were] retained, with some modification of the wording. These three criteria limit the hydrogen generation from metal-water reactions, require maintenance of a coolable core geometry, and provide for long-term cooling of the quenched core.[146]

(It is noteworthy that the AEC's ECCS rulemaking hearing generated a great deal of media attention; for example, on March 12, 1972, *The New York Times* reported: "A.E.C. EXPERTS SHARE DOUBTS OVER REACTOR SAFETY."[147])

Discussing the rulemaking hearing, the Oak Ridge National Laboratory ("ORNL") Review states:

> When protest greeted the AEC's interim criteria for emergency core cooling systems, [AEC Chairman James Schlesinger] convened...quasi-legal hearing[s] for comments from reactor manufactures, electric utility officials, nuclear scientists, environmentalists, and the public.[148]
>
> The hearings pitted the nuclear power industry against the opponents of nuclear power and seriously divided researchers at the AEC and its laboratories. Placed on the witness stand during heated adversarial legal proceedings, some scientists expressed confidence in the interim safety standards, and others did not.

---

[145] This is stated in an e-mail to Petitioner from NRC Public Document Room, October 26, 2010.

[146] Dixy Lee Ray, *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1130.

[147] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, pp. 126, 285 (Notes).

[148] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6.shtml

In a letter to Hans Bethe[149]...ORNL Director Alvin Weinberg,[150] pointed out that emergency cooling systems provided a final defense against [the] melting of [the] fuel in the case of a [LOCA] in the largest light water nuclear reactors. "And it makes me all the more unhappy," Weinberg concluded, "that certain quarters in the AEC have refused to take it seriously until forced by intervenors who are often intent on destroying nuclear energy!"[151]

Now that the AEC and nuclear industry had been called into account on this issue, Weinberg urged [ORNL] staff to offer their expertise fully and without reservation, regardless of whether they agreed with the existing criteria. Schlesinger agreed. Weinberg complained, however, that his staff should have been involved as fully in preparing the criteria as they would be in testifying at the hearings.

Among [ORNL] staff participating in [the] lengthy, sometimes contentious, sometimes tedious hearings were William Cottrell, Philip Rittenhouse, David Hobson, and George Lawson. They and other witnesses were grilled by attorneys for days. More than 20,000 pages of testimony were taken from scientists and engineers, who often expressed sharp dissent on technical matters concerning the adequacy of the safety program. Laboratory experts generally considered that existing criteria for reactor safety were based on inadequate research. ...

The [ORNL's] emphasis on reactor safety and environmental protection made it and Director Weinberg unpopular among some nuclear power advocates and members of the AEC staff—a strange turn of events for Laboratory scientists who had devoted their careers to inventing and advancing practical applications of nuclear energy. ...

Although other events and considerations also played a part, the ECCS hearings of 1972, no doubt influenced major management shifts in 1973 at [ORNL] and [the] AEC. More fundamentally, they influenced the federal government's subsequent decision to dissolve the AEC and to place its regulatory responsibilities and research- and development-related activities into two separate entities.[152]

---

[149] Nobel laureate professor at Cornell University and former director of Los Alamos Scientific Laboratory's Theoretical Division.

[150] Theoretical physicist, Alvin Weinberg, "patented the first design for a water-cooled nuclear reactor;" see Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, p. 25.

[151] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6sb6.htm

[152] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6.shtml

And discussing the rulemaking hearing, in *A Distant Light: Scientists and Public Policy*, Henry W. Kendall states:

> We discovered great vulnerabilities in the emergency systems required in all nuclear power plants. ...
>
> The safety issues we were documenting were quickly raised by intervenors in nuclear power plant construction and licensing hearings at a number of sites in the United States. So that the same safety matters would not be contested in numerous duplicate hearings, we and the AEC agreed that the issues would be pulled out of all local hearings and consolidated in a single [ECCS] rulemaking hearing in Washington. ...
>
> In preparing for the hearing, we very quickly discovered that we had uncovered a hornet's nest. The AEC had engaged in a far more extensive program of suppression of disconcerting safety information than anyone had ever imagined, had censored safety-related information, had pressured their own researchers to keep quiet on key issues, and was sitting on a mass of disquieting research results. In some cases, commission officials made public statements that were contrary to statements they had made in their internal records or reports. ...
>
> In the hearing, [Daniel F.] Ford, who had the instincts and skills of a fine lawyer, although without formal legal training, carried out extensive cross examination both of friendly and of hostile nuclear safety experts and managed, indeed stimulated, a flow of safety documents from whistleblowers in the AEC laboratories that had never been destined to see the light of day. ...
>
> My part was to digest the intricacies of the safety debate, help prepare our technical testimony, and defend it against attack by the 17 lawyers representing the electrical utilities, reactor manufactures, and the AEC who were participants in the hearing as well as having chosen the board who conducted the hearings and sat in judgment. ... I was on the witness stand five days a week for nearly a month, which must be close to a record for this sort of thing. With support from nuclear experts, some known only to us or wholly anonymous, we were able to withstand the numerous attempts to discredit us and our case. Nevertheless, the length and tension involved proved to be very wearing. A sadder consequence of the hearings was that the careers of a number of whistleblowers from the National Laboratories, who were identified during the hearings, were ruined by the AEC.
>
> While the direct result of the ECCS hearings was at best a minor improvement in reactor safety, Ford's work, combined with our written testimony, proved a major embarrassment for nuclear power. The

> testimony he elicited and the safety documents that were released were extraordinarily damaging to the AEC and contributed to the breakup of that agency by the Congress in January 1975.[153]

Discussing dissenting opinions regarding the effectiveness of ECCS and the AEC's and one of its contractor's attempts to intimidate witnesses prior to the rulemaking hearing, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> The testimony that the Hanauer task force[154] prepared for the hearing did not discuss...any of the...internal studies that conflicted with the official optimism about E.C.C.S. performance. It did not mention the fact that two of its members, Morris Rosen[155] and Robert Colmar, had strongly disagreed with its findings. ...
>
> The week before the hearing was to begin, Rosen had been told that the A.E.C. staff was being "reorganized," and he had found himself reorganized out of his job. When he explained to his superiors that this might look suspicious—that they might be accused of stifling dissent—he was given a job in another part of the staff, one where he no longer had any responsibility for E.C.C.S.
>
> The testimony presented by the Hanauer task force...made no reference to the dissenting opinion on the June 1971 policy statement[156] that had been expressed by the [Advisory Committee on Reactor Safeguards]. ...
>
> Shortly before the hearing the management at the Idaho lab [Aerojet Nuclear Company ("Aerojet")] met with staff researchers there and told them that they were free to say whatever they wanted at the hearing. But

---

[153] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer , New York, 2000, pp. 14-15.

[154] The A.E.C.'s task force "was headed by Dr. Stephen Hanauer, who had served on the A.C.R.S. ... The other members of the task force...were Frank Schroeder, Edison Case, Marvin Mann, Victor Stello, Thomas Novak, Norman Lauben, Richard Tedesco, Warren Minners, Denwood Ross, Howard Richings, Paul Norian, Morris Rosen, and Robert Colmar. All of the members of the task force were engineers. Some of them had limited acquaintance with E.C.C.S. problems, but none was recognized as an expert in the field;" Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, pp. 102-103.

[155] "Rosen was noted by Dr. Hanauer as being far more knowledgeable in most areas of ECCS than Hanauer, who headed the Regulatory Staff ECCS task force; Milton Shaw, likewise, referred to Rosen and noted Rosen's synoptic understanding of the ECCS issue;" see Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.31.

[156] The AEC's interim ECCS acceptance criteria for LWRs: "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors—Interim Policy Statement," U.S. Federal Register, Vol. 36, No. 125, June 29, 1971 and No. 244, December 18, 1971.

> the management could not assure them that they would still have a job after the hearing if their testimony displeased the A.E.C.[157]

(It is noteworthy that regarding Oak Ridge staff researchers who would testify, *Meltdown: The Secret Papers of the Atomic Energy Commission* states:

> Before the hearing Rittenhouse and other Oak Ridge staff members who would testify also spoke with Alvin Weinberg, the director of the Oak Ridge lab. Unlike the managers at Idaho, Weinberg told his researchers to "act responsibly and tell the truth."[158])

And discussing the fact that AEC had attempted to intimidate ORNL witnesses prior to the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Despite the strenuous efforts on the part of the Commission to prevent [ORNL] witnesses from presenting their concern regarding ECCS effectiveness and the technical validity of the Interim Acceptance Criteria, and despite efforts at censorship and the suppression of data quite analogous to Commission efforts with respect to Aerojet views, cross-examination of ORNL witnesses by CNI was able to obtain valuable testimony that served to stimulate a number of major subsequent developments in the hearing. One of these developments includes, of course, the later revision in the Regulatory Staff's analysis of its embrittlement criterion.[159]

Discussing the testimony of Philip L. Rittenhouse of ORNL in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [A] safety researcher from Oak Ridge appearing in the hearing was P. L. Rittenhouse, a man who was willing to speak candidly about his own feelings regarding such important ECCS issues as embrittlement and flow blockage but also to share with the public, as an insider, his knowledge concerning the extensive reservations among the AEC's safety researchers regarding ECCS effectiveness. ...
>
> Rittenhouse testified that the AEC Regulatory Staff had presented what were, in his judgment, "arbitrary" and "unreasonable" interpretations of available PWR FLECHT blockage plate experiments and BWR FLECHT test ZR-2, which involved simulation of flow blockage as a result of

---

[157] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, pp. 117, 122.
[158] *Id.*, p. 123.
[159] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.23.

internal pressurization. Rittenhouse also provided extensive criticisms of the grievous misspecification of the embrittlement criterion in the Interim Policy Statement. Rittenhouse was able to provide important insight into the manner in which the AEC Regulatory Staff had avoided the substantive issues associated with flow blockage, had misunderstood the work that he had done, had merely "guessed" what the magnitude of flow blockage might be in a reactor, and had followed a procedure of making extrapolations that he did not believe constituted "good engineering practice." ...

(It is an insight into the preparation of the Interim Policy Statement to note that Rittenhouse, who is identified by the Regulatory Staff as an expert on flow blockage and on embrittlement, the head of AEC research in those areas, was *never* consulted by the Regulatory Staff in connection with the preparation of the Interim Acceptance Criteria [emphasis not added].) ...

[Rittenhouse] spoke with a clarity and candor that established an important precedent for all of the AEC safety research [personnel who] would [testify after him]. He made clear in his presentation, in the face of substantial institutional pressures exerted through written instructions to AEC witnesses to "never disagree with established policy,"[160] that this proceeding was the opportunity for those men to come forward with a full and honest technical evaluation of [ECCS] problems.[161]

(It is noteworthy that regarding written instructions to AEC witnesses, *Meltdown: The Secret Papers of the Atomic Energy Commission* states:

A.E.C. lawyers, at a meeting for A.E.C. staff witnesses a few days before the start of the hearing, also addressed the question of what the staff would say during cross-examination. The A.E.C. witnesses were given a one-page instruction sheet entitled "Hints at Being a Witness." It contained fifteen numbered instructions. ...although they were appearing as expert witnesses and were testifying under oath, item number ten on the list admonished them: "Never disagree with established policy."[162])

And *Meltdown: The Secret Papers of the Atomic Energy Commission* states that at the end of his third day on the witness stand, Rittenhouse testified:

I have worked in the fuel rod failure program [studying] the questions of fuel cladding and swelling, subsequent blockage, and the possible effects of this blockage on cooling effectiveness... As far as these points, in which I am an expert, there is not the information available to objectively

[160] Exhibit 1013
[161] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 4.25, 4.26, 4.28.
[162] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, p. 119.

> confirm, by scientific or technical procedures, what exactly these materials-related phenomena… what effect they may have on the E.C.C.S. in the course of a loss-of-coolant accident.
>
> Beyond that, I can only say that I have talked to a number of [A.E.C. experts], people who work in the area of E.C.C.S., both the materials people, people who work primarily with [computer] codes, people who are experts, if you will, in heat transfer, fluid flow. And I get the genuine feeling from all of these people that they believe there are things we just do not know well enough yet. … They have too many reservations—I believe shared too generally—for me to pass off. These reservations are primarily that certain phenomena, portions of the loss-of-coolant accident—maybe they're not quite sure what's going on. …
>
> Certainly many of the things that we toss around in computer codes and use to predict maximum temperatures or to predict the course of the loss-of-coolant accident…have not been verified experimentally.[163]

"An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," states that Rittenhouse also testified:

> that he believed that there was a consensus that what might occur during a major LOCA is still open to question, and, [after he was asked for the names of colleagues who had expressed doubts about ECCS effectiveness,[164] he] read into the record the names of 28 [colleagues who] had influenced his own views concerning the serious unresolved problems [of ECCS].[165]

The colleagues Rittenhouse read into the record were George Brockett, Morris Rosen, Robert Colmar, George Lawson, Lawrence Ybarrando, Roger Griebe, Rex Shumway, and other nuclear power safety experts.[166] In the following weeks they also testified regarding "their own conclusions about the defects in the A.E.C.'s approval of current emergency-cooling-system designs.[167]

---

[163] *Id.*, pp. 125-126.
[164] *Id.*, pp. 126-127.
[165] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.27.
[166] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, p. 127.
[167] *Id.*

Discussing the testimony of C. George Lawson of ORNL in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> One of the engineers from the Oak Ridge National Laboratory who testified was C. George Lawson, an expert in heat transfer. ...
>
> In response to CNI's question, Lawson affirmed: "There exists at this time [March 19, 1972] such a limited amount of information of the behavior during a loss-of-coolant accident of Zircaloy-clad fuel rods internally pressurized with fission gas that a conclusion of the adequacy of these emergency [core cooling] systems would be speculative."
>
> Lawson rejected "subjective" assessments of ECCS capabilities and stressed the need for experimental demonstrations of system performance. In the light of what Lawson regarded as "objective scientific confirmation" he said that such confirmation of ECCS effectiveness is not available and that ECCS effectiveness, he concludes, is "undemonstrated."
>
> At one point in his testimony Lawson was led to speculate on whether reactors were "safe," but his testimony clearly affirmed that if standards of "reasonable assurance" of ECCS effectiveness were applied then it must be concluded that such "reasonable assurance" does not exist with regard to presently designed emergency core cooling systems and that if one wished to rely on the Interim Acceptance Criteria they must be experimentally verified.[168]

Discussing a letter Alvin Weinberg, Director of Oak Ridge, wrote to the AEC Chairman at the beginning of the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> The concerns expressed by the witnesses from Oak Ridge National Laboratories who appeared in the proceeding were also communicated to the AEC by the respected Director of Oak Ridge, Alvin Weinberg. He affirmed the fundamental doubt of his researchers concerning the lack of adequate experimental proof of ECCS effectiveness in a February 9, 1972 letter to AEC Chairman James Schlesinger.
>
> [Weinberg's letter states:] "...I have a basic distrust of very elaborate calculations of complex situations, especially where the calculations have not been checked by full-scale experiments. As you know, much of our trust in the ECCS depends on the reliability of complex codes. It seems to me—when the consequences of failure are serious—then the ability of the

---

[168] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, pp. 4.23, 4.24-4.25.

> codes to arrive at a conservative prediction must be verified in experiments of complexity and scale approaching those of the system being calculated. I therefore believe that serious consideration should be given first to cross-checking different codes and then to verifying ECCS computations by experiments on large scale and, if necessary, on full scale. This is expensive, but there is precedent for such experimentation—for example, in the full-scale tests on COMET and on nuclear weapons."[169]

Discussing the testimony of Dr. Morris Rosen of the AEC, regarding the ECCS expertise and statements of the employees of Aerojet, in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Dr. Rosen emphasized that the AEC Regulatory Staff does not, in-house, have sufficient technical expertise to do a professional evaluation of the vendor LOCA analysis models and that only with [Aerojet] involvement could a state of the art assessment of the evaluation models be performed.
>
> Dr. Rosen, who supervised [Aerojet] Technical Assistance work for the Regulatory Staff, stated that [Aerojet]: "represents the most significant source of information to the Atomic Energy Commission and the regulatory organization in the field of emergency core cooling.
>
> "I think that as a result of the fact of this hearing, the testimony presented at this hearing, there is an impressive array of the top talent in that organization in my opinion indicating some strong reservations as to the course of the evaluation of emergency core cooling with respect to the large cold leg break.
>
> "My opinion of, let's say, the testimony of George Brockett is, I believe—I don't know his exact title but I think it is manager of development, nuclear safety development at Aerojet, I think he came out strongly indicating that steam binding indeed was a problem.
>
> "I think he indicated perhaps that reductions in operating power levels were required.
>
> "Personal observation about Mr. Brockett: I think in my opinion one would classify him as perhaps one of the leading experts in this country in emergency core cooling, in my opinion, if not the leading expert.
>
> "I think when that man comes out and says there is a problem, I take note of it."[170]

---

[169] *Id.*, pp. 4.28-4.29.
[170] *Id.*, pp. 4.7-4.8.

And discussing the testimony of witnesses from Aerojet collectively in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] has stated that its "fundamental concern" about ECCS is based on the "lack of fundamental data." Witnesses from that laboratory affirmed that reactor designs have gotten ahead of the Commission's understanding of reactor safety problems, because reactor safety research has not kept abreast with the requirements for it in the expanding U.S. nuclear power program.[171]

Discussing the biased manner in which the rulemaking hearing was conducted and its conclusion, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> The industry and A.E.C. staff filed testimony to rebut what Rittenhouse, Brockett, Rosen, Colmar, and the other scientists from the A.E.C. laboratories had said on the witness stand. All of the concerns expressed by these scientists, the Hanauer task force asserted, had been resolved. (The A.E.C. hearing board refused to allow further questioning of these A.E.C. experts, however, to determine whether they agreed with the resolution of their concerns.) At the conclusion of the hearing, the A.E.C. staff, led by Stephen Hanauer, recommended that the A.E.C. reaffirm its June 1971 approval of the existing E.C.C.S. designs—and the Commission did so. Its "final" policy statement on E.C.C.S. left its June 1971 "interim" policy statement essentially intact. ...
>
> While the hearing was still in progress...Edison Case, the Deputy Director of Licensing, told *The New York Times*, in response to an inquiry, that "no costly changes" would be imposed on the industry as a result of the hearings.[172] ...
>
> Case's *faux pas* in disclosing the A.E.C.'s intransigence on E.C.C.S. was the subject of a sardonic private note that Edward J. Bauser, the staff director of the Joint Committee on Atomic Energy, sent to A.E.C. Director of Regulation [L. Manning] Muntzing. "The recent article in *The New York Times* which is enclosed would, in my view, be very disturbing to those who still have faith in the integrity of the administrative process," he wrote. "It could also be useful to anyone who wishes to discredit the integrity of the A.E.C. licensing process."[173]

---

[171] *Id.*, p. 4.8.
[172] *The New York Times*, July 16, 1972.
[173] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, pp. 127, 128.

And discussing personnel changes that occurred after the rulemaking hearing concluded, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> Philip Rittenhouse was removed as head of the fuel-rod-failure program at Oak Ridge under orders from Herbert Kouts, who was the senior A.E.C. official in charge of safety research. Alvin Weinberg...was replaced as director of the [Oak Ridge] laboratory. Rosen and Colmar had already been reassigned prior to the hearing, but Rosen decided subsequently that his chances for advancement were better outside the agency, and he left it. At the Idaho lab, senior personnel who had criticized the "established policy" found themselves, as one of them noted, switched from responsible positions to "nothing jobs." Some of them, like George Brockett, [looked outside of Aerojet for new employment].[174]

(It is noteworthy that before the AEC ECCS rulemaking hearing began, ORNL work on fuel rod failure was scheduled to be terminated at the end of 1971. Regarding this issue, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," states that "[t]he transient test program [for Zircaloy-clad fuel rod clusters in the TREAT facility] is presently inactive because funding is not available"[175] and that "[s]upport of ORNL work on fuel rod failure is now scheduled to be terminated at the end of FY-71."[176]

So the experimental program that conducted "the most realistic loss-of-coolant accident conditions of any experiment to date"[177]—up to 1971—was not provided with funding so investigators could continue researching important safety issues.)

---

174 *Id.*, pp. 128-129.

175 W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.

176 *Id.*, p. ix.

177 Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43.

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,

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

Dated: November 23, 2010

Appendix A Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1[1]

[1] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, p. 14.

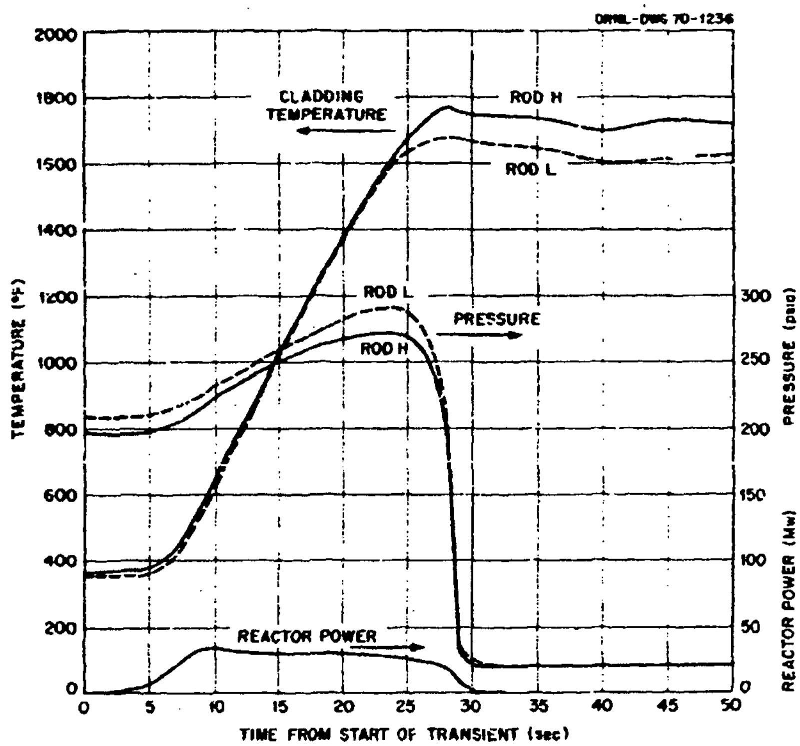


Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1.

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Tue, 23 Nov 2010
23:17:54 -0500
X-Ironport-ID: mail2
X-SBRS: 3.9
X-MID: 30259111
X-fn: Comments November 2010.pdf
X-IronPort-AV: E=Sophos;i="4.59,246,1288584000";
d="pdf'?scan'208";a="30259111"
Received: from mail-wy0-f169.google.com ([74.125.82.169]) by mail2.nrc.gov
with ESMTP; 23 Nov 2010 23:17:51 -0500
Received: by wyf23 with SMTP id 23so9097612wyf.14 for <multiple
recipients>; Tue, 23 Nov 2010 20:17:50 -0800 (PST)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:received:date:message-id
:subject:from:to:cc:content-type;
bh=iBsTUHHOxPtxGTvdX91UftfKAyk4o03SStAh3u3R0+w=;

b=aly8hCZ0MAWX+9NA1Nlwt3oNWH1tr2tBOuJRfh7W5yRGA7Ug51xGGLK6sWQOcvny5g
J6/+j5Nm0d76FYyofhD+NCkiBMXx7Z0Jl4UUZf5rDmd14xtcrhtg0i/FkZJUMZ1UiTWp
K34Gcl/Rup95oE/c+CN45it0im9L7qCVMHI0M=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:cc:content-type;
b=VvotJGKlzELmZQt+uTN8lMs3NWroVWaxvrSj/AqclaJlkAJPK4rpOLJ9m82imR+JzY

MqL+NLZhsHatu8BMFtjNtLNYxn5QpgH1vnttFYWFd8RnGeH1MKb5Fn7mw7kEZspXCCiv
YUpU2WGXBkmGJR7RvCOz4GcnKPgW3jialb1gA=
MIME-Version: 1.0
Received: by 10.227.170.78 with SMTP id c14mr8120735wbz.49.1290572268578; Tue,
23 Nov 2010 20:17:48 -0800 (PST)
Received: by 10.227.72.208 with HTTP; Tue, 23 Nov 2010 20:17:48 -0800 (PST)
Date: Tue, 23 Nov 2010 23:17:48 -0500
Message-ID: <AANLkTi=7Ta0CN9FaAP2YmSB2VBWKk8wghyv_F99BiG1M@mail.gmail.com>
Subject: NRC-2009-0554 (First)
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>, PDR Resource
<pdr.resource@nrc.gov>
CC: Dave Lochbaum <dlochbaum@ucsusa.org>, necnp@necnp.org,
Raymond Shadis <shadis@prexar.com>, "Powers, Dana A" <dapower@sandia.gov>
Content-Type: multipart/mixed; boundary="90e6ba476541a029fa0495c4c5da"
Return-Path: markleyse@gmail.com

Submission ID 22
Mark Leyse
ML103340248

Rulemaking Comments

PRM-50-95
(75FR66007)

2

From: Mark Leyse [markleyse@gmail.com]
Sent: Tuesday, November 23, 2010 11:38 PM
To: Rulemaking Comments; PDR Resource
Cc: Dave Lochbaum; necnp@necnp.org; Raymond Shadis; Powers, Dana A
Subject: NRC-2009-0554 (Second)
Attachments: Comments November 2010 II.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is my second response, dated November 24, 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.

Sincerely,

Mark Leyse

DOCKETED
USNRC

November 24, 2010 (9:15am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Template=SECY-067

DS 10

November 24, 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

TABLE OF CONTENTS

November 24, 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

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-

---

[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.

water reaction considered in ECCS evaluation 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 ("LBPCT") 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.

---

[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 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. Supplementary Information to PRM-50-93 and PRM-50-95

A. Presentation of Robert Leyse and Mark Leyse in Advisory Committee on Reactor Safeguards, Thermal Hydraulic Phenomena Subcommittee Meeting, October 18, 2010

A presentation that Robert Leyse and Mark Leyse gave in Advisory Committee on Reactor Safeguards ("ACRS"), Thermal Hydraulic Phenomena Subcommittee Meeting, on October 18, 2010, helps summarize some of the safety issues raised in PRM-50-93 and PRM-50-95.

The ACRS presentation is quoted below (with changes to some of the punctuation recorded in the transcript and changes to a few words that were improperly recorded):

Mark Leyse: First, I want to thank ACRS for the 10-minute time slot. Ten minutes is not a lot of time, but Bob Leyse and I will summarize some important safety issues. Bob Leyse began working in the nuclear industry in 1950 and worked in nuclear safety at GE, Westinghouse, and EPRI. I am Mark Leyse, author of PRM-50-84, a petition accepted for consideration in NRC's rulemaking process for revisions to 50.46(b) and Appendix K to Part 50. I also wrote PRM-50-93.

PRM-50-93 is the subject of a user need request, dated April 26th, 2010, from Eric Leeds, Director, Office of Nuclear Reactor Regulations, to Brian Sheron, Director, Office of Nuclear Regulatory Research.

NRR's user need request states that, I cite extensive data from numerous multi-rod experiments and that their request is a high priority with a target due date of September 30, 2010.

In PRM-50-93, I argue that NRC's peak cladding temperature limit should be based on data from multi-rod Zircaloy severe fuel damage experiments, because such data demonstrates that the 2200-Fahrenheit limit is non-conservative. I also argue that the Baker-Just and Cathcart-Pawel equations are non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.[7] And I ask that a minimum reflood rate be specified.

[7] Petitioner should have phrased this sentence as, "I also argue that the Baker-Just and Cathcart-Pawel equations are both non-conservative for use in analyses that predict the metal-water reaction rates that would occur in the event of a LOCA.

The page you have lists some of the multi-rod Zircaloy severe fuel damage experiments in which runaway oxidation commenced between 1832 and 2200 degrees Fahrenheit. It is reported that in the LOFT LP-FP-2 experiment—heated with actual decay heat—that runaway oxidation commenced at about 2060 degrees Fahrenheit.

In the Karlsruhe CORA program, there were about 20 experiments. A Karlsruhe paper 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; that is, on bundle insulation. With a good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200 degrees Celsius."

And the page you have has a quote on single-rod quench experiments at Karlsruhe in which there were no temperature excursions during quenching, due to high radiative[8] heat losses.

Now Bob Leyse will discuss the Baker-Just equation.

Bob Leyse: I am Bob Leyse, author of denied PRM-50-76.

The licensing of ECCS in[9] many LWRs under Appendix K specifies Baker-Just. For emphasis, I repeat the licensing of ECCS in[10] many LWRs under Appendix K specifies Baker-Just.

In its technical analysis of PRM-50-76, the NRC fiercely defends Baker-Just. I quote, "The Baker-Just correlation using the current range of parameter inputs is conservative and adequate to assess Appendix K ECCS performance. Virtually every dataset published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800 Fahrenheit." End of quote.

That is an interesting observation in light of data from Zircaloy multi-rod assembles that Mark Leyse has just cited. It is also revealing because the NRC did not even have access to the two key references in the Baker-Just report until April 2010. In response to my persistent demands, NRC acquired the documents and they were placed in ADAMS during April 2010. Short rods of half-inch-diameter Zircaloy 2 were induction heated underwater in Case 1, year 1954, and in steam in Case 2, year 1957.

[8] The word "radiative" was transcribed as "radioactive" in the transcript, p. 187, line 21.
[9] The word "in" was transcribed as "and" in the transcript, on p. 188, line 1.
[10] The word "in" was transcribed as "and" in the transcript, on p. 188, line 3.

Shifting to[11] pages 7, 29, and 31 of the Commissioners' denial of PRM-50-76, I quote, "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." End [of] quote.

Contrary to the Commissioners' observation, it is not sound science to combine the testing of single short rods of zirconium alloy with the testing of multi-rod stainless or Inconel assemblies in order to ascertain the performance of the emergency core cooling systems having thousands of zirconium alloy full-length rods.

Mark Leyse: An Oak Ridge National Laboratory paper discussing the CORA-16 experiment states, "The predicted and observed cladding thermal response are in excellent agreement until application of available Zircaloy oxidation kinetics models causes the low temperature 900 to 1200 degrees Celsius oxidation to be underpredicted."

And another ORNL paper states that, for the CORA-16 experiment, "cladding oxidation was not accurately predicted by available correlations." These papers are from the early 1990s, so the Baker-Just and Cathcart-Pawel equations were among the available Zircaloy oxidation kinetics models that under-predicted oxidation in the 1650-degree to 2200-degree Fahrenheit range.

Severe fuel damage experiments also show that eutectic reactions between fuel assembly components can commence below or at about 2200 degrees Fahrenheit; for example, the chemical reaction between Inconel spacer grids and Zircaloy fuel rods.

In its denial of PRM-50-76, in 2005, the NRC stated that more than 50 Zircaloy tests were conducted at the NRU reactor at Chalk River to evaluate the thermal hydraulic and mechanical deformation behavior of full-length bundles during a large-break LOCA, and that NRC is reviewing the data from that program to determine its value for assessing the current generation of codes such as TRACE. That was from 2005.

But almost all the Zircaloy heat-transfer tests conducted [at] Chalk River had peak cladding temperatures below 2000 degrees Fahrenheit. One test PCT was 2040 degrees Fahrenheit.

Except for the tests conducted at Chalk River, perhaps all [of] the main PWR and BWR heat-transfer experiments (after the original [FLECHT] tests) were conducted with

[11] The word "to" was transcribed as "from" in the transcript, on page 188, line 24.

stainless steel and Inconel 600 fuel rod simulators. Trying to relate this to what would occur in a LOCA in a reactor core with Zircaloy bundles simply does not work.

The NRC needs to conduct realistic heat-transfer experiments with multi-rod Zircaloy bundles in which the bundles would be heated up to at least 2200 degrees Fahrenheit.

The licensing basis PCTs of many plants do not provide necessary margins of safety. For example, the licensing basis PCT of Indian Point Unit 2 is 1937 degrees Fahrenheit, and Oyster Creek's is set at 2150 degrees Fahrenheit. Clearly, NRC's 2200-degree Fahrenheit PCT limit needs to be substantially lowered.

Thank you.

B. BWR Thermal Hydraulic Experiments and Core Spray Cooling

> There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the [General Electric] determination to prove out their ECC systems. ... Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ...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.[12]—J. W. McConnell

22-1

It seems that after the BWR-FLECHT program was concluded about forty years ago that there have not been any BWR heat transfer experiments conducted with parameters realistic enough to conclusively demonstrate that BWR core spray systems would be effective, in the event of a LOCA. Perhaps all of the primary BWR heat transfer experiments conducted after the BWR-FLECHT program was concluded were conducted with multi-rod Inconel 600 bundles.

So it seems that it has also never been conclusively demonstrated that BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if maximum cladding temperatures reached between 1832°F and 2200°F. This is highly problematic, because,

[12] J. W. McConnell, Aerojet internal memoranda; see Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, Union of Concerned Scientists, 1974, p. 5.11.

if a multi-rod Zircaloy bundle were heated up to maximum temperatures between 1832°F and 2150°F, it would (with high probability) incur autocatalytic oxidation. In the event of a LOCA, if autocatalytic oxidation occurred at a LWR, it would lead to a partial or complete meltdown.

Furthermore, to overcome the impression left from the BWR FLECHT program, BWR heat transfer experiments need to be conducted with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Such BWR heat transfer experiments need to be conducted in experiments modeling BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs.

(It is noteworthy that there should be a regulation stipulating minimum allowable amounts of coolant to be supplied to each fuel bundle in the BWR core, in the event of a LOCA.[13])

1. Appendix K BWR Heat Transfer Coefficients

Appendix K to Part 50, ECCS Evaluation Models, I(D)(6), *Post-Blowdown Phenomena, Heat Removal by the ECCS, Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling*, states:

> Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:
>
> a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.
>
> b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot °F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

[13] "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."

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot {}^{\circ}F^{-1}$ shall be applied to all fuel rods.

It is significant that Appendix K convective heat transfer coefficients for BWR Zircaloy fuel rods under spray cooling are based on data from the BWR Full Length Emergency Cooling Heat Transfer ("FLECHT") tests—from tests conducted with stainless steel electrically heated fuel rod simulators.

Regarding the fact that Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are based on BWR FLECHT tests conducted with stainless steel electrically heated fuel rod simulators, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> *From the BWR FLECHT tests there is information on the heat transfer coefficients for both the convective heat flow to the water droplets and steam and for the reflood phase.* The FLECHT tests were made with an electrically heated mock-up of a 7 x 7 rod array complete with its channel box. *The convective heat transfer coefficients were determined from the residue of a thermal balance after all of the known inputs and outputs were calculated.* The factors considered were the electrical heat input, the rate of change of the heat content of the rods as calculated from their temperature history, and the calculated radiation from the rods to each other and to the channel walls. The residue from these inputs and outputs was ascribed to convective heat transfer. The convective heat transfer coefficients so determined could not be very accurate because their calculation involved taking the difference between two large numbers. The coefficients so obtained are small and are about what one would expect from the mechanisms of natural convection and radiation to steam (Exhibit 1113, p. 16-14).
>
> *The values of the calculated convective heat transfer coefficients depend to some extent upon the value used for the thermal emissivity of the stainless steel, since the convective heat transfer is obtained after subtracting the radiative heat transfer from the total.* Theoretically a high value of the emissivity leads to a low calculated convective heat transfer coefficient. Values of the emissivity measured after the tests ranged from 0.6 to 0.9 (Exhibit 461, p. 81 and Exhibit 1113, p. 16-14), and to add conservatism to the calculation, the Interim Policy Statement required the use of the highest measured emissivity, 0.9, for the calculation of the convective heat transfer coefficients. However, it turned out that this resulted in a higher coefficient (less conservative) for the critical inner rods, with a higher estimated standard error (Exhibit 461, Table 2). After reviewing the derivation of the coefficients as given in Exhibit 461, we

believe that those originally listed as best estimates by General Electric are the most credible and should be used. The effect of this change on the peak cladding temperature will be small, about five degrees according to Exhibit 461.

There has been a great deal of criticism of the BWR FLECHT tests, particularly by the Consolidated National Intervenors (Exhibit 1041, Chapter 5), and both General Electric and the Regulatory Staff have defended them (Closing Statements). However, for the purpose of calculating the maximum cladding temperature, only the derived heat transfer coefficients are of any great importance. The values obtained have always been known to have a high statistical error; furthermore, the values are low and reasonable, and there seems little to be gained by renewing the controversy over the manner of conducting and interpreting all features of the tests.

The high but inevitable statistical error of the coefficients for the inner rods (1.5 ± 1.0 BTU/hr·ft^2·°F) is bothersome and leads to an estimated error band of as much as ±200°F in the calculated peak temperature in some circumstances (Exhibit 1113, p. 16-36). *The test bundle SS2N was used to derive the heat transfer coefficients*; another test bundle, SS4N, resulted in cladding temperatures 200°F higher than those of the bundle used as a standard; one half of this discrepancy could be explained by test differences, with the other half left to be attributed to statistical variations (Exhibit 1113, p. 16-38). The problem of these large statistical errors in the convective heat transfer coefficients is compensated to some extent by the fact that the coefficients were determined at atmospheric pressure, whereas the reactor would be at some elevated pressure at which the heat transfer would be improved (Exhibit 1113, p. 16-26).

The evidence for the value 25 BTU/hr·ft^2·°F of the two phase reflooding heat transfer coefficient is sketchy (Exhibit 1032, p. II 6.3-51), but it is applied for only a short time because the high reflood rate would quickly quench the core, and the exact value is of little significance [emphasis added].[14]

So "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[t]he

[14] 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. 1125-1126. 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.

[BWR FLECHT] test bundle SS2N was used to derive the [Appendix K] heat transfer coefficients"[15] for BWR Zircaloy fuel rods.

(In the name "SS2N," "SS" stands for "stainless steel" and "N" stands for "Nichrome.")

And also regarding Appendix K heat transfer coefficients for BWRs, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "the heat transfer coefficients utilized in the GE core spray and reflood calculation model,[16] were derived on the basis of the SS2N test series."[17, 18]

In the BWR FLECHT SS2N test series, conducted from August to October 1969, three steady state tests were conducted with a peak power of 150 kW and coolant rates of 1.0-2.45 gallons/min.; 24 transient tests were conducted with peak powers of 100-250 kW, coolant rates of 2.45-5.0 gallons/min., and initial temperatures of 865-1850°F; eight combined spray and flooding tests were conducted with peak powers of 235-250 kW, coolant rates of 2.0-3.5 gallons/min. and 2.0-6.0 in./sec., and initial temperatures of 1335-1870°F.[19]

In the BWR FLECHT tests, five tests were conducted with Zircaloy electrically heated fuel rod simulators; however, Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are not based on the data from the five Zircaloy tests.

Explaining the purpose of the five BWR FLECHT Zircaloy tests "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> [I]t 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

[15] *Id.*, p. 1126.
[16] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, p. 58.
[17] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.
[18] 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, p. A8-10.
[19] *Id.*, p. A8-5.

> transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.[20]

Discussing the PWR FLECHT tests, "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.[21]

It is significant that the Atomic Energy Commission, also concluded that heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods in BWRs.

Regarding the problems with the heat transfer coefficients derived from the SS2N experiments with stainless steel fuel rod simulators, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> It seems probable that the difference between test and theory results from rigid adherence by GE to a time-dependant model of heat transfer coefficients which were derived from their SS2N tests and adopted as their "design model."[22] The design analysis method, based on the SS2N time history, apparently did not permit accommodation of the idiosyncrasies of the Zr2K test experience with its rod heater failures and [thermocouple] equipment malfunctions. Consequently, the predicted results might not reasonably be expected to correspond well with the reality of the Zr2K test. Whether or not design basis production of LOCA thermal histories would agree well with an actual transient also remains to be shown. Results imply that the GE thermal analysis method may be a weak predictive tool and more effort appears to be needed in model development. However, it does appear that with sufficient analysis, FLECHT results would be adequate to form a basis for demonstrating the development of conservative analytical design methods.[23]

---

[20] *Id.*, p. A8-7.

[21] 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."

[22] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.

[23] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," EQL Report No. 9, pp. A8-27, A8-28.

2. Appendix K BWR Heat Transfer Coefficients for New BWR Fuel Assembly Designs

It is significant that Appendix K specifies that its BWR heat transfer coefficients are to be used for fuel rods in a 7 x 7 fuel assembly array. Since Appendix K was written, new BWR fuel assembly designs have come into use, so Appendix K BWR heat transfer coefficients have been converted so that they can also apply to new BWR fuel assembly designs.

Discussing the application of heat transfer coefficients to various BWR fuel assembly designs, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" states:

> Although the channel size has not changed significantly since the 1970's, the BWR fuel assembly designs have changed in many ways. These changes have resulted in a larger number of smaller diameter fuel rods as well as various non-boiling water channel designs. ... Spray heat transfer tests have been performed (*e.g.*, the BWR FLECHT test program) from which convective spray heat transfer coefficients have been derived. CENPD-283-P-A...provides a summary of these tests *and describes how the spray cooling heat transfer coefficients are applied to various fuel geometries*. ... The BWR FLECHT tests, which simulated a 7x7 array, showed that the convective coefficients are dependant on the location of the fuel rod relative to its proximity to the channel enclosure (corner rod, outer row rod, or interior rod). Table 6-2 lists the heat transfer coefficients that are acceptable for use in an Appendix K analysis of 7x7 fuel [emphasis added].[24]

(Table 6-2, Appendix K Spray Cooling Heat Transfer Coefficients, states that the values for heat transfer coefficients are: for corner rods—17.0 $W/m^2{\cdot}K$, for side rods—19.9 $W/m^2{\cdot}K$, for inner rods—8.5 $W/m^2{\cdot}K$, and for channel—28.4 $W/m^2{\cdot}K$.[25])

It is significant that BWR FLECHT spray heat transfer coefficients for 7x7 fuel assembly arrays have been converted so that they can be used for 8x8 fuel assembly arrays.[26] It certainly stands to reason that BWR FLECHT spray heat transfer coefficients

[24] John A. Blaisdell, Westinghouse, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-NP-A, November 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050390435, p. 30.
[25] *Id.*, p. 31.
[26] *Id.*

for 7x7 fuel assembly arrays have also been converted so that they can be used for 9x9 and 10x10 fuel assembly arrays.

3. Criticisms of the BWR FLECHT Tests

Discussing one of Henry Kendall and Daniel Ford's, of Consolidated National Intervenors ("CNI"),[27] criticisms of the BWR-FLECHT tests, "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].[28]

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].[29]

(It is noteworthy 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

---

[27] 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.
[28] 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.
[29] *Id.*, p. A8-6.

the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[30] 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].[31])

And regarding Aerojet internal memoranda that provide commentary on the BWR-FLECHT program consistent with that presented by CNI, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] internal memoranda provide commentary on the BWR-FLECHT program quite consistent with that presented by CNI. Thus, for example, 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]."[32]

So J. W. McConnell concluded that "the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven."[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

---

[30] 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.
[31] *Id.*, p. 4.4.
[32] 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.
[33] *Id.*

> 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.
>
> 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]

(It is significant that in the ECCS rulemaking hearing, the Atomic Energy Commission ("AEC") Commissioners did not seem concerned about decoupling the zirconium-water reaction from cladding heat transfer mechanisms. The AEC Commissioners merely concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients. Regarding the AEC Commissioners' conclusion, "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*

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

coefficients, which depend on conditions in the coolant outside the rod [emphasis added].[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 fuel rods. (Petitioner discusses the fallacy of the AEC Commissioners' conclusion in the following section.))

It is 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"[36] in the BWR-FLECHT program.

4. The Fallacy of the AEC Commissioners' Conclusion that the Heat Generated from the Exothermic Zirconium-Water Reaction would Not Affect the Coolant Outside Fuel Rods

To discuss the fallacy of the AEC Commissioners' conclusion that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside fuel rods, 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. (The BWR FLECHT Zr2K test is discussed in the following section.)

---

[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] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, p. 5.11.

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."[37]

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 Report"] when compared to the equivalent stainless steel tests."[38] 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."[39]

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:

> The second 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*

---

[37] 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 Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[38] 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.

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

> *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.
>
> *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].[40]

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.*[41]

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.

---

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

[41] 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.

It is significant that within the first 18.2 seconds of FLECHT run 9573,[42] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[43] *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 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 heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[44]

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.[45]

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."[46]

---

[42] 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.
[43] *Id.*, p. 3-98.
[44] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.
[45] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.
[46] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at:

Regarding steam temperatures measured by the seven-foot steam probe, "PWR FLECHT 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).[47]

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 5...thermocouples"[48]—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.

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,"[49] 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"[50] is erroneous. 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.

---

www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

[47] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.

[48] *Id.*, p. 3-98.

[49] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.

[50] 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."

5. More Criticisms of the BWR FLECHT Tests

Regarding the BWR-FLECHT Program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"—section V.G.2., CNI Guidance for Commission on Information Needs for LOCA Analysis, FLECHT Program, BWR-FLECHT—states:

> The BWR-FLECHT Program was carried out by the General Electric Company (GE) under a subcontract of the Idaho Nuclear Corporation, itself a contractor to the test program sponsor, the AEC. Roger Griebe of [Aerojet] coordinated the program, as project engineer, for [Aerojet], GE and the AEC. The UCS devoted a very substantial effort to an independent analysis of the BWR-FLECHT Program and its weaknesses. To CNI's knowledge this has been the only independent review of the program which has been carried out and made available in the public literature. CNI believes that the program failures and weaknesses that are identified in CNI testimony (Exhibit 1041) are overwhelmingly supported by the testimony of the knowledgeable engineers in the AEC contract laboratories who were associated and familiar with the elements of the program. CNI testimony sets out the case in substantial detail. In brief, the program was characterized by narrow scope, limited range of parameters investigated (many inappropriate to the tasks at hand), the use of incorrect materials, crude and incompetent instrumentation and operating techniques (with consequent major equipment malfunctions), and, as a culminating weakness, expansive and overgenerous interpretations. These latter, in CNI's view, misrepresented badly technical information available from the test results. In particular, in a test series of over 150 tests only one, ZR-2, simulated fuel rod swelling and rupture and the associated channel blockage which would be expected to occur under LOCA circumstances.. It was a unique test, a circumstance which should not have occurred, and was highly defective. The importance of test ZR-2 was reflected in the hearing record in the extensive time taken by participants to discuss and to try to illuminate the nature and sources of the test weaknesses and to determine reliability what the test had to say.
>
> J.O. Zane of [Aerojet] did not believe ZR-2 was a demonstration of the ability of BWR ECCS to operate in a LOCA (Tr. 6415-6423).
>
> C.G. Lawson of ORNL said test ZR-2 was borderline and more tests were required employing pressurized fuel rods as in ZR-2 (Tr. 5719-5725).
>
> P.L. Rittenhouse of ORNL stated that it was unreasonable and arbitrary to conclude that test ZR-2 shows flow blockage would not inhibit the spray cooling system (Tr. 4757).

Roger Griebe, the engineer at [Aerojet] with perhaps the most familiarity with the BWR-FLECHT Program, said that General Electric did not have the enthusiasm he felt necessary to conduct the tests (Tr. 6935-6945) and that he could not personally defend the General Electric conclusions (Tr. 7006). He said that GE "overstated" points, became carried away with impressions not verified by the technical data, and that the General Electric conclusion that protection was provided by the ECCS against all break sizes was not completely supported in the FLECHT data (Tr. 7100 et. seq.). In cross-examination he stated that he felt the GE reporting of the data was "tremendously slanted" (Tr. 7117). [Aerojet]-GE-AEC internal memorandum released by CNI bearing on the conduct of the BWR-FLECHT tests tells an even more dismal story of the conduct and interpretation of these tests than is contained either in CNI testimony or in the oral transcript. Based on the careful reading of the memorandum in the light of CNI's analysis of the tests and of the cross-examination of [Aerojet] and GE witnesses, CNI has concluded that in effect GE tried to approach elements of the test program, and attempted to interpret the results, in ways wholly inconsistent with the technical content of the test program.

These [Aerojet] memos, incorporated in Exhibit 1153, note:

"This was not [a] satisfactory demonstration test—the same need exists today—in fact, the need is greater because margin appears to be less than originally expected."

"[GE's] role in this program can only be described as a conflict of interest... Because the GE systems are marginally effective...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."

"...the close coupling between GE-FLECHT Project Group [the testing group] and GE licensing group has precluded pursuing a completely objective experimental program in an expedient manner."

An internal investigation of the failure of some of the GE design test apparatus to function properly, concluded:

"...the 'why' of the situation has come down to the simple fact that we believed GE was doing the job for which they were paid... the GE effort in heater development has been demonstrated to be seriously inadequate."

CNI's conclusion has been that it has proven inappropriate and damaging for the AEC to have established a policy of letting industry do the testing to check out the industry's own claims regarding safety system performance. The inherent conflict of interest has led to a testing program of narrow scope and poor quality.

One should note the letter of June 30, 1970 (Exhibit 1029) from William B. Cottrell, Director of the ORNL Nuclear Safety Program. In this letter to A.J. Pressesky:

"The Commission's position in its support of nuclear safety research is *seriously compromised* by relegating significant portions of the nuclear safety research and development program to the same industry it would license [emphasis added]."

Later in the letter, Cottrell cites examples known to him wherein a reactor vendor when given the responsibility for undertaking safety research on the reactor he was selling failed to get to the heart of the safety problems in question: In regard to fuel rod swelling in LOCA circumstances, the reactor vendors, on the basis of their own in-house R&D, concluded that the diametrical swelling of the fuel rods during the LOCA would be less than 30%. This they later increased to 60%. ORNL experiments subsequently demonstrated that swelling greater than 100% was possible under realistic conditions over significant portions of the core. Additionally, vendors maintained that embrittlement would not occur in the LOCA and hence its consequences need not be considered in evaluating the accident. Cottrell noted that ORNL experiments have been much more pessimistic in this regard.

With regard to [the] BWR-FLECHT Program, CNI concluded that in effect *GE tried to sabotage elements of the program* and attempted to interpret the results in ways utterly inconsistent with the test program's technical content. It is CNI's conclusion that the judgments set forth in its testimony are now even more powerfully supported by the hearing record [emphasis not added]. 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 of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[51]

[51] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 5.37-5.41.

(It is noteworthy that despite the testimony of a number of safety experts that ZR-2 did not demonstrate the ability of BWR ECCS to work effectively, the AEC Commissioners concluded:

> [T]he data of the Zr-2 BWR FLECHT experiment were cited as *evidence for the effectiveness of spray cooling*, although no temperature measurements were made at the positions of maximum bulging. We believe that additional assessments need to be made of these effects.
>
> In addition to the primary heat transfer effects of taking into consideration the swelling and rupture of the cladding, there would be important secondary effects arising from the steam oxidation of the cladding by the steam. Higher temperatures would lead to increased oxidation, which would contribute to a further increase in temperature, and the opening in the cladding would allow oxidation on the inside, again increasing the calculated temperature [emphasis added].[52])

6. Criticisms of GE's BWR Core Spray Tests

Regarding problems with BWR core spray cooling, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing"—section V.I., CNI Guidance for Commission on Information Needs for LOCA Analysis, BWR Core Spray—states:

> CNI identified the weaknesses in simulations of BWR core spray cooling of a full-length bundle as implemented in GE's uniquely defective test ZR-2 (Exhibit 1041, p. 5.39). CNI raised the possible existence of core spray diversion mechanisms that could in a BWR LOCA result in spray starvation of the central and hotter fuel bundles of a core. Similar concerns have been raised by Aerojet (Exhibit 1032, p. 124). GE diligence in resolving these concerns leaves a substantial amount to be desired. Cross-examination established (Tr. 13,925 et. seq.) that GE had done *no experiments* to determine spray droplet size distribution *either* at spray nozzles or at bundle tops (Tr. 13,953-13,956). They have done *no experiments* to determine the degree of superheat of the ejected steam. However, they "assume" the steam is saturated. They have done *no experiments* to determine steam temperature at bundle exit *or* to determine steam velocity at the bundle top. Moreover, the GE analytical model *does not furnish* a prediction of the velocity *nor compute entrainment* of spray

[52] 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. 1106; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

drops by upward streaming steam. Since GE believes superheating "will not occur in the reactor" they carried out no calculations to see if superheating results in velocity increases. GE, however, stated that in some of the ZR FLECHT tests a thermocouple was placed in the exit flue (steam exhaust line) some distance from the bundle exit. Temperatures at that point would surely be lower than at the bundle top. The thermocouple showed temperatures [of up] to 250°F. This information effectively invalidates the GE statement that no experimental information contradicted the saturation assumption especially in view of their not setting forth experimental results confirming the assumption. A hint that the exit steam velocities may be very great is given by Roger Griebe's observations that the steam plume and apparent steam exit velocity from test ZR-3 and 4 were unexpectedly and uniquely large (Staff reference 16.20 in the Regulatory Staff Supplementary Testimony). These observations raise some unresolved questions about the applicability of the GE core spray tests which are discussed next.

GE carried out spray tests using upward *air*flow, *with no heating*, to attempt to simulate spray operation (Tr. 13,919-13,925), but in view of the remarks above, CNI believes the test results to be inapplicable. Lawson of ORNL has criticized the tests because of the difference between steam and air on droplet entrainment (Tr. 5790-5795).

CNI shares the Aerojet view that spray may not work (Exhibit 1032). CNI believes that GE has not made adequate attempts to establish a contrary view, a situation which may reflect the fact that the contrary may not be supportable. The GE attitude toward test conduct and the interpretation of test data is well established with respect to the BWR-FLECHT tests by the [Aerojet]-GE-AEC internal memoranda placed in the record by CNI. These are discussed in [the FLECHT Section], above. It is shown that GE made a poor accommodation to the conflict of interest inherent in their assumption of responsibility for carrying out the tests. GE's diligence in core spray effectiveness tests is no better than in FLECHT, and their conclusions no better supported. As pointed out in [Section VII, "The Regulatory Failure,"] there is no assurance available from the FLECHT program that a BWR bundle can be cooled successfully at the spray rates employed in the tests and with the identified weaknesses in spray injection simulation (Exhibit 1041, p. 5.39) that reduced the conservatism of the test. Spray diversion in a BWR LOCA would reduce even further the controllability of the accident and it is without question a possibility which has not been eliminated. It requires prompt resolution.[53]

[53] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 5.43-5.45.

Regarding core spray distribution, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Another example relates to the distribution of core spray assumed by GE. GE performed tests involving air up-flow with non-heated simulated bundles as the basis for its core spray distribution assumptions. In June of 1972, the Regulatory Staff ask[ed] GE to describe the basis upon which it could conclude that these air up-flow tests are applicable to the reactor situation. These tests were performed many years ago [before 1973] by GE and they have been the basis upon which GE boiling water reactor emergency core cooling systems have been evaluated for several years, and they are the basis upon which the model approved by the Regulatory Staff in June 1971 determines now much emergency cooling water is delivered to the core. In asking this question, the Regulatory Staff raise[d] the most fundamental doubt about the kind of review that it made of the GE LOCA analysis during all [of] these years in which it has been allowing GE reactors to operate. [54]

7. More Recent BWR Thermal Hydraulic Experiments have been Conducted with Inconel 600 Fuel Rod Simulators

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 of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[55]

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 Thermal-Hydraulic Experiment 1 ("TH-1") tests, conducted at Chalk River in the early '80s, where the test planners—"for safety

[54] *Id.*, pp. 7.5-7.6.
[55] *Id.*, p. 5.41.

purposes"—did not want the maximum PCTs of the TH-1 tests to exceed 1900°F[56]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

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,[57] the Rig of Safety Assessment ("ROSA") III facility had electrically heated Inconel 600 fuel rod simulators,[58] and the Full Integral Simulation Test ("FIST") facility had electrically heated Inconel 600 fuel rod simulators.[59]

Additionally, the BWR FIX-II test facility had electrically heated Inconel 600 fuel rod simulators[60] and the NUPEC BWR Full-Size Fine-Mesh Bundle Test ("BFBT") facility had electrically heated Inconel 600 fuel rod simulators.[61]

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

[56] 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. 3-3.

[57] 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.

[58] 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.

[59] 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.

[60] 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, pp. 5-119, 5-129.

[61] 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.

cladding material. The SSTF experiments used steam injection to simulate core heat[62] the maximum temperature of the steam was 800 F.[63]

(It is noteworthy that many of the papers reporting on BWR heat-transfer experiments do not mention what type of cladding material was used in the fuel rod simulators in the experiments they describe. For example, Petitioner has not located any papers that state what type of cladding material is used in the fuel rod simulators in the Purdue University Multidimensional Integral Test Assembly ("PUMA") facility. Most likely, the PUMA facility—currently investigating BWR-related problems—uses Inconel 600 fuel rod simulators: the Rod Bundle Heat Transfer ("RBHT") facility at Penn State University—currently investigating PWR-related problems—uses Inconel 600 fuel rod simulators.[64] Also, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, which describes many experimental facilities, does not mention what type of cladding material was used in the fuel rod simulators at the experimental facilities it describes.)

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.[65]

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

---

[62] 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.

[63] 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.

[64] 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.

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

(type 304), stainless steel (type 309), and Inconel 800HT. Inconel 600 oxidized very little over a period of 1000 hours of cyclic exposure time.

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."[66]

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. To conclusively demonstrate that BWR ECCSs would be effective, in the event of a LOCA, it would be necessary to conduct BWR heat transfer experiments with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Experiments with Inconel 600 fuel rod simulators are inadequate.

Furthermore, 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.

[66] Mr. Kelly, NRC, Advisory Committee on Reactor Safeguards, Transcript of Subcommittee Meeting on Thermal Hydraulic Phenomena, July 7, 2008, p. 168.

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,

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

Dated: November 24, 2010

Received: from mail2.nrc.gov (148.184.176.43) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Tue, 23 Nov 2010
23:38:05 -0500
X-Ironport-ID: mail2
X-SBRS: 4.4
X-MID: 30260064
X-fn: Comments November 2010 II.pdf
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
AI0PAP8k7ExKfVlpkGdsb2JhbACTP48rCBYBAgkJExEDH5gjixCJZlIYhQYuiFkBAQMFhUcEhF
qGBQ
X-IronPort-AV: E=Sophos;i="4.59,246,1288584000";
d="pdf'?scan'208";a="30260064"
Received: from mail-ww0-f41.google.com ([74.125.82.41]) by mail2.nrc.gov with
ESMTP; 23 Nov 2010 23:38:03 -0500
Received: by wwb29 with SMTP id 29so626212wwb.2 for <multiple
recipients>; Tue, 23 Nov 2010 20:38:02 -0800 (PST)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:received:date:message-id
:subject:from:to:cc:content-type;
bh=WIWHQWI0AsIdu5XbxWX9Wp+oVYkpt/TfeIX3UMviEF4=;
b=JrEPogUsavbmucNayCZwD+q34Pf/t7ofK5B0RQjrSVxmIHxa75IPXvHi8HFvtZolmm
Cj03yEpZgAvbA2nzLETv6mYZxYWrsaXzEGk3one2mMTMJ4CxOrGAd/t+/PKy/5CSu5JP
xKtzNzWx7kP7kPJqw5PEYdoHQ5UGsOO+0S53k=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:cc:content-type;
b=mhvgeLWzsWFSuxXgZ+v3YeDnxTQz0CrAsrbXyDcWPT4JVRZrZMwQBYYI0DHDrkOdlj
lwJDZxn6G5/usvN736TKXXfl2tQ6Dr8k91/fiULhdyvKvN09FiBb9i9hw0zXqs/VdKut
SnVPjEI/UmXnH5o8+IleXI5Y10T5YsDkbopqM=
MIME-Version: 1.0
Received: by 10.227.170.78 with SMTP id c14mr8137290wbz.49.1290573480512; Tue,
23 Nov 2010 20:38:00 -0800 (PST)
Received: by 10.227.72.208 with HTTP; Tue, 23 Nov 2010 20:38:00 -0800 (PST)
Date: Tue, 23 Nov 2010 23:38:00 -0500
Message-ID: <AANLkTi=D5BZkTH-Hy2FvB9r1-aXZAZagRVTNpVxV4V1S@mail.gmail.com>
Subject: NRC-2009-0554 (Second)
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>, PDR Resource
<pdr.resource@nrc.gov>
CC: Dave Lochbaum <dlochbaum@ucsusa.org>, <necnp@necnp.org>, Raymond Shadis
<shadis@prexar.com>, "Powers, Dana A" <dapower@sandia.gov>
Content-Type: multipart/mixed; boundary="90e6ba476541dcd0170495c50dd9"
Return-Path: markleyse@gmail.com

Submission ID 23
Aladar Stolmar
ML103340250

3

PRM-50-95
(75FR66007)

DOCKETED
USNRC

November 24, 2010 (11:10am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

November 24, 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

From Aladár Stolmár, Lőrinci, Szabadság tér 3 HU3021 (Hungary)

23-1

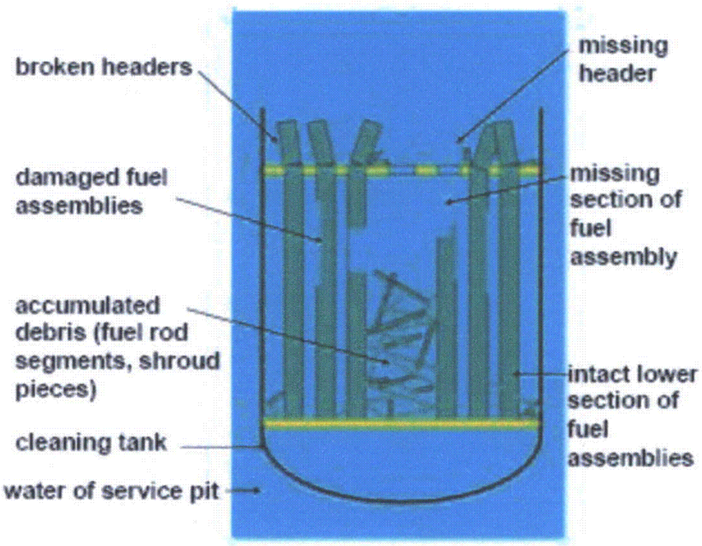
Consideration of the zirconium-steam reaction and the ignition and intense firestorm in nuclear reactor fuel rods is well overdue. Reevaluating the evidence provided by the TMI-2 reactor accident, Chernobyl-4 reactor accident, and Paks Unit 2 fuel washing incident, with consideration of this intense fiery process, will bring us closer to an ultimately safe nuclear power plant design.

For a brief look into the benefits provided by such an effort I am providing two quotes:

1) (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2004/secy2004-0224/2004-0224scy.pdf) The United States Nuclear Regulatory Commission stated:

"Because the Paks incident resulted in conditions more severe than a traditionally analyzed loss-of-coolant accident, yet the fuel remained well below any melting temperature (i.e., it was coolable), this project appears to have the potential to provide significant insights to fuel behavior under accident conditions."

2) (http://www-pub.iaea.org/MTCD/publications/PDF/TDL-002_web.pdf) The OECD-IAEA Paks Fuel Project Final Report describes the final state of the fuel rods



e 2.11. Distribution of damaged fuel in the cleaning tank

Template=SECY-067

DS 10

„2.4 State of damaged fuel
After the incident detailed visual examination was carried out with the help of video cameras. The examination indicated that most of the fuel assemblies suffered damage. Brittle failure and fragmentation of fuel assemblies was observed. Above the upper plate several assembly heads were broken, standing in inclined position (Fig. 2.11). One assembly header was found far from its original place. Many assemblies were broken and fragmented below the upper plate, too. Some assemblies were fractured in their entirety. Fuel rod fragments and shroud pieces accumulated on the lower plate between the assemblies. Many fuel rod pieces and fragments of assembly shroud were spread in the tank. Some fuel pellets fell out of fuel rods, their form remained mainly intact. Heavy oxidation of the zirconium components was identified. Less oxidation was found in the periphery than in the centre. The bottom part of the fuel remained intact.
The visual investigations have also shown that the fuel assemblies positioned closer to the vertical axis suffered heavier damage, in some cases long parts were simply broken out from them. Thanks to the better position for the radiative heat transfer, the outermost assemblies suffered less heavy damage. The broken fuel pins, shrouds and fallen down fuel pellets formed a heap of debris on the bottom positioning and support plate.
There were no signs of melting or formation of zirconium-steel eutectics on the surface of stainless steel components. This fact indicates that the maximum temperature during the incident remained below ≈1400 oC.
The activity concentrations in the coolant and the release through the chimney are regularly measured and such data were available after the incident, too. The incident happened two weeks after reactor shutdown, for this reason the release of isotopes with short half-life was very low. Integrating the activity concentrations over time and coolant volume in the pool, and summarizing the release through the chimney in time, the total activity release from the fuel was determined for several isotopes. Most of the activity remained in the water, since the incident took place under 13 m water level, only the noble gases were released through the chimney. The integrated activity release was compared to the calculated inventories and the release rate was determined. In case of gaseous and volatile isotopes the release rate was roughly 1% (the precise data are given in Table 2.1). The release rate of non-volatile isotopes was much less. The ≈1% iodine, cesium and noble gas release indicated that the temperatures in the cleaning tank could not be very high, otherwise larger release should have been recorded. Considering these release rates the maximum temperature was estimated about 1200-1300 oC. This temperature range can explain as well that the local oxidation reached 100% in some positions.
The hot cell examination of the damaged fuel could not be carried out at the Paks nuclear power plant, since the power plant does not have the necessary equipment and facilities for the detailed investigation of irradiated fuel.
The very brittle state of the damaged fuel was observed during the removal operations. Several fuel assemblies and fuel rods were fragmented when the damaged fuel was removed from the cleaning tank and placed in the containers."

23-2

It is a much overdue duty of NRC and IAEA to evaluate the evidence provided by the TMI-2 accident, Chernobyl-4 accident, Paks-2 incident, and related experiments. Evaluating this evidence, one can see that the ignition of the zirconium fire in the steam occurs at a local temperature of the fuel cladding of around 1000-1200°C, [[and that a self-feeding with steam due to the precipitation of eroded fuel pellets and zirconia reaction product from the hydrogen stream into the water pool, causes intense evaporation.]]

There are insignificant differences in the progression of the firestorms that occurred in the TMI-2 reactor severe accident, Paks washing vessel incident, and Chernobyl-4 reactor accident; the later defined only by the amount of zirconium available for the reaction. At the mean time, there are significant similarities in the processes leading to the ignition of the firestorm. In all three of the compared cases, it took several hours of ill-fated actions or in-actions of the operators to cause the ignition condition. Also, there are similarities in the end result of the firestorm; namely, that the extent of the fuel damage is much less than it was predicted from any other severe fuel damage causing scenarios, introduced for explanations. Therefore the fraction of released fission products is significantly less than was anticipated from the fuel melting or a so called "steam-explosion" scenario. Also, the fiery steam-zirconium reaction results in a much higher than anticipated (from any other scenarios) rate of Hydrogen production, which in turn requires a review of containment designs.

PUBLIC SUBMISSION

As of: November 24, 2010
Received: November 24, 2010
Status: Pending_Post
Tracking No. 80ba1f9f
Comments Due: November 26, 2010
Submission Type: Web

Docket: NRC-2009-0554
Mark Edward Leyse; Calculated Maximum Fuel Element Cladding Temperature

Comment On: NRC-2009-0554-0024
Mark Edward Leyse; Mark Edward Leyse and Raymond Shadis, on Behalf of the New England Coaltion; Petitions for Rulemaking

Document: NRC-2009-0554-DRAFT-0024
Comment on FR Doc # 2010-27164

Submitter Information

Name: Aladar Stolmár
Address:
Szabadsag ter 3
Lorinci, Hungary, HU3021
Organization: retired

General Comment

See attached file(s)

Consideration of the zirconium-steam reaction and the ignition and intense firestorm in nuclear reactor fuel rods is well overdue. Reevaluating the evidence provided by the TMI-2 reactor accident, Chernobyl-4 reactor accident, and Paks Unit 2 fuel washing incident, with consideration of this intense fiery process, will bring us closer to an ultimately safe nuclear power plant design.

For a brief look into the benefits provided by such an effort I am providing two quotes:

1) (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2004/secy2004-0224/2004-0224scy.pdf) The United States Nuclear Regulatory Commission stated:

"Because the Paks incident resulted in conditions more severe than a traditionally analyzed loss-of-coolant accident, yet the fuel remained well below any melting temperature (i.e., it was coolable), this project appears to have the potential to provide significant insights to fuel behavior under accident conditions."

2) (http://www-pub.iaea.org/MTCD/publications/PDF/TDL-002_web.pdf) The OECD-IAEA Paks Fuel Project Final Report describes the final state of the fuel rods.
...

It is a much overdue duty of NRC and IAEA to evaluate the evidence provided by the TMI-2 accident, Chernobyl-4 accident, Paks-2 incident, and related experiments. Evaluating this evidence, one can see that the ignition of the zirconium fire in the steam occurs at a local temperature of the fuel cladding of around 1000-1200°C, [[and that a self-feeding with steam due to the precipitation of eroded fuel pellets and zirconia reaction product from the hydrogen stream into the water pool, causes intense evaporation.]]

There are insignificant differences in the progression of the firestorms that occurred in the TMI-2 reactor severe accident, Paks washing vessel incident, and Chernobyl-4 reactor accident; the later defined onl

Attachments

NRC-2009-0554-DRAFT-0024.1: Comment on FR Doc # 2010-27164

Rulemaking Comments

From: Gallagher, Carol
Sent: Wednesday, November 24, 2010 10:53 AM
To: Rulemaking Comments
Subject: Comment on PRM-50-93/PRM-50-95
Attachments: NRC-2009-0554-DRAFT-0024.pdf

Van,

Attached for docketing is a comment on PRM-50-93/50-95 from Aladar Stolmar that I received via the regulations.gov website on 11/24/10.

Thanks,
Carol

Received: from HQCLSTR01.nrc.gov ([148.184.44.79]) by TWMS01.nrc.gov
([148.184.200.145]) with mapi; Wed, 24 Nov 2010 10:53:06 -0500
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: "Gallagher, Carol" <Carol.Gallagher@nrc.gov>
To: Rulemaking Comments <Rulemaking.Comments@nrc.gov>
Date: Wed, 24 Nov 2010 10:52:37 -0500
Subject: Comment on PRM-50-93/PRM-50-95
Thread-Topic: Comment on PRM-50-93/PRM-50-95
Thread-Index: AcuL751pr4hrO82LRj6Xe9p0ZsLQtQ==
Message-ID:
<6F9E3C9DCAB9E448AAA49B8772A448C546A0E47CBC@HQCLSTR01.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<6F9E3C9DCAB9E448AAA49B8772A448C546A0E47CBC@HQCLSTR01.nrc.gov>
MIME-Version: 1.0

Submission ID 24
John Butler, Nuclear Energy Institute
ML103340251

PRM-50-95
(75FR66007)

4

NUCLEAR ENERGY INSTITUTE

John C. Butler
DIRECTOR
ENGINEERING AND OPERATIONS SUPPORT
NUCLEAR GENERATION DIVISION

DOCKETED
USNRC

November 24, 2010 (3:05pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

November 24, 2010

Ms. Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attn: Rulemakings and Adjudications Staff

Subject: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC–2009–0554

Project Number: 689

Dear Ms. Vietti-Cook:

The attachment to this letter provides comments from the Nuclear Energy Institute (NEI)[1] on behalf of the nuclear energy industry on the Petition for Rulemaking (PRM-50-95). This petition requests that the NRC order the licensee of Vermont Yankee to lower the licensing basis peak cladding temperature in order to provide a necessary margin of safety in the event of a Loss of Coolant Accident (LOCA).

As noted in the October 27, 2010 Federal Register Notice, the requested actions and supporting information addressed in PRM-50-95 are similar to actions requested under PRM-50-93. As such, NEI comments on the earlier petition, provided on April 12, 2010, continue to apply. Neither of the referenced tests cited in support of PRM-50-93 and PRM-50-95, whether reviewed in isolation or in combination with other tests, support the changes sought by the petitioner. NEI recommended that the petitioner's request under PRM-50-93 be denied. This recommendation applies to the actions requested under PRM-50-95.

[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

Template= SECY-067

DS 10

As explained in the attached comments, experimental evidence shows that the current LOCA Peak Cladding Temperature (PCT) licensing limit is sufficient to ensure that the cladding can withstand post-quench LOCA loads in order to maintain a coolable geometry. Additionally, the energy released from the metal-water reaction is currently accounted for in LOCA licensing calculations used to determine PCT values. Evidence shows that with sufficient cooling to account for the heat generation from the metal-water reaction the threat of clad melting is abated. Thus, it is the industry's position that the current regulatory limit of 2200°F (1204°C) PCT is adequate to maintain plant safety in the event of a large break LOCA and the proposed reduction of Vermont Yankee's PCT to 1832°F (1000°C) is not warranted. The petitioner's requests for action under PRM-50-93 and PRM-50-95 should be denied.

If you have any questions regarding this matter, please feel free to contact me at jcb@nei.org; 202-739-8108 or Gordon Clefton at 202-739-8086; gac@nei.org.

Sincerely,

John C. Butler

Attachment

Industry Comments on Petition PRM-50-95

A petition for rulemaking pursuant to Title 10 of the *Code of Federal Regulations* (10CFR) Section 2.206 of the NRC's regulations was filed on June 7, 2010, requesting that the Nuclear Regulatory Commission (NRC) order the licensee of Vermont Yankee Nuclear Power Station to lower the licensing basis peak cladding temperature 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). The petitioner states that his interpretation of data from select multi-rod (assembly) severe fuel damage experiments indicates that the current licensed Peak Cladding Temperature (PCT) of Vermont Yankee of 1960°F (1071°C) does not provide a necessary margin of safety to help prevent a partial or complete meltdown in the event of a LOCA. The petitioner's interpretation of the data concludes that Vermont Yankee's large break PCT should be decreased to a temperature lower than 1832°F (1000°C) in order to provide a necessary margin of safety.

Background

The petitioner uses data from select multi-rod severe accident tests in an attempt to demonstrate that the cladding may reach the autocatalytic zirconium-water regime at temperatures lower than the licensed PCT for Vermont Yankee. In addition, the petitioner calls into question the adequacy of the correlations used in calculating the metal-water reaction rates. It is the Industry's position that the current licensing evaluations for Vermont Yankees PCT and the regulatory limit of 2200°F (1204°C) are valid.

Review of the Selection of 2200°F (1204°C) Criterion in 1973 ECCS Hearings

It is clear from a review of the 1973 Emergency Core Cooling System (ECCS) Hearings that the primary rationale for the selection of the embrittlement criteria (i.e., the 17%- Equivalent Clad Reacted (ECR) oxidation and the 2200°F (1204°C) peak cladding temperature) is retention of cladding ductility at temperatures higher than 275°F (135°C). The criteria are essentially based on the ductile-brittle transition data obtained from Hobson's slow-ring-compression tests performed at 73-302°F (23-150°C) [1].

24-1

The criterion that must be satisfied is that the cladding must have sufficient ductility to survive post-quench LOCA loads. From the results of post-test metallographic analysis of the slow-ring-compression specimens, Hobson [1] observed a good correlation between zero ductility temperature (ZDT) and the fractional thickness of transformed beta layer (or the sum of oxide plus alpha layer thickness) as long as the specimen was oxidized at ≤2200°F (≤1204°C). In spite of comparable thickness of transformed beta layer, specimens oxidized at 2400°F (1315°C) were far more brittle. This observation was explained on the basis of excessive solid-solution hardening of transformed-beta phase at high oxygen (O) concentrations that are characteristic of oxidation at the high temperature. Because of the solubility limit of oxygen in the beta phase, this high O concentration cannot be reached at 2200°F (1204°C) but can be reached at 2400°F (1315°C). Thus, embrittlement is not simply a function of the extent of oxidation alone, but is related to the exposure temperature. 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 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).

The petition calls into question the Baker-Just correlation that is specified in Appendix K of 10CFR50.46 for the calculation of the energy release rate due to oxidation, hydrogen generation, and ECR. The Baker-Just correlation using the current range of parameter inputs has been shown to be conservative and adequate to assess Appendix K ECCS performance. Virtually every data set published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F (982°C). Recent tests conducted at ANL have demonstrated that the correlation over-predicts the zirconium-water reaction by as much as 30% at the limiting temperature 2200°F (1204°C) with no observable zirconium-water autocatalytic reactions. Thus, use of the Baker-Just correlation is still appropriate.

24-2

The 1989 USNRC Regulatory Guide 1.157 allowed the use of a best-estimate correlation to calculate the zirconium-water reaction for temperatures greater than 1900°F (1038°C) and recommended the use of the Cathcart-Pawel correlation (NUREG-17). The NRC, foreign organizations such as JAEA in Japan and CEA in France, and the United States nuclear industry are currently conducting and evaluating experimental and analytical programs on fuel cladding behavior under LOCA conditions. The research includes the effects of various types of zirconium-based cladding, high burnup, mixed oxides, ZrO_2 phase change hysteresis, and system pressures. These tests including both well-characterized isothermal high temperature oxidation tests and integral rodlet tests conducted at temperatures up to 2200°F (1204°C) have confirmed predictive capability of the Cathcart-Pawel correlation with no observable zirconium-water autocatalytic reactions. Thus, use of the Cathcart-Pawel correlation is still appropriate.

As pointed out by the petitioner, prevention of runaway oxidation was a consideration when limiting peak cladding temperatures to 2200°F (1204°C). Since heat generation from a metal-water reaction could become excessive and an autocatalytic type of situation could occur at high cladding temperatures, design considerations still address the heat balance near this temperature.

24-3

The effects of the exothermic zirconium-water reaction are considered in the ECCS design because of their potential influence on the thermal and mechanical behavior of the system. A review of available literature concludes that the zirconium-water reaction is relatively slow and corrosion-like under most conditions; however, at very high temperatures a self-sustaining reaction with steam can occur. The term autocatalytic oxidation has been misused by the industry for some time to identify the situation in which the heating rate resulting from the metal-water reaction is so rapid that any reasonable cooling process cannot arrest the cladding heatup. At any temperature approaching the 10CFR50.46 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.

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. A range 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

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 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.

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. This is reinforced from calculations conducted in support of the Quench-06 test [2]. 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).

Thus, 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. A proposed limit of 1832°F (1000°C) to prevent the initiation of the oxidation phenomenon as requested by the petitioner is not justified.

24-4

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.

High-temperature oxidation behavior has been investigated by numerous investigators including prototypic LOCA tests in TREAT and PBF test reactors and by ANL investigators. During TREAT-FRF2 test, a seven-rod cluster was oxidized at ~2399°F (~1315°C) and quenched [3]. There was no reported evidence of melting during these tests due to autocatalytic oxidation even though the tests were conducted at temperatures in excess of the regulatory limit. This information is summarized in Figure 3 [from Ref. 4 and Ref. 5]. Thus, there is further evidence to support that a cladding temperature limit of 1832°F (1000°C) to prevent the initiation of the control blade melting is not justified.

Conclusions

Experimental evidence shows that the current LOCA PCT licensing limit is sufficient to ensure that the cladding can withstand post-quench LOCA loads in order to maintain a coolable geometry. Additionally, the energy released from the metal-water reaction is currently accounted for in LOCA licensing calculations used to determine PCT values. Evidence shows that with sufficient cooling to account for the heat generation from metal-water reaction the threat of clad melting is abated. Thus, it is the Industry's position that the current regulatory limit of 2200°F (1204°C) PCT is adequate to maintain plant safety in the event of a large break LOCA and the proposed reduction of Vermont Yankee's PCT to 1832°F (1000°C) is not warranted.

References

1. D. O. Hobson, "Ductile-brittle behavior of Zircaloy fuel cladding," Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, March 26, 1973, pp. 274-288.

2. W. Hering, et. al., "Comparison and Interpretation Report of the OECD International Standard Problem No. 45 Exercise (Quench-06)," FZKA 6722, Forshchungszentrum Karlsruhe GmbH, Karlsruhe, 2002.

3. R. A. Lorenz, "Fuel Rod Failure under Loss-of-Coolant Conditions in TREAT," Nucl. Tech. 11 (1971) 502-520.

4. F. M. Haggag, "Zircaloy Cladding Embrittlement Criteria: Comparison of In-Pile and Out-of-Pile Results," NUREG/CR-2757, July 1982.

5. H. M. Chung and T. F. Kassner, "Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light-Water Reactors," NUREG/CR-1344, ANL-79-48, Argonne National Laboratory, January 1980.

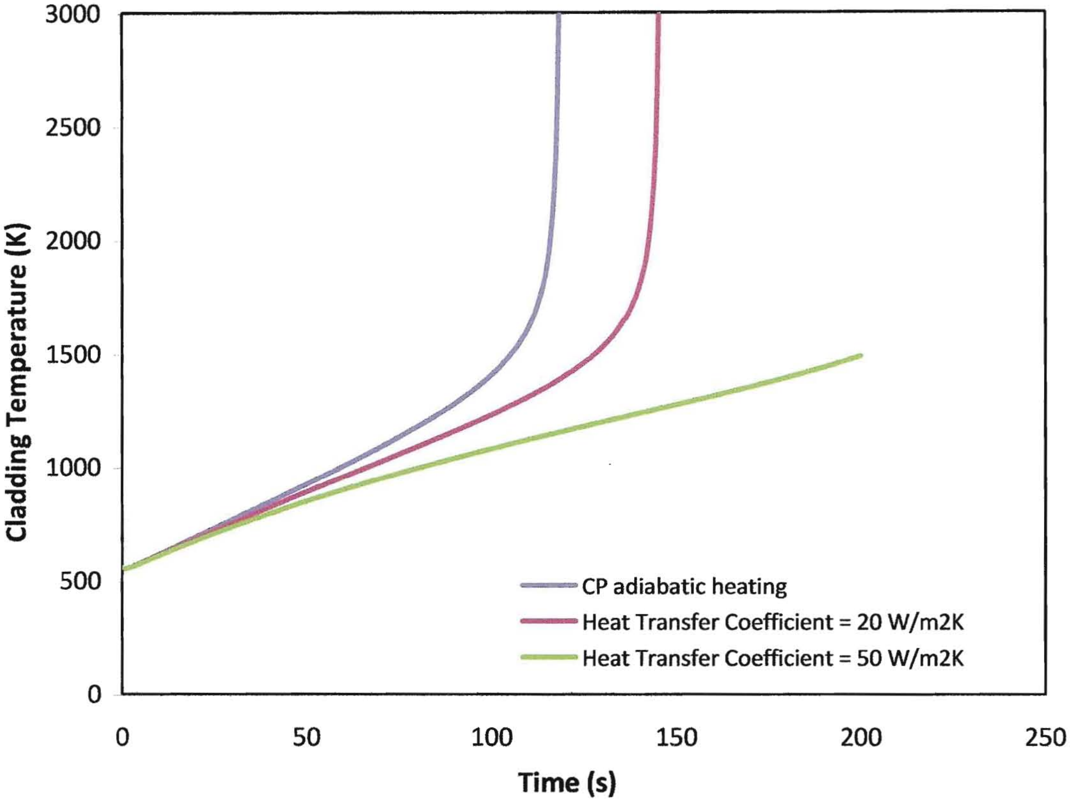


Figure 1 - PCT evolution of a rod due to decaying and two-sided oxidation heat assuming different cooling rates (heat transfer coefficients).

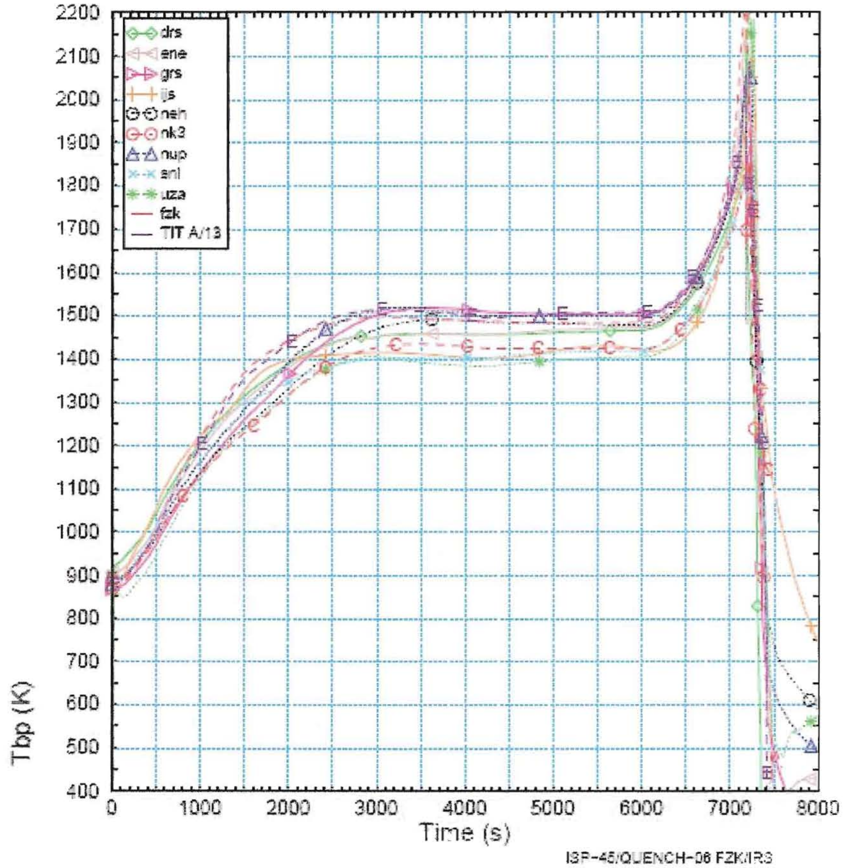


Figure 2 - Maximum bundle temperature calculated during the open phase for Quench-06 [3].

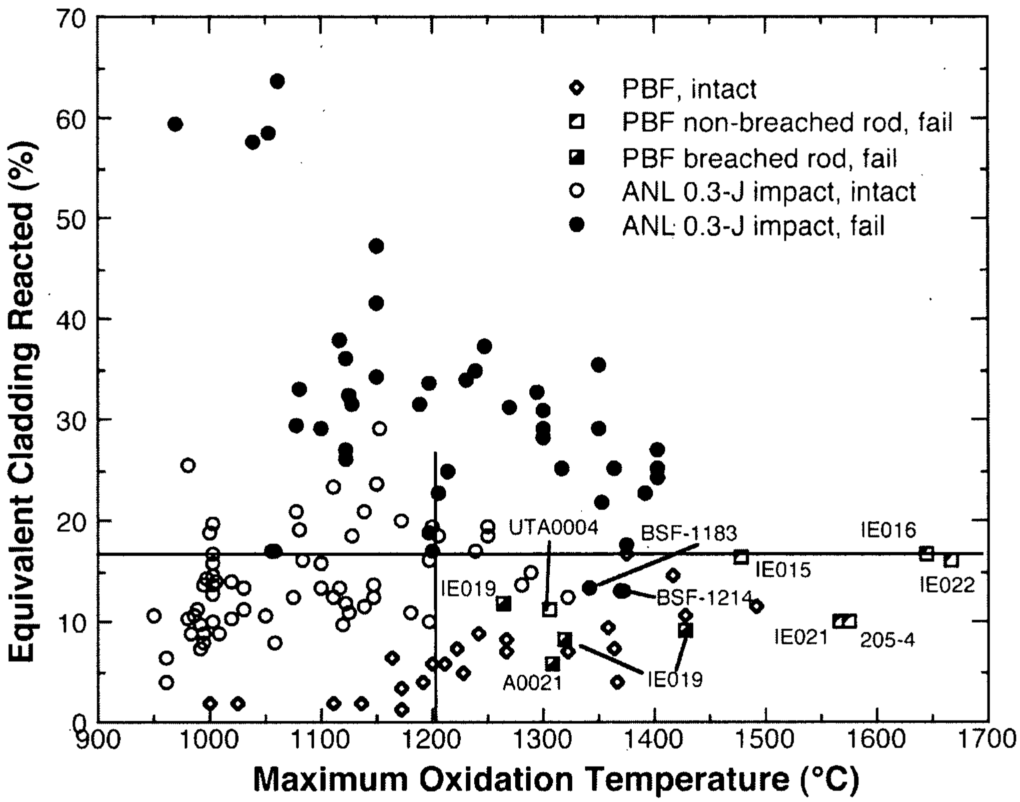


Figure 3 - Comparison of Data from Hot-Cell Handling Failure of Zircaloy Rods Exposed to High Temperature in Power Burst Facility (Ref. 4) and 0.3-J Impact Tests in ANL (Ref. 5)

Docket, Hearing

From:	Vietti-Cook, Annette
Sent:	Wednesday, November 24, 2010 2:24 PM
To:	Docket, Hearing
Subject:	FW: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC Order Vermont Yankee to Lower the Licensing Basis PCT.
Attachments:	11-24-10_NRC_Industry Comments on PRM-50-95.pdf; 11-24-10_NRC_Industry Comments on PRM-50-95_Attachment.pdf

---

From: BELL, Denise [mailto:dxb@nei.org] **On Behalf Of** BUTLER, John
Sent: Wednesday, November 24, 2010 2:13 PM
Subject: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC Order Vermont Yankee to Lower the Licensing Basis PCT.

November 24, 2010

Ms. Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attn: Rulemakings and Adjudications Staff

Subject: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC Order Vermont Yankee to Lower the Licensing Basis PCT. Docket ID NRC–2009–0554

Project Number: 689

Dear Ms. Vietti-Cook:

The attachment to this letter provides comments from the Nuclear Energy Institute (NEI) on behalf of the nuclear energy industry on the Petition for Rulemaking (PRM-50-95). This petition requests that the NRC order the licensee of Vermont Yankee to lower the licensing basis peak cladding temperature in order to provide a necessary margin of safety in the event of a Loss of Coolant Accident (LOCA).

As noted in the October 27, 2010 Federal Register Notice, the requested actions and supporting information addressed in PRM-50-95 are similar to actions requested under PRM-50-93. As such, NEI comments on the earlier petition, provided on April 12, 2010, continue to apply. Neither of the referenced tests cited in support of PRM-50-93 and PRM-50-95, whether reviewed in isolation or in combination with other tests, support the changes sought by the petitioner. NEI recommended that the petitioner's request under PRM-50-93 be denied. This recommendation applies to the actions requested under PRM-50-95.

As explained in the attached comments, experimental evidence shows that the current LOCA Peak Cladding Temperature (PCT) licensing limit is sufficient to ensure that the cladding can withstand post-quench LOCA loads in order to maintain a coolable geometry. Additionally, the energy released from the metal-water reaction is currently accounted for in LOCA licensing calculations used to determine PCT values. Evidence shows that with sufficient cooling to account for the heat generation from the metal-water reaction the threat of clad melting is abated. Thus, it is the industry's position that the current regulatory limit of 2200°F (1204°C) PCT is adequate to maintain plant safety in the event of a large break LOCA and the proposed reduction of Vermont Yankee's PCT to 1832°F (1000°C) is not warranted. The petitioner's requests for action under PRM-50-93 and PRM-50-95 should be denied.

If you have any questions regarding this matter, please feel free to contact me at jcb@nei.org; 202-739-8108 or Gordon Clefton at 202-739-8086; gac@nei.org.

Sincerely,

John C. Butler

Attachment

John C. Butler
Director, Engineering & Operations Support

Nuclear Energy Institute
1776 I Street NW, Suite 400
Washington, DC 20006
www.nei.org

P: 202-739-8108
F: 202-533-0113
M: 202-391-2970
E: jcb@nei.org

nuclear. clean air energy.

Received: from HQCLSTR02.nrc.gov ([148.184.44.77]) by OWMS01.nrc.gov
([148.184.100.43]) with mapi; Wed, 24 Nov 2010 14:24:24 -0500
Content-Type: application/ms-tnef; name="winmail.dat"
Content-Transfer-Encoding: binary
From: "Vietti-Cook, Annette" <Annette.Vietti-Cook@nrc.gov>
To: "Docket, Hearing" <Hearing.Docket@nrc.gov>
Date: Wed, 24 Nov 2010 14:24:24 -0500
Subject: FW: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC
Order Vermont Yankee to Lower the Licensing Basis PCT.
Thread-Topic: Industry Comments on Petition for Rulemaking (PRM-50-95), NRC
Order Vermont Yankee to Lower the Licensing Basis PCT.
Thread-Index: AcuMC4swFELms9ULQyi2eFPhf0JL/QAAaURw
Message-ID:
<2C5246E2C48F77418DF2EE22F3C7DE97076312CAC2@HQCLSTR02.nrc.gov>
Accept-Language: en-US
Content-Language: en-US
X-MS-Has-Attach: yes
X-MS-Exchange-Organization-SCL: -1
X-MS-TNEF-Correlator:
<2C5246E2C48F77418DF2EE22F3C7DE97076312CAC2@HQCLSTR02.nrc.gov>
MIME-Version: 1.0

Submission ID 25
Robert Leyse
ML103340252

PRM-50-95
(75FR66007)

5

November 26, 2010

DOCKETED
USNRC

November 26, 2010 (1:30pm))

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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

NRC should not authorize Plant License Renewals or Power Uprates prior to its resolution of PRM-50-93 and PRM-50-95.

The 2200 degree Fahrenheit PCT limit is too high. The 2200 PCT limit is based on embitterment criteria. The Baker-Just equation was placed into 50.46 and it has been convenient in licensing. Its current use is fiercely defended by the NRC.

According to analyses funded by NRC, when the Baker-Just correlation is applied, the predicted thermal runaway starts at 2600 degrees Fahrenheit, while the alternative Cathcart-Pawel correlation of Reg. Guide 1.157 yields runaway at 2700. This is detailed on page 28 of PRM-50-93.

At a joint meeting of three ACRS subcommittees on May 31, 2002, there is the following pertinent exchange:

MR. LAUBEN: That's it. Sure. No.
That's an easy and quantifiable way to compare it. It just gives you a minimum measure because what's really true because of the slope changes so much is that you can see a much bigger difference. **In general I would say I could never achieve turn-around much above 2300 in the limited 100 calculations I did with Baker-Just but I could reach something as close to 2800 with Cathcart-Pawel. Now that's –**

MEMBER WALLIS: Maybe you need to show these calculations. Something more convincing than what we heard today --

Template=SECY-067

DS 10

25-1

At another point in that joint meeting of three ACRS subcommittees on May 31, 2002:

MEMBER WALLIS: 2200 has a very iffy basis. The only justification really is that it is worked over 30 or 40 years. If you are going to change it you're going to have to have some really good arguments.

However, Member Wallis is wrong. There is nothing "iffy" about 2200. At Karlsruhe it had already been clearly demonstrated that 2200 is too high and there is nothing "iffy" about the fact that 2200 is too high. An array of experiments having multirod assembles of rods with zirconium alloy cladding reveal that thermal runaway begins well below the 2600 to 2700 range. Perhaps the most impressive is LOFT LP-FP-2 where thermal runaway of the fuel bundle was initiated in the 2060 to 2240 degree Fahrenheit range. And, the series of CORA experiments at Karlsruhe with bundled electrically heated rods having Zirconium alloy cladding and uranium fuel pellets, yielded thermal runaway over a range from about 1800 to 2200 degrees Fahrenheit.

Although PRM-50-93 is dated November 2009, there is little evidence that the NRC has pursued its evaluation. On April 26, 2010, NRR issued a USER NEED REQUEST FOR TECHNICAL ANALYSIS OF PETITION FOR RULEMAKING ON 10 CFR 50.46 (PRM-50-93) and at that time the activity was (finally) assigned a high priority. Quoting from the User Need Request, *The requested deliverable for this user need is a technical letter report. Your office provided an outstanding technical analysis [reference 2] of a similar rulemaking petition, and we request the final deliverable for this user need be in this same format. We also request that a draft of your report be provided for comment by August 31, 2010 and the final report by September 30, 2010. We will provide comments on the draft within one week of receipt.*

However: On October 27, 2010, the NRC published for public comment a notice of consolidation of petitions for rulemaking. *The PRMs to be consolidated are PRM-50-93 filed by Mark Edward Leyse on November 17, 2009, and PRM-50-95 filed on June 7, 2010, by Mark Edward Leyse and Raymond Shadis, on behalf of the New England Coalition.* What Mark Leyse filed on June 7, 2010 was not a PRM, it was a 2.206 petition. It appears that by consolidating these actions by Mark Leyse, the NRC has substantially extended the deadline for producing a Technical Letter Report regarding PRM-50-93. Nevertheless, the priority is established by the technical facts that are in the record and high priority attention by the NRC reviewers remains warranted.

In fact, Mark Edward Leyse first learned about the extended deadline when the ACRS Thermal Hydraulics Phenomena Subcommittee briefly discussed the matter on Monday, October 18, 2010. Mark Leyse and Robert Leyse had jointly made a 10 minute presentation, and at the end of the meeting the subcommittee discussed the matter as follows:

CONSULTANT KRESS: I found it very unusual

17 that public comments are made to the subcommittee.

18 Those usually go to the full committee. I don't know

19 what your obligation is with respect to those.

20 CHAIR BANERJEE: I think to report it to

21 the full committee and ask if -

22 CONSULTANT KRESS: Just report it to the

23 full committee.

24 CHAIR BANERJEE: ask if they wish it to be

25 made to the full committee. I don't think that we can

act on it.

2 CONSULTANT KRESS: No. That was my point.

3 It has to be acted by the full committee.

CONSULTANT WALLIS: But if you want a comment, it looked as if there could be a significant point here, I mean it's something that is not trivial to look at and see is there a question here and what's the evidence for -

CHAIR BANERJEE: Has the comments been made to the staff or is it just to the subcommittee?

MR. BAJOREK: This is Steve Bajorek. Actually there are two petitions in play right now. The petition they talked about brings up the point that they Baker-Just is possibly not conservative. He has the same comment on Cathcart-Pawel. Asks to look at some of these other test data that he claims we have not looked at before.

He also submitted -

CHAIR BANERJEE: Particularly bundle data.

MR. BAJOREK: Bundle, yes. The staff has put together a small group to start to evaluate these concerns. We started to take a look at it and another petition came in, this one on the behalf of Connecticut or Yankee, it's a plant that's been up for relicensing. There are --

CONSULTANT WALLIS: Vermont Yankee?

MR. BAJOREK: Vermont Yankee, that's right. Vermont Yankee is being relicensed. They have also put in a petition on their behalf where they cite many of the same concerns. Because these petitions are over lapping, the staff decided they were not going to look at them individually, they were going to put them together. We went through our OGC. They said that was an appropriate thing to do and now the window of time for evaluating those petitions and those concerns has been reopened and I think we have another -- I think we have a year to go through and reevaluate everything. So there's a group that is looking at that.

CHAIR BANERJEE: So I think we can report that to the full committee.

CONSULTANT WALLIS: But just report that.

That's all we have to do.

MEMBER ABDEL-KHALIK: And I think from the committee's perspective, we await the staff's evaluation and we will review the staff's evaluation.

MR. BAJOREK: He did make the point that while there was a user need letter, point out and the research was supposed to have responded by I think the end of August. That was the original schedule. But because they amended their own petition, and submitted another petition, OGC decided to lump it together and that window of time has moved out.

CHAIR BANERJEE: Okay. Well with that, I think I'd like to thank you all and adjourn the meeting.

Now, it is unlikely that the combined review of PRM-50-93 and PRM-50-95 adds sufficient complexity and data to justify a one year extension to the deadline for producing the Technical Analysis that is to be the basis of a recommendation to the NRC Commissioners for action on PRM-50-93 and PRM-50-95. Certainly, a substantial amount of review of PRM-50-93 should have been already completed prior to the merging of PRM-50-93 with the recent PRM-50-95.

Robert H. Leyse
P. O. Box 2850
Sun Valley, ID 83353

Rulemaking Comments

From: Bobleyse@aol.com
Sent: Friday, November 26, 2010 12:16 AM
To: Rulemaking Comments; Inverso, Tara
Subject: Comment PRM-50-93 and PRM-50-95
Attachments: The 2200 degree Fahrenheit PCT limit is too high.doc

Comment is attached.

Robert H. Leyse

Received: from mail1.nrc.gov (148.184.176.41) by TWMS01.nrc.gov
(148.184.200.145) with Microsoft SMTP Server id 8.1.393.1; Fri, 26 Nov 2010
00:15:35 -0500
X-Ironport-ID: mail1
X-SBRS: 3.0
X-MID: 27281131
X-fn: The 2200 degree Fahrenheit PCT limit is too high.doc
X-IronPort-Anti-Spam-Filtered: true
X-IronPort-Anti-Spam-Result:
Aj4DAJvR7kxADM4ngmdsb2JhbACCAqENFQEBCwslBxMDH78NhUcE
X-IronPort-AV: E=Sophos;i="4.59,259,1288584000";
d="doc'32?scan'32,208,217,32";a="27281131"
Received: from imr-ma01.mx.aol.com ([64.12.206.39]) by mail1.nrc.gov with
ESMTP; 26 Nov 2010 00:15:36 -0500
Received: from imo-ma04.mx.aol.com (imo-ma04.mx.aol.com [64.12.78.139]) by
imr-ma01.mx.aol.com (8.14.1/8.14.1) with ESMTP id oAQ5FZxc019597; Fri, 26 Nov
2010 00:15:35 -0500
Received: from Bobleyse@aol.com by imo-ma04.mx.aol.com (mail_out_v42.9.) id
f.f16.4d53eff (37085); Fri, 26 Nov 2010 00:15:33 -0500 (EST)
Received: from magic-d21.mail.aol.com (magic-d21.mail.aol.com
[172.19.155.137]) by cia-db06.mx.aol.com (v129.7) with ESMTP id
MAILCIADB067-90dd4cef4274370; Fri, 26 Nov 2010 00:15:32 -0500
From: <Bobleyse@aol.com>
Message-ID: <25f5b.288ee37c.3a209c74@aol.com>
Date: Fri, 26 Nov 2010 00:15:32 -0500
Subject: Comment PRM-50-93 and PRM-50-95
To: rulemaking.comments@nrc.gov, tara.inverso@nrc.gov
MIME-Version: 1.0
Content-Type: multipart/mixed;
boundary="part1_25f5b.288ee37c.3a209c74_boundary"
X-Mailer: AOL 9.1 sub 5009
X-AOL-IP: 68.105.215.152
X-AOL-VSS-CODE: clean
X-AOL-VSS-INFO: 5400.1158/0
X-Spam-Flag: NO
X-AOL-SENDER: Bobleyse@aol.com
Return-Path: Bobleyse@aol.com

Submission ID 26
Mark Leyse
ML110050023

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

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.

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.

Respectfully submitted,

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

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

TABLE OF CONTENTS

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

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

---

[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.

---

[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.

26-1

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.*

---

[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.

26-2

> ... 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.

---

[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].

---

[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

---

[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]

---

[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.

26-3

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.

[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.

---

[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]

---

[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

---

[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.
>
> *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]

---

[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

[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

---

[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.

---

[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]

[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]

26-4

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

---

[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).

---

[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.)

---

[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.

26-5

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.*

26-6

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]

[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]

---

[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.

26-7

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])

---

[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.

26-8

(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.)

---

[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.

---

[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

[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.

---

[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.

26-9

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.)

26-10

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*

---

[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]

---

[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

---

[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.)

26-11

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,

---

[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.)

---

[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.

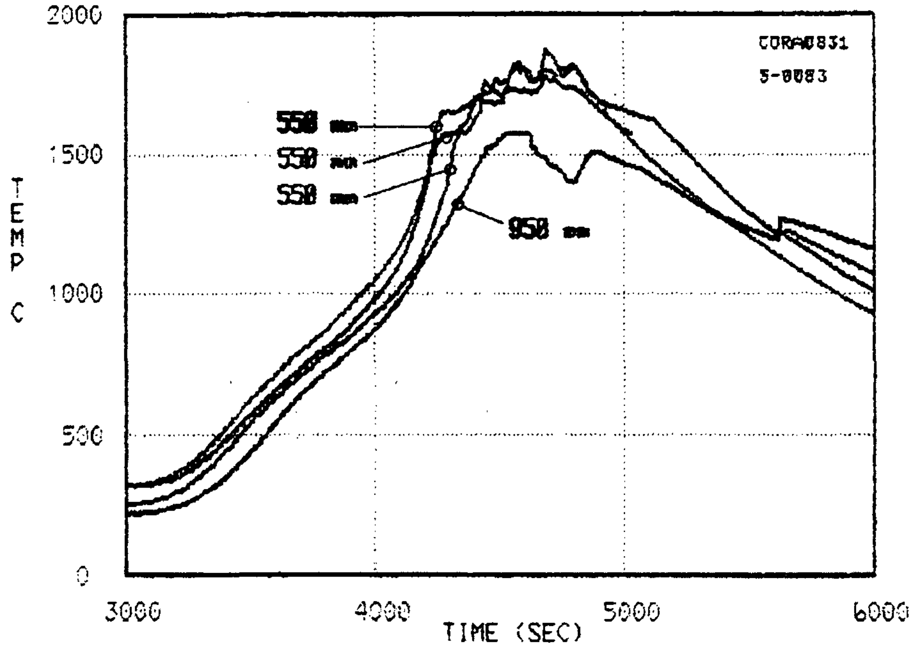
Respectfully submitted,

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

Dated: December 27, 2010

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.



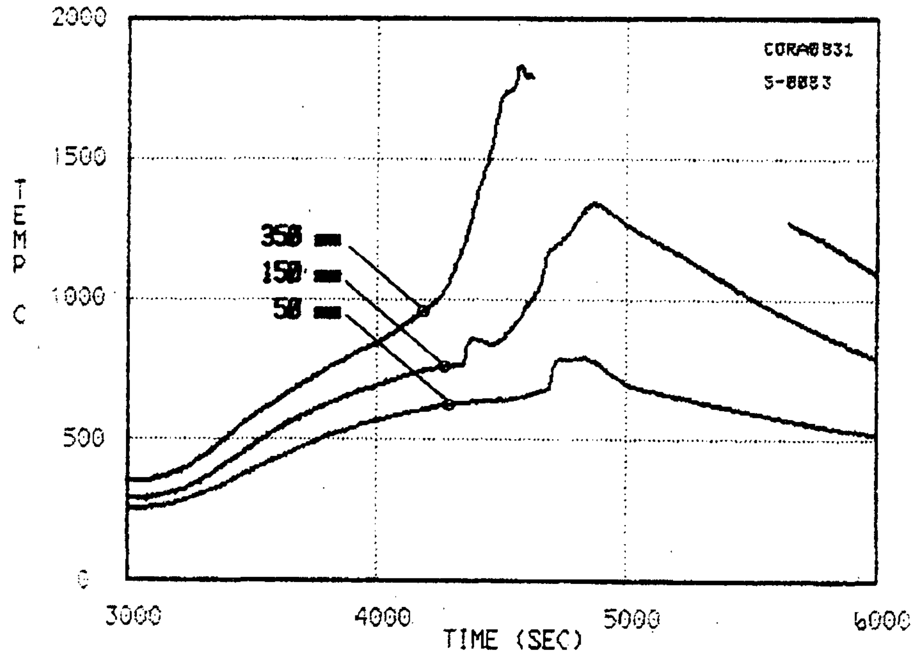
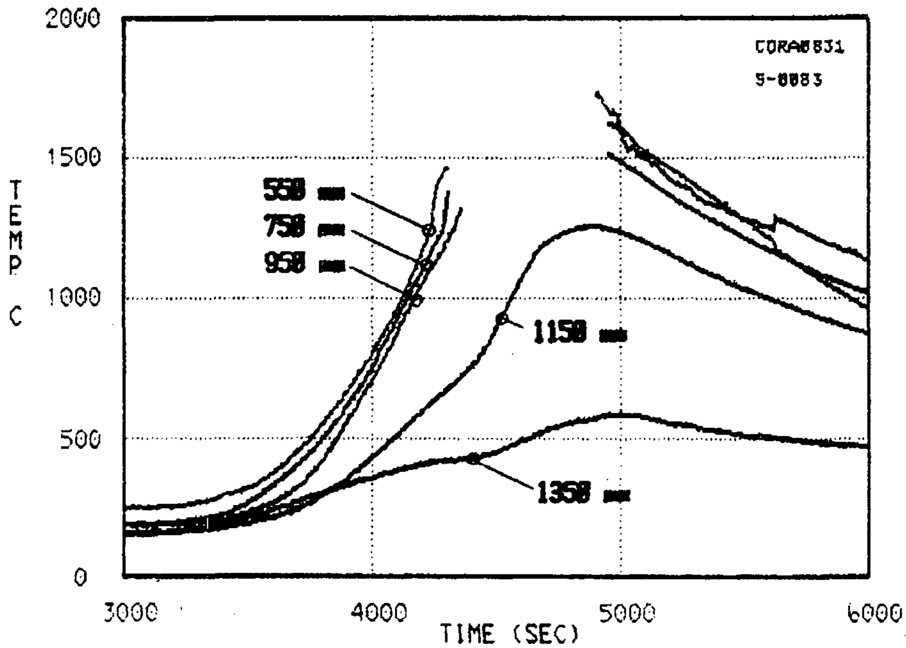


Fig. 16: CORA-16; Temperatures of unhheated rods



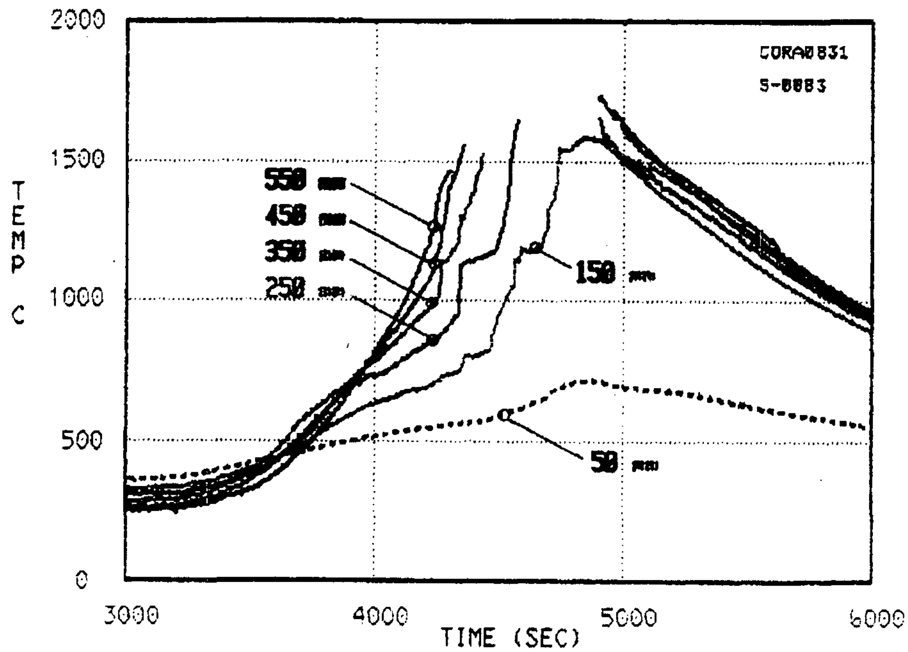


Fig. 17: CORA-16; Temperatures of the absorber blade

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.2.247.2; Mon, 27 Dec 2010
19:51:11 -0500
X-Ironport-ID: mail2
X-SBRS: 4.4
X-MID: 31664414
X-fn: Comment III Response to NEI.pdf
X-IronPort-AV: E=Sophos;i="4.60,237,1291611600";
d="pdf'?scan'208";a="31664414"
Received: from mail-wy0-f169.google.com ([74.125.82.169]) by mail2.nrc.gov
with ESMTP; 27 Dec 2010 19:51:08 -0500
Received: by wyj26 with SMTP id 26so8927359wyj.14 for <multiple
recipients>; Mon, 27 Dec 2010 16:51:08 -0800 (PST)
DKIM-Signature: v=1; a=rsa-sha256; c=relaxed/relaxed;
d=gmail.com; s=gamma;
h=domainkey-signature:mime-version:received:received:date:message-id
:subject:from:to:cc:content-type;
bh=SIRRLfMeqq9AHRG6vcP26QJ+2aElNcBizuBhMVRCBR4=;
b=fk5I5rfQwME51d4mNUU6M3XjxYkqD6sTUFb+8ugs/ymSbttoLdL9JiSiMrWw3fTd5E
A4PoxJGM+m1lkWnj8ST65AnD/Cw8gl2xSjFmY0Ozq5mPyotvDUwOjOp/wlpv2bR3UE2D
WoXYI5VH3rLitLTbod90AxlV7dlg//9JDNN/8=
DomainKey-Signature: a=rsa-sha1; c=nofws;
d=gmail.com; s=gamma;
h=mime-version:date:message-id:subject:from:to:cc:content-type;
b=bx4Fm1YnplMV9nu0EVxEAzwbYbqDiCVRjsxtaHEIuQabmbampriq3d8w94LLb7VGMB
A6W1S5QIOEP36UgZZBZnNk1Ql4uljpp9ioToVLns7URAFJPXJLctrZr7m6w/oOJjIZt5
IDBldnhUA/IWSQNVtFt7kgTgs7GlVsPgvjy48=
MIME-Version: 1.0
Received: by 10.227.199.84 with SMTP id er20mr7515670wbb.194.1293497466927;
Mon, 27 Dec 2010 16:51:06 -0800 (PST)
Received: by 10.227.195.195 with HTTP; Mon, 27 Dec 2010 16:51:06 -0800 (PST)
Date: Mon, 27 Dec 2010 17:51:06 -0700
Message-ID: <AANLkTi=YPcBokoHdrpG6tambhb_Ov0ffuwcqKvxED3j2@mail.gmail.com>
Subject: NRC-2009-0554 (Third)
From: Mark Leyse <markleyse@gmail.com>
To: Rulemaking Comments <rulemaking.comments@nrc.gov>, PDR Resource
<pdr.resource@nrc.gov>
CC: Dave Lochbaum <dlochbaum@ucsusa.org>, necnp@necnp.org,
Raymond Shadis <shadis@prexar.com>, "Powers, Dana A" <dapower@sandia.gov>
Content-Type: multipart/mixed; boundary="000e0cd6e32608bc6a04986dd992"
Return-Path: markleyse@gmail.com

Submission ID 27
Mark Leyse
ML111020046

PRM-50-93 (75FR03876)
PRM-50-95 (75FR66007)

7

DOCKETED
USNRC

April 11, 2011 (11:50 am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 7, 2011

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

Note:

Appendix A to this letter: Report by Nuclear Energy Agency Groups of Experts entitled "In-Vessel and Ex-Vessel Hydrogen Sources" is currently a separate document in ADAMS and non-publically available pending NRC determination that there are no legal restrictions precluding the NRC from making the document available to the public.

Office of the Secretary

TABLE OF CONTENTS

April 7, 2011

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

---

[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 from Thermal-Hydraulic Experiment 1, conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, 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.

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 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 an enforcement action 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 ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a 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.

---

[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 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.

27-1

II. Supplementary Information to PRM-50-93 and PRM-50-95

A. NRC Does Not Acknowledge the Existence of Reports which Explicitly State that Analyses Using the Baker-Just and Cathcart-Pawel Correlations Under-Predict Hydrogen Production in Multi-Rod Bundle Severe Fuel Damage Experiments

"In-Vessel and Ex-Vessel Hydrogen Sources," Part I, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," published in 2001, explicitly states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[7]

In more detail, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I states:

> Reflooding and quenching of the uncovered core is the most important accident management measure to terminate a severe accident transient. If the core is overheated, this measure can lead to increased oxidation of the Zircaloy cladding which in turn can trigger a temperature escalation. Relatively short flooding and quenching times can thereby lead to high hydrogen source rates which must be taken into account in risk analysis and in the design of hydrogen mitigation systems.
>
> Until recently, the experimental database on quenching phenomena was rather scarce. The available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the few available tests (CORA, LOFT LP-FP-2).[8]

This indicates that available Zircaloy-steam oxidation correlations—including the legally-required Baker-Just and Cathcart-Pawel correlations—are not adequate for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

The LOFT LP-FP-2 experiment, conducted in 1985, is considered "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."[9] In the LOFT LP-FP-2 experiment, "[t]he first recorded and qualified rapid temperature rise

---

[7] Report by Nuclear Energy Agency ("NEA") Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I, B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9 (hereinafter: "In-Vessel and Ex-Vessel Hydrogen Sources," Part I).

[8] *Id.*

[9] 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.

associated with the rapid reaction between Zircaloy and water occurred at about...[2060°F]"[10]—approximately 140°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(It is noteworthy that " "GAMA Perspective Statement on In-Vessel Hydrogen Sources," [was] prepared by B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), on the basis of information collected from GAMA [Working Group on the Analysis and Management of Accidents] members and the previous Principal Working Group on Coolant System Behaviour (PWG2). It was endorsed by GAMA in April 2001 and approved for publication by CSNI [Committee on the Safety of Nuclear Installations] in June 2001."[11])

(It is also noteworthy that "[GAMA] is mainly composed of technical specialists in the areas of coolant system thermal-hydraulics, in-vessel protection, containment protection, and fission product retention. Its general functions include the exchange of information on national and international activities in these areas, the exchange of detailed technical information, and the discussion of progress achieved in respect of specific technical issues. Severe accident management is one of the important tasks of the group."[12])

In 2005, NRC denied PRM-50-76,[13] which addressed the fact that the Baker-Just and Cathcart-Pawel correlations are deficient because they were not developed to consider how heat transfer would affect Zircaloy-steam reaction kinetics in the event of a LOCA.[14]

---

[10] 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. 30.
[11] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, p. 5.
[12] *Id.*, p. 3.
[13] NRC, "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-76)," Attachment 1, Federal Register Notice, June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359 (hereinafter "Denial of PRM-50-76," Attachment 1).
[14] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.

In 2005, regarding the fact that data from isothermal tests are used for the development of oxidation correlations, NRC stated:

> For the development of oxidation correlations, limited by oxygen diffusion into the metal, well-characterized isothermal tests are more important than the complex thermal hydraulics suggested by [Robert H. Leyse]. [Robert H. Leyse's] suggested use of complex thermal-hydraulic conditions would be counter-productive in reaction kinetics tests because temperature control is required to develop a consistent set of data for correlation development. Isothermal tests allow this needed temperature control. *It is more appropriate to apply the developed correlations to more prototypic transients (including complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations are indeed limiting.* This is what was done by Westinghouse in WCAP-7665, by Cathcart and Pawel in NUREG-17 and by the NRC in its technical safety analysis of PRM-50-76[15] [emphasis added].

"Denial of PRM-50-76," Attachment 1 states that the Baker-Just and Cathcart-Pawel correlations were used in analyses of prototypic transients (including those with complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations were limiting. Obviously, NRC overlooked the fact that it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[16]

Regarding Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to all four of the FLECHT Zircaloy-clad experiments reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,"[17] "Denial of PRM-50-76," Attachment 1, states:

> The Baker-Just correlation using the current range of parameter inputs is conservative and adequate to assess Appendix K ECCS performance. Virtually every data set published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F.

---

[15] NRC, "Denial of PRM-50-76," Attachment 1, pp. 21-22.
[16] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.
[17] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083 (hereinafter: "PWR FLECHT Final Report").

> [Robert H. Leyse] did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. [Robert H. Leyse] also ignored the Westinghouse application of the Baker-Just correlation to these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients. The NRC applied the Cathcart-Pawel oxide thickness equation to 15 of their transient temperature experiments. The equation was conservative or best-estimate for 13 experiments and non-conservative for the remaining two. This result is consistent with the application of the Cathcart-Pawel equations, which are intended for use in best-estimate LOCA calculations in accordance with [Regulatory Guide] 1.157.[18]

First, as mentioned in PRM-50-93, there is no metallurgical data from the locations of run 9573 that incurred runaway oxidation, because Westinghouse did not obtain such data. So neither Westinghouse nor the NRC applied the Baker-Just correlation to metallurgical data from the locations of run 9573 that incurred runaway oxidation; furthermore, the NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of run 9573 that incurred runaway oxidation.

Second, as discussed in Petitioner's comments on PRM-50-93, dated March 15, 2010, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments, in which local steam starvation conditions are postulated to have occurred[19]—during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so

[18] NRC, "Denial of PRM-50-76," pp. 20-22.

[19] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

So Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to oxide layers on the bundle from FLECHT run 9573 were to locations that most likely were steam starved: those are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in ECCS evaluation calculations.

It is unfortunate that NRC performed such an inadequate technical analysis of PRM-50-76. NRC ignored data from multi-rod bundle severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) that 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. And, as stated above, NRC overlooked the fact it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[20]

Furthermore, NRC ignored ORNL reports from 1990 and 1991, discussing the CORA-16 experiment, which explicitly state that "[c]ladding oxidation was not accurately predicted by available correlations"[21] and that "[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."[22]

---

[20] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

[21] 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.

[22] 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.

27-2

1. Brief Overview of the Isothermal Experiments Used for the Development of the Baker-Just and Cathcart-Pawel Correlations

The Baker-Just correlation—used in Appendix K to Part 50 ECCS evaluation calculations—is primarily based on data from Lemmon and Bostrom's experiments,[23] conducted with inductively heated Zircaloy-2 specimens. In Lemmon's experiments, "Lemmon measured the rates of reaction between Zircaloy-2 and steam in the temperature range 1000-1700°C by inductively heating specimens in steam at 50 psia and measuring the rate of hydrogen evolution."[24] (Bostrom's experiments were conducted in a temperature range above that of design basis accidents: 1300-1860°C.[25]) Lemmon's specimen was a Zircaloy-2 cylinder that was 2 inches long and 0.5 inches in diameter.[26]

(It is noteworthy that in the course of producing his public comments on Petitioner's PRM-50-93, Robert H. Leyse became aware that NRC staff had never studied the basic references of ANL-6548,[27] the report regarding the Baker-Just correlation. In NRC's technical review of PRM-50-76, NRC staff did not review the basic references of ANL-6548. Robert H. Leyse's actions in prompting NRC to acquire the basic references[28] of ANL-6548 are well documented: see the letter from T. J. McGinty to Robert H. Leyse, dated April 16, 2010 (ADAMS Accession Number: ML100950085). Robert H. Leyse submitted Comment 13 on PRM-50-93 (ADAMS Accession Number: ML101020563), emphasizing that PRM-50-93 is based on sound science and that NRC staff had not had access to the reports (discussing experiments that the Baker-Just correlation is primarily based on) cited in ANL-6548, until March 2010.

---

23 G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, 2003, p. 2.

24 V. F. Urbanic and T. R. Heidrick, "High-Temperature Oxidation of Zircaloy-2 and Zircaloy-4 in Steam," Journal of Nuclear Materials 75, 1978, p. 252.

25 *Id.*

26 Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100570218, p. C-4.

27 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.

28 One of which is the report by Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957.

Based on his analysis of the key reports referenced in ANL-6548, that NRC staff had never studied, Robert H. Leyse stated to the ACRS Subcommittee on Plant License Renewal, September 8, 2010 (ADAMS Accession Number: ML102530135), that "[i]t is absurd to license the emergency cooling of tons of zirconium alloy, having thousands of square feet of interfacial surface area, based on the limited investigations that yielded the Baker-Just equation.")

The Cathcart-Pawel correlation—used in best-estimate ECCS evaluation calculations—is based on data from "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies."[29] Cathcart and Pawel's experiments were conducted in two different furnaces with Zircaloy-4 PWR tube specimens. In the MaxiZWOK furnace, the specimen was 18 inches long (only a small segment of that tube—in close proximity to the thermocouple stations—served as the specimen); in the MiniZWOK furnace, the specimen was about 1.2 inches long.[30]

III. CONCLUSION

27-3

It is unfortunate that NRC has overlooked the fact it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments,"[31] and overlooked the fact that ORNL reports from 1990 and 1991 explicitly state that analyses using the available Zircaloy oxidation kinetics models under-predicted the low-temperature (1652-2192°F) oxidation in the CORA-16 experiment.

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 would help improve public and plant-worker safety.

[29] 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.

[30] *Id.*, pp. 12, 15.

[31] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

Respectfully submitted,

Mark Edward Leyse

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

Dated: April 7, 2011

Rulemaking Comments

From:	Mark Leyse [markleyse@gmail.com]
Sent:	Thursday, April 07, 2011 10:24 PM
To:	Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul
Cc:	Robert H. Leyse; Dave Lochbaum; Deborah Brancato; Phillip Musegaas; Raymond Shadis; necnp@necnp.org; Powers, Dana A; Ed Lyman
Subject:	NRC-2009-0554 (Fourth)
Attachments:	NRC-2009-0554 (Fourth).pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Edward Leyse's, Petitioner's, fourth response, dated April 7, 2011, 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.

Sincerely,

Mark Edward Leyse

Submission ID 28
Mark Leyse
ML11209C489

8

Rulemaking Comments

From:	Dudley, Richard
Sent:	Wednesday, July 27, 2011 2:23 PM
To:	Rulemaking Comments; Ngbea, Evangeline
Cc:	Helton, Shana; Bladey, Cindy; Mizuno, Geary; Mensah, Tanya
Subject:	FW: Supplementary Information to the PRB Meeting (July 11, 2011)
Attachments:	Supplement to PRB Teleconference July 11, 2011.pdf

DOCKETED
USNRC

July 27, 2011 (2:55 pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Please post this information as a late comment on the PRM-50-93/95 docket.

Dick Dudley

From: Mark Leyse [mailto:markleyse@gmail.com]
Sent: Thursday, July 21, 2011 2:25 AM
To: Boska, John
Cc: Deborah Brancato; Phillip Musegaas; Dudley, Richard
Subject: Supplementary Information to the PRB Meeting (July 11, 2011)

Dear Mr. Boska:

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

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

Thank you,

Mark Leyse

Template=SECY-067

DS 10

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

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

28-1

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

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

---

[1] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, Abstract.

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

[3] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

[4] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, p. 2152.

[5] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350642, p. 2298.

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

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

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

28-2

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

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

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

---

[6] *Id.*, p. 2299.
[7] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.
[8] *Id.*, p. ix.
[9] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43. See also Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2300.
[10] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2299.

It is reasonable to assume that—as in the CORA-2 and CORA-3 experiments, in which local steam starvation conditions are postulated to have occurred[11]—during PWR FLECHT runs 8874 and 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions occurred, which cannot be detected in the post-test investigation.

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

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

28-3

In the early 1980s, the NRC contracted with National Research Universal at Chalk River, Ontario, Canada to run a series of tests, including the Thermal-Hydraulic Experiment 1, to evaluate the thermal-hydraulic behavior of a full-length Zircaloy 32-rod UO_2 fuel bundle during the heatup, reflood, and quench phases of a large-break LOCA,[12] in the NRU reactor. The TH-1 experiment was conducted with low-level fission heat to simulate decay heat:[13] the average fuel rod power for the tests was 0.37 kW/ft[14] and the peak power was 0.55 kW/ft.[15]

---

[11] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[12] NRC, "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-76)," Attachment 1, Federal Register Notice, June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 18-19.

[13] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic

In a comparison between the data from TH-1 and an example of a prediction (using the Baker-Just correlation) of the behavior of Zircaloy UO_2 fuel rods under LOCA conditions, which is discussed in Westinghouse's "PWR FLECHT Final Report,"[16] it is evident that analyses using the Baker-Just correlation under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128.

In TH-1 test no. 128, with a peak power of 0.55 kW/ft,[17] a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1604°F, the overall PCT was 1991°F (an increase of 387°F).[18] And in the "PWR FLECHT Final Report" example, the UO_2 Zircaloy fuel assembly, with a peak power of 1.24 kW/ft, a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1600°F, was predicted to have an overall PCT of approximately 1880°F (an increase of approximately 280°F).[19]

So with similar parameters (but with a lower fuel rod power) TH-1 test no. 128 had an overall PCT increase that was more than 100°F greater than the overall PCT increase predicted in the UO_2 Zircaloy fuel assembly example discussed in "PWR FLECHT Final Report." This indicates that analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions.

At the same temperatures, analyses using the Cathcart-Pawel correlation predict a lower heat generation rate than analyses using the Baker-Just correlation predict. Therefore, analyses using the Cathcart-Pawel correlation would also under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128.

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

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

[15] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024, pp. 6-13, 6-15 (hereinafter "Safety Analysis Report").

[16] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 4-2, 4-3, 4-4 (hereinafter "PWR FLECHT Final Report").

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

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

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

Analyses using the Baker-Just and Cathcart-Pawel correlations would also most likely under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 130, which I discussed in the last PRB meeting.

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

It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown.

So data from thermal hydraulic experiments indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of loss-of-coolant accidents.

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

Submission ID 29
Mark Leyse
ML11213A211

9

PRM 50-95
(75FR66007)

DOCKETED
USNRC

August 1, 2011 (9:59am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

July 30, 2011

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

Template = SECY-067

DS10

TABLE OF CONTENTS

July 30, 2011

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

---

[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 from Thermal-Hydraulic Experiment 1, conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, 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.

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 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 an enforcement action 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 ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a 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.

II. Supplementary Information to PRM-50-93 and PRM-50-95

29-1 In these comments on PRM-50-93 and PRM-50-95, Petitioner discusses data that indicates that analyses using the Baker-Just and Cathcart-Pawel correlations under-

[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 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.

predict the amount of heat that Zircaloy oxidation generated in Thermal-Hydraulic Experiment 1 ("TH-1") test no. 128, a thermal hydraulic experiment simulating LOCA conditions, conducted with a full-length Zircaloy 32-rod UO_2 fuel bundle.[7]

Analyses using the Baker-Just and Cathcart-Pawel correlations would also most likely under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 130. 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 cladding temperature was 2040°F.[8] So the peak cladding temperature increased by 190°F after the reactor shutdown, because of the heat generated from the metal-water reaction.

It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown.

The data from TH-1 test no. 128 (and most likely also TH-1 test no. 130) is another piece of evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

Perhaps it could be argued that there may have been a problem in TH-1 test no. 128 (although no problems were reported in TH-1 test no. 128[9]) and that therefore it is not certain that analyses using the Baker-Just and Cathcart-Pawel correlations under-

---

[7] NRC, "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-76)," Attachment 1, Federal Register Notice, June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 18-19 (hereinafter "Denial of PRM-50-76," Attachment 1).

[8] 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, p. 13 (hereinafter "Prototypic Thermal-Hydraulic Experiment").

[9] Sometimes thermal hydraulic experiments simulating LOCA conditions reach different overall PCTs, even when they are conducted with similar test parameters. However, "Prototypic Thermal-Hydraulic Experiment" does not report that there were any problems with TH-1 test no. 128 and the data from the TH-1 tests is fairly consistent. See C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882.

predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128. However, any such claims *cannot* be substantiated without conducting more tests with the same parameters as TH-1 test no. 128, to find out if the overall PCT would in fact be more than 100°F lower than it was in TH-1 test no. 128, as analyses using the Baker-Just and Cathcart-Pawel correlations would predict.

Furthermore, in the interest of conservatism and to uphold NRC's congressional mandate to protect the lives, property, and environment of the people of the United States of America, NRC needs to consider the data from TH-1 test no. 128 (and most likely also TH-1 test no. 130), as evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

It is clear that NRC needs to conduct pressurized water reactor ("PWR") thermal hydraulic experiments, simulating LOCA conditions, with a realistic range of different reflood rates, with full-length zirconium alloy[10] multi-rod bundles (comprised of either fuel rods sheathing UO_2 fuel or realistic, pressurized[11] and non-pressurized fuel rod simulators), to investigate the behavior of such bundles when reaching local cladding temperatures of up to 2200°F or higher.

The conductors of such experiments would be able to measure Zircaloy oxidation rates.

A. Analyses Using the Baker-Just Correlation Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in Thermal-Hydraulic Experiment 1 Test No. 128

In this section Petitioner compares data from the TH-1 tests—conducted at National Research Universal ("NRU") at Chalk River, Ontario, Canada to evaluate the

[10] Zircaloy fuel cladding is a particular type of zirconium alloy fuel cladding. ZIRLO and M5 fuel cladding materials are also zirconium alloys; however, they are different zirconium alloys than Zircaloy.

[11] It is noteworthy that "Prototypic Thermal-Hydraulic Experiment" states that experiments with pressurized fuel rod simulators for materials deformation tests would "concentrate on evaluating not only ballooning and rupture but also the added effects on the thermal hydraulic behavior of flow blockage." See C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 2.

thermal-hydraulic behavior of a full-length Zircaloy 32-rod UO_2 fuel bundle during the heatup, reflood, and quench phases of a large-break LOCA[12]—with an example of a prediction (using the Baker-Just correlation) of the behavior of Zircaloy UO_2 fuel rods under LOCA conditions, which is discussed in "PWR FLECHT Final Report."[13] In this comparison, it is evident that analyses using the Baker-Just correlation under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128.

In TH-1 test no. 128, with a peak power of 0.55 kW/ft,[14] a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1604°F, the overall PCT was 1991°F (an increase of 387°F).[15] And in the "PWR FLECHT Final Report" example, the UO_2 Zircaloy fuel assembly, with a peak power of 1.24 kW/ft, a reflood rate of 2.0 in./sec., and a PCT at the onset of reflood of 1600°F, was predicted to have an overall PCT of approximately 1880°F (an increase of approximately 280°F).[16]

So with similar parameters (but with a lower fuel rod power) TH-1 test no. 128 had an overall PCT increase that was more than 100°F greater than the overall PCT increase predicted in the UO_2 Zircaloy fuel assembly example discussed in "PWR FLECHT Final Report." This indicates that analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions.

Data from TH-1 test no. 128 is another piece of evidence that indicates the Baker-Just correlation is not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

---

[12] NRC, "Denial of PRM-50-76," Attachment 1, pp. 18-19.

[13] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 4-2, 4-3, 4-4 (hereinafter "PWR FLECHT Final Report").

[14] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024, pp. 6-13, 6-15 (hereinafter "Safety Analysis Report").

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

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

1. Analyses Using the Cathcart-Pawel Correlation would also Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in TH-1 Test No. 128

As discussed in section II.A., analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128. This means that analyses using the Cathcart-Pawel correlation would also under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128.

Discussing the Baker-Just and Cathcart-Pawel correlations 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," the NRC states:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F...[17]

(It is noteworthy that data from TH-1 test no. 128 indicates that the Baker-Just correlation is *not* substantially conservative at 2200°F. In fact, data from TH-1 test no. 128 indicates that the Baker-Just correlation is non-conservative.)

So at the same temperatures, analyses using the Cathcart-Pawel correlation predict a lower heat generation rate than analyses using the Baker-Just correlation predict. Therefore, analyses using the Cathcart-Pawel correlation would under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions.

Data from TH-1 test no. 128 is another piece of evidence that indicates the Cathcart-Pawel correlation is not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

---

[17] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 3; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.

2. It is Probable that in Addition to TH-1 Test No. 128, Analyses Using the Baker-Just and Cathcart-Pawel Correlations would Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in Other Tests in Thermal-Hydraulic Experiment 1

One of the guidelines for the TH-1 tests was that the fuel cladding temperatures would *not* exceed 1900°F[18]—300°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In three of the TH-1 tests the overall PCTs exceeded 1900°F, exceeding the PCTs predicted for the tests. The overall PCTs of TH-1 test nos. 127, 128, and 130 were 1991°F, 1991°F, and 2040°F, respectively. So it is probable that the Baker-Just and Cathcart-Pawel correlations would under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test nos. 127 and 130.

As discussed in section II., 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 cladding temperature was 2040°F.[19]

a. TH-1 Test No. 130

In Atomic Energy Commission ("AEC") responses to questions submitted by Anthony Z. Roisman, pertaining to the IP-2 licensing hearing, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[20]

Indeed, computer codes using the Baker-Just correlation may calculate that the Zircaloy-steam reaction is negligible below 1900°F; however, experimental data from

[18] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, p. 3-3.
[19] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[20] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

multi-rod thermal hydraulic experiments demonstrates that the Zircaloy-steam reaction is substantial below 1900°F.

Data from TH-1 test no. 130 demonstrates that the Zircaloy-steam reaction is *not* negligible below 1900°F.

In TH-1 test no. 130, there was a peak power of 0.55 kW/ft[21] and a reflood rate of 0.74 in./sec.[22] At the onset of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[23]

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 cladding temperature was 2040°F.[24] 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 runaway oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

(TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average and peak power of TH-1 test no. 130 would have been 0.37 kW/ft[25] and 0.55 kW/ft,[26] respectively, 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.

---

[21] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.
[22] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, Abstract, p. v. The Abstract states that the lowest reflood rate in the TH-1 tests was 1.88 cm/ sec (0.74 in./sec); the Summary states that the lowest reflood rate in the TH-1 tests was 0.74 in./sec; page 13 states that the reflood rate of TH-1 test no. 130 was 0.7 in./sec: so the value of "0.7 in./sec," given on page 13, was rounded off from 0.74 in./sec.
[23] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[24] *Id.*
[25] *Id.*, p. 10.
[26] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.

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.[27]

(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.)

Analyses using the Baker-Just and Cathcart-Pawel correlations would most likely under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 130. It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown.

The data from TH-1 test no. 130 is most likely another piece of evidence that indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

---

[27] 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.

B. A More Detailed Discussion of the Example of a Prediction (Using the Baker-Just Correlation) of the Behavior of Zircaloy UO_2 Fuel Rods under LOCA Conditions

Regarding the example—discussed in section II.A. above—of a prediction (using the Baker-Just correlation) of the behavior of Zircaloy UO_2 fuel rods under LOCA conditions, "PWR FLECHT Final Report" states:

> Figure 4-1[28] shows a comparison of the temperature response of boron nitride-stainless steel (BN-SS), boron nitride-Zircaloy (BN-Zr), and uranium dioxide-Zircaloy (UO_2-Zr) rods for 6 and 2 in./sec. flooding rates. The curves were generated by a conduction code using heat transfer coefficients obtained from stainless steel PWR FLECHT tests.[29] The gap coefficients for the BN and UO_2 cases were 10,000 and 500 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot {}^\circ F^{-1}$, respectively. Initial temperature distributions were assumed to be uniform in the BN cases, whereas a 59°F difference between peak pellet and initial clad temperature was used in the UO_2 cases. Metal-water reaction was predicted in the Zircaloy cases using the Baker-Just parabolic rate equation (reference 4).[30] The BN-SS curves are generally representative of the behavior of Group I and II PWR FLECHT heater rods and were found to be in good agreement with the measured temperature response for the same run conditions. The BN-Zr curves are representative of the behavior of Group III PWR FLECHT rods while the UO_2-Zr curves are representative of reactor fuel rod response, assuming the BN-SS heat transfer coefficients apply.[31]

Figure 4-1., "Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods,"[32] depicts temperature plots of the BN-SS, BN-Zr, and UO_2-Zr representative

---

[28] See Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods.

[29] The heat transfer coefficients obtained from stainless steel PWR FLECHT tests are used in Appendix K to Part 50 ECCS evaluation calculations. Appendix K to Part 50—ECCS Evaluation Models (I)(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized Water Reactors*, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the stainless steel] FLECHT results [reported in "PWR FLECHT Final Report"]."

[30] 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, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050550198.

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

[32] See Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods.

examples for the flooding rates of 6 in./sec. and 2 in./sec. Petitioner will only discuss the representative examples with the 2 in./sec. flooding rate.

For the BN-SS, BN-Zr, and UO_2-Zr representative examples, with the 2 in./sec. flooding rate, the maximum overall PCTs of all three representative examples are lower than 1900°F.

The UO_2-Zr representative example (with a temperature plot drawn with a solid line) has the middle overall PCT value of approximately 1880°F.

The BN-SS and BN-Zr representative examples have overall PCTs of approximately 1840°F and 1890°F, respectively.

Figure 4-1., "Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods" lists some of the parameters for the BN-SS, BN-Zr, and UO_2-Zr representative examples: initial PCT of 1600°F, flooding rate of 2 in./sec., pressure of 60 psia, peak power of 1.24 kW/ft, inlet coolant temperature of 150°F.

C. A Comparison between the TH-1 Tests and the PWR FLECHT Tests

Regarding the fact that the TH-1 tests can be compared to the PWR FLECHT tests, "Safety Analysis Report" states:

> The largest body of information bearing on fuel rewetting or quench is that of the Westinghouse FLECHT experimental series. Cadek (1972) and Rosal (1978) have written reports that describe the experiments and results and cover the same range of reflood rates as in the [TH-1, TH-2, and TH-3] tests proposed for NRU. ...
>
> The NRU [TH-1, TH-2, and TH-3] LOCA [tests] will be quite similar to that of the [PWR] FLECHT tests, with the following major exceptions:
>
> 1) NRU LOCA has nuclear-heated rods; FLECHT has electrically-heated rods.
>
> 2) NRU has Zircaloy-clad rods; FLECHT has stainless steel-clad rods.
>
> 3) NRU has peak-to-average axial power distribution of 1.51; FLECHT's peak-to-average axial power distribution is 1.66.
>
> 4) The NRU test has 32 rods. The tests have different rod surface-to-shroud surface ratios.

> 5) Pre-transient steam cooling in the NRU tests distorts the initial axial temperature distribution.[33]

"Safety Analysis Report" also points out another major difference between the NRU Th-1 tests and the PWR FLECHT tests: "At high cladding temperatures the steam will react with the Zircaloy cladding as given by: $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ [+ ΔH_R]."[34] In which the heat of reaction, ΔH_R, is 559kJ (143 kcal) per mole Zr.[35]

However, the main point of "Safety Analysis Report" is that the TH-1 tests can be compared to the PWR FLECHT tests.

And discussing the goals of the TH-1 tests and other NRU experiments, "Prototypic Thermal-Hydraulic Experiment" states:

> The data [from the NRU experiments] will be used to assess various calculational models for reactor safety analyses and conclusions derived from the large series of electrically heated tests and smaller scale in-pile tests being conducted elsewhere. The test data provide information for evaluating cooling degraded cores as a result of either an accident or an off-normal operating transient. ...
>
> The results of the program will be used to provide data for model calibration or to help define the primary heat transfer mechanisms for new analytical models. The geometry, mass flux, heat capacity, and materials are all prototypic, which eliminates much of the uncertainty of prior test results from other programs. Major concerns of other programs, such as length of fuel bundle or type of heating, [electrical instead of nuclear], should be answered by these test results.[36]

So one of the goals of the TH-1 tests was to use the test data to "assess...conclusions derived from the large series of electrically heated tests."[37]

1. A Comparison of the Results of TH-1 Test No. 107 and PWR FLECHT Run 3724: Tests with Lower PCTs at the Onset of Reflood

It is informative to compare the results of TH-1 test no. 107 and PWR FLECHT run 3724.

[33] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 9-31, 9-32.
[34] *Id.*, p. 9-40.
[35] *Id.*, p. 9-41.
[36] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, pp. 2-3.
[37] *Id.*, p. 2.

Parameters of the two tests:

1) TH-1 test no. 107 (Zircaloy) had a peak power of 0.55 kW/ft,[38] reflood rate of 1.9 in./sec., PCT at the onset of reflood of 1154°F and an overall PCT of 1578°F (an increase of 424°F);[39]

2) PWR FLECHT run 3724 (stainless steel) had a peak power of 1.24 kW/ft, reflood rate of 1.9 in./sec., a PCT at the onset of reflood of 1187°F, and an overall PCT of 1614°F (an increase of 427°F).[40]

TH-1 test no. 107 and PWR FLECHT run 3724, with lower PCTs at the onset of reflood of 1154°F and 1187°F, respectively, had cladding temperature increases of 424°F and 427°F, respectively. So at temperatures where the oxidation of Zircaloy does not produce much heat, the results of TH-1 test no. 107 and PWR FLECHT run 3724 are similar. This indicates that the results of the TH-1 tests can be compared with the results of PWR FLECHT tests.

D. When NRC Denied PRM-50-76, it Overlooked Data which Indicates that Analyses Using the Baker-Just and Cathcart-Pawel Correlations Under-Predict the Amount of Heat that Zircaloy Oxidation Generated in TH-1 Test No. 128

In 2005, NRC denied PRM-50-76,[41] which addressed the fact that the Baker-Just and Cathcart-Pawel correlations are deficient because they were not developed to consider how heat transfer would affect Zircaloy-steam reaction kinetics in the event of a LOCA.[42]

In 2005, regarding the fact that data from isothermal tests are used for the development of Zircaloy-steam oxidation correlations, NRC stated:

> For the development of oxidation correlations, limited by oxygen diffusion into the metal, well-characterized isothermal tests are more important than the complex thermal hydraulics suggested by [Robert H. Leyse]. [Robert H. Leyse's] suggested use of complex thermal-hydraulic conditions would be counter-productive in reaction kinetics tests because temperature

[38] C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208, pp. 6-13, 6-15.
[39] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882, p. 13.
[40] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-5.
[41] NRC, "Denial of PRM-50-76," Attachment 1.
[42] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009.

> control is required to develop a consistent set of data for correlation development. Isothermal tests allow this needed temperature control. *It is more appropriate to apply the developed correlations to more prototypic transients (including complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations are indeed limiting.* This is what was done by Westinghouse in WCAP-7665, by Cathcart and Pawel in NUREG-17 and by the NRC in its technical safety analysis of PRM-50-76[43] [emphasis added].

So "Denial of PRM-50-76," Attachment 1 states that the Baker-Just and Cathcart-Pawel correlations were used in analyses of prototypic transients (including those with complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations were limiting.

29-2

First, as pointed out in Petitioner's comments on PRM-50-93 and PRM-50-95, dated April 7, 2011, NRC overlooked the fact that it was reported in 2001, in an OECD Nuclear Energy Agency report, that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[44]

29-3

Second, NRC overlooked the fact that ORNL reports from 1990 and 1991, discussing the CORA-16 experiment, explicitly state that "[c]ladding oxidation was not accurately predicted by available correlations"[45] and that "[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."[46]

The fact that analyses using the available Zircaloy-steam oxidation correlations under predict the oxidation rates that occur in large-scale integral severe fuel damage

---

[43] NRC, "Denial of PRM-50-76," Attachment 1, pp. 21-22.

[44] Report by Nuclear Energy Agency ("NEA") Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I, B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9.

[45] 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.

[46] 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.

29-4

experiments indicates that the available Zircaloy-steam oxidation correlations are also inadequate for use in ECCS evaluation models predicting the oxidation rates that would occur in the event of a LOCA.

Third, as discussed in sections II., II.A., and II.A.1. above, NRC overlooked the fact that analyses using the Baker-Just and Cathcart-Pawel correlations under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions. TH-1 test no. 128 (with a lower fuel rod power but with otherwise similar parameters) had an overall PCT increase that was more than 100°F greater than the overall predicted PCT increase of the UO_2 Zircaloy fuel assembly example, discussed in "PWR FLECHT Final Report."

Data from TH-1 test no. 128 is another piece of evidence that indicates the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of a design basis accident.

(It is noteworthy that NRC's technical safety analysis of PRM-50-76 lists "Prototypic Thermal-Hydraulic Experiment," which reports on the data of the TH-1 tests, as reference 17.[47]

And noteworthy that NRC's technical safety analysis of PRM-50-76 states:

> NRC has continued to study complex thermal hydraulic effects on ECCS heat transfer processes during accident conditions related to LOCAs[48] consistent with Commission direction. The NRC funded more than 50 Zircaloy clad bundle reflood experiments at the NRU reactor [the program the TH-1 tests were part of].[49, 50]

It is also noteworthy that, in NRC's "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-

---

47 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. 12.

48 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.

49 C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882 and C. L. Mohr, *et al.*, "Safety Analysis Report," NUREG/CR-1208.

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, p. 10.

76)," NRC stated that it was "reviewing...data from [the early '80s, from the program the TH-1 tests were part of,] to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[51])

29-5

Fourth, there is no metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation, because Westinghouse did not obtain such data. So neither Westinghouse nor NRC applied the Baker-Just correlation to metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation; furthermore, NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of the Zircaloy bundles from PWR FLECHT runs 8874 and 9573 that incurred runaway oxidation.[52]

29-6

Fifth, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments, in which local steam starvation conditions are postulated to have occurred[53]—during PWR FLECHT runs 8874 and 9573, the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

So Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to oxide layers on the bundles from PWR FLECHT runs 8874 and 9573 were to locations that most likely were steam starved: those are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in ECCS evaluation calculations.

III. CONCLUSION

It is unfortunate that NRC has ignored data from TH-1 test no. 128, a multi-rod bundle thermal hydraulic experiment, that indicates that the Baker-Just and Cathcart-

---

[51] NRC, "Denial of PRM-50-76," Attachment 1, p. 19.

[52] Westinghouse obtained a total of 13, 2, 15, and 3 metallurgical samples from PWR FLECHT Zircaloy runs 2443, 2544, 8874, and 9573, respectively. See F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," pp. B-2, B-3.

[53] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

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.

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 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: July 30, 2011

Appendix A Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods[1]

[1] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 4-3.

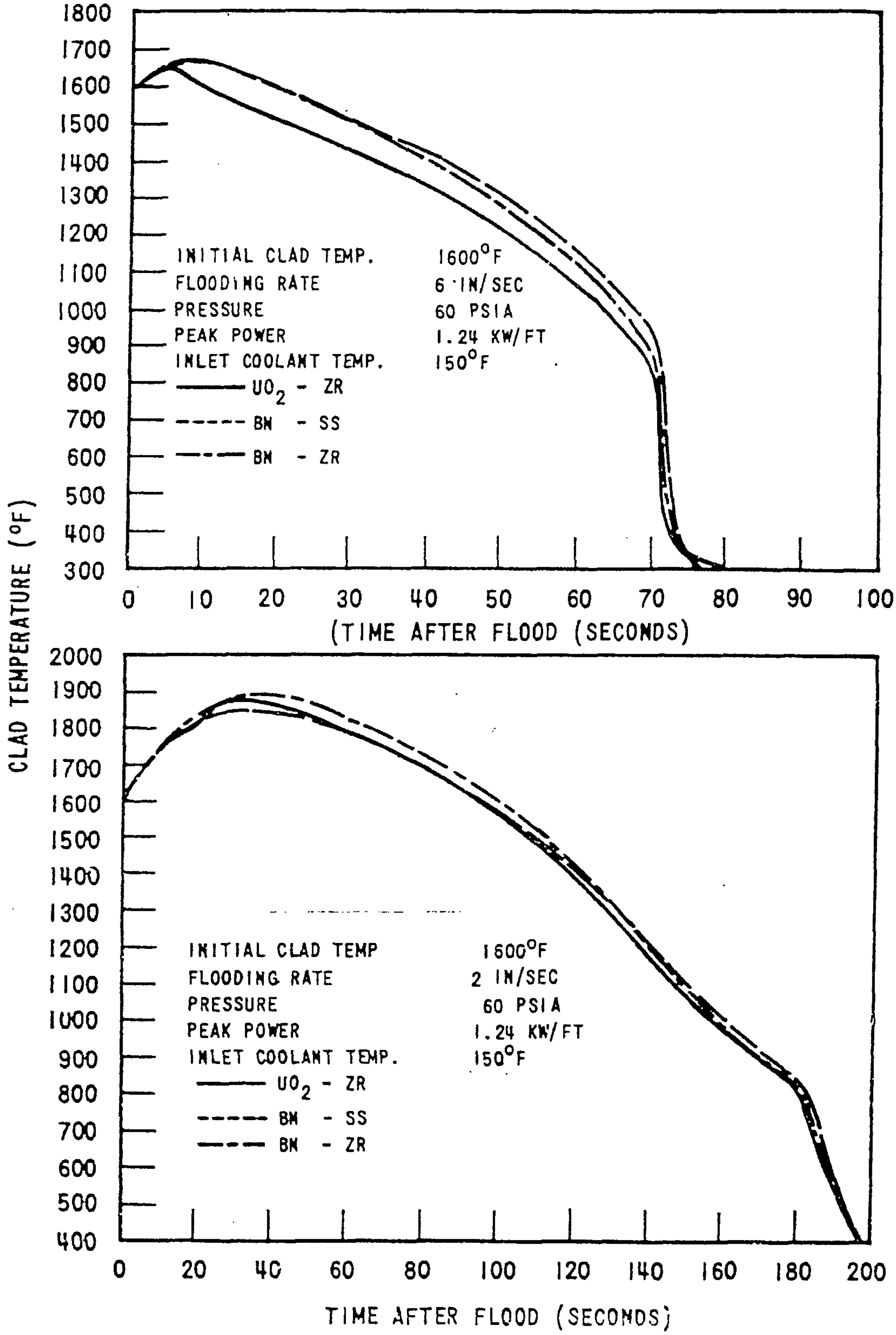


Figure 4-1. Comparison of Thermal Response of PWR-FLECHT and Reactor Fuel Rods

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Saturday, July 30, 2011 6:45 PM
To: Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul
Subject: NRC-2009-05 54 (Fifth)
Attachments: Comment on PRM-50-93 and PRM-50-95 July 2011.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Edward Leyse's, Petitioner's, fifth response, dated July 30, 2011, 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.

Sincerely,

Mark Edward Leyse

Submission ID 30
Mark Leyse
ML12109A084

PRM-50-93
(75FR03876)

PRM-50-95
(75FR66007)

April 16, 2012

DOCKETED
USNRC

April 17, 2012 (9:30 am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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

Template = SECY-067

DS 10

TABLE OF CONTENTS

April 16, 2012

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 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

---

[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 from Thermal-Hydraulic Experiment 1, conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, 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.

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 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 an enforcement action 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 ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a 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.

II. Supplementary Information to PRM-50-93 and PRM-50-95

In section II.A. of these comments on PRM-50-93 and PRM-50-95, Petitioner responds to NRC's "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA

[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 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.

Tests,"[7] regarding the fact that it has been postulated that cladding strain was a factor in increasing the zirconium-steam reaction rates that occurred in the boiling water reactor ("BWR") CORA-16 experiment.[8]

In sections II.A. and II.C. of these comments, Petitioner provides information indicating that cladding strain either had a negligible effect or no effect on increasing the zirconium-steam reaction rates that occurred in the BWR CORA-16 and pressurized water reactor ("PWR") CORA-2 experiments, for which computer safety models using the available zirconium-steam reaction correlations under-predicted zirconium-steam reaction rates.

In section II.B. of these comments, Petitioner provides information indicating that computer safety models using the available zirconium-steam reaction correlations did not accurately predict the oxidation rates that occurred in BWR CORA experiments (in addition to CORA-16), at temperatures above approximately 1922°F and greater.

In section II.D. of these comments, Petitioner provides information from a 2011 IAEA report, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," which states that the zirconium-steam reaction correlations used in computer safety models have limitations; one being that the correlations were derived from experiments that tested zirconium in isothermal conditions—in conditions in which the zirconium specimens were kept at a constant temperature.[9]

30-1

A. Response to NRC's Recent Evaluation of the CORA-16 Experiment

There is experimental data that indicates that currently used zirconium-steam reaction correlations are inadequate for predicting the reaction rates that would occur in a LOCA. For example, when investigators compared the results of the CORA-16 experiment—a BWR severe fuel damage test, simulating a meltdown, conducted with a

[7] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML112211930.

[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] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 11.

multi-rod zirconium alloy bundle—with the predictions of computer safety models, they found that the zirconium-steam reaction rates that occurred in the experiment were under-predicted. The investigators concluded that the "application of the available Zircaloy oxidation kinetics models [zirconium-steam reaction correlations] causes the low-temperature [1652-2192°F] oxidation to be underpredicted."[10]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in the CORA-16 experiment.[11] In NRC's 2011 evaluation of the CORA-16 experiment, NRC stated that an ORNL paper, "In-Vessel Phenomena—CORA," noted that in CORA-16, "cladding strain could be a factor and that cladding strain and significant oxidation occurred simultaneously."[12]

However, NRC erroneously observed that "In-Vessel Phenomena—CORA" "provided an analytical adjustment that improved the timing prediction with respect to the measured temperatures."[13]

In fact, the ORNL paper's authors employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16.[14] There are three graphs in the ORNL paper depicting cladding temperature plots from different cladding elevations (550 mm, 750 mm, and 950 mm) of "heated rod 5.3" in CORA-16:[15] each plot illustrates that cladding temperatures were greater in the experiment than computer safety models—using the available zirconium-steam reaction correlations—initially predicted (*without employing a multiplicative factor*), indicating that zirconium-steam reaction rates were also under-predicted. Each graph also depicts predicted cladding temperature plots that were computer generated by using a simple *multiplier* to *enhance* the predicted zirconium-steam reaction rates (and the amount of heat the zirconium-steam reaction yielded). By using the multiplier the predicted reaction rates

---

[10] 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.
[11] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[12] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[13] *Id.*
[14] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[15] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.

were matched closer to the reaction rates that occurred in the experiment; hence, the multiplier also helped the predicted cladding temperatures match the cladding temperatures that occurred in the experiment.

NRC also erroneously stated that "In-Vessel Phenomena—CORA," did not report that computer safety models under-predicted zirconium-steam reaction rates in CORA-16:[16] a simple glance at the three graphs described above[17] reveals that the paper reported that reaction rates were under-predicted. As explained above, the cladding temperatures were initially under-predicted; hence, the authors of the paper employed a multiplier to enhance the predicted reaction rates. Besides a second ORNL paper explicitly states that the low-temperature (1652°F to 2192°F) oxidation that occurred in CORA-16 was under-predicted.[18] (Petitioner has quoted the second ORNL paper, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," in a number of different comments on PRM-50-93/95 that Petitioner has sent to NRC.)

"In-Vessel Phenomena—CORA" reports that in CORA-16, the *estimated* cladding strain was in the "range of 0.005 to 0.11," at 4200 seconds into the experiment, at locations of the cladding where temperatures were between 1832°F and 2372°F. This certainly does not explain why zirconium-steam reaction rates in the cladding temperature range from 1652°F to 1832°F were under-predicted by computer safety models (as the second ORNL paper reports). It is also unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1832°F to 2192°F.

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, NRC pointed out that an NRC report, NUREG/CR-4412,[19] "explain[s] that under *certain* conditions ballooning and

---

[16] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[17] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[18] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.
[19] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML083400371.

deformation of the cladding can increase the available surface area for oxidation, thus enhancing the apparent oxidation rate" [emphasis not added].[20]

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA or [severe accident] will permit [fuel] cladding deformation (ballooning and possibly rupture) to occur because the fuel rod internal pressure may be greater than the external (coolant) pressure. In this case, oxidation and deformation can occur simultaneously. This in turn may result in an apparent enhancement of oxidation rates because: 1) ballooning increases the surface area of the cladding and permits more oxide to form per unit volume of Zircaloy and 2) the deformation may crack the oxide and provide increased accessibility of the oxygen to the metal. However deformation generally occurs before oxidation rates become significant; *i.e.*, below [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[21]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[22] NUREG/CR-4412 also explains that "it is possible to make a very crude estimate of the expected average enhancement of oxidation kinetics by deformation;"[23] the report provides a graph of the "rather sparse"[24] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[25]

NUREG/CR-4412 has a brief description of the rather sparse data; in one case, two investigators (Furuta and Kawasaki), who heated specimens up to temperatures between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[26]

NUREG/CR-4412 provides other examples of tests by different investigators: 1) tests were conducted in which pressurized tubes were exposed to steam at 1652°F for

---

[20] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[21] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 27.
[22] *Id.*, pp. 27, 30.
[23] *Id.*, p. 30.
[24] *Id.*
[25] *Id.*, p. 29.
[26] *Id.*, p. 30.

up to 30 minutes;[27] 2) tests were conducted in which pressurized tubes were exposed to steam at 1310°F and 1472°F for between five to seven hours;[28] 3) tests were conducted in which specimens that were heated up to between 752°F and 887°F;[29] and 4) Zircaloy-2 ring compression tests were conducted in flowing steam between 1292°F and 2372°F.[30]

Only one pair of investigators (Bradhurst and Heuer) conducted tests that encompassed the entire cladding temperature range—1652°F to 2192°F—in which zirconium-steam reaction rates were reported to be under-predicted for CORA-16. Bradhurst and Heuer reported that "[m]aximum enhancements occurred at slower strain rates. … However, the overall weight gain or average oxide thickness in [the Zircaloy-2 specimens] was only minimally increased because of the localization effects of cracks in the oxide layer." [31] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[32] (In CORA-16, in the cladding temperature range from 1652°F to 2192°F, cladding strain would have occurred over a very brief period of time, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain *may* only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater (in a LOCA environment in which local cladding temperatures would be increasing rapidly).[33]

The graphs in "In-Vessel Phenomena—CORA" depicting cladding temperature plots from different cladding elevations of "heated rod 5.3" in CORA-16 show that the temperature differences between the lower predicted (with no enhancement) and higher

---

[27] *Id.*
[28] *Id.*
[29] *Id.*, p. 27.
[30] *Id.*
[31] *Id.*
[32] F. J. Erbacher, S. Leistikow, "A Review of Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident," Kernforschungszentrum Karlsruhe, KfK 3973, September 1985, p. 6.
[33] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 30.

actual cladding temperatures have a general trend of increasing above approximately 1832°F—up to above 2552°F.[34] Hence, the CORA-16 analysts' use of "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16,[35] primarily enhances predicted zirconium-steam reaction rates at cladding temperatures of approximately 1832°F and greater—up to above 2552°F. As stated above, the graph in NUREG/CR-4412 of the relatively sparse database on the phenomenon of cladding strain enhancing zirconium-steam reaction rates indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[36]

One phenomenon NRC did not consider in its 2011 analysis of CORA-16 is that "[t]he swelling of the [fuel] cladding...alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,"[37] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

(In its comments on the CORA-16 experiment, NRC explains that "[t]he mechanisms causing [oxidation] enhancement are highly unlikely to occur for typical pre-pressurized [fuel] rods, which will deform and rupture before the oxidation rate is significant."[38] NRC is correct; for example, in a PWR LB LOCA, the ballooning of the

---

[34] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[35] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[36] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 29.
[37] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041950059, p. 43.
[38] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.

cladding could commence at a local cladding temperature of approximately 1225°F[39] and rupture at local cladding temperatures between 1290°F and 1470°F,[40] temperatures below those at which zirconium-steam reaction rates become rapid. In CORA-16, the test rods' internal pressures were in a range from 4.6 to 6.1 bar (66.7 to 88.5 psi) (far lower than the internal pressures of fuel rods in a reactor core) and the external system pressure, outside of the test rods was 2.2 bar (31.9 psi); hence, there was not much of a difference between the internal and external pressures, which explains why cladding strain and rupture occurred at higher temperatures in CORA-16.)

A plausible explanation for why zirconium-steam reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1652°F to 2192°F by computer safety models would be that the currently used zirconium-steam reaction correlations are inadequate for use in computer safety models.

ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F cladding temperature range for CORA-16. The analysts seem to have merely assumed that the available zirconium-steam reaction correlations could not possibly be inadequate for use in computer safety models; hence, they did not seem to think it was necessary to support and confirm their *estimates* of cladding strain enhanced zirconium-steam reaction rates with solid experimental data.

In NRC's 2011 evaluation of CORA-16, NRC concluded that the fact zirconium-steam reaction rates were under-predicted by computer safety models—using the available zirconium-steam reaction correlations—"is inadequate as a basis to revise regulations or invalidate the use of [the] Baker-Just and Cathcart-Pawel [correlations] for

---

[39] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041950059, p. 43.

[40] NRC, "Acceptance Review of Proposed Generic Issue on Dispersal of Fuel Particles During a Loss-of-Coolant Accident," October 21, 2011, Enclosure, "Fuel Dispersal During a LOCA: Generic Issue Proposal," located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML112910156, p. 2.

design basis calculations of oxidation."[41] (The Baker-Just and Cathcart-Pawel correlations are among the available zirconium-steam reaction correlations.)

NRC's conclusion is unsubstantiated, as the information presented in this section indicates. When NRC chooses to invalidate experimental data, which is important for simulating accidents, with unsubstantiated postulations, NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative computer safety models.[42]

B. Concurrent Cladding Strain and Oxidation has Been Reported to have Occurred in the BWR CORA Experiments as a Whole

In NRC's 2011 evaluation of the CORA-16 experiment, NRC notes that "In-Vessel Phenomena—CORA," states that computer safety models using the available zirconium-steam reaction correlations accurately predicted the zirconium-steam reaction (oxidation) rates that occurred in the CORA-17 experiment.[43] NRC is correct; "In-Vessel Phenomena—CORA" also states that "cladding strain *was not* a factor in the CORA-17 experiment" [emphasis not added].[44] However, a 1997 ORNL paper states that *concurrent cladding strain and oxidation* occurred in the BWR CORA experiments as a whole. The 1997 ORNL paper discusses *all* of the BWR CORA experiments—CORA-16, -17, -18, -28, -31, and -33—and states that "concurrent cladding strain and oxidation in the β Zircaloy phase regime [which commences above approximately 1922°F] must be considered in the experimental analysis" of the BWR CORA experiments.[45] In other words, the 1997 ORNL paper claims that concurrent cladding strain and oxidation caused oxidation rates to be *enhanced* in the BWR CORA experiments as a whole.

---

[41] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[42] Charles Miller, *et al.*, NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," SECY-11-0093, July 12, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML111861807, p. 3.
[43] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[44] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[45] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," ORNL, 1997, p. 4.

(Perhaps CORA-17 was an exception (not having enhanced oxidation) or perhaps CORA-17 was reanalyzed and found to have had enhanced oxidation, after all. It would also seem that the CORA-33 experiment might be an exception, because CORA-33 was conducted under relatively steam-starved conditions; that is, with "minimal steaming;" in CORA-33, there was "[n]o temperature escalation as a result of [the] limited steam input."[46] Nonetheless, in CORA-33, "[c]oncurrent cladding strain and oxidation in the zircaloy β phase [which commences above approximately 1922°F] occurred"[47] and "the computed cladding strain was significant over 400 mm [15.75 inches] of the rod length."[48])

If concurrent cladding strain and oxidation caused oxidation rates to be *enhanced* in some of the BWR CORA experiments (in addition to CORA-16), that indicates computer safety models using the available zirconium-steam reaction correlations did *not* accurately predict the oxidation rates that occurred in BWR CORA experiments (in addition to CORA-16), at temperatures above approximately 1922°F and greater. Furthermore, if "concurrent cladding strain and oxidation in the β Zircaloy phase regime [which commences above approximately 1922°F] must be considered in the experimental analysis" of the BWR CORA experiments, it follows that the BWR CORA experiment analyses (in addition to the analysis of CORA-16) employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation."[49]

C. For the PWR CORA-2 Experiment, the Thickness of Oxide Layers Was Under-Predicted at Locations that Did Not have Cladding Ballooning

A computer safety model (a CORA experiment-specific, modified version of SCDAP/MOD1[50]) using available zirconium-steam reaction correlations, under-predicted the thickness of oxide layers that occurred at different locations of the multi-rod bundle

[46] L. J. Ott, Siegfried Hagen, "Interpretation of the Results of the CORA-33 Dry Core Boiling Water Reactor Test," Nuclear Engineering and Design, 167, 1997, p. 291.
[47] *Id.*, p. 297.
[48] *Id.*, p. 298.
[49] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[50] K. Minato *et al.*, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," KfK 4827, July 1991, p. 10.

that was used in the PWR CORA-2 experiment.[51] A 1991 Kernforschungszentrum Karlsruhe ("KfK") report, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," discussing CORA-2, states that "[a] comparison...shows that the measured oxide layers were thicker than those of the calculation."[52]

(It is important to clarify that the computer simulation of CORA-2 predicted the growth of the oxide layer thicknesses up to 5010 seconds, the point at which, in the simulation, the growth of the oxide layers ceased,[53] because "[d]ue to the preceding melt relocation, a complete consumption of the [Zircaloy] stopped the oxidation."[54]

It is possible that the predicted oxide layer thicknesses would have been thicker in the computer simulation if the melt relocation had not consumed the Zircaloy and stopped the oxidation at 5010 seconds. However, in a scenario without a complete consumption of the Zircaloy, if the computer simulation accurately simulated the quantity of steam that would have been available in CORA-2 after 5010 seconds, it is likely that the oxidation would have either been insignificant or would have stopped, anyway, because there would not have been much (if any) available steam. In the actual CORA-2 experiment at 5010 seconds, the oxidation of the Zircaloy would have either been insignificant or would have stopped, because there would not have been much (if any) available steam. In CORA-2, the steam flow rate of 6 grams per second was terminated at 4600 seconds.[55]

The progression of steam availability in CORA-2 is as follows: in the beginning phase of CORA-2 there was an argon flow rate of 10 grams per second through the test bundle; at 3300 seconds into the experiment, there was an argon flow rate of 4 grams per second and steam flow rate of 6 grams per second; at 4600 seconds into the experiment, the steam flow was turned off and the argon flow rate increased to 10 grams per second;

---

[51] *Id.*, Appendix E, Figures 10, 11, and 12.
[52] *Id.*, p. 10.
[53] *Id.*, Appendix E, Figure 12.
[54] *Id.*, p. 10.
[55] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 54.

at 4900 seconds the electrical power of the bundle was shutoff and the cool-down phase commenced.[56])

For CORA-2, the *calculated* maximum thickness of oxide layers was approximately 0.40 mm at the 350 mm elevation of the test bundle.[57] And for CORA-2, on non-ballooned locations of the test bundle, the thicknesses of oxide layers were *measured* at 0.52 and 0.54 mm (at the 268 mm elevation), 0.54 mm (at the 298 mm elevation), and 0.52 mm (at the 480 mm elevation).[58] The oxide layers which were measured at 0.54 mm thick were 35 percent thicker than the predicted the maximum thickness of 0.40 mm. Furthermore, the *calculated* maximum thicknesses of oxide layers were less than 0.40 mm at the locations/elevations at which oxide layers were *measured* at 0.52 and 0.54 mm thick.

Therefore, a computer safety model using available zirconium-steam reaction correlations, significantly under-predicted oxide layer thicknesses at a number of non-ballooned locations of the CORA-2 bundle. However, there are potential problems with making a comparison of the measured and predicted oxide layer thicknesses of CORA-2, because, as mentioned above, in the computer simulation, at 5010 seconds, the growth of the oxide layers ceased, because a melt relocation consumed the Zircaloy and stopped the oxidation. Yet, in the actual CORA-2 experiment, after 5010 seconds, it is likely that the oxidation would have either been insignificant or would have stopped, because there would not have been much (if any) available steam.

Furthermore, because a computer safety model using available zirconium-steam reaction correlations, significantly under-predicted oxide layer thicknesses at a number of non-ballooned locations of the CORA-2 bundle, it means that the zirconium-steam reaction rates were also significantly under-predicted at a number of non-ballooned locations of the CORA-2 bundle. The CORA-2 data indicates that it cannot be legitimately claimed that cladding stain increased zirconium-steam reaction (oxidation) rates at a number of locations of the CORA-2 bundle.

---

[56] *Id.*
[57] K. Minato *et al.*, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," p. 10.
[58] *Id.*, Appendix E, Figure 11.

The CORA-2 data calls into question the validity of the postulation that cladding strain was a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16 and the BWR CORA experiments as a whole. The CORA-2 data also calls into question the validity of the claim that currently used zirconium-steam reaction rate correlations are adequate for use in computer safety models. Furthermore, the CORA-2 data calls into question the validity of the claim that data from single rod tests—in which a tiny specimen is held at a constant temperature—is adequate for deriving the zirconium-steam reaction correlations used in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA.

1. Additional Information on the PWR CORA-2 Experiment

The 1991 KfK report states that in CORA-2, there was a slight circumferential elongation that occurred at the non-ballooned locations of the test bundle. The intact (non-ballooned) cladding at the 268 mm, 298 mm, and 480 mm elevations had "a circumferential elongation widening the gap between the pellet and the cladding."[59] It is postulated that the "elongation [was] caused by the high inner rod pressure and/or the volume growth due to the oxidation of the [zirconium alloy] cladding."[60] Therefore, it is possible that the slight circumferential elongation that occurred at the non-ballooned locations of the CORA-2 bundle was *only* due to "the volume growth due to the oxidation of the...cladding."[61] In CORA-2, it is also possible that the cladding deformation was caused either solely by the inner rod pressure or caused by both the volume growth due to oxidation and the inner rod pressure.

The 1991 KfK report also states that the "process of [ballooning] starts with a slight circumferential elongation over nearly the whole length of the rod, widening the gap between pellet and cladding. Then the cladding balloons in the hot region of the rod until it ruptures due to mechanical stress at the hottest azimutal position."[62] (The 1991 KfK report neither states the temperature at which the pellet-cladding gap widening would commence nor states the temperature at which the pellet-cladding gap widening

[59] *Id.*, p. 12.
[60] *Id.*
[61] *Id.*
[62] *Id.*, p. 29.

would mostly cease, at non-ballooned locations of the of the rod. The 1991 KfK report also does not state what the time duration would be between the time the pellet-cladding gap widening commenced and mostly ceased, at non-ballooned locations of the of the rod.)

In CORA-2, the initial value of the outer diameter of the cladding was 10.75 mm. After the CORA-2 experiment was conducted, for four rods that were intact at certain elevations, the outer diameter of the cladding was *measured* at 11.3-12.5 mm and 11.5-12.0 mm on two separate rods (at the 268 mm elevation); at 11.5-12.0 mm and 11.3-12.3 mm on two separate rods (at the 298 mm elevation); and at 11.5-12.3 mm on one rod (at the 480 mm elevation). This means that the measured percentage increase of the circumferential elongation for intact cladding was limited between values of 5 and 16 percent.[63]

D. A 2011 IAEA Report States that the Zirconium-Steam Reaction Correlations Used in Computer Safety Models have Limitations

30-2

A 2011 IAEA report states that the zirconium-steam reaction correlations used in computer safety models have limitations; one being that the correlations were derived from experiments that tested zirconium in isothermal conditions—in conditions in which the zirconium specimens were kept at a constant temperature.[64]

Such experiments use tiny zirconium specimens heated in a steam environment to investigate the reaction rates of zirconium in steam. The experiments are termed "single rod tests," which is a misnomer, because the specimens used in the experiments are tiny segments of fuel cladding, usually about one or two inches long.

To perform a test at a constant temperature (at higher temperatures—above 1800°F—at which the zirconium-steam reaction generates a great deal of heat), it is necessary that the specimen be a single rod, because a single rod will have radiative heat losses to its surrounding cooler environment, which removes the heat generated by the zirconium-steam reaction. This allows the specimen to be held at a constant temperature when the specimen temperature exceeds 1800°F, because in such experiments "any

[63] *Id.*, p. 48.

[64] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.

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 [specimen]."[65]

It would not be possible to conduct an experiment under isothermal conditions—at a constant temperature—with a multi-rod zirconium alloy bundle at temperatures above 1800°F, because the heat generated from the exothermic reaction of the zirconium in steam would be overpowering and cause the local bundle temperatures to increase rapidly. Although it is not explicitly stated, one could argue that the 2011 IAEA report essentially points out that one of the limitations of the experiments used to derive zirconium-steam reaction correlations is that they were single rod experiments, not more realistic multi-rod bundle experiments.

(It is noteworthy that in 1971, Daniel Ford of Union of Concerned Scientists pointed out that computer safety models use a zirconium-steam correlation[66] "derived from experimental data...completely outside of the context of nuclear systems"[67]—from small-scale experiments conducted with tiny zirconium alloy specimens that were held at a constant temperature.)

The 2011 IAEA report also states that a consequence of the zirconium-steam reaction correlations—used in computer safety models—being derived from experiments that tested zirconium in isothermal conditions is that "[t]he temperature gradient is less than [9°F per second] to use the correlation[s] for transient conditions."[68] If this is true, it means that computer safety models using such correlations could perhaps only accurately predict zirconium-steam reaction rates for LOCA conditions in which the local fuel cladding temperature would increase at a rate of *less* than 9°F per second. Unfortunately, the 2011 IAEA report does not discuss how inaccurately such correlations predict

---

[65] 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, pp. 118-119.

[66] The Baker-Just correlation.

[67] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, p. 2551.

[68] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.

reaction rates for LOCA conditions in which the local fuel cladding temperature would increase at a rate of *greater* than 9°F per second.

The 2011 IAEA report states that "post test calculations of temperature-transient experiments...have confirmed the use of [zirconium-steam reaction] correlations under...conditions [in which the local fuel cladding temperature would increase at a rate of less than 9°F per second];"[69] however, it does not provide information on either how accurate or inaccurate calculations were for experiments in which the local fuel cladding temperature increased at a rate of greater than 9°F per second. (The report merely states that "the reaction rate...can differ with time/temperature during the transient."[70])

In a PWR LB LOCA, the maximum local cladding temperature could possibly increase as rapidly as 30°F per second[71] (caused by the residual heat (stored energy) in the fuel, decay heating, and at higher temperatures, heat generated by the zirconium-steam reaction); for example, a computer simulation of a LB LOCA occurring at Indian Point Unit 2, *predicted* that cladding temperatures would increase from about 600°F to above 2100°F in about 50 seconds, indicating an *average* cladding temperature increase of about 30°F per second. The information in the 2011 IAEA report seems to imply that if cladding temperatures were to increase at a rate of 30°F per second that computer safety model predictions of reaction rates would be inaccurate.

An example of "the reaction rate...differ[ing] with time/temperature during [a] transient,"[72] is the progression of the zirconium-steam reaction rates that occurred in the LOFT LP-FP-2 experiment, the only severe fuel damage experiment conducted with actual decay heat.[73] The initial heat up rate of the fuel cladding in LOFT LP-FP-2 was

---

[69] *Id.*

[70] *Id.*

[71] A plot of maximum cladding temperatures derived from a computer simulation of a LB LOCA occurring at Indian Point Unit 2, depicts cladding temperatures increasing from about 600°F to above 2100°F in about 50 seconds, indicating an *average* cladding temperature increase of about 30°F per second; see Entergy, "Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate (TAC MC1865)," August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, Attachment 1, p. 2.

[72] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.

[73] 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.

approximately 1.8°F per second.[74] At high cladding temperatures at which the zirconium-steam reaction became rapid, the local heat up rate of the fuel cladding began increasing. For example, at one location on the central fuel bundle (at the 42-inch elevation) when cladding temperatures had reached just below 2200°F, the fuel cladding heat up rate had increased to approximately 21.4°F per second;[75] at the same location, between approximately 2200°F and 2780°F, the *average* heat up rate was approximately 36.3°F per second.[76] Fuel cladding temperatures were also rapidly increasing at other locations, indicating that zirconium-steam reaction rates were increasing at a number of locations in the fuel bundle.[77]

It should be clarified that the 2011 IAEA report does opine that the two zirconium-steam reaction correlations—the Baker-Just and Cathcart-Pawel correlations—that computer safety models use for NRC's legally-binding simulations of LOCAs are reliable for intact fuel cladding.[78] However, there is experimental data that indicates that currently used zirconium-steam reaction correlations are inadequate for use in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA. (Such data is discussed above in sections II.A. and II.C. of these comments.)

For example, the fact that computer safety models using the available zirconium-steam reaction correlations under-predicted the reaction rates that occurred in CORA-16, in the cladding temperature range from 1652°F to 2192°F, indicates that the currently used correlations are inadequate for use in computer safety models that simulate LOCAs. Furthermore, the results of CORA-16 indicate that data from single rod tests—in which a tiny specimen is held at a constant temperature—is inadequate for deriving the zirconium-steam reaction correlations used in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA.

---

[74] *Id.*
[75] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," 2011, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML112650009, p. 4.
[76] *Id.*, p. 5.
[77] *Id.*, pp. 3-5.
[78] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 8.

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,

/s/

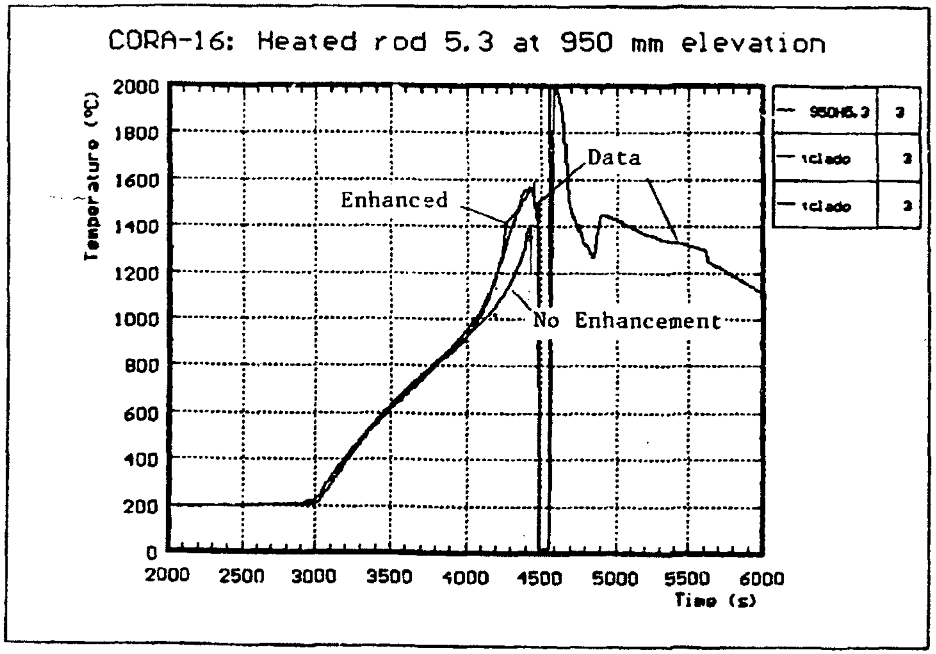
Mark Edward Leyse
P.O. Box 1314
New York, NY 10025
markleyse@gmail.com

Dated: April 16, 2012

Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations[1]

[1] 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.

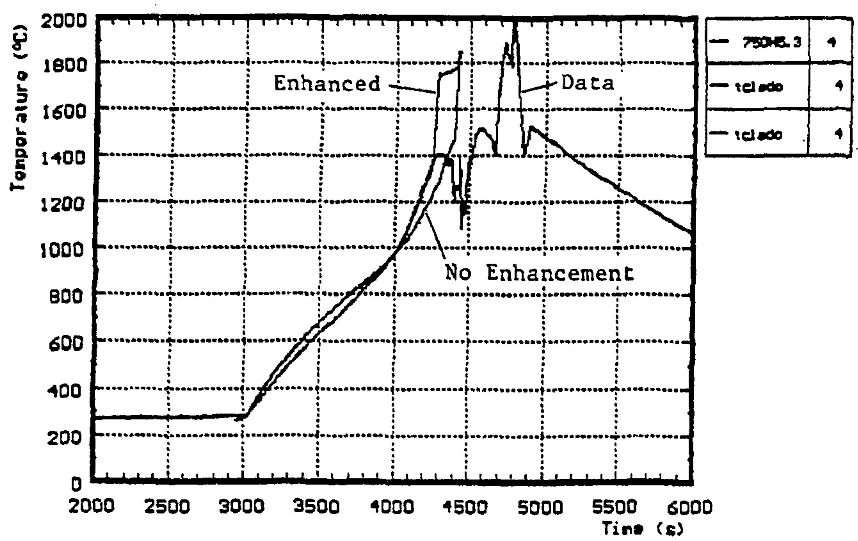
Use Of A Simple Multiplicative Factor (Function Of Strain) To Enhance The Zircaloy Oxidation Yields Reasonable Predictions For CORA-16



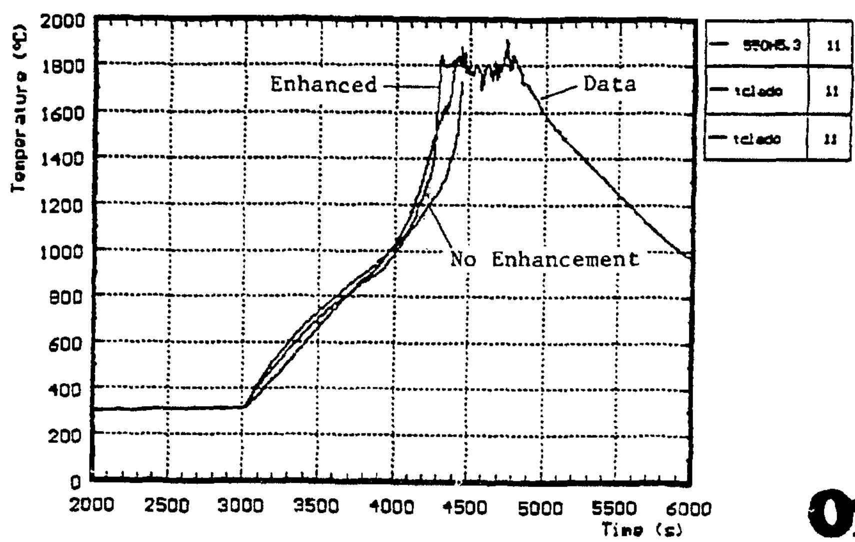
ornl

Strain Enhanced Zircaloy Oxidation
(Continued)

CORA-16: Heated rod 5.3 at 750 mm elevation



CORA-16: Heated rod 5.3 at 550 mm elevation



ornl

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Monday, April 16, 2012 11:45 PM
To: Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul; CHAIRMAN Resource
Cc: Robert H. Leyse; Christopher Paine; Thomas B. Cochran; Weaver, Jordan; Matthew G. McKinzie; Nuclear; Dave Lochbaum; Ed Lyman
Subject: NRC-2009-0□554 (Sixth)
Attachments: Final April 2012 COMMENTS ON PRM-50-93 and PRM-50-95.pdf

Dear Ms. Vietti-Cook:
Attached to this e-mail is Mark Leyse's, Petitioner's, sixth response, dated April 16, 2012, 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, among other things, Petitioner responds to NRC's "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," regarding the fact that it has been postulated that cladding strain was a factor in significantly increasing the zirconium-steam reaction rates that occurred in the BWR CORA-16 experiment. As discussed in these comments, there appears to be no data to support such a postulation.
Sincerely,
Mark Leyse

Submission ID 31
Mark Leyse
ML13031A698

PRM-50-93 & 50-95
(75FR66007)

DOCKETED - USNRC
January 30, 2013 (6:39 a.m.)
OFFICE OF THE SECRETARY - RULEMAKINGS AND ADJUDICATIONS STAFF

Speiser, Herald

From: Mark Leyse [markleyse@gmail.com]
Sent: Wednesday, January 30, 2013 6:39 AM
To: RulemakingComments Resource; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul; Doyle, Daniel; Bladey, Cindy
Cc: Robert H. Leyse; Christopher Paine; Thomas B. Cochran; Weaver, Jordan; Matthew G. McKinzie; Nuclear; Dave Lochbaum; Ed Lyman; shadis@prexar.com; necnp@necnp.org; Paul Gunter; Deborah Brancato; Phillip Musegaas
Subject: NRC-2009-0554 (Seventh)
Attachments: January 30, 2013 COMMENTS ON PRM-50-93 AND PRM-50-95.pdf

Dear Ms. Vietti-Cook:
Attached to this e-mail is Mark Leyse's, Petitioner's, seventh response, dated January 30, 2013, 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.

Sincerely,
Mark Leyse

SECY-067

DS10

January, 30, 2013

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

31-1

A Few Issues Raised in PRM-50-93 the Technical Staff Has Overlooked, Covered Briefly

I. Runaway Oxidation (Thermal Runaway of Fuel Cladding Temperatures) Has Commenced below 2200°F

Regarding the 2200°F 10 C.F.R. § 50.46(b)(1) fuel peak cladding temperature ("PCT") limit, in NRC's October 2012 Draft Interim Review of PRM-50-93/95, NRC concludes:

> [A]utocatalytic reactions have not occurred at temperatures less than 2200 degrees F. Accordingly, the 2200 degree F regulatory limit is sufficient provided the correlations used to determine the metal-water reaction rate below 2200 degrees F are suitably conservative such that excessive reaction rates do not occur below that value.[1]

In PRM-50-93/95 and in comments on PRM-50-93/95, Petitioner submitted information stating that runaway (autocatalytic) zirconium-steam reactions ("runaway oxidation") *have* commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit. For example, PRM-50-93 (pages 46-47) quotes an OECD Nuclear Energy Agency report, which states that runaway oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater.[2] In NRC's October 2012 Draft Interim Review of PRM-50-93/95, NRC neither discusses nor mentions such information.

Interestingly, an NRC document, "Perspectives on Reactor Safety," states that in a postulated station blackout scenario at Grand Gulf, runaway zirconium oxidation would commence at 1832°F.[3] (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

---

[1] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12265A277, p. 2.

[2] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," Executive Summary, February 2000, p. 9. (Regarding the statement that runaway (autocatalytic) oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater, "Degraded Core Quench: Summary of Progress 1996-1999" explicitly states that "[a] notable feature of the [QUENCH] experiments was the occurrence of temperature excursions starting in the unheated region at the top of the shroud, from temperatures of 750-800°C, which is more than 300 K lower than excursion temperatures associated with runaway oxidation by steam.")

[3] NRC, "Perspectives on Reactor Safety," NUREG/CR-6042, Rev. 2, March 2002, available at: NRC's ADAMS Documents, Accession Number: ML021080117, pp. 3.7-4, 3.7-5, 3.7-29.

31-2

Furthermore, in NRC's own September 2011 Draft Interim Review of PRM-50-93/95, NRC presented data demonstrating that runaway oxidation commenced in the LOFT LP-FP-2 experiment when fuel-cladding temperatures were lower than 2200°F. (In PRM-50-93 (pages 27, 33, 41, 42), Petitioner quoted a Pacific Northwest Laboratory paper, which states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec [18°F/sec], signal[s] the onset of an autocatalytic oxidation reaction."[4] This is for cases in which there would be relatively low initial heatup rates—for example, 1.0 K/sec (1.8°F/sec)—followed by substantially higher heatup rates, caused by the contribution of heat generated by the exothermic oxidation reaction.) In NRC's September 2011 Draft Interim Review of PRM-50-93/95, NRC presented data stating that in LOFT LP-FP-2, when local temperatures reached 1477 K (2199.2°F), the heatup rates at two fuel-cladding locations (TE-5C07-042 and TE-5D13-042) were 10.3 K/sec (18.5°F/sec) and 11.9 K/sec (21.4°F/sec), respectively.[5]

Hence, NRC's October 2012 Draft Interim Review of PRM-50-93/95 overlooks data that NRC provided in September 2011 demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F PCT limit. Clearly, NRC needs to correct its erroneous conclusion that runaway oxidation has not commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

It is noteworthy that a report regarding best-estimate predictions for LOFT LP-FP-2 states that runaway oxidation would commence if fuel-cladding temperatures were to start increasing at a rate of 3.0 K/sec (5.4°F/sec);[6] this is for cases in which there would be relatively low initial heatup rates. (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

---

[4] 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, available at: NRC's ADAMS Documents, Accession Number: ML042230126, p. 282.

[5] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," September 2011, available at: NRC's ADAMS Documents, Accession Number: ML112650009, p. 4.

[6] S. Guntay, M. Carboneau, Y. Anoda, "Best Estimate Prediction for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3803, June 1985, available at: NRC's ADAMS Documents, Accession Number: ML071940361, p. 38.

II. Computer Safety Models Are Unable to Determine the Increased Hydrogen Production Which Occurred in the CORA and LOFT LP-FP-2 Experiments

31-3

Regarding the CORA severe accident experiments and the Cathcart-Pawel and Baker-Just correlations, in NRC's August 2011 Draft Interim Review of PRM-50-93/95, NRC concludes:

> The results of [the] CORA [experiments] do not suggest that the Cathcart-Pawel or Baker-Just correlations are non-conservative. The assertions made by the petition with regards to Cathcart-Pawel and Baker-Just are not substantiated by the CORA data."[7]

And regarding the LOFT LP-FP-2 severe accident experiment and the Cathcart-Pawel and Baker-Just correlations, in NRC's September 2011 Draft Interim Review of PRM-50-93/95, NRC concludes:

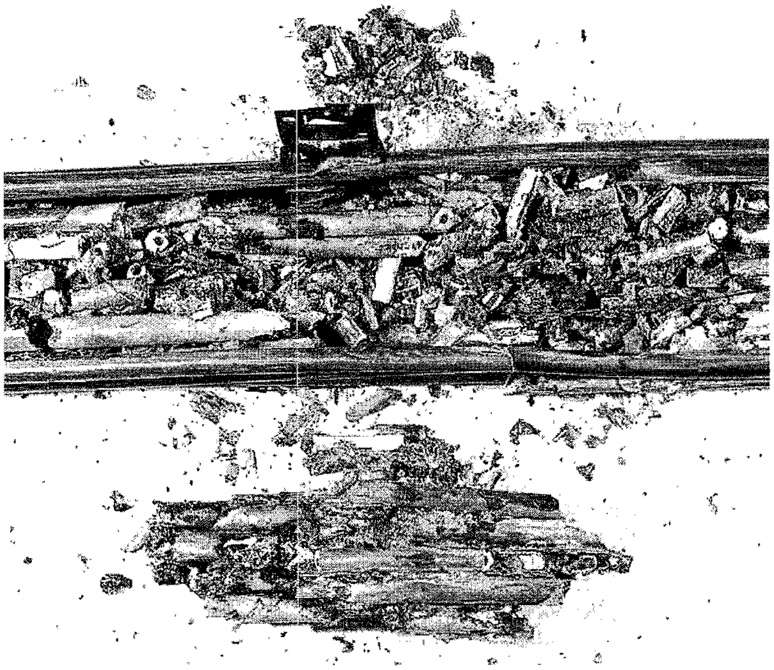
> The results of LOFT Test LP-FP-2 do not...suggest that the Cathcart-Pawel or Baker-Just correlations are non-conservative. The assertions made in PRM-50-93/95 with regards to Cathcart-Pawel and Baker-Just are not substantiated by the results of this LOFT test."[8]

In Petitioner's comments on PRM-50-93/95 (page 5), dated April 7, 2011,[9] Petitioner quoted an OECD Nuclear Energy Agency report, published in 2001, which explicitly states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[10] Yet NRC's draft interim reviews of PRM-50-93/95 on the CORA and LOFT LP-FP-2 experiments neither discuss nor mention the Nuclear Energy Agency statement—instead NRC claims that the CORA data and LOFT LP-FP-2 data confirm that the Cathcart-Pawel and Baker-Just correlations are conservative for use in computer safety models.

---

[7] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 2011, available at: NRC's ADAMS Documents, Accession Number: ML112290888, p. 3.
[8] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," p. 5.
[9] Mark Leyse, Comments on PRM-50-93/95, April 7, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111020046.
[10] Report by Nuclear Energy Agency ("NEA") Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNIIR(2001)15, October 1, 2001, Part I, B. Clement (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9.

III. NRC's TRACE Simulations of FLECHT Run 9573 Are Invalid because They Did Not Simulate the Section of the Test Bundle That Incurred Runaway Oxidation



Section of the Bundle from FLECHT Run 9573

31-4

In NRC's October 2012 Draft Interim Review of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573 that it performed.[11] (FLECHT run 9573 was a design basis accident experiment.) NRC provides results of its TRACE simulations for the 2, 4, 6, 8, and 10-foot elevations of the FLECHT run 9573 bundle, which were the elevations where thermocouples were located on the bundle.[12]

Unfortunately, in FLECHT run 9573 there were no thermocouples located at the section of the bundle which incurred runaway oxidation—"within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[13] (There was a steam probe thermocouple located at the 7-foot elevation.[14]) Hence, NRC's TRACE simulations did not include the section of the FLECHT run 9573 bundle that incurred runaway oxidation.

---

[11] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," pp. 7-8.

[12] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, available at: NRC's ADAMS Documents, Accession Number: ML070780083, p. 2-10.

[13] *Id.*, p. 3-97.

[14] *Id.*, p. 2-13.

As stated in PRM-50-93 (pages 59, 60), Westinghouse reported, regarding the FLECHT run 9573 bundle, that a "[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[15] And, as stated in PRM-50-93 (page 60), Westinghouse reported that "[t]he remainder of the [FLECHT run 9573] bundle was in excellent condition."[16]

(Appendix A of PRM-50-93 has photographs of the "locally severe damage zone," which incurred runaway oxidation, of the bundle from FLECHT run 9573.)

It is reasonable to assume that—as in CORA-2, in which local steam starvation conditions are postulated to have occurred[17]—in FLECHT run 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, NRC's TRACE simulations for FLECHT run 9573, using the Baker-Just and Cathcart-Pawel correlations, encompassed locations—the 2, 4, 6, 8, and 10-foot elevations of the bundle—that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, NRC's TRACE simulations are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

A. NRC's TRACE Simulations of FLECHT Run 9573 Did Not Include Data Taken from the Seven-Foot Elevation of the Bundle

The highest predicted temperature in NRC's TRACE simulations of FLECHT run 9573 was 1598.4 K (2417.7°F) at the 6-foot elevation, *at 18 seconds* after flooding commenced: predicted by the TRACE simulation using the Baker-Just correlation. As stated in PRM-50-93 (pages 10-11, 59, 63), Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds*

[15] *Id.*, p. 3-97.
[16] *Id.*
[17] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

after flooding commenced in FLECHT run 9573.[18] And, as stated in PRM-50-93 (pages 59-60, 60-61), Westinghouse reported that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[19] Therefore, at locations at which heater rods stated to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 82°F higher than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation.

As stated in PRM-50-93 (pages 66-67), Westinghouse reported, regarding the FLECHT run 9573 bundle that "[t]he steam probe thermocouple located one foot above midplane [at the 7-foot elevation] in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding."[20] (Appendix I of PRM-50-93 is a Westinghouse memorandum, dated December 14, 1970, reporting that the steam heatup rate exceeded 300°F/sec, at the 7-foot elevation.)

Hence, there is yet another reason why NRC's TRACE simulations FLECHT run 9573 were not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models. NRC's TRACE simulations did not include data taken from the 7-foot elevation of the FLECHT run 9573 bundle, where a steam probe thermocouple measured steam temperature heatup rates that exceeded 300°F/sec.

It is unfortunate that NRC has overlooked the *new information* on FLECHT run 9573—not discussed in PRM-50-76—that Petitioner provided in PRM-50-93 and in comments on PRM-50-93/95.

---

[18] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[19] *Id.*
[20] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.

Respectfully submitted,

/s/

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

Submission ID 32
Mark Leyse
ML14104B253

PRM-50-93 & 50-95
(75FR66007)

Docketed - USNRC
April 14, 2014
Office of the Secretary
Rulemakings and Adjudications Staff

From:	Mark Leyse
To:	RulemakingComments Resource; PDR Resource; Inverso, Tara; Bladey, Cindy; Clifford, Paul; Dudley, Richard; Doyle, Daniel
Cc:	Robert H. Leyse; Christopher Paine; Matthew G. McKinzie; Nuclear; Dave Lochbaum; Ed Lyman; shadis@prexar.com; Clay Turnbull; Deborah Brancato; Phillip Musegaas
Subject:	NRC-2009-0554 (Eighth)
Date:	Saturday, April 12, 2014 6:02:12 PM
Attachments:	NRC-2009-0554 (Eighth) Mark Leyse (April 12, 2014).pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Leyse's, Petitioner's, eighth response, dated April 12, 2014, 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.

Sincerely,

Mark Leyse

April 12, 2014

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

Attention: Rulemakings and Adjudications Staff

Comments on

Nuclear Regulatory Commission's Draft Interim Reviews of

Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554

Submitted by

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

TABLE OF CONTENTS

Mark Edward Leyse's Comments on Nuclear Regulatory Commission's Draft Interim Reviews of Two Petitions for Rulemaking: PRM-50-93 and PRM-50-95; NRC-2009-0554

In these comments, Mark Edward Leyse ("Petitioner") comments on the U.S. Nuclear Regulatory Commission's ("NRC") Draft Interim Reviews ("DIR") of two petitions for rulemaking: PRM-50-93[1] and PRM-50-95[2] ("PRM-50-93/95"). Petitioner highlights some of the pertinent information, submitted by Petitioner in PRM-50-93/95 and in public comments on PRM-50-93/95, which NRC did not consider in its DIRs. Problems with NRC's TRACE simulations of FLECHT run 9573 are also discussed.

32-1

I. NRC has Overlooked Specific Data Cited by Petitioner from Experiments in which Runaway Oxidation Commenced at Temperatures Lower than the 10 C.F.R. § 50.46(b)(1) 2200°F Peak Fuel-Cladding Temperature Limit

> *The heat evolved from the zircaloy-[steam] reaction at temperatures above 2000°F is significant and produces an autocatalytic effect.*[3]— General Electric, 1959

Regarding the 2200°F 10 C.F.R. § 50.46(b)(1) peak fuel-cladding temperature ("PCT") limit, in NRC's October 2012 DIR of PRM-50-93/95, NRC concludes:

> [A]utocatalytic reactions have not occurred at temperatures less than 2200 degrees F. Accordingly, the 2200 degree F regulatory limit is sufficient provided the correlations used to determine the metal-water reaction rate below 2200 degrees F are suitably conservative such that excessive reaction rates do not occur below that value.[4]

---

[1] Mark Leyse, PRM-50-93, November 17, 2009, available at: NRC's ADAMS Documents, Accession Number: ML093290250.

[2] Mark Leyse, PRM-50-95, June 7, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101610121. (PRM-50-95 was originally a 10 C.F.R. § 2.206 enforcement action petition that Petitioner wrote on behalf of New England Coalition ("NEC"), dated June 7, 2010. In October 2010, NRC published a notice in the Federal Register stating that it had determined the NEC petition met the requirements for a petition for rulemaking under 10 C.F.R. § 2.802.)

[3] J. I. Owens, R. W. Lockhart, D.R. Iltis, K. Hikido, General Electric Company, "Metal-Water Reactions: VIII. Preliminary Consideration of the Effects of a Zircaloy-Water Reaction during a Loss-of-Coolant Accident in a Nuclear Reactor," GEAP-3279, September 30, 1959, p. 34.

[4] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12265A277, p. 2.

In PRM-50-93/95 and in comments on PRM-50-93/95, Petitioner submitted information stating that runaway (autocatalytic) zirconium-steam reactions ("runaway oxidation") *have* commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit. For example, PRM-50-93 (pages 46-47) quotes an OECD Nuclear Energy Agency report, which states that runaway oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater.[5] The NRC's October 2012 DIR of PRM-50-93/95 fails to respond to or even acknowledge the existence of this information.

In its October 2012 DIR of PRM-50-93/95, NRC neither acknowledges nor discusses the fact that Dr. Robert E. Henry, in presentation slides from "TMI-2: A Textbook in Severe Accident Management," postulated that in the Three Mile Island Unit 2 ("TMI-2") accident, the heat produced by the exothermic zirconium-steam reaction caused thermal runaway to commence in the reactor core when fuel-cladding temperatures reached approximately 1000°C (1832°F).[6] Dr. Henry's postulation is discussed in Petitioner's comments on PRM-50-93/95, dated November 23, 2010, (pages 11-14).[7]

Interestingly, a March 2002 NRC document, "Perspectives on Reactor Safety," states that in a postulated station blackout scenario at Grand Gulf, runaway zirconium oxidation would commence at 1832°F.[8] (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

---

[5] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," Executive Summary, February 2000, p. 9. (Regarding the statement that runaway (autocatalytic) oxidation occurs at temperatures of 1050-1100°C (1922-2012°F) or greater, "Degraded Core Quench: Summary of Progress 1996-1999" explicitly states that "[a] notable feature of the [QUENCH] experiments was the occurrence of temperature excursions starting in the unheated region at the top of the shroud, from temperatures of 750-800°C, which is more than 300 K lower than excursion temperatures associated with runaway oxidation by steam.")

[6] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society 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, available at: NRC's ADAMS Documents, Accession Number: ML102140405, Attachment 2.

[7] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.

[8] NRC, "Perspectives on Reactor Safety," NUREG/CR-6042, Rev. 2, March 2002, available at: NRC's ADAMS Documents, Accession Number: ML021080117, pp. 3.7-4, 3.7-5, 3.7-29.

Furthermore, in NRC's own September 2011 DIR of PRM-50-93/95, NRC presented data demonstrating that runaway oxidation commenced in the LOFT LP-FP-2 experiment when fuel-cladding temperatures were lower than 2200°F. (In PRM-50-93 (pages 27, 33, 41, 42), Petitioner quoted a Pacific Northwest Laboratory paper, which states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec [18°F/sec], signal[s] the onset of an autocatalytic oxidation reaction."[9] This is for cases in which there would be relatively low initial heatup rates—for example, 1.0 K/sec (1.8°F/sec)—followed by substantially higher heatup rates, caused by the contribution of heat generated by the exothermic zirconium-steam reaction.) In NRC's September 2011 DIR of PRM-50-93/95, NRC presented data stating that in LOFT LP-FP-2, when local temperatures reached 1477 K (2199.2°F), just under the regulatory limit, the heatup rates at two fuel-cladding locations (TE-5C07-042 and TE-5D13-042) were already 10.3 K/sec (18.5°F/sec) and 11.9 K/sec (21.4°F/sec), respectively.[10]

Hence, NRC's October 2012 DIR of PRM-50-93/95 overlooks data that NRC itself provided in September 2011 demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F PCT limit. Clearly, NRC needs to correct, and explore the safety implications of its erroneous conclusion that runaway oxidation has not commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

It is noteworthy that a report regarding best-estimate predictions for LOFT LP-FP-2 states that runaway oxidation would commence if fuel-cladding temperatures were to start increasing at a rate of 3.0 K/sec (5.4°F/sec);[11] this is for cases in which there would be relatively low initial heatup rates. (This information was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

---

[9] 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, available at: NRC's ADAMS Documents, Accession Number: ML042230126, p. 282.

[10] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," September 2011, available at: NRC's ADAMS Documents, Accession Number: ML112650009, p. 4.

[11] S. Guntay, M. Carboneau, Y. Anoda, "Best Estimate Prediction for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3803, June 1985, available at: NRC's ADAMS Documents, Accession Number: ML071940361, p. 38.

NRC's September 2011 DIR of PRM-50-93/95 failed to report that in LOFT LP-FP-2, at one location, due to the rapid Zircaloy-steam reaction on a Zircaloy guide tube, the temperature increased from 1400 K to 1800 K (2060.6°F to 2780.6°F) in 21 seconds.[12] The September 2011 DIR of PRM-50-93/95 also failed to note the heatup rate at the Zircaloy guide tube location (TE-5H08-027) when temperatures reached 1477 K (2199.2°F)—most likely the heatup rate exceeded 10 K/sec. At that location (TE-5H08-027), the *average* heatup rate was 19 K/sec (approximately 34.3°F/sec) from 1400 K to 1800 K (2060.6°F to 2780.6°F) over a period of 21 seconds.

The NRC's September 2011 DIR of PRM-50-93/95, states that a report, "Quick Look Report on OECD LOFT Experiment LP-FP-2," concluded that "rapid oxidation of zircaloy started at approximately 1480 seconds" and that "thermocouples [temperature measuring devices] at the 42-inch elevation confirms this, as the[ir measurements] exceed[ed] 1477 K (2200°F) by 1460 seconds."[13] NRC is incorrect: the report actually states that "[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 [2060°F];"[14] furthermore, the report states that *recorded* temperatures on a Zircaloy guide tube reached 1800 K (2780.6°F) at 1451 seconds and that *recorded* temperatures on fuel cladding reached 1800 K (2780.6°F) at 1475 seconds.[15]

The "Quick Look Report"" also states:

> The first recorded (and qualified) rapid temperature rise caused by the exothermic reaction between the steam and the zircaloy is at about 1430 s[econds] on guide tube thermocouple TE-5H08-027. (Thermocouple TE-5EIl-027 was judged to have failed at 1311 s[econds], but the mode of failure suggests that temperatures reached 1800 K (2780°F) at some location in the core by 1381 s[econds].) The rapid temperature rise began from approximately 1400 K (2060°F).[16]

[12] Adams, J. P., *et al*., "Quick Look Report on OECD LOFT Experiment LP-FP-2," OECD LOFT-T-3804, September 1985, available at: NRC's ADAMS Documents, Accession Number: ML071940358, pp. 30, E-4, E-8.
[13] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," p. 4.
[14] Adams, J. P., *et al*., "Quick Look Report on OECD LOFT Experiment LP-FP-2," p. 30.
[15] *Id*., p. E-8.
[16] *Id*., p. E-4.

In PRM-50-93 (page 39), Petitioner quoted a report that stated that "[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 on a guide tube at the 0.69-m (27-in.) elevation."[17] And Petitioner, in PRM-50-93 (page 40), quoted the same report, which stated that "[i]t 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)."[18] NRC overlooked the fact that the very same sentence is on page 30 of the report it referenced: "Quick Look Report on OECD LOFT Experiment LP-FP-2.")

LOFT LP-FP-2 combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water;[19] and "[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."[20]

(See Appendix A for information about the BWR FLECHT Zr2K test and Thermal Hydraulic 1 test 130: design basis accident experiments in which runaway oxidation (most likely) commenced and almost commenced, respectively, at fuel-cladding temperatures that were lower than the 2200°F PCT limit. Although neither mentioned in PRM-50-93/95 nor in comments on PRM-50-93/95, the PHEBUS B9R-2 test is also discussed.)

---

[17] 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, available at: NRC's ADAMS Documents, Accession Number: ML062840091, p. 30.

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

[19] 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.

[20] 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.

I.A. NRC Overlooked an Experiment in which Runaway Oxidation either Commenced at a Temperature Lower than the 2200°F PCT Limit or at a Temperature Not High Enough above 2200°F to Provide a Necessary Margin of Safety

NRC's October 2012 DIR of PRM-50-93/95 falsely claims that Petitioner omitted "some important information from the "Compendium of ECCS Research for Realistic LOCA Analysis," [which] discusses conservatism in the regulatory criteria, and provides some justification." [21]

The October 2012 DIR of PRM-50-93/95 quotes the "important information" from "Compendium of ECCS Research for Realistic LOCA Analysis":

> The MT-6B test conducted in June 1984 showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant. In the FLHT-test, completed in March 1985, 12 ruptured zircaloy clad rods were subjected to an autocatalytic temperature excursion. From the measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500 – 2600°F (1371 – 1427°C) temperature region.[22]

The first sentence from the quote above, regarding the MT-6B test (Materials Test 6B) was already quoted in PRM-50-93 (pages 31, 35). And PRM-50-93 discussed the MT-6B test (pages 30-31, 35). One of the things that PRM-50-93 points out is that three publications report *different* peak fuel-cladding temperature values for the MT-6B test: the PCT was reported variously as 2060°F (1400 K),[23] 2200°F (1477 K),[24] and 2336°F (1553 K).[25]

---

[21] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 2.

[22] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 2; the source of this quote is NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, available at: NRC's ADAMS Documents, Accession Number: ML053490333, p. 8-2.

[23] 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. viii.

[24] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

[25] G. M. Hesson, *et al.*, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," 1993, p. 2.

The second and third sentences from the quote above, regarding the FLHT-test (actually the FLHT-1 test: Full-Length High-Temperature Severe Fuel Damage Test 1) were also already quoted in PRM-50-93 (page 37). And PRM-50-93 discusses the FLHT-1 test (pages 31-38); and Appendix E of PRM-50-93 has graphs depicting cladding temperature values for the maximum temperature region of the FLHT-1 test fuel assembly; the FLHT-1 test is also discussed in Petitioner's comments on PRM-50-93/95, dated December 27, 2010, (pages 31-36).[26] PRM-50-93 already highlighted that it is highly likely that in the FLHT-1 test, runaway oxidation commenced at cladding temperatures of approximately 1520°K (2277°F) or lower. Even if it were determined that runaway oxidation commenced at 77°F above NRC's 2200°F PCT limit, this would indicate that the 2200°F PCT limit is non-conservative, because the limit would not provide a necessary margin of safety in the event of a loss-of-coolant accident ("LOCA").

In PRM-50-93 (pages 34-35), Petitioner explains why he believes that in the FLHT-1 test, the cladding temperature excursion began at a temperature of approximately 1520°K (2277°F) or lower.

In PRM-50-93 (page 34), a quote is provided that describes the procedure the conductors of the FLHT-1 test followed. Regarding the test procedure, "Full-Length High-Temperature Severe Fuel Damage Test 1" states:

> 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.[27]

PRM-50-93 argues (pages 34-35) that it is obvious from the description in the quote above and from the cladding-temperature plots provided in Appendix E of PRM-50-93 that when cladding temperatures reached approximately 1475°K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was *not* achieved. (The slopes of the lines of the cladding-temperature value plots of the FLHT-1 test

---

[26] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

[27] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.6.

become nearly vertical, after the cladding-temperature values reach approximately 1520°K (2277°F), indicating that only a short time period passed before temperatures increased to approximately 2275°K (3636°F).) In fact, cladding temperatures continued to increase. This is clearly stated in the quote above, which states that increased "flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly…"[28]

Clearly, the conductors of the FLHT-1 test *could not terminate* the cladding-temperature increase after peak cladding temperatures reached approximately 1475°K (2200°F); they increased the coolant flow rates yet still could not prevent the runaway zirconium-steam reaction from commencing. Peak cladding temperatures increased from approximately 1520°K (2277°F) or lower to approximately 2275°K (3636°F), within approximately 85 seconds.[29]

It is unfortunate that NRC overlooked the information provided in PRM-50-93 on the FLHT-1 test and did not review the FLHT-1 test.

32-2

II. NRC Has Not Considered the Problems with the Metallurgical Data from the Four Zircaloy PWR-FLECHT Experiments

Regarding the metallurgical data from the four Zircaloy PWR-FLECHT experiments, in NRC's October 2012 DIR of PRM-50-93/95, NRC states:

> Furthermore, while PRM-50-93 takes issue and disagrees with parts of the NRC's evaluation of petition PRM-50-76, it fails to consider that in the NRC evaluation there were calculations of oxygen uptake and ZrO_2 thickness for the four FLECHT Zircaloy experiments (Cadek *et al*., 1971). The calculations showed Cathcart-Pawel to be best-estimate and Baker-Just to be conservative.[30]

[28] *Id.*
[29] *Id.*, pp. v, 4.6.
[30] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 6.

When NRC performed its technical safety analysis of PRM-50-76,[31] NRC was evidently unaware of the serious problems with the metallurgical data that Westinghouse took and analyzed from the four FLECHT Zircaloy experiments.

In NRC's October 2012 DIR of PRM-50-93/95, NRC overlooked *new information*—not discussed in PRM-50-76—that Petitioner provided in PRM-50-93 (pages 49-50) and in comments on PRM-50-93/95, dated November 23, 2010 (pages 45-47),[32] dated March 15, 2010 (pages 32-34),[33] dated April 7, 2011 (pages 7-9),[34] which indicates Westinghouse's metallurgical data from Zircaloy PWR FLECHT run 9573 is invalid. And in comments on PRM-50-93/95, dated July 30, 2011 (page 18),[35] Petitioner provided new information indicating that the metallurgical data from Zircaloy PWR FLECHT run 8874 is also invalid; see Section II.A.

Appendixes A and B of PRM-50-93 have photographs of the sections of the test bundles from FLECHT runs 9573 and 8874 that incurred runaway oxidation, respectively.

Furthermore, although neither discussed in PRM-50-93 nor in comments on PRM-50-93/95, there are also significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from Zircaloy PWR FLECHT runs 2443 and 2544; see Section II.B.

II.A. NRC Overlooked Problems with the Metallurgical Data from FLECHT Runs 8874 and 9573

In PRM-50-93 and in comments on PRM-50-93/95, Petitioner *emphasized* that there are significant problems with Westinghouse's examinations of the metallographic cross-

[31] 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, available at: NRC's ADAMS Documents, Accession Number: ML041210109.
[32] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[33] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[34] Mark Leyse, Comments on PRM-50-93/95, April 7, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111020046.
[35] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

sections that were taken from test rods from Zircaloy PWR FLECHT run 9573, because Westinghouse did not obtain metallurgical data from the locations of the rods from run 9573 that incurred runaway oxidation.[36] Then, in comments on PRM-50-93/95, Petitioner stated that Zircaloy PWR FLECHT run 8874 had also incurred runaway oxidation and that Westinghouse did not obtain metallurgical data from the locations of the rods from run 8874 that incurred runaway oxidation. It is probable that the locations of the test bundles from runs 8874 and 9573 that Westinghouse did examine were steam starved: the examined locations had limited oxidation because they had been exposed to a limited amount of steam.

It is reasonable to assume that—as in CORA-2, in which local steam starvation conditions are postulated to have occurred[37]—in FLECHT runs 8874 and 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, Westinghouse's application of the Baker-Just zirconium-steam correlation (used in computer safety models) to the oxide layers on the test bundles from FLECHT runs 8874 and 9573 were to locations that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, that is not a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models.

Subsequently, NRC applied the Baker-Just and Cathcart-Pawel correlations to the metallurgical data from the four FLECHT Zircaloy experiments:[38] unfortunately, NRC did not apply the Baker-Just and Cathcart-Pawel correlations to metallurgical data from the locations of FLECHT runs 8874 and 9573 that incurred runaway oxidation. Hence,

---

[36] Runaway oxidation was not expected to occur in any of Westinghouse's PWR FLECHT tests. "PWR FLECHT Final Report" does not mention that the bundles from PWR FLECHT runs 8874 and 9573 incurred runaway oxidation.

[37] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, p. 41.

[38] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, available at: NRC's ADAMS Documents, Accession Number: ML050250359, pp. 21-22.

NRC's analyses are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

It is unfortunate that NRC has overlooked the information Petitioner provided which indicates that Westinghouse's metallurgical data from FLECHT runs 8874 and 9573 is invalid.

(See Appendixes B and C for photographs of the sections of the test bundles from FLECHT runs 9573 and 8874 that incurred runaway oxidation, respectively.)

II.B. Problems with the Metallurgical Data from FLECHT Runs 2443 and 2544

Although neither discussed in PRM-50-93/95 nor in comments on PRM-50-93/95, there are also significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from Zircaloy PWR FLECHT runs 2443 and 2544.

A Westinghouse report states that two of the PWR FLECHT experiments—runs 2443 and 2544—with Zircaloy test bundles had unintended internal gas pressure increases, at the middle sections of the bundles, which caused the Zircaloy cladding to balloon and move away from the heat source of the internally heated rods and from the location of the thermocouples.[39] The actual temperatures of the Zircaloy cladding of the test bundles at the middle section were lower than the temperatures Westinghouse recorded. Therefore, the quantity of oxidation which occurred at the middle sections of the test bundles from FLECHT runs 2443 and 2544, occurred at lower temperatures than Westinghouse claimed.

Westinghouse would have accurately measured the thickness of each oxide layer; however, Westinghouse concluded that the thicknesses of the oxide layers from the middle sections of the test bundles from FLECHT runs 2443 and 2544 had been produced at higher temperatures than they were actually produced at. Hence, the metallurgical data was erroneously associated with cladding temperatures that were too high. Clearly, Westinghouse's metallurgical data from FLECHT runs 2443 and 2544 is not valid for

[39] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, available at: NRC's ADAMS Documents, Accession Number: ML070780083, p. 3-95.

performing a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models. NRC's subsequent analyses—which used data from FLECHT runs 2443 and 2544—are also not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

(Interestingly, in Westinghouse's comparison of eight metallurgical samples from run 2443, taken from two feet above and below the midplane location, *all* of the measured oxide thicknesses *exceeded* the predicted oxide thicknesses.[40])

III. NRC's TRACE Simulations of FLECHT Run 9573 Are Invalid because They Did Not Simulate the Section of the Test Bundle that Incurred Runaway Oxidation

32-3

In NRC's October 2012 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573 that it performed.[41] NRC provides results of its TRACE simulations for the 2, 4, 6, 8, and 10-foot elevations of the FLECHT run 9573 test bundle, which were the elevations where thermocouples were located on the bundle.[42]
Unfortunately, in FLECHT run 9573 there were no thermocouples located at the section of the test bundle which incurred runaway oxidation—around the 7 ft elevation. (There was a steam probe thermocouple located at the 7-foot elevation.[43]) Hence, NRC's TRACE simulations of FLECHT run 9573 did not include the section of the test bundle that incurred runaway oxidation.

As already stated in PRM-50-93 (pages 59, 60), Westinghouse reported, regarding the FLECHT run 9573 bundle, that a "[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[44] (See Figure 1.) And, as previously stated in PRM-50-93 (page 60), Westinghouse reported that "[t]he remainder of the [FLECHT run 9573] bundle was in

[40] In all eight cases measured oxide thicknesses were less than 0.1 x 10^{-3} inches thick; however, all the predicted thicknesses were zero inches. See F. D. Kingsbury, J. F. Mellor, A. P. Suda, Westinghouse Electric Corporation, Appendix B, "Materials Evaluation," of "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. B-9.
[41] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," pp. 7-8.
[42] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 2-10.
[43] *Id.*, p. 2-13.
[44] *Id.*, p. 3-97.

excellent condition."[45] (Appendix A of PRM-50-93 has photographs of the "locally severe damage zone," which incurred runaway oxidation, of the test bundle from FLECHT run 9573.)

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

As stated in Section II.A, it is reasonable to assume that—as in CORA-2, in which local steam starvation conditions are postulated to have occurred[46]—in FLECHT run 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, NRC's TRACE simulations of FLECHT run 9573, using the Baker-Just and Cathcart-Pawel correlations, encompassed locations—the 2, 4, 6, 8, and 10-foot elevations of the test bundle—that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Petitioner contends on the basis of this evidence that NRC's TRACE

---

[45] *Id.*

[46] 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)," p. 41.

simulations are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.

(See Appendix B for photographs of the section of the test bundle from FLECHT run 9573 that incurred runaway oxidation.)

III.A. NRC's TRACE Simulations of FLECHT Run 9573 Did Not Include Data Taken from the Seven-Foot Elevation of the Test Bundle

The highest predicted temperature in NRC's TRACE simulations of FLECHT run 9573 was 1598.4 K (2417.7°F) at the 6-foot elevation, *at 18 seconds* after flooding commenced: predicted by the TRACE simulation using the Baker-Just correlation.[47] As stated in PRM-50-93 (pages 10-11, 59, 63), Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573.[48] And, as stated in PRM-50-93 (pages 59-60, 60-61), Westinghouse reported that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[49] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F higher than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F higher than the highest temperature predicted using the Cathcart-Pawel correlation.

In NRC's October 2012 DIR of PRM-50-93/95, NRC states that "it should be noted that over the first 18 seconds of FLECHT run 9573, the heatup rate was below the 15 K/sec that is considered in the petition to be an indication of an "autocatalytic reaction" rate.[50] In fact, as stated in Section I, PRM-50-93 quotes a paper stating that "a rapid [cladding] temperature escalation, [greater than] *10 K/sec* [18°F/sec], signal[s] the

---

47 NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.
48 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
49 *Id*.
50 NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.

onset of an autocatalytic oxidation reaction"[51] [emphasis added]. (This is for cases in which there would be relatively low initial heatup rates—for example, 1.0 K/sec (1.8°F/sec)—followed by substantially higher heatup rates, caused by the contribution of heat generated by the exothermic zirconium-steam reaction.) The NRC staff response misrepresents a statement made in the petition.

Regarding the heatup rates, NRC states:

> At the elevations where cladding oxidation was significant ([4, 6, and 8 feet]), both the Cathcart-Pawel and the Baker-Just correlations resulted in an over-prediction of the measured heatup rate. Heatup rates with the Baker-Just correlation were greater than those obtained with the Cathcart-Pawel correlation, and were significantly greater than the heatup rates observed in the experimental data. At the peak power elevation ([6 feet]), the heatup rate using the Baker-Just correlation exceeded the experimental value by 41 percent.[52]

As already stated in PRM-50-93 (pages 66-67), Westinghouse reported, regarding the FLECHT run 9573 test bundle that "[t]he steam probe thermocouple located one foot above midplane [at the 7-foot elevation] in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding."[53] (Appendix I of PRM-50-93 is a Westinghouse memorandum, dated December 14, 1970, reporting that the steam heatup rate exceeded 300°F/sec, at the 7-foot elevation.)

Hence, there is yet another reason why NRC's TRACE simulations FLECHT run 9573 were not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models. NRC's TRACE simulations did not include data taken from the 7-foot elevation of the FLECHT run 9573 test bundle, where a steam probe thermocouple measured steam temperature heatup rates that exceeded 300°F/sec. Surely, at the 7-foot elevation, at 18 seconds after flooding

---

[51] F. E. Panisko, N. J. Lombardo, "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," p. 282.

[52] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.

[53] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.

commenced, there were local cladding temperature heatup rates that exceeded 16.1 K/sec (29°F/sec): the maximum heatup rate predicted by NRC's TRACE simulation using the Baker-Just correlation.[54]

It is unfortunate that NRC has overlooked the *new information* on FLECHT run 9573—not discussed in PRM-50-76—that Petitioner provided in PRM-50-93 and in comments on PRM-50-93/95.

(See Appendix D for information about experiments in which zirconium-steam reaction rates occurred that are under-predicted by computer safety models.)

III.B. Results of NRC's TRACE Simulations of FLECHT Run 9573 Were Not Compared to the Highest Cladding Temperatures and Heatup Rates

There are serious problems with the fact that NRC compared the results of its TRACE simulations of FLECHT run 9573 to the *average* value of different thermocouple measurements—data taken from the FLECHT run 9573 test bundle at the 2, 4, 6, 8, and 10-foot elevations, at 18 seconds after flooding commenced. NRC compared its TRACE results regarding cladding temperatures to "the average of the available thermocouple measurements at a particular elevation;"[55] and compared its TRACE results regarding cladding temperature heatup rates to "the average of the available thermocouple measurements at each elevation."[56] The values of the averages of the cladding temperatures and heatup rates would be lower than the maximum values of the cladding temperatures and heatup rates at each elevation. Assessing the Baker-Just and Cathcart-Pawel correlations for use in computer safety models by comparing TRACE results with averaged thermocouple measurements is not a legitimate assessment.

Furthermore, in comments on PRM-50-93/95, dated April 12, 2010 (pages 26-27), Petitioner pointed out that in the PWR FLECHT tests—including run 9573—there were radiative heat losses from the test bundles to the bundle housing, which "constituted a

[54] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 8.
[55] *Id*., p. 7.
[56] *Id*., p. 8.

700°F cold spot;"[57] therefore, especially, the peripheral rods of the FLECHT run 9573 bundle would have radiated heat to the surrounding bundle housing.

Regarding the fact that the FLECHT run 9573 test bundle's interior rods were hotter than the peripheral rods, NRC's October 2012 DIR of PRM-50-93/95 states:

> In FLECHT run 9573 there were three thermocouples that registered temperatures greater than 2200 degrees F at a time of 18 seconds. … These were thermocouples numbered 3D3, 2D2, and 4E3. Each of these three thermocouples was on the interior of the bundle and shielded from the housing by at least one row of heater rods. Because of the low thermal radiation view factor, the [bundle] housing is not expected to have had a large influence on local heat transfer coefficients on the interior of the bundle.[58]

Hence, NRC acknowledges that temperatures were hotter in the interior of the test bundle; nonetheless, NRC decided to compare its TRACE results to the average value of different thermocouple measurements—hotter interior temperatures averaged with the cooler temperatures of the bundle's peripheral rods.

(In a LOCA, the concern would be that the *maximum fuel element cladding temperature* did not exceed the 2200°F 10 C.F.R. § 50.46(b)(1) PCT limit: the PCT limit pertains to the "hot spot," not to the average of cladding temperatures at a particular elevation.)

IV. NRC Overlooked Information Pertaining to PWR FLECHT Run 9573 Heat Transfer Coefficients

Regarding Petitioner's comments on PRM-50-93/95 dated March 15, 2010 (pages 5-9),[59] concerning FLECHT run 9573 heat transfer coefficients, NRC's October 2012 DIR states:

> The comments discuss the negative heat transfer coefficients near the mid-plane elevation in FLECHT run 9573 and that, as pointed out in the data

[57] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, Union of Concerned Scientists, 1974, p. 5.31.
[58] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 5.
[59] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.

> report [WCAP-7665] (Cadek *et al.*, 1971),[60] this occurred at approximately the time when heater rods began to fail in the bundle and the cladding temperatures were 2200-2300 degrees F. The comments also noted that heat transfer coefficients in this test were lower than those in other FLECHT tests with Zircaloy cladding. The petitioner, however, failed to recognize or acknowledge that this aspect of FLECHT run 9573 was addressed in the NRC technical evaluation of PRM-50-76 where this anomaly was attributed to the data reduction process. (See page 7 of NRC, 2004.)[61]

In the passage above, NRC has made an incorrect statement and overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients. It needs to be clarified that, as previously and correctly stated in PRM-50-93 (pages 59-60, 60-61), WCAP-7665 reports that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F"[62]—*i.e.*, they were not caused by temperatures in the range of 2200 to 2300°F.

First, NRC incorrectly described the statement from its own technical evaluation of PRM-50-76. NRC's technical evaluation does not say that the "anomaly," regarding heat transfer coefficients, was *definitely* attributed to the data reduction process. NRC's technical evaluation states that "*[s]ome* of the anomaly [lower 'measured' heat transfer coefficients] *can probably be explained* due to a deficiency in the data reduction process" [emphasis added].[63]

(More importantly, NRC needs to acknowledge that additional information regarding FLECHT run 9573 was provided in PRM-50-93 and that NRC's technical evaluation of PRM-50-76 is seriously flawed. For example, NRC's technical evaluation of PRM-50-76 does not mention the fact that the FLECHT run 9573 test bundle incurred runaway oxidation—there is still no NRC analysis of the sections of the bundle that incurred runaway oxidation.)

---

[60] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665.
[61] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 5.
[62] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[63] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 CFR Part 50 and Regulatory Guide 1.157," p. 7.

In fact, Westinghouse's 1971 report, WCAP-7665, states that "anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. *These may have been related to the high steam probe temperatures measured at the 7 ft elevation*" [emphasis added].[64] (The high steam probe temperatures "exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure."[65])

Second, NRC has overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients that Petitioner provided in PRM-50-93 (pages 9-11, 59-70) and comments on PRM-50-93/95, dated March 15, 2010 (pages 5-9),[66] dated November 23, 2010 (pages 29-34),[67] and dated December 27, 2010 (pages 15-21).[68] As stated, Westinghouse postulated that the negative heat transfer coefficients observed in FLECHT run 9573 "may have been related to the high steam probe temperatures measured at the 7 ft elevation."[69] In PRM-50-93 and comments on PRM-50-93/95, Petitioner argues that the high steam temperatures were in fact the cause of the negative heat transfer coefficients; the negative heat transfer coefficients were a result of heat transfer from the steam—measured at temperatures exceeding 2500°F—to the test bundle rods.

Regarding FLECHT run 9573, in October 2002, Westinghouse stated, "[t]he high fluid [steam] temperature 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."[70] Hence, the heat generated by the zirconium-steam reaction is what heated the steam to temperatures exceeding 2500°F—a phenomenon that could occur in a large break LOCA.

---

[64] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.
[65] *Id.*, p. 3-97.
[66] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[67] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[68] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.
[69] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.
[70] H. A. Sepp, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, available at: NRC's ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

32-4

IV.A. NRC's Incorrect Claim that Its TRACE Simulations of FLECHT Run 9573 Demonstrate that Conservative Heat Transfer Models Can Be Developed from Data Obtained Primarily from Experiments Conducted with Stainless Steel Rods

In its October 2012 DIR of PRM-50-93/95, the NRC Staff claims:

> The TRACE simulations…demonstrate that it is possible to develop heat transfer models based on data obtained primarily from stainless steel rods and conservatively simulate FLECHT run 9573. When either the Cathcart-Pawel or Baker-Just correlations are used to determine the metal-water reaction rate, TRACE was found to conservatively predict the cladding temperatures at each elevation. … The staff concludes that there is nothing in the petition that [indicates] use of stainless steel clad rod data is inaccurate or insufficient for development of heat transfer models. [71]

As discussed in Section III, NRC's TRACE simulations of FLECHT run 9573 are invalid because they did not simulate the section of the test bundle that incurred runaway oxidation. The simulations of FLECHT run 9573 encompassed locations of the test bundle that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Furthermore, the simulations did not include data taken from the 7-foot elevation of the test bundle, where a steam probe thermocouple measured steam temperature heatup rates that exceeded 300°F/sec. There are also serious problems with the fact that NRC compared the results of its TRACE simulations of FLECHT run 9573 to the *average* value of different thermocouple measurements taken at each elevation and not to the maximum values of the cladding temperatures measured at each elevation (the 2, 4, 6, 8, and 10-foot elevations of the test bundle, at 18 seconds after flooding commenced).

Clearly, NRC's TRACE simulations are neither legitimate simulations of FLECHT run 9573 nor legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models. Hence, the TRACE simulations do *not* "demonstrate that it is possible to develop heat transfer models based on data obtained primarily from stainless steel rods." [72]

[71] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 9.
[72] *Id*., p. 9.

As stated in Section IV, NRC has overlooked information pertinent to PWR FLECHT run 9573 heat transfer coefficients that Petitioner provided in PRM-50-93 (pages 9-11, 59-70) and comments on PRM-50-93/95, dated March 15, 2010 (pages 5-9),[73] dated November 23, 2010 (pages 29-34),[74] and dated December 27, 2010 (pages 15-21).[75] The information Petitioner provided supports the claim that Appendix K to Part 50 Section 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.

32-5

V. NRC's Conclusions Regarding Reflood Rates Are Invalid because They Are Based on NRC's TRACE Simulations of FLECHT Run 9573, which Did Not Simulate the Section of the Test Bundle that Incurred Runaway Oxidation

In its March 2013 DIR of PRM-50-93/95, NRC's conclusions regarding reflood rates are based on NRC's TRACE simulations of FLECHT run 9573. As discussed in Section III, NRC's TRACE simulations of FLECHT run 9573 are invalid because they did not simulate the section of the test bundle that incurred runaway oxidation. In fact, NRC's TRACE simulations of FLECHT run 9573 encompassed locations of the test bundle that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

In its March 2013 DIR of PRM-50-93/95, NRC *incorrectly* concludes that its "TRACE simulation of Test 9573 showed reasonable agreement with available data, with TRACE exceeding the measured maximum cladding temperature 18 seconds into the

[73] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[74] Mark Leyse, Comments on PRM-50-93/95, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[75] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

test."[76] As discussed in Section III.A, Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573[77] and that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[78] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F *higher* than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F *higher* than the highest temperature predicted using the Cathcart-Pawel correlation.

V.A. Comparisons of NRC's TRACE Simulations of FLECHT Run 9573 with Actual Experimental Data

In order to reach its conclusions regarding reflood rates for its DIR of PRM-50-93/95, NRC relies on invalid TRACE simulations of FLECHT run 9573. Different conclusions would be reached by objectively reviewing actual experimental data from tests *conducted with zirconium alloy bundles*. (Interestingly, the TRACE simulations of FLECHT run 9573 (the ones done in order to reach conclusions regarding reflood rates) seem to have only used the Cathcart-Pawel correlation; apparently, the Baker-Just correlation was not used in any of the simulations.[79])

[76] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," March 8, 2013, available at: NRC's ADAMS Documents, Accession Number: ML13067A261, p. 4.
[77] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[78] *Id.*
[79] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," pp. 4, 7.

V.A.1. TRACE Simulations of Reflood Cooling Compared to Actual Experimental Data

In its March 2013 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573 in which:

> In each case, the initial axial cladding temperature profile was scaled to that of Test 9573 to obtain the desired maximum cladding temperature at the start of each simulation. The reflood rate was assumed to be 1.1 inch/sec, consistent with Test 9573. At maximum initial cladding temperatures less than approximately 1200 degrees F (922 K), typical of those expected following the blowdown period of a LOCA, the peak cladding temperature[s] remain below 1800 degrees F (1255 K).[80]

In FLECHT run 9573 the *actual* PCT at the onset of reflood was 1970°F;[81] however, for the NRC TRACE simulations discussed in this section (V.A.1), FLECHT run 9573 was assigned PCTs at the onset of reflood that were less than approximately 1200°F. These TRACE simulations each resulted in FLECHT run 9573 having an overall PCT that was less than 1800°F. But there are problems with these TRACE simulations because there is data from *actual* thermal hydraulic LOCA experiments conducted with zirconium alloy bundles that indicates these simulations under-predict the overall PCT that FLECHT run 9573 would have had if its PCT at the onset of reflood had been 1200°F or lower. NRU Thermal Hydraulic 1 ("TH-1") test nos. 109 and 125 were conducted with reflood rates of 1.3 inches/second (in/sec) and 1.4 in/sec, respectively. TH-1 test no. 109 had a PCT at the onset of reflood of 1158°F and an overall PCT of 1881°F; and TH-1 test no. 125 had a PCT at the onset of reflood of 1138°F and an overall PCT of 1802°F.[82]

TH-1 test nos. 109 and 125 both had greater reflood rates than FLECHT run 9573. The greater reflood rates of TH-1 test nos. 109 and 125 would have had more of an effect on mitigating the overall PCT increases in those tests than the lower reflood rate of FLECHT run 9573 had on mitigating run 9573's overall PCT increase. (As discussed in

[80] *Id*., p. 4.
[81] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.
[82] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, available at: NRC's ADAMS Documents, Accession Number: ML101960414, p. 13.

Section V.B.1, the flooding rate is the most influential parameter that affects the overall PCT in thermal hydraulic LOCA experiments.)

And the TH-1 tests had an average fuel rod power of 0.38 kW/ft;[83] the peak rod power of FLECHT run 9573 was 1.24 kW/ft.[84] The lower fuel rod power of the TH-1 tests would not have affected the overall PCT increases as much as the greater fuel rod power of FLECHT run 9573 affected run 9573's overall PCT increase. (Regarding low power runs of thermal hydraulic LOCA experiments, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report" states that "[the] temperature rises…are smaller for the low power [runs] since lower energy removal rates and temperature differences are needed to remove the generated energy."[85]) Nonetheless, TH-1 test nos. 109 and 125, which both had initial PCTs that were less than 1200°F, had overall PCTs that exceeded 1800°F. (NRC's TRACE simulations of FLECHT run 9573—conducted with assigned initial PCTs of less than 1200°F for run 9573—predicted that run 9573's overall PCT would remain below 1800°F.) Such actual experimental data is further evidence that NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

V.A.2. TRACE Simulations of Steam Cooling Compared to Actual Experimental Data

In its March 2013 DIR of PRM-50-93/95, NRC discusses TRACE simulations of FLECHT run 9573; NRC states:

> Consider the TRACE model of the Zircaloy clad bundle that represented the bundle used in FLECHT Test 9573. Assuming an initial temperature profile with a maximum temperature of 1200 degrees F (922 K), a simulation was conducted with no liquid injection but with steam-only cooling of the bundle. [The] steam-only mass flow rate [was] 0.114 kg/s through the bundle. The peak cladding temperature obtained [was] 1325.7

[83] 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, available at: NRC's ADAMS Documents, Accession Number: ML083470834, p. 9-40.

[84] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.

[85] G. P. Lilly, H. C. Yeh, L. E. Hochreiter, N. Yamaguchi, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report," WCAP-8838, March 1977, available at: NRC's ADAMS Documents, Accession Number: ML070780090, p. 3-5.

K (1927 degrees F). No liquid injection can be interpreted as a reflooding rate of 0.0 in/sec. Cooling was accomplished not by reflood of the bundle, but only by convective cooling to the steam. The cladding exceeded 1000 C (1832 degrees F), and thus metal-water reaction became a significant source of heat. Nevertheless, the peak cladding temperature remained below 2200 degrees F and an "autocatalytic" (runaway) oxidation did not occur.[86]

Again, there is data from *actual* thermal hydraulic LOCA experiments conducted with zirconium alloy bundles that indicates NRC's TRACE simulations under-predict the overall PCT that FLECHT run 9573 would have had if its PCT at the onset of reflood had been 1200°F and its reflood rate had been 0.0 in/sec, with a steam-only mass flow rate of 0.114 kilograms/second ("kg/sec") through the test bundle.

In FLECHT run 9573, a steam-only mass flow rate of 0.114 kg/sec would be approximately equal to a reflood rate of 0.68 in/sec, if the steam were condensed.[87] In NRC's TRACE simulation, the steam was assigned an inlet temperature of approximately 307°F.[88]

TH-1 test nos. 127 and 130 were conducted with reflood rates of 1.0 in/sec and 0.74 in/sec, respectively. TH-1 test no. 127 had a PCT at the onset of reflood of 966°F and an overall PCT of 1991°F; and TH-1 test no. 130 had a PCT at the onset of reflood of 998°F and an overall PCT of 2040°F.[89]

In TH-1 test no. 130, the reactor *actually* tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures increased

[86] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," pp. 5-6.

[87] In the four PWR FLECHT facility tests with zirconium alloy (7 x 7) bundles, the bundle housing was square with internal dimensions of 4.200 inches (in) and there were 42 test rods with a diameter of 0.422 inch, six control rod thimbles a diameter of 0.545 inch, and one instrument tube with a diameter of 0.463 inch. See F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," pp. 2.1, 2.11, 3.8. In FLECHT run 9573, the cross-sectional flow area was 10.198 in^2, which is calculated by subtracting the total of $42\pi(.211\ in^2) + 6\pi(.2725\ in^2) + \pi(.2315\ in^2)$ from $(4.2\ in^2)$. A mass of 0.114 kg of water has a volume of 6.957 in^3. In FLECHT tests with 7 x 7 bundles, a volume of 6.957 in^3 of water—with a cross-sectional area of 10.198 in^2—would have had a height of 0.68 in.

[88] In NRC's TRACE simulation of steam-only cooling of FLECHT run 9573, the steam was saturated steam at a pressure of 0.42 MPa. See NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 6. Saturated steam at a pressure of 0.42 MPa (60.9 pounds per square inch) would have a temperature of approximately 307°F.

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

by 190°F, because of the heat generated by the zirconium-steam reaction (of course, there would have also been a slight amount of actual decay heat[90]) and the peak measured cladding temperature was 2040°F.[91] In TH-1 test no. 130, if the reactor had not shutdown when the PCT was approximately 1850°F, the overall PCT would have exceeded 2040°F. In fact, if the reactor had not shutdown when the PCT was approximately 1850°F it is possible that the combination of the simulated decay heat and heat generated by the zirconium-steam reaction would have caused the test bundle to incur runaway oxidation; in such a case, the PCT would have increased to greater than 3300°F.

(TH-1 test no. 130 is discussed on pages 24-25 of Petitioner's comments on PRM-50-93/95, dated December 27, 2010,[92] on page 5 of Petitioner's comments on PRM-50-93/95, dated July 27, 2011,[93] and on pages 9-11 of Petitioner's comments on PRM-50-93/95, dated July 30, 2011.[94])

TH-1 test nos. 127 and 130 both had greater coolant inlet rates (reflood rates of 1.0 in/sec and 0.74 in/sec, respectively) than the steam-only mass flow rate of 0.114 kg/sec (approximately equal to a reflood rate of 0.68 in/sec, if the steam were condensed) that was assigned to FLECHT run 9573 for NRC's TRACE simulations. The greater coolant inlet rates of TH-1 test nos. 127 and 130 would have had more of an effect on mitigating the overall PCT increases in those tests than the lower coolant inlet rate assigned to FLECHT run 9573 had on mitigating run 9573's overall PCT increase. And the TH-1 tests had an average fuel rod power of 0.38 kW/ft;[95] the peak rod power of

[90] TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft. See C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.

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

[92] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

[93] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.

[94] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

[95] C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.

FLECHT run 9573 was 1.24 kW/ft.[96] The lower fuel rod power of the TH-1 tests would not have affected their overall PCT increases as much as the higher fuel rod power of FLECHT run 9573 affected its overall PCT increase. Furthermore, TH-1 test nos. 127 and 130 both had initial PCTs that were less than 1000°F; and 1200°F was the initial PCT assigned to FLECHT run 9573 for NRC's TRACE simulations. Nonetheless, TH-1 test nos. 127 and 130 had overall PCTs of 1991°F and 2040°F, respectively. (NRC's TRACE simulations of FLECHT run 9573 predicted that run 9573's overall PCT would be 1927°F.) Such actual experimental data is yet further evidence that NRC's TRACE simulations are not legitimate verifications of NRC's conclusions regarding reflood rates.

NRC has incorrectly concluded that "[t]he [TRACE] steam-only cooling calculation demonstrates that it is possible to cool a Zircaloy clad bundle without reflooding."[97] NRC should review actual experimental data and not rely on invalid TRACE simulations of FLECHT run 9573, which did not simulate the section of the test bundle that incurred runaway oxidation.

32-6

V.B. Information Pertaining to LOCA-Reflood Phenomena that NRC Overlooked

V.B.1. NRC Overlooked the Significant Role that Reflood Rates have in Determining the PCT in a LOCA

Regarding reflood LOCA hydraulics, in its March 2013 DIR of PRM-50-93/95, NRC states that "[b]ecause numerous parameters have an effect on reflood hydraulics, no single parameter completely controls the peak cladding temperature for a particular transient."[98] While NRC's assertion is correct as far as it goes, it does not go far enough. As previously stated in PRM-50-93 (page 13), regarding the significance that coolant flood rates played in the PWR FLECHT test program, the "PWR FLECHT Final Report" states, "[i]n general, the effect on heat transfer coefficient[s] of varying system parameters was clearly discernable, *with flooding rate being by far the most influential*

---

[96] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-8.
[97] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 6.
[98] *Id.*, p. 2.

parameter investigated"[99] [emphasis added]. Hence, reflood rates would have a significant role in determining the PCT in a LOCA; and thus there needs to be a new regulation stipulating minimum allowable core reflood rates in the event of a LOCA, as requested in PRM-50-93.

V.B.2. NRC Overlooked the Role that the Heat Generated by the Exothermic Zirconium-Steam Reaction has in Increasing Fuel-Cladding Temperatures in a LOCA

Regarding fuel-cladding temperature increases of over 1000°F that were observed in NRU reflood tests conducted with Zircaloy fuel-cladding, in its March 2013 DIR of PRM-50-93/95, NRC states:

> Part of the basis for the petition's request for a limit on reflood rate, is the significant temperature increases observed in the NRU reflood tests. Starting from initial cladding temperatures less than 1000 degrees F, several NRU tests produced temperature increases of over 1000 degree F. The petition cites NRU test 127 and 130 as examples. The petition appears to imply that similar temperature increases would occur if the initial cladding temperatures had been 1200 degrees F or more. *This is not correct, however*[100] [emphasis added].

PRM-50-93/95 does in fact state that it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower would not, with high probability, prevent zirconium alloy fuel cladding with peak cladding temperatures of approximately 1200°F or greater at the onset of reflood, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. NRC claims that this is incorrect (NRC's argument is quoted below in this section (V.B.2)); however, NRC has overlooked the role that the heat generated by the exothermic zirconium-steam reaction has in increasing fuel-cladding temperatures in a LOCA.

As already discussed in section V.A.2, in TH-1 test no. 130 (conducted with zirconium alloy fuel cladding), the reactor shutdown when the PCT was approximately 1850°F and after the reactor shutdown, cladding temperatures increased by 190°F, because of the heat generated by the zirconium-steam reaction (of course, there would

[99] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 5-1.
[100] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

have also been a slight amount of actual decay heat[101]) and the peak measured cladding temperature was 2040°F.[102] If the reactor had not shutdown when the PCT was approximately 1850°F, the overall PCT would have exceeded 2040°F; and it is highly probable that the test bundle would have incurred runaway oxidation and that the PCT would have increased to greater than 3300°F.

NRC needs to consider that if TH-1 test no. 130 had been conducted with an initial PCT of 1200°F and the reactor did not shutdown when the PCT was approximately 1850°F, with high probability, the overall PCT would have exceeded 2200°F, because of the heat generated by the zirconium-steam reaction.

Regarding the results of LOCA tests *conducted with stainless steel bundles* in three experimental programs—PWR FLECHT SEASET,[103] PWR FLECHT Cosine,[104] and PWR FLECHT Skewed[105]—in its March 2013 DIR of PRM-50-93/95, NRC states:

> Thermal radiation becomes more important in transferring heat away from hot spots, and as rod temperatures increase the temperature difference between the cladding and the coolant increases. Figure 1…shows the effect of initial cladding temperature on temperature rise from tests in three experimental facilities. As the initial cladding temperature increases, the overall temperature rise *decreases*[106] [emphasis not added].

It is important to recognize that *only* thermal hydraulic LOCA experiments conducted with stainless steel bundles demonstrate the phenomenon of higher cladding temperature increases for tests with lower PCTs at the onset of reflood (in the entire

[101] TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft. See C. L. Mohr *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.
[102] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.
[103] Lee, N., Wong, S., Yeh, H.C., and Hochreiter, L.E., "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Evaluation and Analysis Report," WCAP-9891, NUREG/CR-2256, February 1982, available at: NRC's ADAMS Documents, Accession Number: ML070740214.
[104] G. P. Lilly, H. C. Yeh, L. E. Hochreiter, N. Yamaguchi, "PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report," WCAP-8838.
[105] Lilly, G.P. *et al.*, "PWR FLECHT Skewed Profile Low Flooding Rate Test Series Evaluation Report," WCAP-9183, November 1977, available at: NRC's ADAMS Documents, Accession Number: ML070780095.
[106] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," p. 3.

design basis accident cladding temperature range, below 2200°F). And, of course, nuclear power plants use zirconium alloy fuel rod cladding—not stainless steel fuel rod cladding.

At lower temperatures thermal hydraulic LOCA experiments conducted with Zircaloy bundles also demonstrate the phenomenon of higher cladding temperature increases for tests with lower PCTs at the onset of reflood; however, the results of experiments conducted with Zircaloy bundles are *different* at higher temperatures. In the temperature range at which the oxidation of Zircaloy becomes significant, the heat generated by the zirconium-steam reaction causes higher cladding temperature increases, as PCTs at the onset of reflood increase.

This trend is seen in four Zircaloy tests—TH-1 test nos. 105, 107, 110, and 128—conducted with an average fuel rod power of 0.38 kW/ft;[107] the first three tests had a reflood rate of 1.9 in/sec; the fourth test had a reflood rate of 2.0 in/sec. TH-1 test no. 105 had a PCT at the onset of reflood of 907°F and an overall PCT of 1364°F (an increase of 457°F); TH-1 test no. 107 had a PCT at the onset of reflood of 1154°F and an overall PCT of 1578°F (an increase of 424°F); TH-1 test no. 110 (Zircaloy) had a PCT at the onset of reflood of 1314°F and an overall PCT of 1665°F (an increase of 351°F); and TH-1 test no. 128 (Zircaloy) had a PCT at the onset of reflood of 1604°F and an overall PCT of 1991°F (an increase of 387°F).[108]

TH-1 test nos. 105, 107, and 110, demonstrate the phenomenon of higher cladding temperature increases for tests that had lower PCTs at the onset of reflood (for thermal hydraulic experiments conducted with Zircaloy bundles *at lower temperatures*). However, in TH-1 test no. 128, with a PCT at the onset of reflood of 1604°F, the overall PCT increase is 36°F greater than the overall PCT increase in TH-1 test no. 110, with a PCT at the onset of reflood of 1314°F. The overall PCT increased more in TH-1 test no. 128—with a slightly higher reflood rate—because of the heat that was generated by the zirconium-steam reaction.

---

[107] C. L. Mohr *et al.,* "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, p. 9-40.
[108] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.

(Unfortunately, an extremely limited amount of tests have been conducted with zirconium alloy bundles, so there is not much experimental data available to discuss.) NRC is incorrect in its conclusion that "[a]s the initial cladding temperature increases, the overall temperature rise *decreases*"[109] [emphasis not added]. Incredibly, NRC has *only* considered data from thermal hydraulic LOCA experiments conducted with stainless steel bundles and *overlooked* data from experiments conducted with the industry-standard zirconium alloy bundles.

VI. Conclusion

NRC's October 2012 DIR of PRM-50-93/95 actually overlooks experimental data NRC itself provided in its September 2011 DIR demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F PCT limit.[110] Clearly, the NRC Staff needs to correct its erroneous conclusion that runaway oxidation has not commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

It is unfortunate that NRC has also overlooked the new information Petitioner provided which indicates that Westinghouse's metallurgical data from FLECHT run 9573 is invalid. There are significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from FLECHT run 9573, because Westinghouse did not obtain metallurgical data from the locations of the rods from run 9573 that incurred runaway oxidation.

Additionally, NRC's TRACE simulations of FLECHT run 9573 did not include the section of the test bundle that incurred runaway oxidation. In fact, NRC's TRACE simulations encompassed locations of the test bundle that were most likely steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, NRC's TRACE simulations are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in

---

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

[110] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," p. 4.

computer safety models, and not legitimate verifications of NRC's conclusions regarding reflood rates.

The highest predicted temperatures in NRC's TRACE simulations of FLECHT run 9573 at 18 seconds after flooding commenced, using the Baker-Just correlation and Cathcart-Pawel correlation, were 2417.7°F and 2338.2°F, respectively.[111] Westinghouse reported that steam temperatures (measured by the seven-foot steam probe) exceeded 2500°F *at 16 seconds* after flooding commenced in FLECHT run 9573.[112] And Westinghouse reported that "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F."[113] Therefore, at locations at which heater rods started to fail at approximately 18 seconds after flooding commenced, the localized temperatures were in excess of 2500°F—more than 80°F higher than the highest temperature predicted by NRC's TRACE simulation using the Baker-Just correlation; and more than 160°F higher than the highest temperature predicted using the Cathcart-Pawel correlation. Hence, NRC's TRACE simulations of FLECHT run 9573 indicate that the Baker-Just and Cathcart-Pawel correlations are not sufficiently conservative for use in computer safety models.

(See Appendix A for information about the BWR FLECHT Zr2K test and TH-1 test 130, design basis accident experiments in which runaway oxidation (most likely) commenced and almost commenced, respectively, at fuel-cladding temperatures that were lower than the 2200°F PCT limit. And see Appendix D for information about experiments in which zirconium-steam reaction rates occurred that are under-predicted by computer safety models.)

It is also important to recognize the limitations of thermal hydraulic LOCA experiments that were conducted with stainless steel bundles. Of course, nuclear power plants use zirconium alloy fuel-cladding—not stainless steel fuel-cladding.

---

111 NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.

112 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.

113 *Id*.

Respectfully submitted,

/s/

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

Dated: April 12, 2014

32-7

Appendix A Experiments in which Runaway Oxidation (Most Likely) either Commenced or Almost Commenced at Fuel Cladding Temperatures Lower than the 2200°F PCT Limit

I. An Experiment in which Runaway Oxidation Most Likely Commenced at a Temperature Lower than the 2200°F PCT Limit: The BWR FLECHT Zr2K Test

NRC's October 2012 Draft Interim Review ("DIR") of PRM-50-93/95 concluded that "autocatalytic reactions have not occurred at temperatures less than [the 2200°F PCT limit];"[1] however, the NRC's DIR overlooked information Petitioner presented on the BWR FLECHT Zr2K test. (The BWR FLECHT Zr2K test is discussed on pages 35-45 of Petitioner's comments on PRM-50-93, dated March 15, 2010,[2] with information in Appendix F of the March 15, 2010 comments; and discussed on pages 39-49 of PRM-50-95, with information in Appendix G of PRM-50-95.)

In the Atomic Energy Commission's ("AEC") emergency core cooling systems ("ECCS") rulemaking hearing, conducted in the early 1970s, Dr. Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[3] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[4] conducted with a pressurized Zircaloy multi-rod bundle. Among other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [Zircaloy-steam] reactions"[5] and that the test "was

---

[1] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12265A277, p. 2.

[2] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.

[3] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists ("UCS").

[4] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-17; this paper cites UCS, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

[5] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-18.

saved only as a 'consequence of the extensive heater failures that occurred'."[6] In the hearing, Dr. Roger Griebe, the Aerojet Nuclear Company (Aerojet) project engineer who coordinated the BWR-FLECHT program, testified that "there is *no* convincing proof available from [Zr2K] test data to demonstrate that near-thermal runaway definitely did not exist" in the Zr2K test [emphasis not added].[7, 8]

(Petitioner would argue that actual thermal runaway—not *near* thermal runaway—occurred in the BWR FLECHT Zr2K test, because local test bundle cladding temperatures increased from lower than 2200°F to greater than 2900°F in approximately 40 seconds.[9])

General Electric ("GE") argued that the exothermic Zircaloy-steam reaction was insignificant in the thermal response of the Zircaloy heater rods and estimated that the energy from the exothermic Zircaloy-steam reaction was between 5 and 10% of the total energy input.[10] However, it is probable that GE was incorrect: in some of the BWR CORA experiments, conducted years later, in the 1980s, the Zircaloy-steam reaction contributed between 33 and 48% of the total energy input, once cladding temperatures reached approximately 2200°F.[11]

Thermocouple (a temperature measuring device) measurements taken during the Zr2K test, recorded that at between approximately 2100 and 2200°F, local cladding temperatures began to rapidly increase, leading to increases of tens of degrees Fahrenheit per second: in some intervals (approximately 20 seconds long), there were local

---

[6] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-24; this paper cites UCS, "An Evaluation of Nuclear Reactor Safety," p. 5.63, as the source of this information.

[7] Official Transcript of the AEC's Emergency Core Cooling Systems Rulemaking Hearing, pp. 7138-7139.

[8] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, UCS, 1974, p. 5.11.

[9] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Figure 12, as the source of this information.

[10] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-18, A8-19.

[11] 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. 5.

temperature increases of several hundred degrees Fahrenheit.[12] The thermocouples recorded that local cladding temperatures increased to greater than 2900°F.

GE argued that the thermocouple measurements of the rapid cladding-temperature increases taken in the Zr2K test were not valid, claiming "that the 'erratic thermocouple outputs[13] do not represent actual cladding temperatures, but are the result of equipment malfunctions'[14] associated with the Zr2K test."[15] In the rulemaking hearing, the AEC agreed with GE that the thermocouple measurements of the rapid cladding-temperature increases taken in the Zr2K test were not valid; the AEC stated that "[i]n [the Zr2K test], the maximum cladding temperature was approximately 2250°F."[16]

However, it is highly probable that GE and the AEC were incorrect: the thermocouple measurements taken in the Zr2K test resemble thermocouple measurements taken in BWR severe fuel damage experiments, in which there were rapid cladding-temperature increases that commenced below 2200°F, leading to increases of

---

[12] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-25, A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13l97, June 1971, Figures A-11, A-12, and J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Figure 12, as the sources of this information.

[13] A California Institute of Technology report which analyzed data from the Zr2K test, concluded that the observed thermocouple measurements were not erratic; see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-21, A8-23.

[14] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Appendix D, p. 107.

[15] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-24, A8-27.

[16] 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. 1104-1105. This document is available at: NRC's ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999; the source of this information is Exhibit 1069, pp. 53-54, from the rulemaking hearing.

tens of degrees Fahrenheit per second. Local cladding temperatures in such experiments exceeded 2900°F.[17]

In the ECCS rulemaking hearing, Dr. Kendall and Ford contended in their direct testimony that "GE's interpretation of [the Zr2K test] is based on a…maximum cladding temperature curve that…constituted false reporting of the test data;" [18] and Dr. Griebe testified "that GE 'tremendously slanted' BWR-FLECHT data "towards the lower temperatures and towards the interpretation GE obviously presented in their report'."[19]

(In their final decision on the issues raised in the ECCS rulemaking hearing, the AEC commissioners observed that "[t]he conditions in [the BWR FLECHT Zr-2 test] were stated to be significantly more severe than the conditions reasonably expected to prevail during a postulated BWR LOCA, even for the 'hot' bundle."[20])

II. An Experiment that Most Likely Would have Incurred Runaway Oxidation if the Reactor had Not Shutdown When Maximum Fuel Cladding Temperatures Were Approximately 1850°F: Thermal Hydraulic 1 Test 130

In NRC's October 2012 DIR of PRM-50-93/95, NRC states that "[b]ecause of the initial high temperature in FLECHT run 9573, the conditions exceeded design basis LOCA conditions and were more typical of a severe accident test."[21] Indeed, FLECHT run 9573 had high initial cladding temperatures (the BWR FLECHT Zr-2 test also exceeded design basis LOCA conditions, as noted in Section I of Appendix A). However, a different PWR LOCA test (NRU Thermal Hydraulic 1 ("TH-1") test 130), which in some ways resembles FLECHT run 9573, did not have high initial cladding temperatures; TH-1 test no. 130 was also conducted with a relatively low power level.

---

[17] L. Sepold *et al.*, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," FZKA 7447, pp. I, 1.

[18] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," pp. 5.12, 5.14.

[19] *Id*.

[20] Dixy Lee Ray *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," pp. 1104-1105; the source of this information is Exhibit 1148, p. P-15, from the rulemaking hearing.

[21] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," p. 7.

(TH-1 test no. 130 is discussed on pages 24-25 of Petitioner's comments on PRM-50-93/95, dated December 27, 2010,[22] on page 5 of Petitioner's comments on PRM-50-93/95, dated July 27, 2011,[23] and on pages 9-11 of Petitioner's comments on PRM-50-93/95, dated July 30, 2011.[24])

In TH-1 test no. 130, there was a reflood rate of 0.74 in./sec.[25] At the onset of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[26] (TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 was 0.38 kW/ft.[27])

In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures increased by 190°F, because of the heat generated from the zirconium-steam reaction (of course, there would have also been a slight amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[28]

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 exceeded 2040°F. In fact, it is highly probable that the multi-rod bundle in the TH-1 test no. 130, would have incurred runaway oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

---

[22] Mark Leyse, Comments on PRM-50-93/95, December 27, 2010, available at: NRC's ADAMS Documents, Accession Number: ML110050023.

[23] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.

[24] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.

[25] C. L. Mohr *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, available at: NRC's ADAMS Documents, Accession Number: ML101960414, Abstract, p. v. The Abstract states that the lowest reflood rate in the TH-1 tests was 1.88 cm/ sec (0.74 in./sec); the Summary states that the lowest reflood rate in the TH-1 tests was 0.74 in./sec; page 13 states that the reflood rate of TH-1 test no. 130 was 0.7 in./sec: so the value of "0.7 in./sec," given on page 13, was rounded off from 0.74 in./sec.

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

[27] 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, available at: NRC's ADAMS Documents, Accession Number: ML083470834, p. 9-40.

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

32-8

III. In the PHEBUS B9R-2 Test, a Rapid Fuel-Cladding Temperature Escalation Commenced at Approximately 1880°F

(The information discussed in this section was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[29] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[30]

Discussing PHEBUS B9R-2, the 1996 European Commission report states:

> The B9R-2 test (second part of B9R) illustrates the oxidation in different cladding conditions representative of a pre-oxidized and fractured state. … During B9R-2, an unexpected strong escalation of the oxidation of the remaining Zr occurred when the bundle flow injection was switched from helium to steam while the maximum clad temperature was equal to 1300 K [1027°C (1880°F)].[31]

According to an October 2000 OECD Nuclear Energy Agency report, the initial heatup rate in PHEBUS B9R-2 was less than 0.1°C/sec up to 727°C (1340°F) (during the pure helium phase of the experiment).[32] However, according to a graph with a plot of fuel-cladding temperature values at the 0.6 meter "hot level" of the PHEBUS B9R-2 test bundle, the initial heatup rate in PHEBUS B9R-2 was approximately 1.0°C/sec up to 727°C (1340°F); however, the heatup rate decreases to lower than 0.2°C/sec between

---

[29] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 311.

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

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

[32] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix Update 1996-1999," NEA/CSNI/R(2000)21, October 2000, p. 97.

approximately 877°C (1610°F) and 1002°C (1835°F).[33] (See Figure 1.) As stated, the cladding-temperature escalation commenced at approximately 1027°C (1880°F).

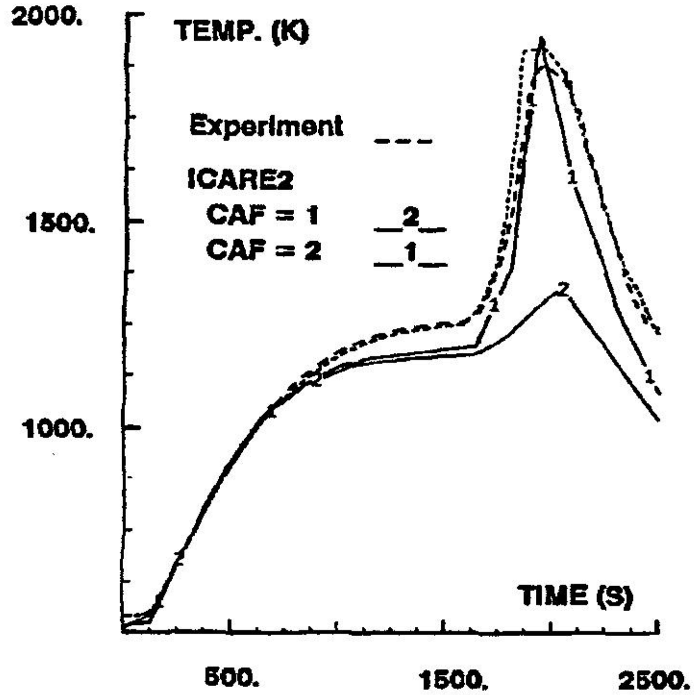
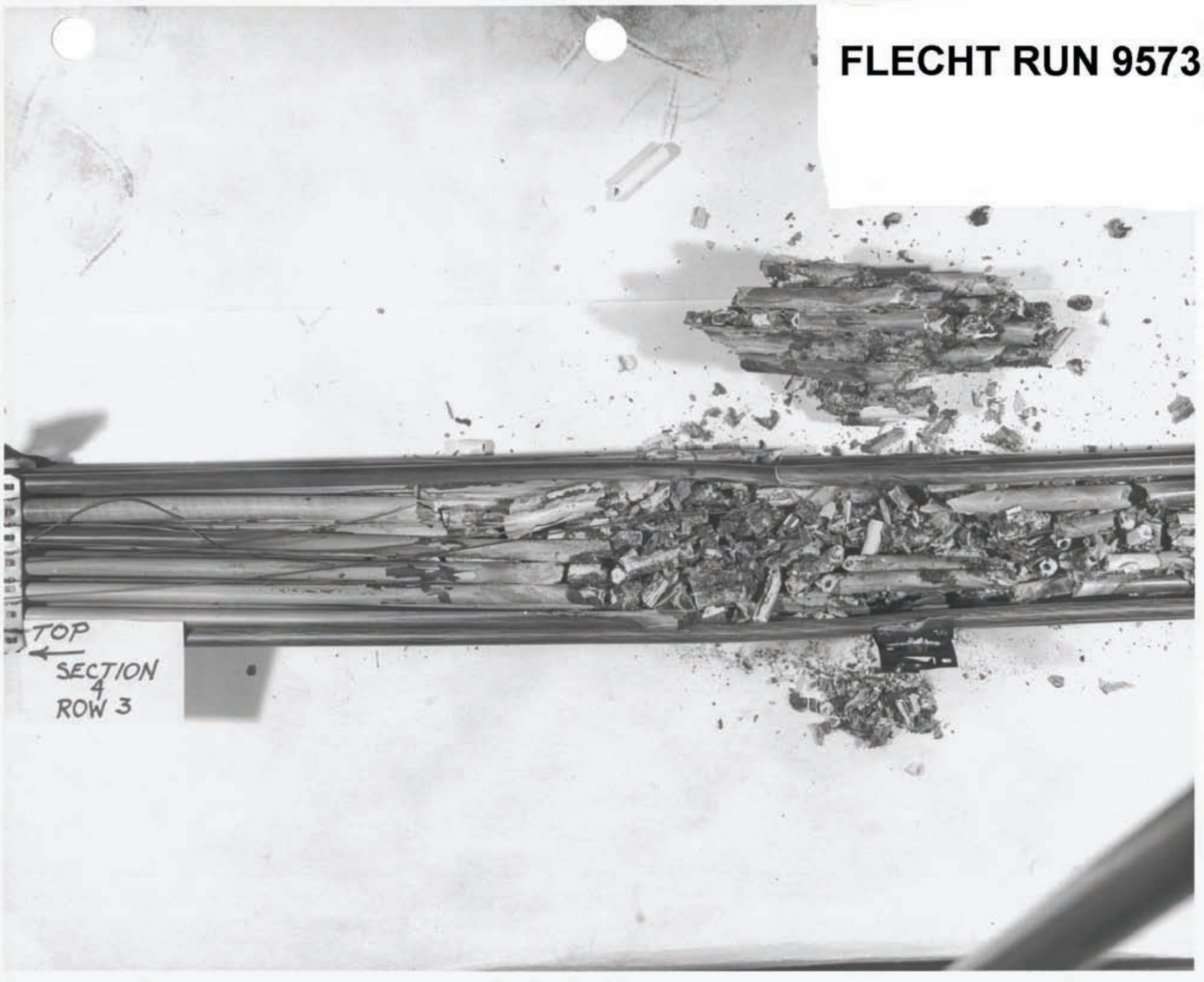


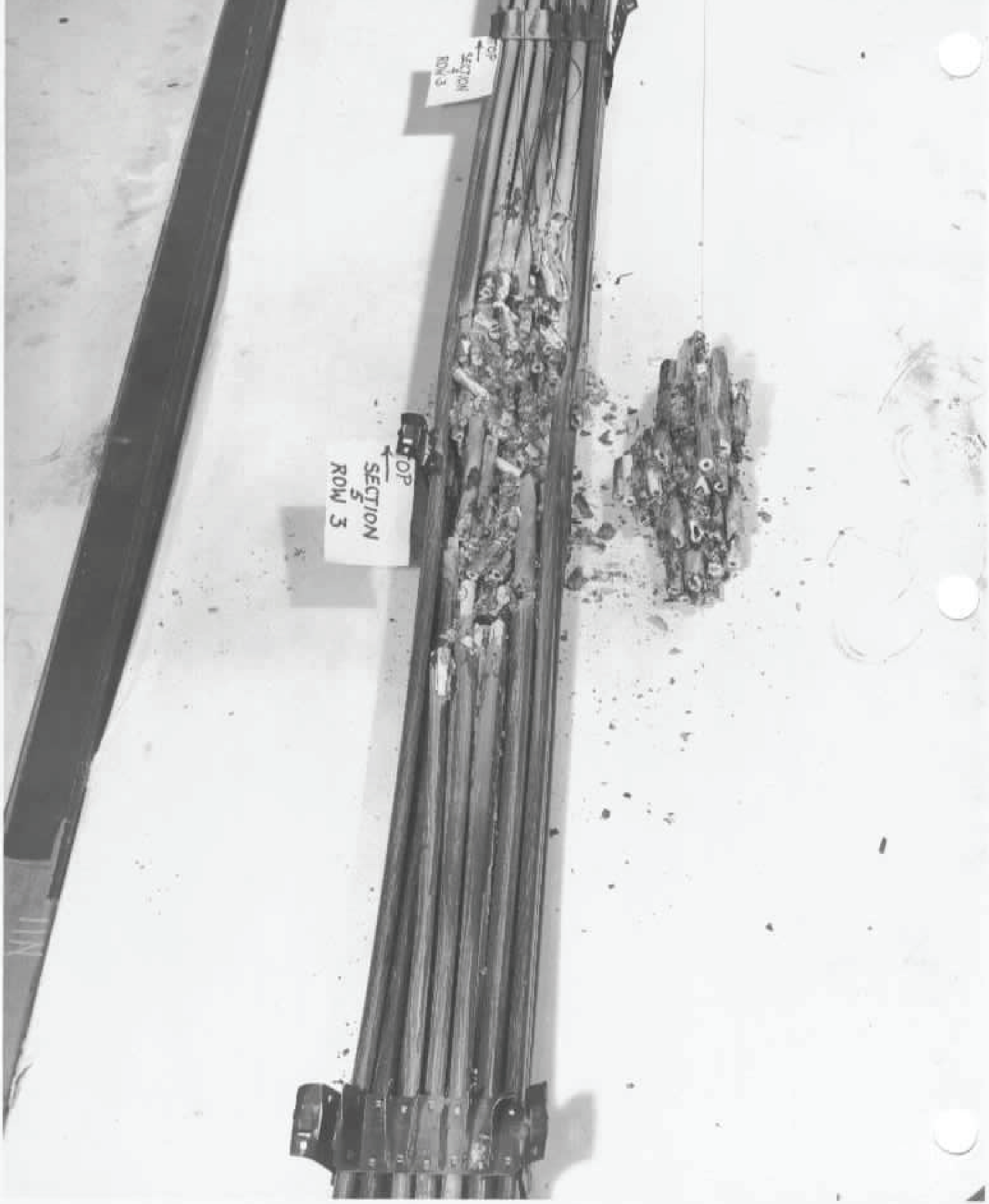
Figure 1. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[34]

[33] G. Hache, R. Gonzalez, B. Adroguer, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, p. 312.
[34] *Id*.

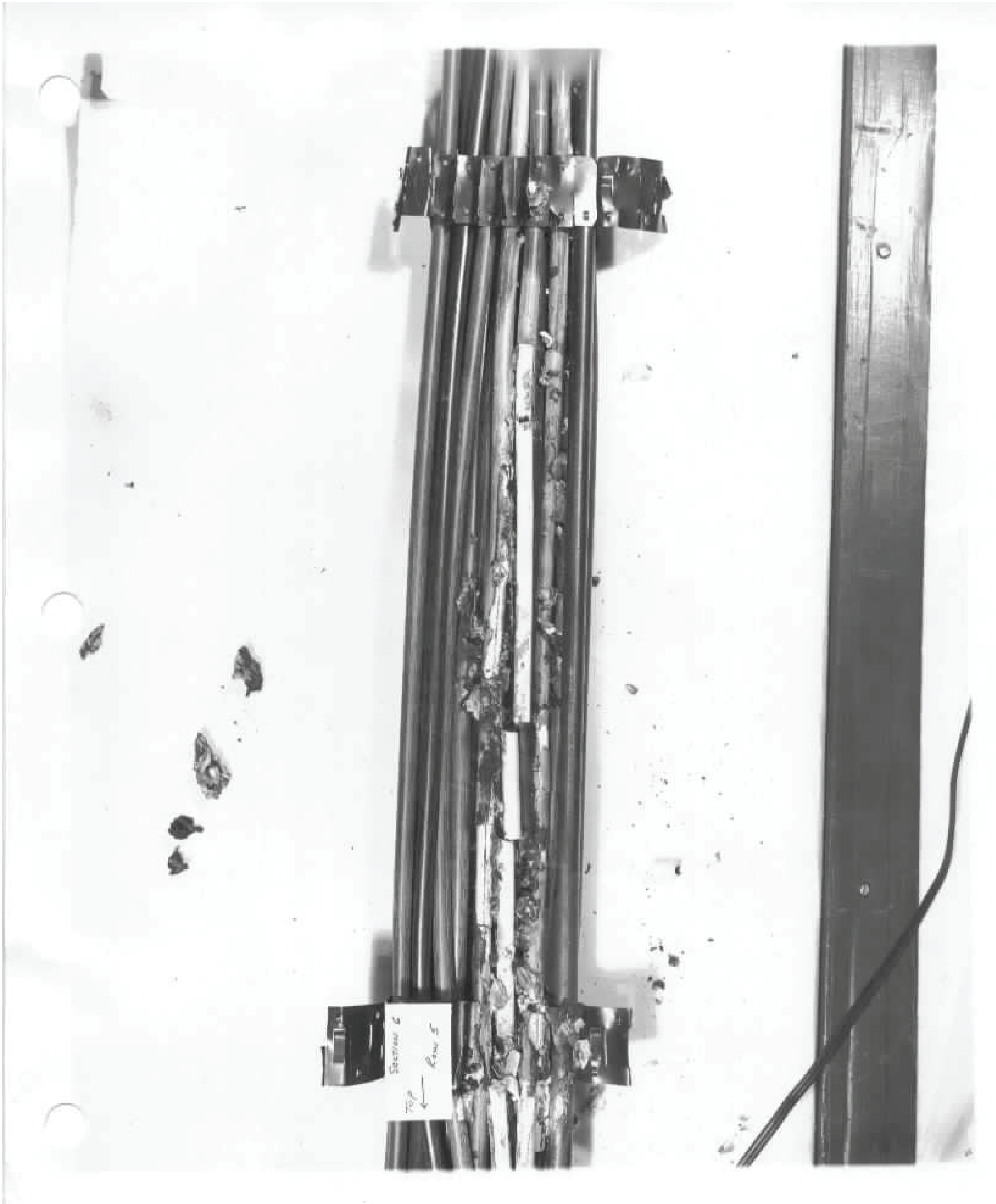
Appendix B Photographs of the Section of the Test Bundle from FLECHT Run 9573 that Incurred Runaway Oxidation

FLECHT RUN 9573




SECTION 4 ROW 3
SECTION 5 ROW 3

Appendix C Photograph of the Section of the Test Bundle from FLECHT Run 8874 that Incurred Runaway Oxidation


Section 6
Row 5
Top

Appendix D Experiments in which Zirconium-Steam Reaction Rates Occurred that Exceed the Rates Predicted by Computer Safety Models

I. Severe Accident Experiments in which Hydrogen Generation Rates Occurred that Exceed the Rates Predicted by Computer Safety Models

32-9 In Petitioner's comments on PRM-50-93/95 (page 5), dated April 7, 2011,[1] Petitioner quoted an OECD Nuclear Energy Agency report, published in 2001, which explicitly states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[2] PRM-50-93/95 argues that computer safety models using either the Baker-Just correlation or Cathcart-Pawel correlation—both among the available Zircaloy-steam oxidation correlations—under-predict the zirconium-steam reaction rates that would occur in loss-of-coolant accidents and severe accidents. However, NRC's draft interim reviews of PRM-50-93/95 on the CORA and LOFT LP-FP-2 experiments neither discuss nor mention Nuclear Energy Agency's statement, which pertains to the Baker-Just and Cathcart-Pawel correlations.

In fact, NRC's August 2011 Draft Interim Review ("DIR") of PRM-50-93/95, NRC concludes:

> The results of [the] CORA [experiments] do not suggest that the Cathcart-Pawel or Baker-Just correlations are non-conservative. The assertions made by the petition with regards to Cathcart-Pawel and Baker-Just are not substantiated by the CORA data.[3]

And NRC's September 2011 DIR of PRM-50-93/95, NRC concludes:

> A close examination of thermocouple data for LOFT LP-FP-2 found that the heatup rates below 2200ºF did not indicate presence of an exothermic "autocatalytic" reaction. The results of LOFT Test LP-FP-2 do not therefore suggest that the Cathcart-Pawel or Baker-Just correlations are

---

[1] Mark Leyse, Comments on PRM-50-93/95, April 7, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111020046.

[2] Report by Nuclear Energy Agency ("NEA") Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNIIR(2001)15, October 1, 2001, Part I, B. Clement (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9.

[3] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: NRC's ADAMS Documents, Accession Number: ML112211930, p. 3.

non-conservative. The assertions made in PRM-50-93/95 with regards to Cathcart-Pawel and Baker-Just are not substantiated by the results of this LOFT test.[4]

(As discussed in Section I of Petitioner's letter with comments on NRC's DIRs of PRM-50-93/95, NRC has overlooked data that NRC provided in September 2011 demonstrating that runaway oxidation commenced in LOFT LP-FP-2 when fuel-cladding temperatures were lower than the 2200°F peak cladding temperature ("PCT") limit.)

It is unfortunate that NRC overlooked the Nuclear Energy Agency's statement that the available Zircaloy-steam oxidation correlations—which the Baker-Just and Cathcart-Pawel correlations are among—are not suitable for use in computer safety models to determine the increased hydrogen production in the CORA and LOFT LP-FP-2 experiments.

The Nuclear Energy Agency's statement pertains to the increased hydrogen production that would occur in severe accidents during a reflooding of an overheated reactor core.[5] A 1999 paper explains that "[n]o models are yet available to predict correctly the quenching processes in the CORA and LOFT LP-FP-2 tests. …the increased hydrogen production during quenching cannot be determined on the basis of the available Zircaloy/steam oxidation correlations."[6]

The Nuclear Energy Agency's statement does not pertain to the design basis accident temperature range. However, PRM-50-95—originally a 10 C.F.R. § 2.206 enforcement action petition, *which NRC decided to make into a petition for rulemaking*[7]—discusses boiling water reactor ("BWR") severe accident phenomena, in

4 NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," September 2011, available at: NRC's ADAMS Documents, Accession Number: ML112650009, p. 5.

5 Report by Nuclear Energy Agency ("NEA") Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNIIR(2001)15, October 1, 2001, Part I, B. Clement (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9.

6 Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, Vol. 270, 1999, pp. 207-208.

7 Mark Leyse, PRM-50-95, June 7, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101610121. (PRM-50-95 was originally a 10 C.F.R. § 2.206 enforcement action petition that Petitioner wrote on behalf of New England Coalition (NEC), dated June 7, 2010. In

addition to phenomena which would occur in the design basis accident temperature range: fuel cladding temperatures lower than the 2200°F PCT limit. Given that the Fukushima Dai-ichi accident occurred in March 2011 and that NRC has since performed simulations of BWR severe accidents with the MELCOR computer safety model, it would seem appropriate for NRC to acknowledge that MELCOR under-predicts the hydrogen generation rates that occur during a reflooding of an overheated reactor core.

II. Computer Safety Models Fail to Accurately Predict the Onset of the Fuel-Cladding Temperature Escalation that Commenced in the LOFT LP-FP-2 Experiment (in the Design Basis Accident Temperature Range)

As discussed in Section I of Petitioner's letter with comments on NRC's DIRs of PRM-50-93/95, the onset of the fuel-cladding temperature escalation commenced in the LOFT LP-FP-2 experiment when fuel-cladding temperatures were lower than the 2200°F PCT limit.

Computer safety models have failed to accurately predict the onset of the fuel-cladding temperature escalation that occurred in the LOFT LP-FP-2 experiment. Regarding a fairly recent computer safety model (ASTEC V1.3 code) simulation of the LOFT LP-FP-2 experiment, a 2010 paper, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17), but the onset of temperature escalation in the upper part of the CFM [center fuel module] was delayed.[8]

In "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17, the graph of the cladding-temperature values in the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment depicts that the onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature greater than 1700 K (2600°F); figure 17 also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature well

October 2010, NRC published a notice in the Federal Register stating that it had determined that the NEC petition, met the requirements for a petition for rulemaking under 10 C.F.R. § 2.802.)

[8] G. Bandini *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," Progress in Nuclear Energy, 52, 2010, p. 155.

below 1500 K (2240°F).[9] Hence, the difference between the calculated and actual experimental value for the onset of the temperature escalation (at the 1.067 m elevation) is greater than 200 K (360°F)—a significant difference.

(It is noteworthy that, regarding the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment during reflood, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflooding were not reproduced by ASTEC due to lack of adequate modeling.[10])

III. An Experiment for which the Quantity of Hydrogen Produced by the Zircaloy-Steam Reaction at about 1800°F Is Under-Predicted by Computer Safety Models: The FRF-1 Experiment

The FRF-1 experiment—conducted in the TREAT facility[11]—was not a large-scale experiment yet Union of Concerned Scientists and the authors of a report on the FRF-1 experiment[12] claimed that, as of 1971, it simulated "the most realistic loss-of-coolant accident conditions of any experiment to date."[13]

(The FRF-1 experiment is discussed in Petitioner's comments on PRM-50-93/95, dated November 23, 2010 (pages 37-45),[14] and dated July 27, 2011 (pages 1-2);[15] and in Appendix A to Petitioner's comments on PRM-50-93/95, dated November 23, 2010, there is a graph depicting the maximum cladding temperatures which occurred in the FRF-1 experiment.)

---

[9] *Id*.
[10] *Id*.
[11] The First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") was conducted in the Transient Reactor Test Facility ("TREAT").
[12] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971.
[13] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer-Verlag, New York, 2000, p. 43.
[14] Mark Leyse, Comments on PRM-50-93, November 23, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340249.
[15] Mark Leyse, Comments on PRM-50-93/95, July 27, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11209C490.

Data from the FRF-1 experiment indicates that computer safety models under predict the quantity of hydrogen produced by the Zircaloy-steam reaction. In the experiment, at fuel rod temperatures of about 1800°F, the Zircaloy-steam reaction generated 1.2 ± 0.6 liters of hydrogen. In the Indian Point Unit 2 ("IP-2") licensing hearing, Westinghouse Electric, which had performed experimental simulations of loss-of-coolant accidents, and conducted computer simulations of such accidents, testified that their computer safety models predicted that there would be no zirconium-steam reaction at 1800°F—that no hydrogen would be produced in a loss-of-coolant accident if local temperatures of the fuel rods were to reach 1800°F.[16]

In the IP-2 licensing hearing, Dr. Jack Roll of Westinghouse contended that data from the FRF-1 experiment was not reliable, because "the measurement of the extent of [zirconium-steam] reaction was in fact by an inferred route, and there were no direct measurements taken," that "[t]here was a large uncertainty in the measurement of total hydrogen evolution during the experiment," and that there was "an uncertainty in the temperatures of the fuel [rods] during the experiment."[17] Westinghouse concluded that it is not possible to know if the data from the FRF-1 experiment actually demonstrated that the extent of the zirconium-steam reaction was higher (or much higher) than would be predicted by computer safety models.

Unfortunately, there was not a means to confirm if Westinghouse's claims were correct or not, because the Atomic Energy Commission decided to discontinue funding for the TREAT facility loss-of-coolant accident experimental program.[18] The FRF-1 experiment could not be replicated; its results could not be confirmed.

---

[16] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, available at: NRC's ADAMS Documents, Accession Number: ML100350644, pp. 2152-2153.

[17] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, available at: NRC's ADAMS Documents, Accession Number: ML100350642, pp. 2297-2299.

[18] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.

IV. Problems with the Explanation for Why Low-Temperature Oxidation Rates Are Under-Predicted for the CORA-16 Experiment

As stated in PRM-50-95 (pages 12, 13, 26, 27) and in Petitioner's comments on PRM-50-93/95, March 15, 2010 (page 30),[19] dated April 12, 2010 (page 8),[20] dated November 24, 2010 (page 7),[21] dated July 30, 2011 (page 16),[22] and April 16, 2012 (pages 6, 7, 9, 11, 20),[23] when investigators compared the results of the CORA-16 experiment—a BWR severe fuel damage test, simulating a meltdown, conducted with a multi-rod zirconium alloy bundle—with the predictions of computer safety models, they found that the zirconium-steam reaction rates that occurred in the experiment were under-predicted. The investigators concluded that the "application of the available Zircaloy oxidation kinetics models [zirconium-steam reaction correlations] causes the low-temperature [1652-2192°F] oxidation to be underpredicted."[24]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in the CORA-16 experiment.[25] (In Petitioner's comments on PRM-50-93/95, dated April 16, 2012 (pages 5-13),[26] Petitioner provided information indicating that it is *unlikely* that cladding strain increased the zirconium-steam reaction rates that occurred in the CORA-16 experiment; it is certainly *unsubstantiated* that cladding strain increased reaction rates.)

---

[19] Mark Leyse, Comments on PRM-50-93, March 15, 2010, available at: NRC's ADAMS Documents, Accession Number: ML100820229.
[20] Mark Leyse, Comments on PRM-50-93/95, April 12, 2010, available at: NRC's ADAMS Documents, Accession Number: ML101020564.
[21] Mark Leyse, Comments on PRM-50-93/95, November 24, 2010, available at: NRC's ADAMS Documents, Accession Number: ML103340248; NRC dates these comments November 23, 2010.
[22] Mark Leyse, Comments on PRM-50-93/95, July 30, 2011, available at: NRC's ADAMS Documents, Accession Number: ML11213A211.
[23] Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12109A084.
[24] 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.
[25] 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.
[26] Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, available at: NRC's ADAMS Documents, Accession Number: ML12109A084.

In NRC's 2011 evaluation of the CORA-16 experiment, NRC stated that an ORNL paper, "In-Vessel Phenomena—CORA," noted that in CORA-16, "cladding strain could be a factor and that cladding strain and significant oxidation occurred simultaneously."[27] However, NRC erroneously observed that "In-Vessel Phenomena—CORA" "provided an analytical adjustment that improved the timing prediction with respect to the measured temperatures."[28]

In fact, the ORNL paper's authors employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16.[29] There are three graphs in the ORNL paper depicting cladding temperature plots from different cladding elevations (550 mm, 750 mm, and 950 mm) of "heated rod 5.3" in CORA-16:[30] each plot illustrates that cladding temperatures were greater in the experiment than computer safety models—using the available zirconium-steam reaction correlations—initially predicted (*with no enhancement*), indicating that zirconium-steam reaction rates were also under-predicted. Each graph also depicts predicted cladding temperature plots that were computer generated by using a simple *multiplier* to *enhance* the predicted zirconium-steam reaction rates (and the amount of heat the zirconium-steam reaction produced). By using the multiplier the predicted reaction rates were matched closer to the reaction rates that occurred in the experiment; hence, the multiplier also helped the predicted cladding temperatures match the cladding temperatures that occurred in the experiment.

NRC also erroneously stated that "In-Vessel Phenomena—CORA," did not report that computer safety models under-predicted zirconium-steam reaction rates in CORA-16:[31] a simple glance at the three graphs described above[32] reveals that the paper reported that reaction rates were under-predicted. And a second ORNL paper explicitly states that

[27] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: NRC's ADAMS Documents, Accession Number: ML112211930, p. 3.
[28] *Id*.
[29] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[30] See Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[31] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[32] See Mark Leyse, Comments on PRM-50-93/95, April 16, 2012, Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.

the low-temperature (1652°F to 2192°F) oxidation that occurred in CORA-16 was under-predicted.[33] (Petitioner has quoted the second ORNL paper in a number of different comments on PRM-50-93/95 that Petitioner has sent to NRC.)

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, NRC pointed out that an NRC report, NUREG/CR-4412,[34] "explain[s] that under *certain* conditions ballooning and deformation of the cladding can increase the available surface area for oxidation, thus enhancing the apparent oxidation rate" [emphasis not added].[35]

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA or [severe accident] will permit [fuel] cladding deformation (ballooning and possibly rupture) to occur because the fuel rod internal pressure may be greater than the external (coolant) pressure. In this case, oxidation and deformation can occur simultaneously. This in turn may result in an apparent enhancement of oxidation rates because: 1) ballooning increases the surface area of the cladding and permits more oxide to form per unit volume of Zircaloy and 2) the deformation may crack the oxide and provide increased accessibility of the oxygen to the metal. However deformation generally occurs before oxidation rates become significant; *i.e.*, below [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[36]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[37] NUREG/CR-4412 also explains that "it is possible to make a very crude estimate of the expected average enhancement of oxidation kinetics by deformation;"[38] the report provides a graph of the "rather sparse"[39] data. The graph indicates that the general trend

---

[33] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.
[34] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, available at: NRC's ADAMS Documents, Accession Number: ML083400371.
[35] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[36] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 27.
[37] *Id*., pp. 27, 30.
[38] *Id*., p. 30.
[39] *Id*.

is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[40]

NUREG/CR-4412 has a brief description of the rather sparse data; in one case, two investigators (Furuta and Kawasaki), who heated specimens up to temperatures between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[41]

In fact, NUREG/CR-4412 states that only one pair of investigators (Bradhurst and Heuer) conducted tests that encompassed the temperature range—1652°F to 2192°F—in which zirconium-steam reaction rates were under-predicted for CORA-16. Bradhurst and Heuer reported that "[m]aximum enhancements occurred at slower strain rates. … However, the overall weight gain or average oxide thickness in [the Zircaloy-2 specimens] was only minimally increased because of the localization effects of cracks in the oxide layer." [42] A second report states that "Bradhurst and Heuer…found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[43] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a very brief period of time, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain *may* only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater[44] (in a LOCA environment in which local cladding temperatures would be increasing rapidly). Furthermore, ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain

---

[40] *Id*., p. 29.
[41] *Id*., p. 30.
[42] *Id*.
[43] F. J. Erbacher, S. Leistikow, "A Review of Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident," Kernforschungszentrum Karlsruhe, KfK 3973, September 1985, p. 6.
[44] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 30.

actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F temperature range for CORA-16.

There is also one phenomenon NRC did not consider in its 2011 analysis of CORA-16: "[t]he swelling of the [fuel] cladding…alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,"[45] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

In NRC's 2011 evaluation of CORA-16, NRC concluded that the fact zirconium-steam reaction rates were under-predicted by computer safety models—using the available zirconium-steam reaction correlations—"is inadequate as a basis to revise regulations or invalidate the use of [the] Baker-Just and Cathcart-Pawel [correlations] for design basis calculations of oxidation."[46] (The Baker-Just and Cathcart-Pawel correlations are among the available zirconium-steam reaction correlations.) NRC's conclusion is unsubstantiated, as the information presented in this section indicates. When NRC chooses to invalidate experimental data, which is important for simulating accidents, with unsubstantiated postulations, NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative computer safety models.[47]

A plausible explanation for why zirconium-steam reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F by computer

[45] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, available at: NRC's ADAMS Documents, Accession Number: ML041950059, p. 43.

[46] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.

[47] Charles Miller, *et al*., NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," SECY-11-0093, July 12, 2011, available at: NRC's ADAMS Documents, Accession Number: ML111861807, p. 3.

safety models would be that the currently used zirconium-steam reaction correlations are inadequate for use in computer safety models.

V. Oxidation Models Are Not Able to Predict the Fuel-Cladding Temperature Escalation that Commenced at Approximately 1880°F in the PHEBUS B9R-2 Test

(The information discussed in this section was neither provided in PRM-50-93/95 nor in comments on PRM-50-93/95.)

32-10

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[48] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region; the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[49]

Discussing PHEBUS B9R-2, the 1996 European Commission report states:

> The B9R-2 test (second part of B9R) illustrates the oxidation in different cladding conditions representative of a pre-oxidized and fractured state. This state results from a first oxidation phase (first part name B9R-1, of the B9R test) terminated by a rapid cooling-down phase. During B9R-2, an unexpected strong escalation of the oxidation of the remaining Zr occurred when the bundle flow injection was switched from helium to steam while the maximum clad temperature was equal to 1300 K [1027°C (1880°F)]. *The current oxidation model was not able to predict the strong heat-up rate observed* even taking into account the measured large clad deformation and the double-sided oxidation (final state of the cladding from macro-photographs).
>
> … *No mechanistic model is currently available to account for enhanced oxidation of pre-oxidized and cracked cladding*[50] [emphasis added].

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to the cladding of high burnup fuel rods. The

---

[48] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 311.

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

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

PHEBUS B9R-2 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just and Cathcart-Pawel correlations, are inadequate for use in computer safety models.

Submission ID 33
Mark Leyse
ML15288A506

PRM-50-93 & 50-95
(75FR66007)

Docketed - USNRC
October 12, 2015

From: Mark Leyse
To: Doyle, Daniel; CHAIRMAN Resource; CMRSVINICKI Resource; CMROSTENDORFF Resource; CMRBARAN Resource; RulemakingComments Resource; PDR Resource
Cc: bobleyse@aol.com; shadis@prexar.com; Burnell, Scott; Bladey, Cindy; Inverso, Tara; Dave Lochbaum; Matthew G. McKinzie; Thomas B. Cochran; Deborah Brancato; Geoffrey Fettus; Diane Curran; Paul Gunter; Paul Gallay; Alemayehu, Bemnet; michal_freedhoff@markey.senate.gov; Ed Lyman
Subject: [External_Sender] Re: Status of PRM-50-93/95
Date: Sunday, October 11, 2015 7:22:06 AM

Dear Mr. Doyle:

Thank you for your update. I appreciate that the NRC Staff is considering comments I made to the Commissioners on January 31, 2013 in its review of PRM-50-93 and PRM-50-95.

On April 12, 2014, I sent the Staff additional comments that reiterate and further expand on issues I raised in my presentation to the Commissioners. I request that the Staff also consider and respond to my April 12, 2014 comments. The April 12, 2014 comments are in ADAMS at ML14104B253.

I also appreciate that you said I may contact you and ask questions. I would like to ask several questions. They relate to issues I raised with the Commissioners on January 31, 2013.

My questions concern the TRACE computer code simulation of FLECHT Run 9573 that was performed for the Staff's review of PRM-50-93 and PRM-50-95. The TRACE simulation is discussed in the "Draft Interim Review" in ADAMS at ML12265A277.

First) Why was the severely damaged section of the test bundle of FLECHT Run 9573 *not simulated* in the TRACE simulation?

Second) Was the severely damaged section of the test bundle of FLECHT Run 9573 *intentionally omitted* from the TRACE simulation?

33-1

Third) Why were the results of the TRACE simulation of FLECHT Run 9573 compared to the *average* of the available thermocouple measurements at each particular elevation and not to the *highest* thermocouple measurement at each particular elevation?

Fourth) Given that the highest cladding temperature (the PCT) is the concern in LOCA analysis (key to power uprate calculations), do you believe the Staff erred by not comparing the *highest* thermocouple measurement at each particular elevation to the results of its TRACE simulation of FLECHT Run 9573?

Fifth) How much money did the NRC spend on its TRACE simulation of FLECHT Run 9573, including interpreting and reporting the simulation's results?

A photograph of the severely damaged section of the test bundle of FLECHT Run 9573.

I devoted many well-referenced pages to discussing FLECHT Run 9573. I provided quotes and information from a Westinghouse report that discusses FLECHT Run 9573. The Westinghouse report is referred to in the NRC's "Draft Interim Review." The Westinghouse report is in ADAMS at ML070780083.

As I stated before: Westinghouse reported, regarding FLECHT Run 9573, that a “[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation.”

(The quote is from page 3.97 of Westinghouse's “PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,” WCAP-7665, April 1971.)

My main point in discussing FLECHT Run 9573 was that a section of the test bundle overheated and heavily oxidized. Yet Staff members "simulated" the test without including what Westinghouse called the "severe damage zone."

I would appreciate it if you would answer my questions.

Thank you,

Mark Leyse

P.S. Please place this e-mail in ADAMS.

On Tue, Jun 23, 2015 at 10:51 AM, Doyle, Daniel <Daniel.Doyle@nrc.gov> wrote:

> Mr. Leyse,
>
> I am writing to provide an update on your letters dated November 17, 2009, and June 7, 2010, in which you submitted petitions to the U.S. Nuclear Regulatory Commission (NRC). In your letter dated November 17, 2009, you requested that

the NRC amend the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and Appendix K to Part 50 to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments. In addition, you requested that the NRC create a new regulation to establish a minimum allowable core reflood rate in the event of a loss-of-coolant accident (LOCA). In your letter dated June 7, 2010, you requested that the NRC order Vermont Yankee Nuclear Power Station (Vermont Yankee) to lower the licensing basis peak cladding temperature to 1,832 degrees F in order to provide a necessary margin of safety in the event of a LOCA.

The NRC docketed your November 17, 2009, letter as petition for rulemaking (PRM) 50-93. A notice of receipt and request for public comment on PRM-50-93 was published in the *Federal Register* on January 25, 2010 (75 FR 3876). Your letter dated June 7, 2010, was submitted as a petition for enforcement action under 10 CFR 2.206. On August 6, 2010, the NRC denied your § 2.206 petition because it did not demonstrate that Vermont Yankee was in violation of any NRC regulations. Because your § 2.206 petition asserted that there were generic inadequacies in NRC regulations, the NRC decided to review it under 10 CFR 2.802 as a petition for rulemaking and docketed it as PRM-50-95. Because PRM-50-93 and PRM-50-95 address similar issues, the NRC consolidated these two petitions for review as a single petition for rulemaking activity. Another *Federal Register* notice was published on October 27, 2010 (75 FR 66007), and the comment period was reopened. The public comment period ended on November 26, 2010. Thirty-two public comments have been received to date on the combined petitions. These comments have been posted at regulations.gov (ID: NRC-2009-0554).

The NRC staff is considering the merits of your PRM and the public comments received. As described in the NRC's letter to you dated August 25, 2011, the NRC has decided to increase the visibility to the public of the NRC's review of these particular petitions. The NRC will publicly release its draft interim reviews regarding each group or category of issues on a periodic basis as the review progresses. These draft interim reviews will be posted on regulations.gov. So far, the NRC has publicly released four draft interim reviews:

- Evaluation of CORA test series (8/23/11)
- Evaluation of LOFT LP-FP-2 (9/27/11)
- Evaluation of conservatism of 2200F, metal-water reaction rate correlations, and "the impression left from run 9573" (10/16/12)
- Evaluation of request to establish minimum reflood rate (3/8/13)

The NRC staff will consider and respond to the comments you made regarding PRM-50-93 and PRM-50-95 at the Commission briefing on public participation in NRC regulatory decision-making on January 31, 2013, in the review of these petitions.

The NRC is considering the remaining issues and will notify you as the draft interim reviews are completed. Once the petitions have been resolved, a notice will be published in the *Federal Register* explaining the Commission's finding. You will also receive a letter at that time notifying you of the action that the Commission has taken.

Please feel free to contact me at Daniel.Doyle@nrc.gov or 301-415-3748 if you have questions.

Sincerely,

Dan Doyle

Project Manager

U.S. Nuclear Regulatory Commission

daniel.doyle@nrc.gov

(301) 415-3748