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 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

COMMENTS ON PRM-50-93; NRC-2009-0554....................................4

I. Introduction....................................4

II. Background....................................5

II. 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 5

II. Supplementary Information to PRM-50-93 Section III.B. Reflood Rates....................................9

1. Reflood Rates and the AEC's ECCS Rulemaking Hearing....................................9

2. Reflood Rates, Cladding Temperatures at the Onset of Reflood, and TRAC-M (TRACE)....................................13

3. Downcomer Boiling and Reflood Downcomer Bypass....................................19

II. Supplementary Information to PRM-50-93 Section III.C.1.d. The LOFT LP-FP-2 Experiment....................................20

II. Supplementary Information to PRM-50-93 Section III.C.1.e. The CORA Experiments....................................23

1. Three Papers on the CORA Experiments....................................23

2. The 1990 CORA Workshop at Kernforschungszentrum Karlsruhe....................................29

II. Supplementary Information to PRM-50-93 Section III.C.1.g. The QUENCH-04 Test 31

II. Supplementary Information to PRM-50-93 Section III.C.1.h. Examining the Autocatalytic Metal-Water Reaction that Occurred during FLECHT RUN 9573....................................32

II. Supplementary Information to PRM-50-93 Section III.D.4. A Comparison of the High Temperature Oxidation Behavior of Zircaloy and Stainless Steel Assemblies....................................34

II. Examining the Autocatalytic Metal-Water Reaction that Occurred during the BWR FLECHT Zr2K Test....................................35

III. Conclusion....................................46

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

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

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-50-1, 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-50-1, 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.

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.

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]

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.

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.

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.

> 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.

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.

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.

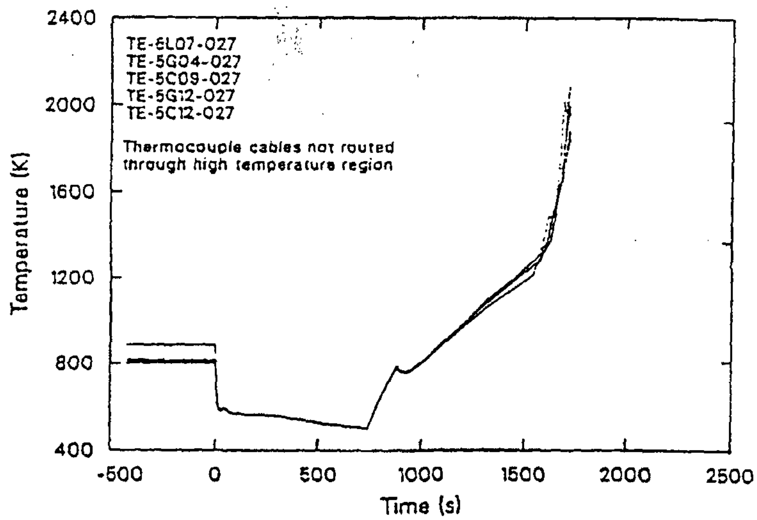


Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

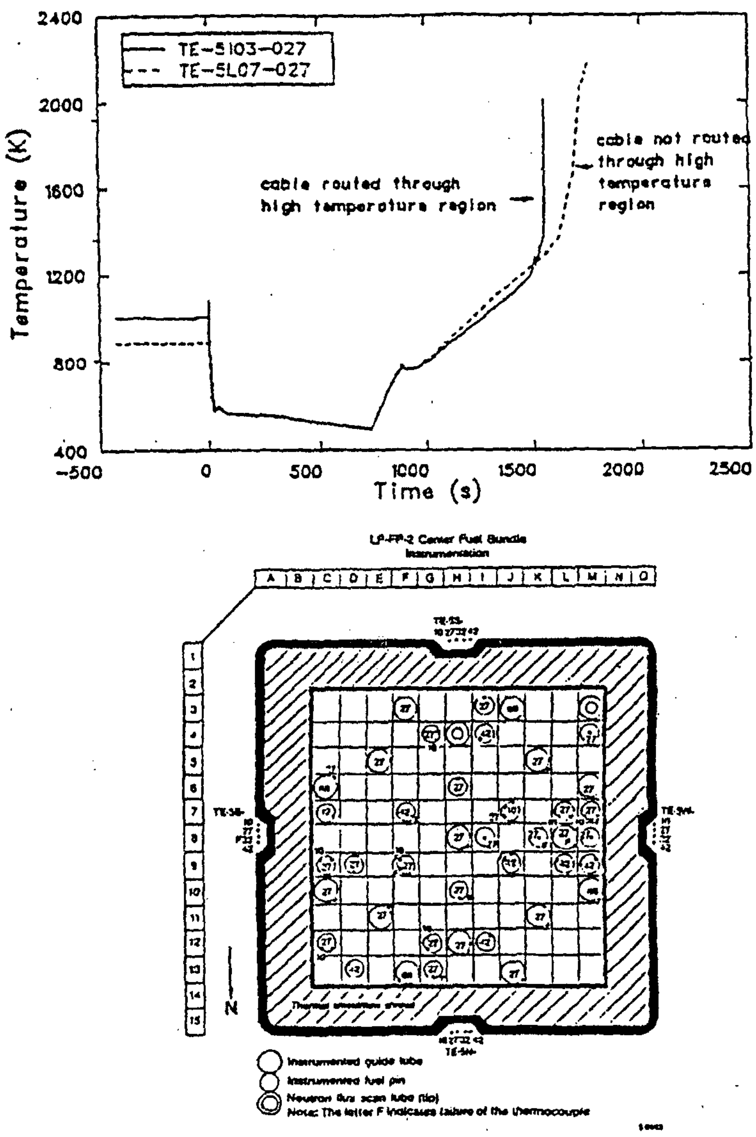
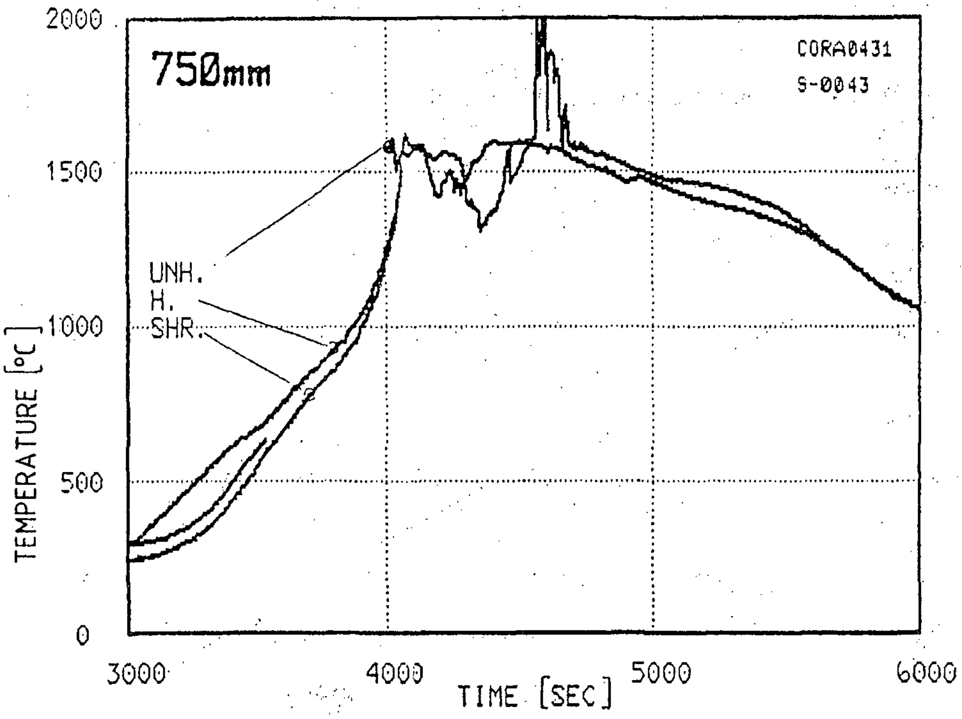


Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

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.



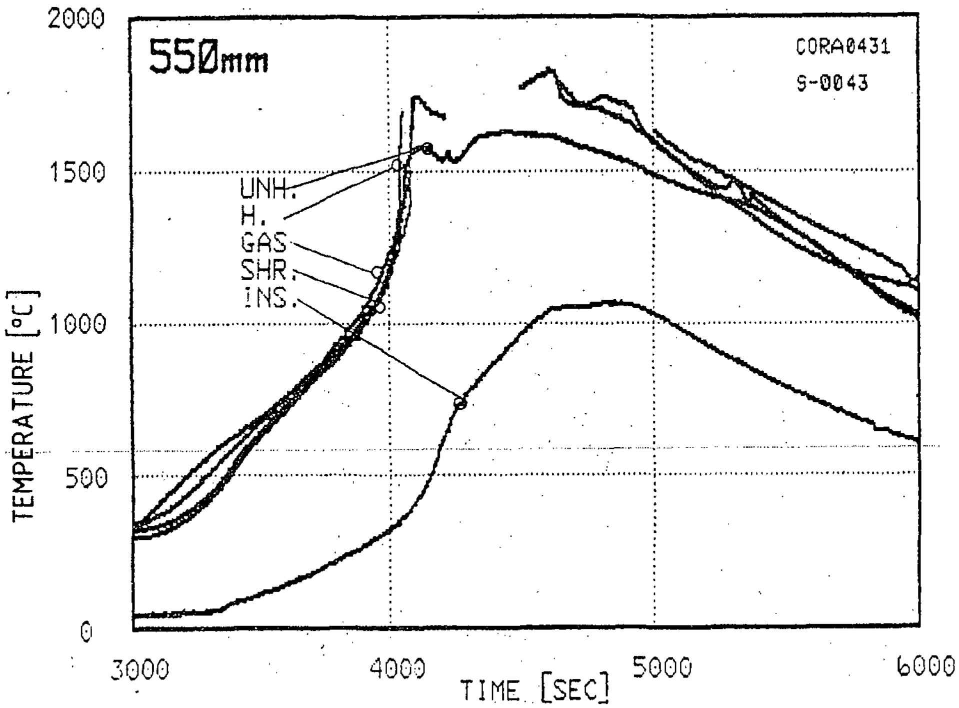
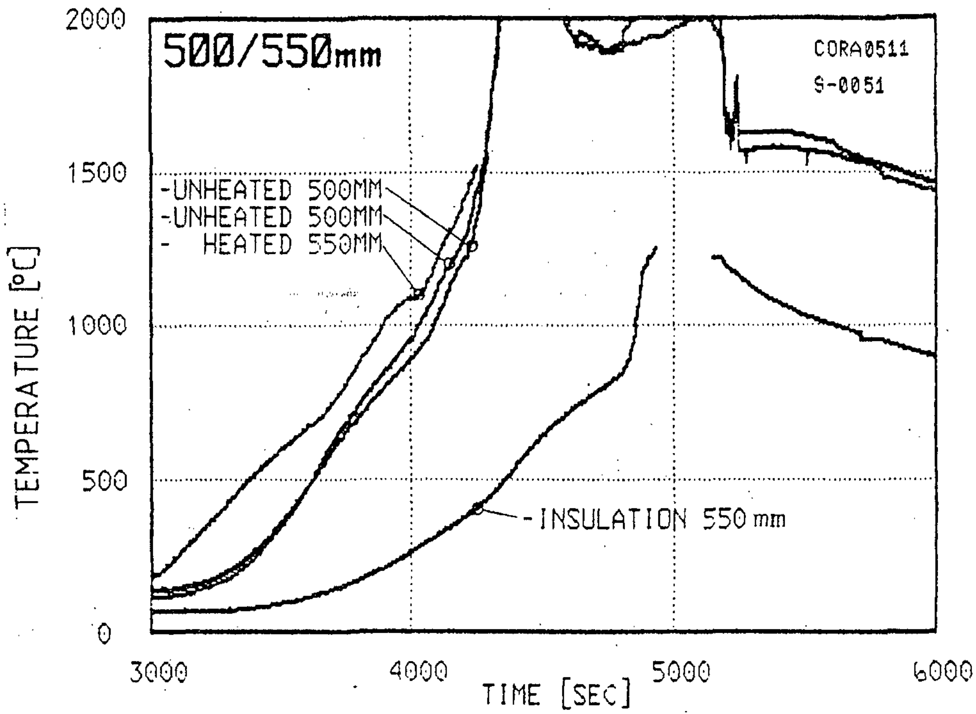


Fig. 12. **Temperatures during test CORA-2 at 500 mm and 750 mm elevation. Temperatures of heated (H) and unheated rod (UNH), atmosphere (gas), shroud (SHR), and outer surface of shroud insulation (INS)**



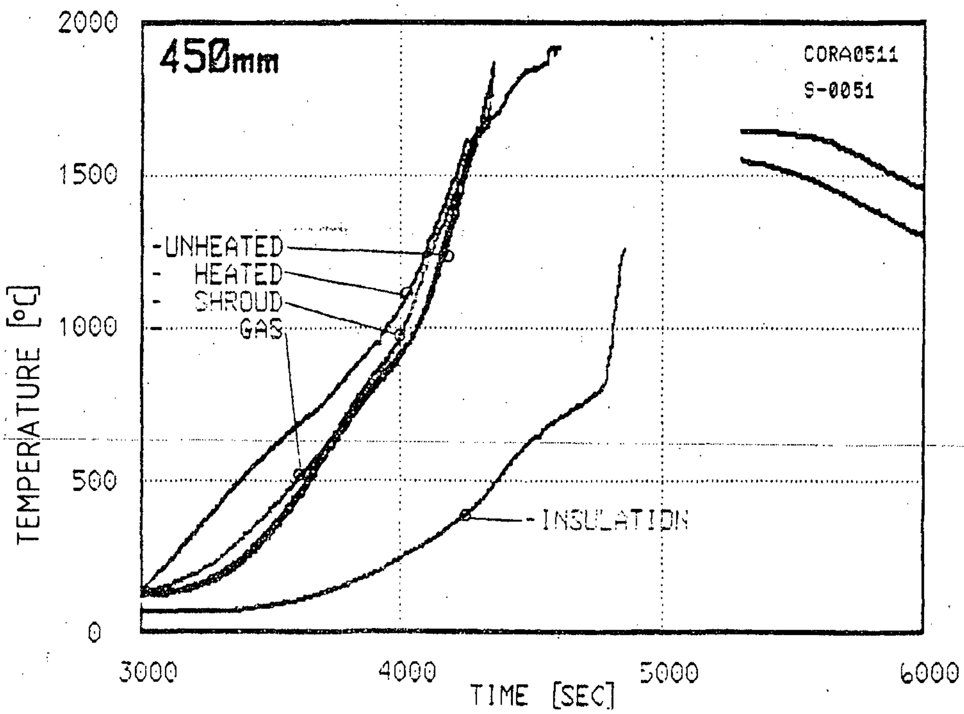


Fig. 13. Temperatures measured during test CORA-3 at 450 mm and 550 mm elevation

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.

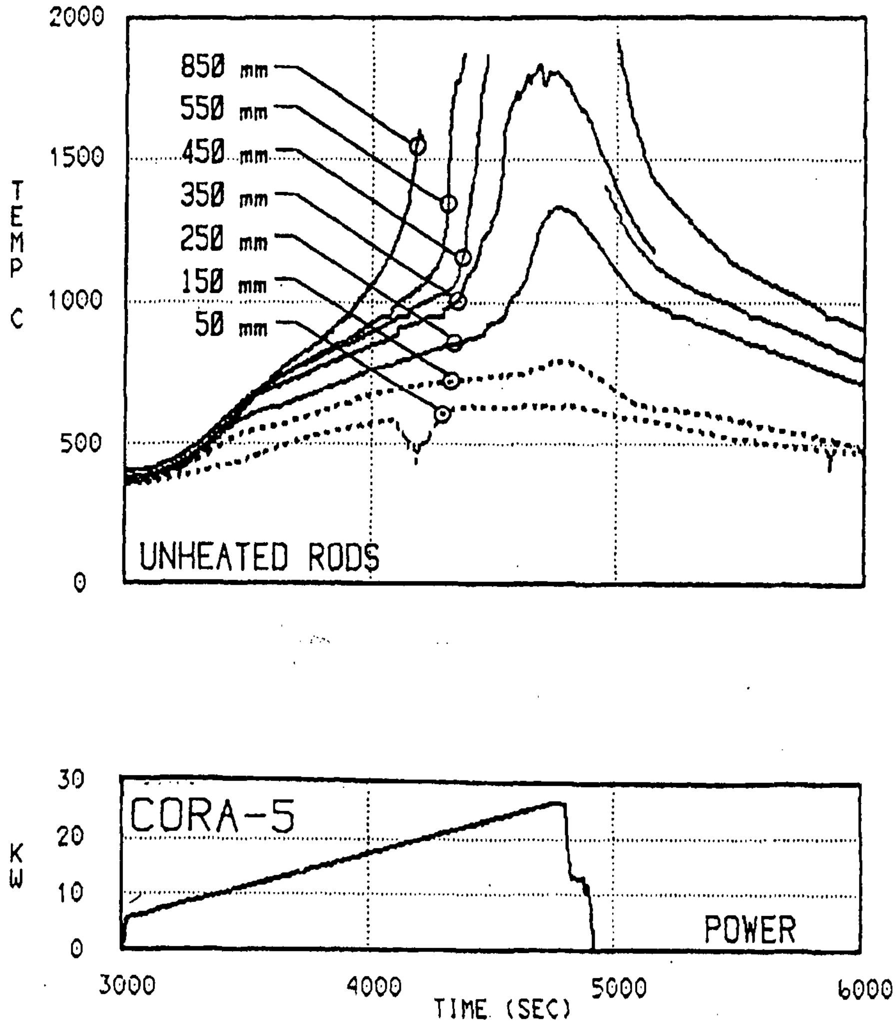


Fig.15: Temperatures of unheated rods and power history of CORA-5

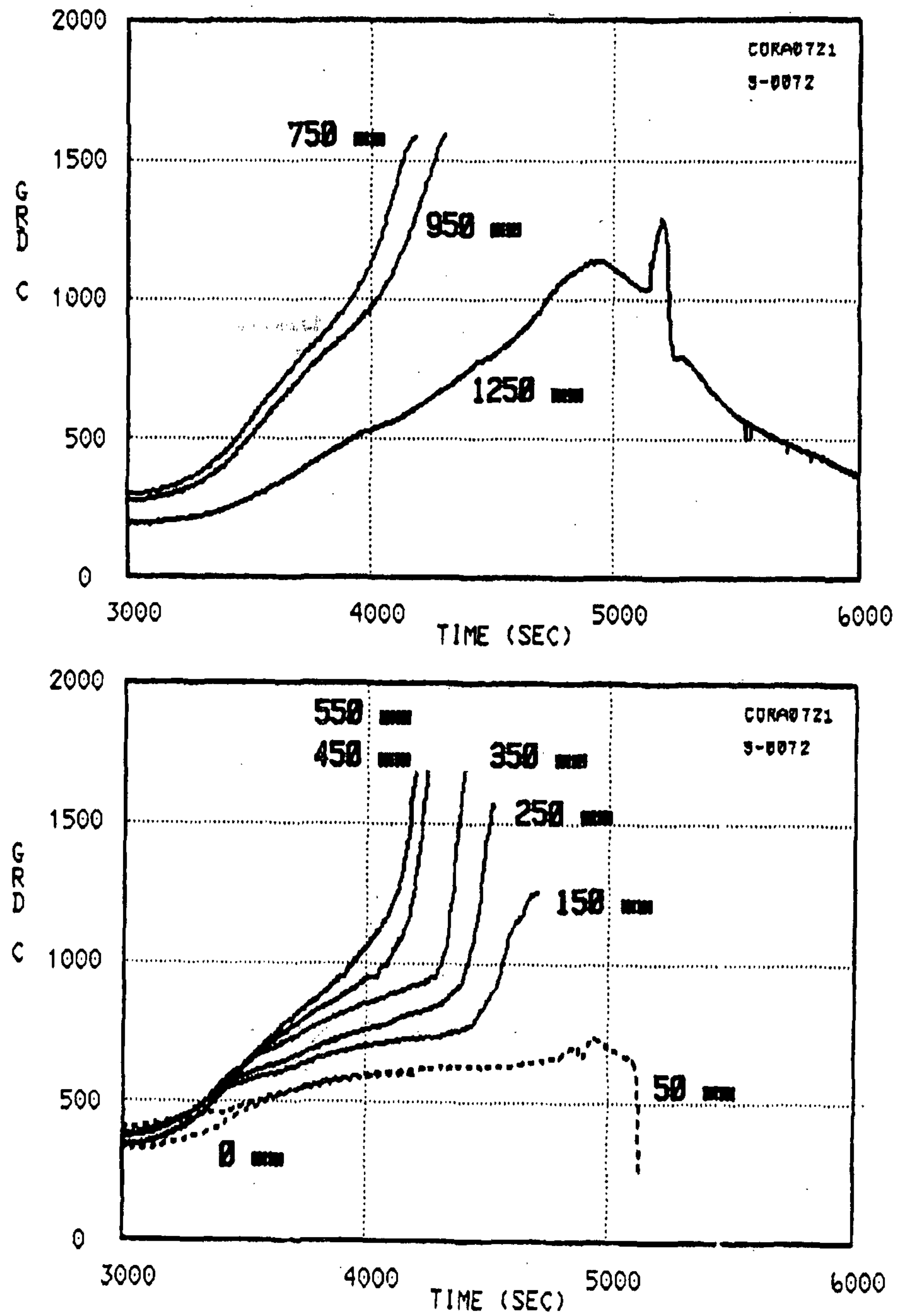


Fig.16: Temperatures of unheated rods during CORA-12

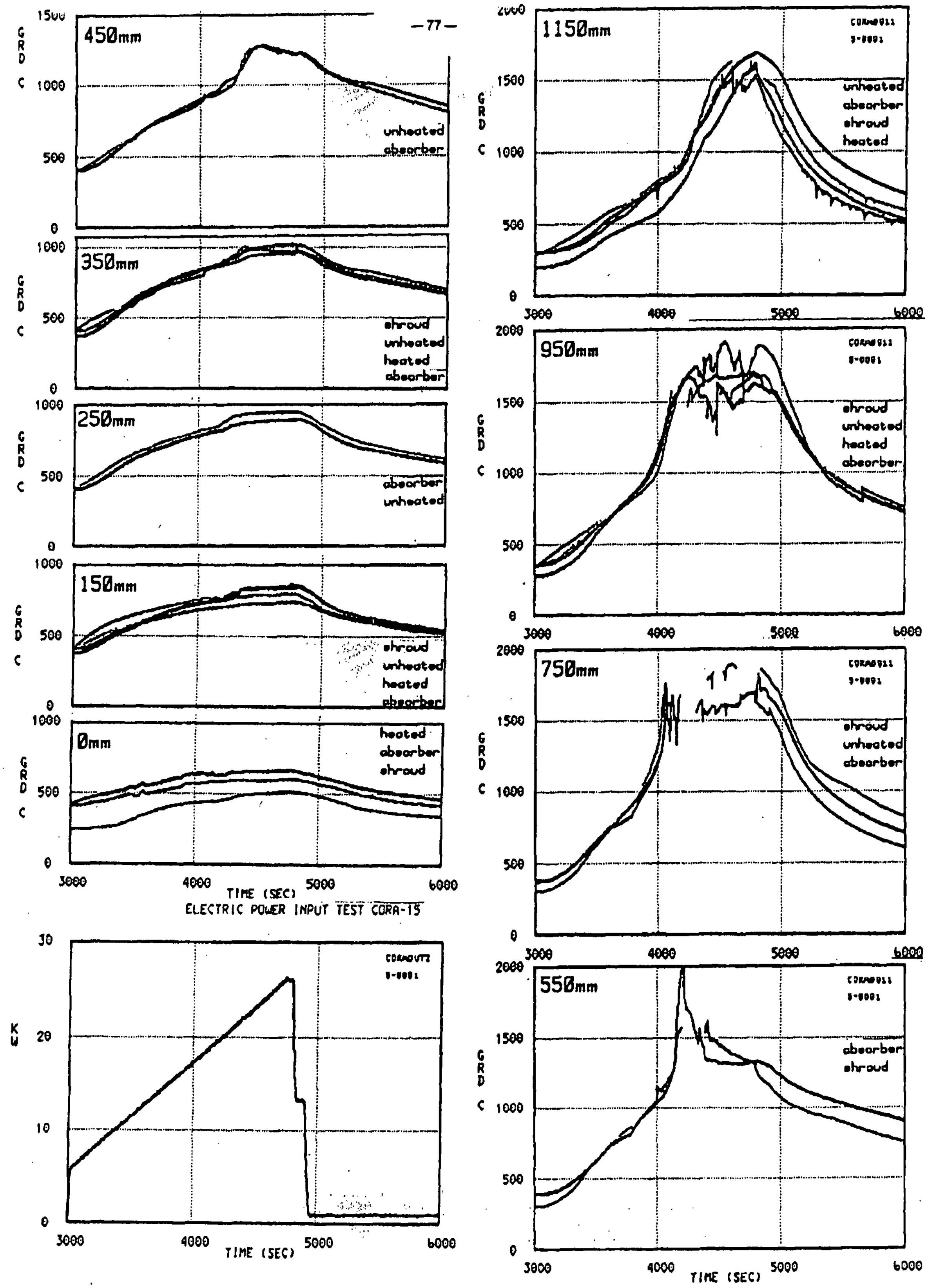
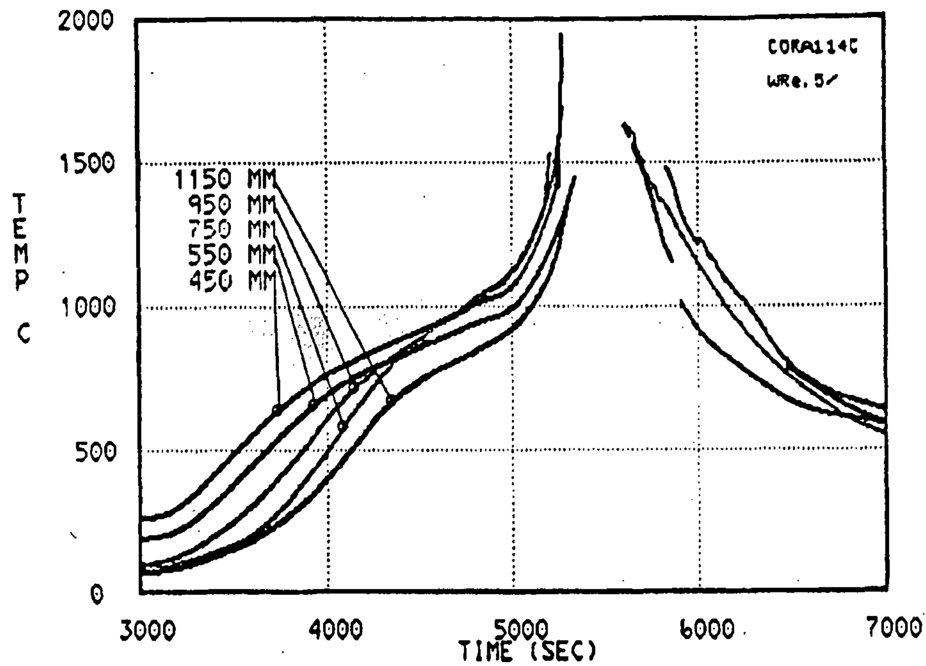


Fig.17: Temperatures at different elevations during CORA-15



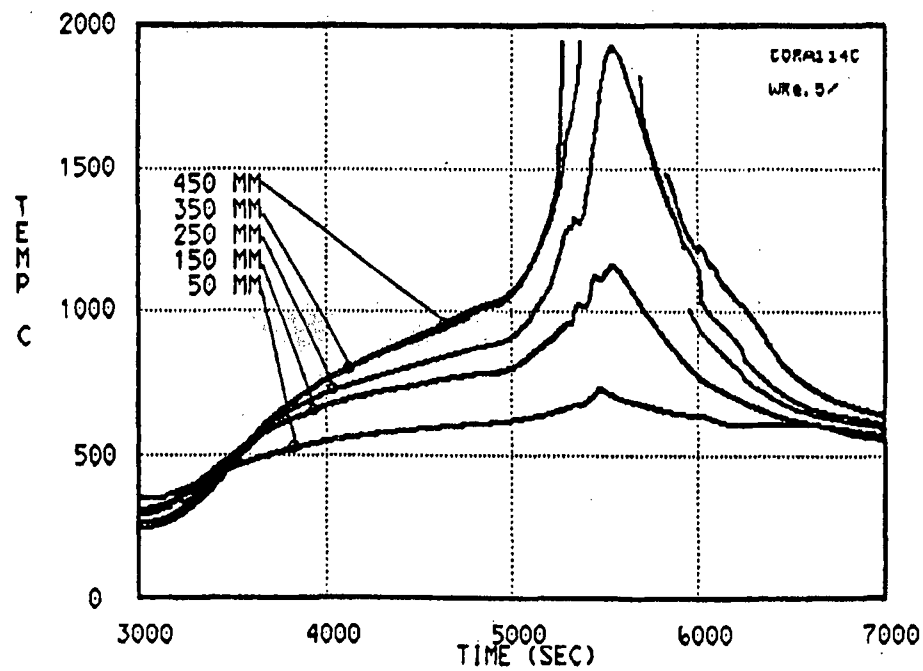
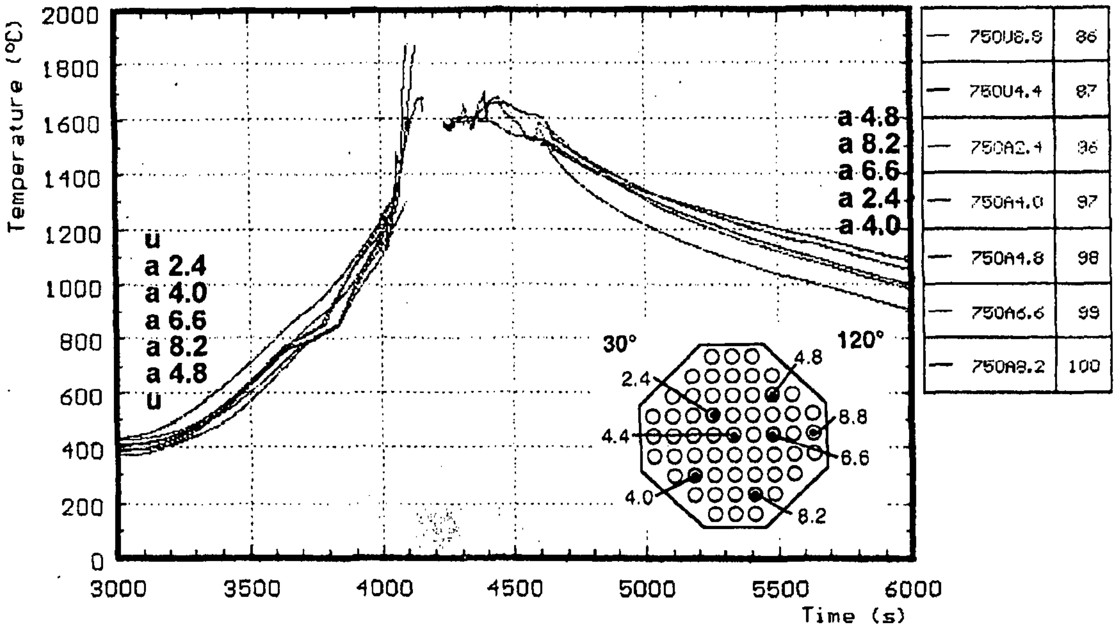
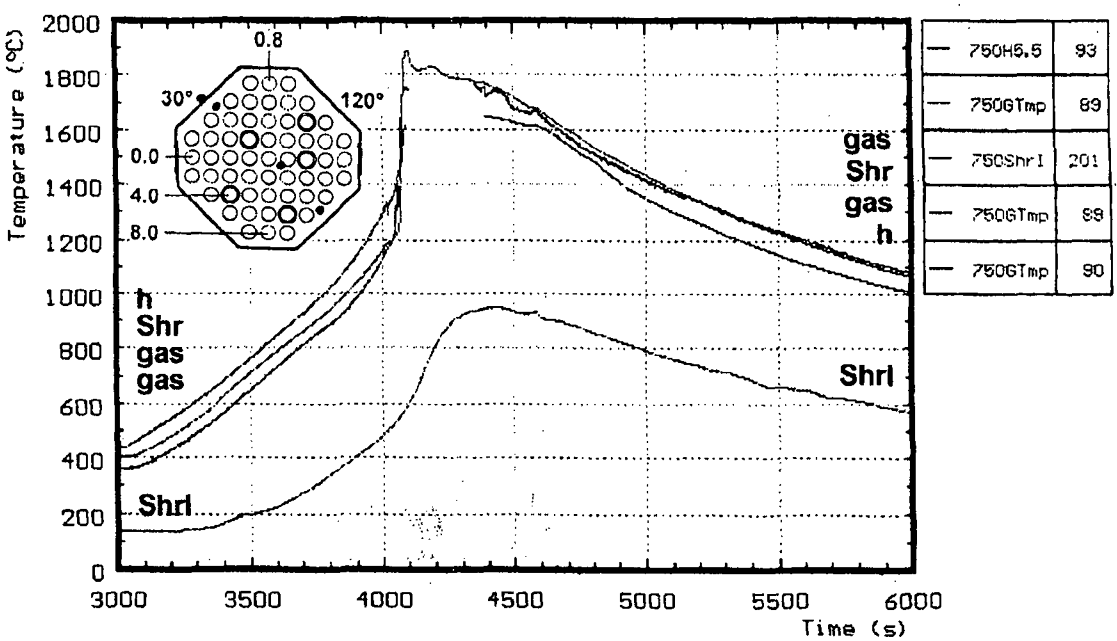


Fig.18: Temperatures of unheated rods during CORA-9





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)

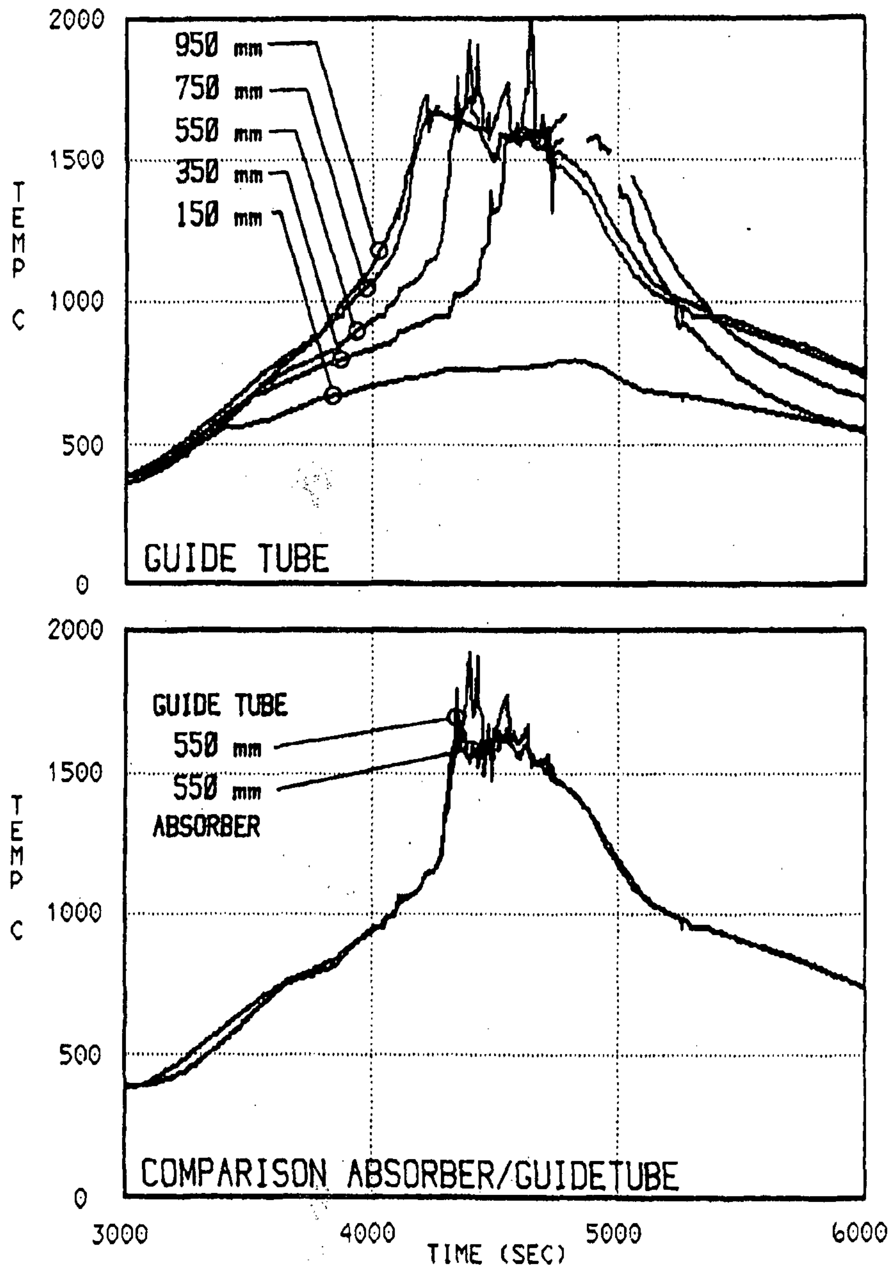


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

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.

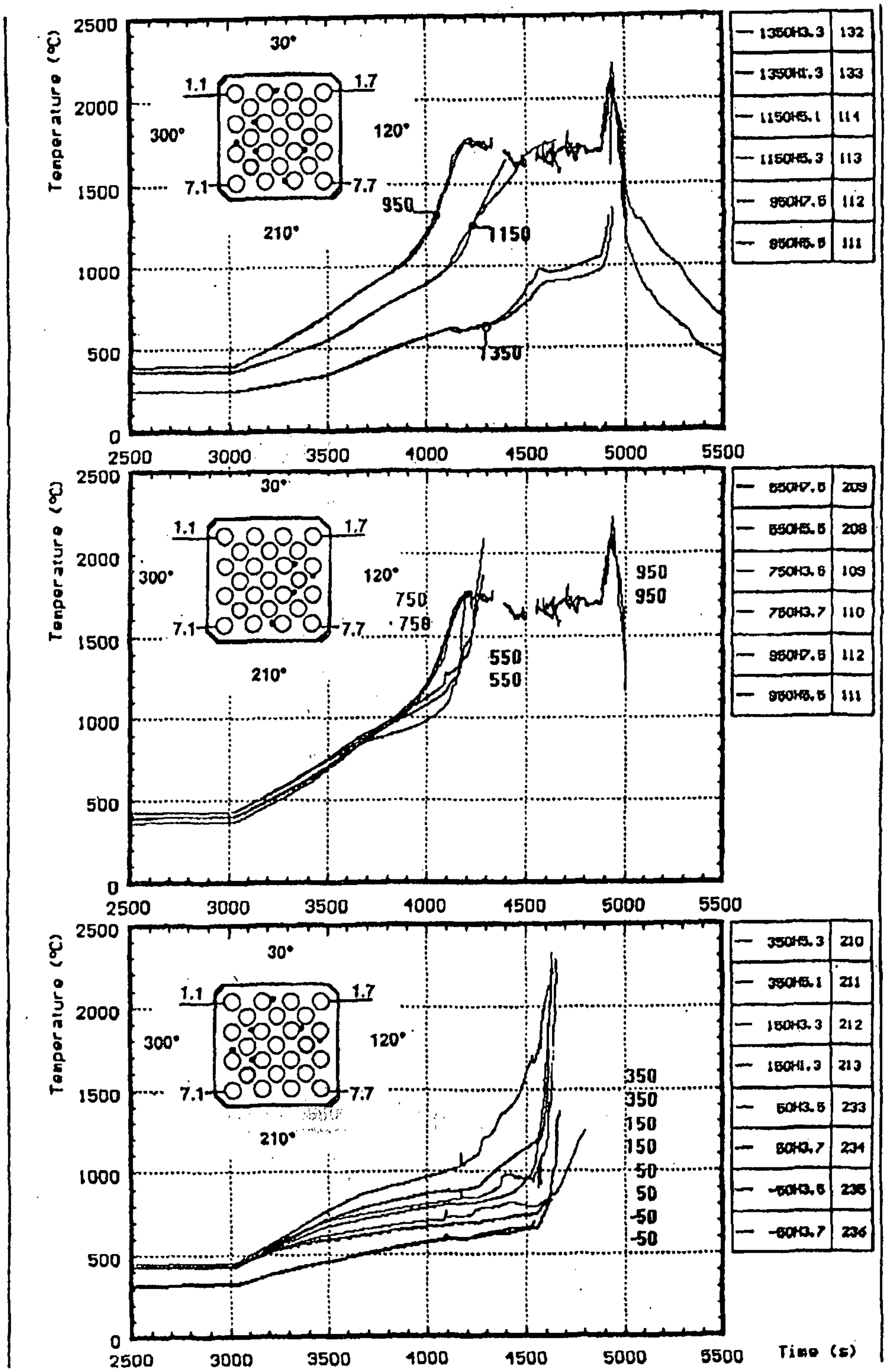


Fig. 37: Temperatures of the heated rods (CORA-13)

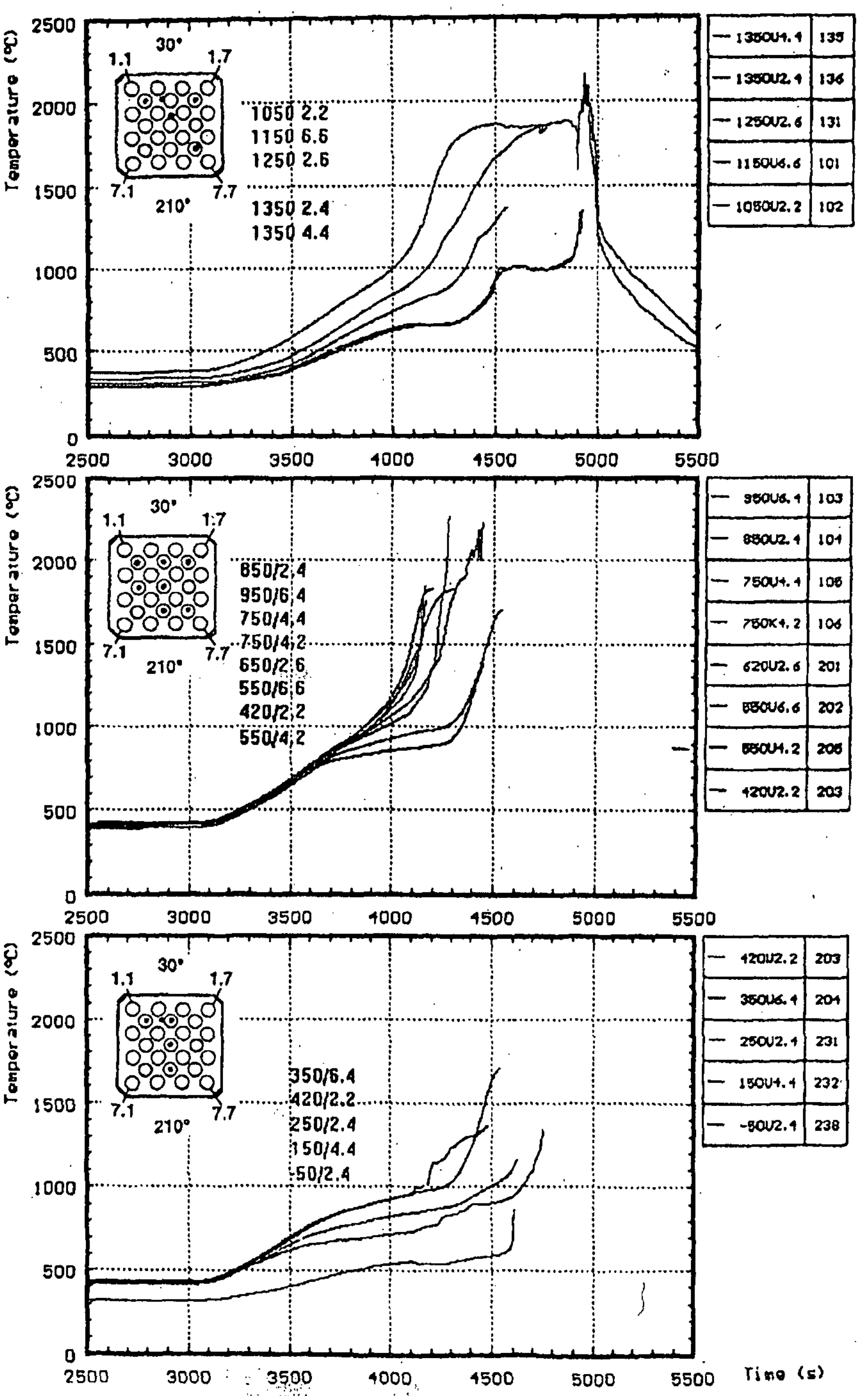


Fig. 39: Temperatures of the unheated rods (CORA-13)

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

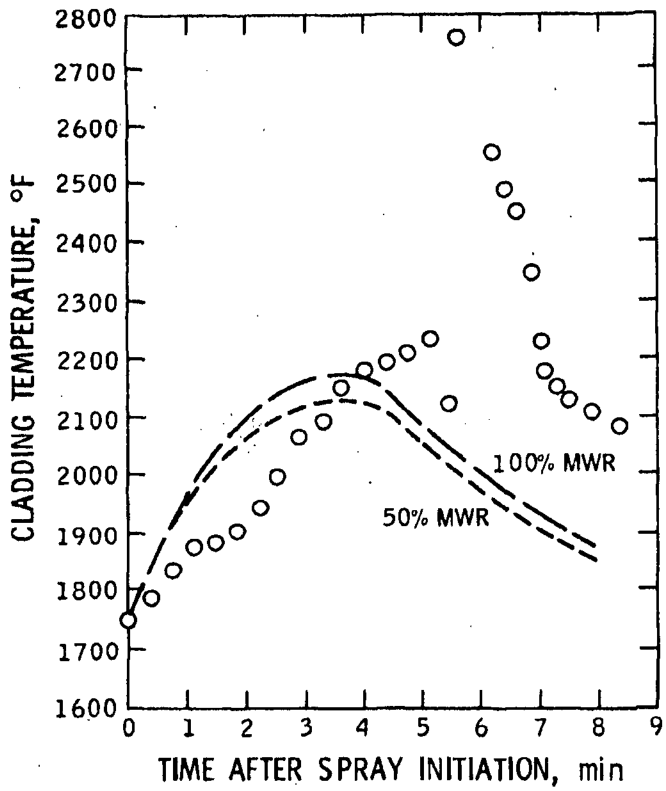
[a] Percentage of total energy, i.e. chemical reaction power and electric power input

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

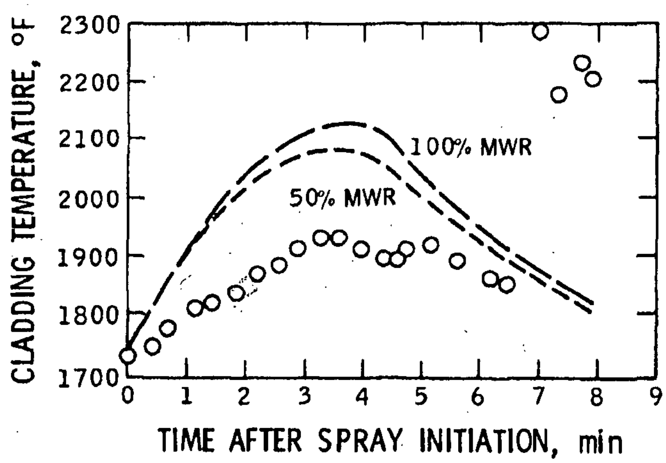
Appendix F Figure A8.9. Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[6] and Figure A8.10. Analysis of Zr2K Thermal Response[7]

[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

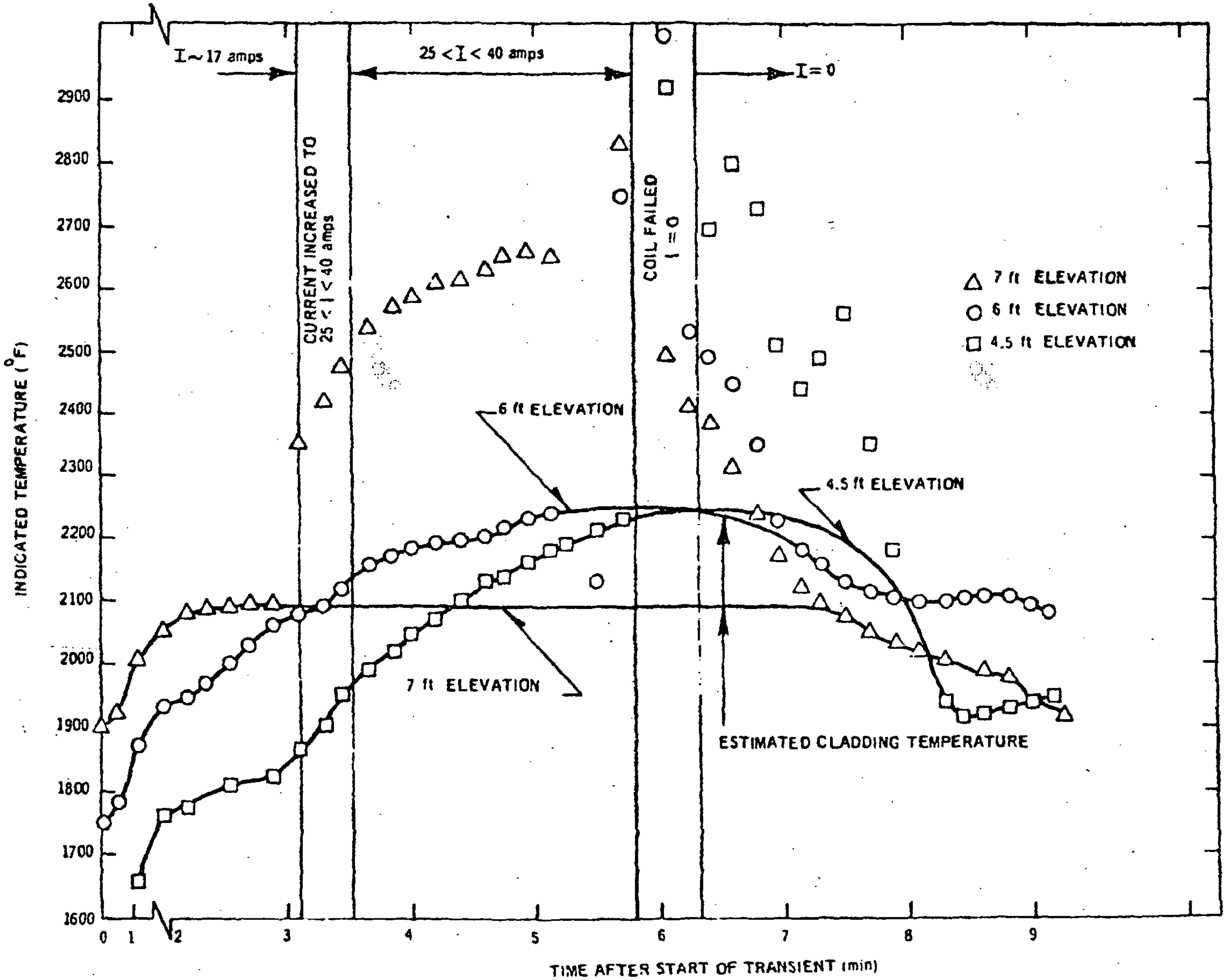


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

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+I1CqfsalPLJmNC5TH4=;
b=unjpCpdm4P6tdZtLEJBlXfCzuVELkoALZOv+4wSF2O/VgXPdSJDTwiTCOtArqeWh5L

jWbZMkEY7mt3XtmVLzWZJl1tMC3ZOZg1dQDgyhnnl/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
OLGFf7csb5n52v23Yr8ipseSTldG03aQPFjtURwD+Q8sBpbuZf1XUNjt1ys9M4W1ibpu
Y49lPwlctpGirzeRHBuZhzW1GEkLqYOMeWnSE=
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="000e0cd6051084680504825997 66"
Return-Path: markleyse@gmail.com