PRM-50-93

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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

November 17, 2009

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

Attention: Rulemakings and Adjudications Staff

Subject: Petition for Rulemaking, Submitted Pursuant to 10 C.F.R. § 2.802

Dear Ms. Vietti-Cook:

Enclosed is a petition for rulemaking, dated November 17, 2009, submitted pursuant to 10 C.F.R. § 2.802. Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments.

Petitioner also requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments. 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.[1]

Additionally, Petitioner requests that the NRC make a new regulation stipulating minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").

Petitioner is submitting this petition, because Petitioner is aware that data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. Data from such experiments also indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

Additionally, 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

[1] Best-estimate ECCS evaluation models used in lieu of Appendix K calculations are described in NRC Regulatory Guide 1.157.

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lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

If implemented, the regulations proposed in the enclosed petition for rulemaking would help improve public and plant-worker safety.

Respectfully submitted,

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

November 17, 2009

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

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PETITION FOR RULEMAKING

TABLE OF CONTENTS

Appendix C Table B-1. Group III Test Results (The Four FLECHT Zircaloy Tests)

Appendix D Table 1. Experimental Heat Cladding Temperatures (The 28 Tests from Thermal-Hydraulic Experiment 1)

Appendix E Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region (Graphs of Cladding Temperature Values During the FLHT-1 Test)

Appendix F Figure 3.7. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 and Figure 3.10. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 with Saturation Temperature (Graphs of Cladding Temperature Values During the LOFT LP-FP-2 Experiment)

Appendix G Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF)

Appendix H Figure 2.1. Temperature Regimes for Extensive Liquid Phase Formation and Relocation

Appendix I Memorandum RD-TE-70-616, FLECHT Monthly Report, December 14, 1970

Appendix J Memorandum RD-THD-17, FLECHT Technical Review Meeting Minutes No. 58, December 18, 1970

Appendix K Memorandum PA-TE-70-419, Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874), July 24, 1970

Appendix L Memorandum RD-ED-THE-33, Report of Events Leading to FLECHT 10 x 10 Bundle Test, April 23, 1969

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

Attention: Rulemakings and Adjudications Staff

PETITION FOR RULEMAKING

I. NEEDED REGULATIONS

This petition for rulemaking is submitted pursuant to 10 C.F.R. § 2.802 by Mark Edward Leyse. Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments.[1]

Petitioner also requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models, Sources of Heat during the LOCA, Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[2] 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.[3]

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[1] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

[2] 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.

[3] Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.

Additionally, Petitioner requests that the NRC make a new regulation stipulating minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[4]

II. STATEMENT OF PETITIONER'S INTEREST

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

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

PRM-50-84 was summarized briefly in the American Nuclear Society's *Nuclear News*'s June 2007 issue[5] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[6] by the Union of Concerned Scientists.

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[4] It can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

[5] American Nuclear Society, *Nuclear News*, June 2007, p. 64.

[6] David Lochbaum, Union of Concerned Scientists, "Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84)," July 31, 2007, located at:

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

Petitioner is submitting this petition, dated November 17, 2009, because Petitioner is aware that 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. Data from such experiments also indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

Additionally, it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

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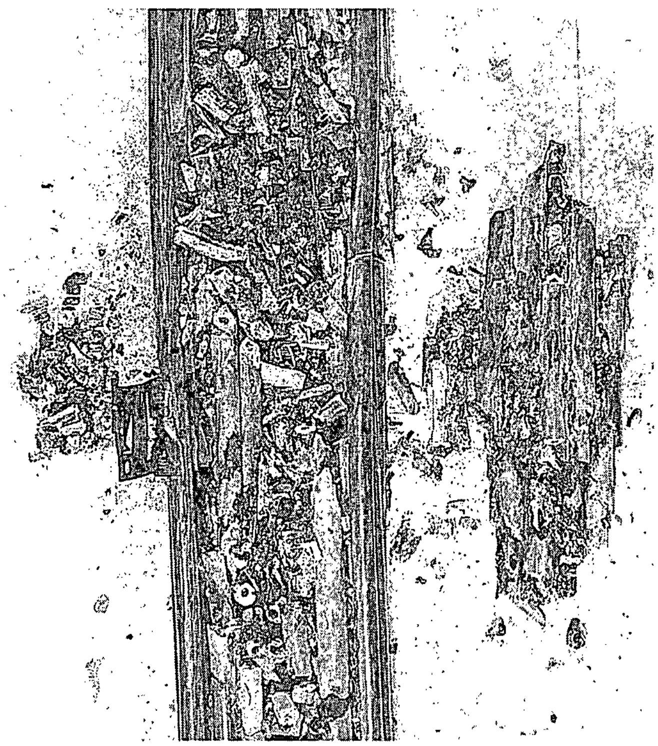
www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072130342, p. 3.

III. BACKGROUND

A. Introduction



A section of the assembly from FLECHT run 9573[7]

In 1973, the Commissioners of the Atomic Energy Commission ("AEC") stated, "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[8] Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.

"PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" ("PWR FLECHT Final Report") states that, "[t]he objective of the PWR FLECHT...test program was to obtain experimental reflooding heat transfer data under simulated loss-of-

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[7] See Appendix A for more photographs of the assembly from FLECHT Run 9573; see also Appendix B for a photograph of the assembly from FLECHT Run 8874.

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

coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[9] An autocatalytic oxidation reaction was not expected to occur in any of the FLECHT tests.[10]

The data reported in "PWR FLECHT Final Report" is important for ECCS evaluation calculations, required for all holders of operating licenses for nuclear power plants. Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models, Post-Blowdown Phenomena, Refill and Reflood Heat Transfer for Pressurized Water Reactors*, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]."

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT Final Report"] when compared to the equivalent stainless steel tests."[11] 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."[12]

1. Why "The Impression Left from Run 9573" Cannot be Separated from Zirconium-Water Reaction Models

In fact, within the first 18.2 seconds of FLECHT run 9573,[13] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[14] *i.e.*,

[9] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 1-1.
[10] "PWR FLECHT Final Report" does not mention that an autocatalytic oxidation reaction occurred during FLECHT run 9573.
[11] 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.
[12] *Id.*, p. 17.
[13] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[14] *Id.*, p. 3-98.

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."[15]

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.[16]

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."[17]

And 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

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

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

[17] 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.

exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure).[18]

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"[19]—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.

2. Petitioner's Argument

In this petition for rulemaking, Petitioner will argue that data from severe fuel damage experiments conducted with Zircaloy fuel assemblies (*e.g.*, the LOFT LP-FP-2 experiment) 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. In such tests Zircaloy cladding incurred autocatalytic (runaway) oxidation at temperatures far below where the Baker-Just and Cathcart-Pawel equations predict autocatalytic oxidation to occur. Petitioner will also argue that data from such experiments indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

Additionally, Petitioner will argue that it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be a variable reflood rate throughout the core; however, at times the reflood

[18] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[19] *Id.*, p. 3-98.

rate could be approximately one inch per second or lower at different locations throughout the core.

Petitioner believes that the "the impression left from run 9573" includes the fact that run 9573 had a low coolant flood rate; it had the lowest flood rate of the four FLECHT Zircaloy tests. It also had the lowest initial cladding temperature, before flood, of the four Zircaloy tests. Therefore, it is highly probable that run 9573 incurred autocatalytic oxidation, because it had a low flood rate.

Unfortunately, contrary to the claims of the NRC,[20] it has not been empirically established that "the impression left from run 9573" has ever been overcome by subsequent experiments with Zircaloy cladding.

B. Reflood Rates

1. The Low Flood Rate of Run 9573

In "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157" ("Technical Safety Analysis of PRM-50-76"), the NRC states:

> At this time [2004] we know that high temperature tests similar to run 9573 would require rod bundle powers well outside the range of operation of any current or proposed PWRs. Also, no realistic transient experiments or analyses have indicated cladding temperatures at the beginning of reflood anywhere near the 1970°F achieved in run 9573. If run 9573 were repeated the results would probably be the same, the high temperatures and high power would quickly catapult the cladding into the severe metal-water reaction regime, destroying the bundle and producing very little useful heat transfer information.[21]

Indeed, it is reasonable to postulate that if run 9573 were repeated that the fuel assembly would once again be destroyed by autocatalytic oxidation; however, this would be as a consequence of the low flood rate of the coolant (1.1 in./sec.) as well as the high initial cladding temperatures and high power of the assembly.. In "Technical Safety Analysis of PRM-50-76," the NRC neglected to mention the fact that run 9573 had a low

[20] NRC, "Denial of Petition for Rulemaking (PRM-50-76)."
[21] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," April 29, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041210109, p. 8.

coolant flood rate. Regarding the significance that coolant flood rates played in the PWR FLECHT test program, "PWR FLECHT Final Report" states, "[i]n general, the effect on heat transfer coefficient[s] of varying system parameters was clearly discernable, *with flooding rate being by far the most influential parameter investigated*" [emphasis added].[22] The NRC's "Compendium of ECCS Research for Realistic LOCA Analysis" reiterates that in the PWR FLECHT test program, flooding rates were the most influential parameter for affecting heat transfer coefficients.[23]

It is significant that run 9573 had a lower initial cladding temperature than, and the same power level as, other Zircaloy tests conducted in the PWR FLECHT test program that did not incur autocatalytic oxidation. It is also significant that run 9573 had the lowest flood rate of the four Zircaloy tests (see Appendix C Table B-1. Group III Test Results). Additionally, it is noteworthy that "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 rate was about one inch per second."[24]

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

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22 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 5-1.

23 NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.4-14.

24 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."

2. The More than 50 Zircaloy Assembly Tests Performed at the NRU Reactor

In "Denial of Petition for Rulemaking (PRM-50-76)" the NRC states:

> 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).[25]

It is interesting that the NRC merely mentions the fact that more than 50 tests were performed in the NRU reactor, as if the fact that the tests were conducted is proof that the impression left from FLECHT run 9573 has been overcome by subsequent experiments with Zircaloy cladding. It is significant that almost all of the thermal-hydraulic and mechanical deformation tests conducted in the NRU reactor had peak cladding temperatures ("PCT") of the fuel assemblies that did not exceed 2000°F—only one test had a PCT that exceeded 2000°F; it was 2040°F. 10 C.F.R. § 50.46(b)(1) stipulates that in the event of a LOCA, the PCT must not exceed 2200°F. So all but one of the NRU reactor tests had PCTs that were more than 200°F below the regulated limit. In other words, the NRU reactor tests did not simulate LOCA conditions that were severe enough to overcome the impression left from run 9573.

The more than 50 NRU reactor thermal-hydraulic and mechanical deformation tests were conducted in a series of experiments: Thermal-Hydraulic Experiment 1 ("TH-1"), Thermal-Hydraulic Experiment 2 ("TH-2"), Thermal-Hydraulic Experiment 3 ("TH-3"), Materials Test 1 ("MT-1"), Materials Test 2 ("MT-2"), Materials Test 3 ("MT-3"), Materials Test 4 ("MT-4"), and Materials Test 6A ("MT-6A"). In "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor" ("LOCA Simulations in the NRU Reactor"), there is an overview of 50 tests

[25] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 18-19.

that were planned at the NRU reactor—45 thermal hydraulic tests and five cladding materials tests.[26]

Discussing the thermal hydraulic tests, "LOCA Simulations in the NRU Reactor" states:

> [O]ne assembly will be used for thermal-hydraulic testing for a maximum of 45 test runs...
>
> All rods will be unpressurized; consequently, no severe cladding deformation will occur. ...
>
> The current design for the thermal-hydraulic tests is based on using one heatup rate to minimize reactor control problems and experimental perturbations. The reflood rate and reflood injection times will be used as the prime independent variables and will in various combinations be used to reverse the temperature transient at the desired peak cladding temperature limit.[27]

"LOCA Simulations in the NRU Reactor" also states that the planned heatup rate for all the tests was 15°F/sec.,[28] that the highest predicted PCTs were 1900°F,[29] for seven of the 45 tests, and that "for safety purposes," the maximum PCTs of the tests would be 1900°F.[30] So it is obvious that the NRU reactor tests were not planned to simulate LOCA conditions severe enough to overcome the impression left from FLECHT run 9573.

One may be sympathetic toward the test planners who "for safety purposes" did not want the maximum PCTs of the tests to exceed 1900°F; however, in reality, at a nuclear power plant, in the event of a LOCA, the PCT would not necessarily be limited to 1900°F. Furthermore, thermal hydraulic tests planned to have PCTs of only 1900°F, would not provide valid data for calculating heat transfer coefficients for cladding temperatures greater than 1900°F. Regarding this point, the NRC states that "[h]eat transfer coefficients are not directly measurable quantities. They must be calculated from

[26] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024.
[27] *Id.*, p. 3-1.
[28] *Id.*, pp. 3-2, 3-3.
[29] *Id.*, p. 3-3.
[30] *Id.*

measured temperatures, known heat sources, and known thermal properties" [emphasis added].[31]

a. Thermal-Hydraulic Experiment 1

In TH-1, a total of 28 tests were conducted. The TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[32] The TH-1 tests had the highest cladding temperatures of the more than 50 thermal-hydraulic and mechanical deformation behavior tests conducted in the NRU reactor—three of the tests had PCTs that exceeded 1900°F[33]—that the NRC claimed were conducted on the scale necessary to overcome the impression left from FLECHT run 9573.

Unfortunately, the TH-1 tests were conducted with parameters that would not be severe enough to overcome the impression left from run 9573. The PCTs reached in the TH-1 tests ranged from 1223°F to 2040°F (see Appendix D Table 1. Experimental Heat Cladding Temperatures). The TH-1 tests had reflood rates ranging from 0.7 in./sec. to 10.5 in./sec. and delay times to initiate reflood ranging from 3 sec. to 66 sec.[34] And the TH-1 tests had PCTs at the start of reflood ranging from 800°F to 1666°F.[35]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[36] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[37] "low enough that superheated steam conditions [would] exist at the loop inlet instrument location. The superheat requirement [was] imposed so that meaningful steam temperatures [could] be measured."[38])

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[31] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," p. 7.
[32] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119.
[33] *Id.*, p. 12.
[34] *Id.*, p. 13.
[35] *Id.*
[36] *Id.*, p. 10.
[37] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 6-5.
[38] *Id.*

It is significant that the three TH-1 tests (no. 126, no. 127, and no. 130) with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 5 seconds or lower and PCTs at the start of reflood that were 998°F or lower. In other words, the TH-1 tests were conducted with parameters that would prevent the fuel assemblies' overall PCTs from rising much above 2000°F. In fact, the highest predicted PCTs for the TH-1 tests were 1900°F (no. 127 and no. 129); test no. 130 apparently did not have a predicted PCT. As discussed above, the test planners—"for safety purposes"—did not want the maximum PCTs of the tests to exceed 1900°F.

It is significant that "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor" states:

> A loss-of-coolant accident (LOCA) in a commercial light water reactor (LWR) consists of four distinct phases: blowdown, heatup, reflood, and quench. Each of these phases has a path-dependent process that is a function of 1) the type of event that initiated the accident and 2) the reactor's operating conditions at the time the LOCA was initiated. *No single set of conditions would exist at the time of a LOCA; rather, a broad range of operating parameters could exist in one of many possible combinations* [emphasis added].[39]

And noteworthy that "Degraded Core Quench: A Status Report" states:

> In general, based on best-estimate or conservative assumptions during design-basis accidents, the boundary and initial conditions for reflooding tests can be established during the design-basis accidents. The variation of some of the main parameters can be summarized as: system pressure 0.1-1.0 MPa, flooding velocities 1.5-30 cm/sec. (including natural reflood velocities), mass fluxes 7-300 $kg/m^2sec.$, heater rod peak power 0.7-3 kW/m.[40]

Indeed, "[n]o single set of conditions would exist at the time of a LOCA; rather, a broad range of operating parameters could exist in one of many possible combinations."[41] For this reason, the TH-1 tests—conducted with strictly controlled parameters that

[39] *Id.*, p. 1-1.

[40] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 10; this paper cites N. Aksan, *et al.*, "OECD/NEA-CSNI Separate Effects Test Matrix for Thermal-Hydraulic Code Valuation," Vols. I and II, OCDE/GD (94) 82, OECD/NEA Publication, September 1994, as the source of this information.

[41] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 1-1.

prevented the fuel assemblies' PCTs from rising much higher than 2000°F—are not realistic tests for simulating a wide variety of possible LOCAs; *e.g.*, LOCAs with long reflood delay times and low reflood rates.

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.

Rather than "overcome the impression left from [FLECHT] run 9573," the TH-1 tests, with high probability, confirm Petitioner's claim that if run 9573 were repeated—with the same or a lower coolant flood rate, yet with lower initial cladding temperatures (that in the event of a LOCA, would occur at the beginning of reflood at current and/or proposed PWRs) and a lower power level (within the operational range of current and/or proposed PWRs)—that the fuel assembly would still incur autocatalytic oxidation.

Indeed, it is likely that such a test would also produce a substantial amount of valuable heat transfer information.

b. Thermal-Hydraulic Experiments 2 and 3

In TH-2, a total of 14 tests were conducted. The TH-2 tests are reported on in "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)" ("Data Report for TH-2").[42]

The TH-2 tests and TH-3 tests were conducted with parameters that were not severe enough to "overcome the impression left from [FLECHT] run 9573." "Data Report for TH-2" states:

> The primary objective of TH-2 was to develop reliable cladding temperature control of a simulated LOCA. Peak cladding temperatures were to range from 1033 to 1089°K (1400 to 1500°F) for at least 150 s, using variable rate reflood water coolant.[43]

Additionally, the Abstract for "Data Report for TH-2," states:

> A full-length test bundle containing nonpressurized water reactor fuel rods was used to develop reflood control parameters and procedures that [would] produce a reduced heatup rate or a "flat top" transient for extended periods of time. Variable reflood rates were used, and experimentally determined control system logic parameters were developed. Using these concepts, fuel cladding temperatures from 1033 to 1274°K (1400 to 1834°F) were produced for 283 sec.[44]

In TH-3, a total of three tests were conducted. The TH-3 tests are reported on in "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)" ("Data Report for TH-3").[45]

The Abstract for "Data Report for TH-3" states:

> The objective of TH-3 was to further refine the feedback control parameters developed in the TH-2 experiment and to re-establish the

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[42] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)," NUREG/CR-2526, 1982, located in ADAMS Public Legacy, Accession Number: 8212220265.
[43] *Id.*, p. 2.
[44] *Id.*, p. v.
[45] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)," NUREG/CR-2527, 1983, located in ADAMS Public Legacy, Accession Number: 8304120660.

> operability of the loop prior to the subsequent materials deformation and rupture test (MT-3). The TH-3 and MT-3 experiments were planned for the same reactor window and were run within two days of each other. The TH-3 test results insured the success of MT-3 and provided the opportunity to demonstrate the reactor control improvements and to evaluate a new desuperheater concept that would allow the test to run for extended times at high temperatures. The control system improvements and the addition of the new desuperheater resulted in fuel cladding temperatures above 1033°K (1400°F) for 340 s.[46]

It is significant that in the TH-2 tests the highest PCT was 1834°F,[47] 366°F lower than the 10 C.F.R. § 50.46(b)(1) limit; in the TH-3 tests the highest PCT was 1912°F,[48] 288°F lower than the 10 C.F.R. § 50.46(b)(1) limit. Regardless of the achievements of the TH-2 tests and TH-3 tests in developing "reflood control parameters and procedures that [produced] a reduced heatup rate or a 'flat top' transient for extended periods of time,"[49] it is obvious that they did not simulate LOCA conditions that were severe enough to overcome the impression left from FLECHT run 9573.

c. Materials Tests 1, 2, 3, 4, and 6A

Discussing plans for the first four materials tests, "LOCA Simulations in the NRU Reactor" states:

> The [four] fuel cladding performance tests will be considered for selected conditions based on the results obtained during the thermal-hydraulic tests. The objective of the tests will be to use a constant heatup rate and vary the reflood rate and reflood delay time to obtain peak cladding temperatures between 1033°K (1400°F) and 1255°K (1800°F).[50]

Clearly, the first four NRU reactor materials tests were not planned to simulate LOCA conditions severe enough to overcome the impression left from FLECHT run 9573.

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[46] *Id.*, p. v.

[47] C. L. Mohr, *et al.*, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)," pp. v, 17.

[48] C. L. Mohr, *et al.*, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)," p. 14.

[49] C. L. Mohr, *et al.*, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)," p. v.

[50] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 3-1. A fifth materials test (MT-5) was proposed to the NRC but never approved.

d. Materials Test 1

Discussing the MT-1 test, "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A" ("Materials Test MT-6A") states:

> The first materials experiment (MT-1), i.e., the test on the expansion of Zircaloy fuel cladding...was performed in April 1981, using a cruciform of 11 rods pressurized to 3.21 MPa (465 psia) and [one] water tube surrounded by 20 guard rods[51] sealed at atmospheric pressure. The objective of this test was to assess the rate at which the expanded cladding can be cooled, based on evaluations of the rates of heatup and quenching and the measurements of post-test cladding strain. The delay time and the rate of reflood were selected to duplicate one of the experiments at high temperatures, specifically TH-1.10, in which the fuel cladding reached a peak temperature of 1145°K (1600°F). These conditions were achieved: [six] of the 11 rods ruptured and all 11 pressurized test rods expanded significantly. The average peak rupture strain was 43%; the average time to rupture was 43 sec.; and the average temperature at rupture was 1145°K (1600°F).[52]

So the MT-1 test PCT was approximately 600°F lower than the 10 C.F.R. § 50.46(b)(1) limit.

e. Materials Test 2

Discussing the MT-2 test, "Materials Test MT-6A" states:

> In the second materials experiment (MT-2), ...performed in July 1981, the MT-1 guard rods and shroud assembly were reconstituted underwater and reused with a new cruciform test bundle. One of the objectives of the test was to perform a low-temperature, 1090°K (1500°F), test using variable rates of reflooding. The 12 test rods were pressurized to 3.21 MPa (465 psia). A malfunction of the reflood system, however, resulted in higher temperatures than desired and [eight] of the 11 rods ruptured. The average peak rupture strain was 43%, the average time to rupture was 65 sec., and the average temperature at rupture was 1160°K (1625°F).[53]

So the MT-2 test PCT was more than 500°F lower than the 10 C.F.R. § 50.46(b)(1) limit.

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[51] The guard rods are unpressurized fuel rods that surround the periphery (guard) of the test fuel rods to minimize radial heat loss from the test fuel rods.

[52] C. L. Wilson, G. M. Hesson, J. P. Pilger, L. L. King, F. E. Panisko, Pacific Northwest Laboratory, "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A," 1993, p. ix.

[53] *Id.*

f. Materials Test 3

Discussing the MT-3 test, "Materials Test MT-6A" states:

> The primary objective of the third materials experiment (MT-3)...was to determine the expansion and restrictions on the flow channel for a flat-top temperature transient using pressurized fuel rods. Peak temperatures of the cladding were maintained above 1035°K (1400°F) for 180 sec. The MT-3 experiment repeated the test conditions demonstrated by the TH-3.03 test using a completely new test train with 12 fuel rods pressurized to 3.9 MPa (565 psia) and 20 guard rods. All 12 test rods ruptured during the active two-phase cooling regime. The average peak rupture strain was 46%, the average time of rupture was 133 sec., and the average temperature at rupture was 1070°K (1460°F). The MT-3 experiment had a lower average temperature at rupture and a longer time until rupture than any of the other materials experiments because of the significant amount of reflood water that was introduced early in the transient (the delay time for reflooding was 7 sec.). The active strain region was spread over ~2-m (80-in.) length, and no loss of cooling because of coplanar blockage or liftoff[54] was observed.[55]

So the MT-3 test PCT was more than 700°F lower than the 10 C.F.R. § 50.46(b)(1) limit.

g. Materials Test 4

Discussing the MT-4 test, "Materials Test MT-6A" states:

> The fourth materials experiment (MT-4)...was conducted in May 1982. Its primary objective was to evaluate the expansion and rupture of cladding during heatup in the temperature range from 1035 to 1200°K (1400 to 1700°F). The 12 test rods in the 32-rod bundle were initially pressurized to 4.62 MPa (670 psia) at 295°K (70°F) to assure rupture in the correct temperature range. The MT-4 experiment was most similar to the MT-2 experiment; three differences existed: 1) MT-4 rods were pressurized to 4.62 MPa (670 psia), whereas MT-2 rods were pressurized to 3.21 MPa (465 psia); 2) After the temperature turnaround following the heatup transient, the peak temperatures of the cladding were stabilized to measure the characteristics of the heat transfer of the expanded and ruptured fuel rods, whereas during MT-2 the peak temperatures of the cladding were not stabilized; and 3) self-powered neutron detectors (SPNDs) mounted on the shroud were moved to grid elevations to

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[54] Liftoff is a thermal decoupling of the cladding from the fuel that results in cooling of the cladding during deformation.

[55] C. L. Wilson, *et al.*, "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A," pp. ix-x.

> minimize distortion of axial fission power, whereas MT-2 had the SPNDs mounted away from the Inconel grids. During the test all 12 test rods ruptured with an average peak rod strain of 72.1%. The active strain region was spread over 0.189 m (7.42 in.), the average time of rupture was 55 sec.; and the average temperature at rupture was 1094°K (1511°F).
>
> The MT-4 experiment used a new cruciform bundle of 12 pressurized test fuel rods and the guard fuel rods and shroud previously used in MT-3. Test operations most closely followed the operating conditions of the TH-1.16, during which cooling by reflooding was used to terminate the transient temperature of the heatup at ~1200°K (1700°F). Stabilized operations at the post-transient stage closely followed the operating conditions used in the MT-3 experiment.[56]

So the MT-4 test PCT was approximately 500°F lower than the 10 C.F.R. § 50.46(b)(1) limit.

h. Materials Test 6A

The MT-6A test is discussed in "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A" ("Materials Test MT-6A"). After a reflood delay of approximately 70 seconds—controlled by the data acquisition and control system ("DACS")—the MT-6A test had varying reflood rates: 8 in./sec. for 3 sec., 7 in./sec. for 3 sec., and 2 in./sec. for approximately 170 sec.[57] "Materials Test MT-6A" states that after the reflood rate was held at 2 in./sec. for 3 sec. "the DACS was supposed to take over reflood control to maintain fuel temperatures [that were] approximately constant. An anomaly in the reflood control prevented the DACS from taking control once the reflood rate reached [2 in./sec]. The continued reflood at this rate caused the fuel to cool and quench, ending the test."[58]

In the MT-6A test, the PCT was approximately 1750°F,[59] or 450°F lower than the 10 C.F.R. § 50.46(b)(1) limit. It is obvious that the MT-6A test, and the other NRU reactor materials tests, did not simulate LOCA conditions that were severe enough to overcome the impression left from FLECHT run 9573.

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[56] *Id.*, p. x.
[57] *Id.*, pp. 6, 21.
[58] *Id.*, p. 6.
[59] *Id.*, pp. B.10, B.11.

3. Conclusion of the Reflood Rates Section

It has been demonstrated in the Reflood Rates Section that the Zircaloy cladding tests, performed in the early 1980s, at the NRU reactor—"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"[60]—did not simulate LOCA conditions severe enough to "overcome the impression left from [FLECHT] run 9573."[61]

Furthermore, it can be extrapolated from data from the NRU thermal-hydraulic and mechanical deformation tests that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be a variable reflood rate throughout the core; however, at times the reflood rate could be approximately one inch per second or lower at different locations throughout the core.

It is noteworthy that 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)."[62]

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.

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[60] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.

[61] 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.

[62] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.

It will be demonstrated in the following Metal-Water Reaction Rate Section that, in the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[63] and 2240°F,[64] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[65]

Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[66] the melting point of Zircaloy is approximately 3308°F.[67]

C. The Metal-Water Reaction Rate

10 C.F.R. § 50.46(b)(1) stipulates that in the event of a LOCA, the peak cladding temperature ("PCT") must not exceed 2200°F. Discussing the 2200°F PCT limit and autocatalytic (runaway) zircaloy oxidation, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

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

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63 J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML062840091, pp. 30, 33.

64 R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," NUREG/CP-0114, Vol. 2, 1990, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042250131, p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

65 *Id.*

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

67 NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

> increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.[68]

And in the following paragraph, "Compendium of ECCS Research for Realistic LOCA Analysis" describes a method for assessing the conservatism of the 2200°F PCT limit:

> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. This type of comparison implicitly includes...complex heat transfer mechanisms...and the effects of fuel rod ballooning and rupture on coolability... Analysis of experiments performed in the Power Burst Facility, in the Annular Core Research Reactor, and in the NEILS-CORA (facilities in West Germany) program have shown that temperatures above 2200°F are required before the zircaloy-steam reaction becomes sufficiently rapid to produce an autocatalytic temperature excursion. Another group of relevant experimental data were produced from the MT-6B and FLHT-LOCA and Coolant Boilaway and Damage Progression tests conducted in the NRU Reactor in Canada. ...even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists.[69]

It is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" states that assessing the conservatism of the 2200°F PCT limit, as a boundary that would prevent autocatalytic oxidation from occurring, can be accomplished by analyzing data from multi-rod severe accident tests, because such data indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative.

There is also experimental data from multi-rod severe accident tests that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. For example, the paper, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," states:

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

[68] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[69] *Id.*

A maximum heating rate of 15°K/sec. indicates that an autocatalytic oxidation reaction commenced. "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[70] So at the point when peak cladding temperatures increased at a rate of greater than 10°K/sec. during the CORA experiments, autocatalytic oxidation reactions commenced—at cladding temperatures between 2012°F and 2192°F.

It is noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis," published in 1988, does not mention that autocatalytic oxidation occurred during the LOFT LP-FP-2 experiment, conducted in 1985, at cladding temperatures greater than either 1400°K (2060°F)[71] or 1500°K (2240°F).[72]

1. The Cladding Temperatures at which Autocatalytic Oxidation Occurred during Severe Fuel Damage Experiments

In this section, Petitioner will analyze papers that report on the results of multi-rod severe fuel damage experiments, conducted in the aftermath of the Three Mile Island Unit 2 ("TMI-2") accident. Petitioner will demonstrate that data from such experiments indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the cladding temperatures at which an autocatalytic oxidation reaction would occur, in the event of a LOCA.

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

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

72 R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

Discussing the Baker-Just and Cathcart-Pawel equations in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K," the NRC states:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F... Thus, with regard to runaway temperature escalation, the peak cladding temperature could be raised to 2300°F without affecting this sensitivity and without reducing the margin that the Commission would have perceived in 1973.
>
> To explore this sensitivity further, we performed more than 50 LOCA calculations with RELAP5/Mod3. In about half of the cases, the Baker-Just equation was used for the metal-water heat generation rate, and in the other half, the Cathcart-Pawel equation was used. Reactor power just prior to the LOCA was varied parametrically to simulate incremental variations in decay heat. The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F. Each series of calculations done with the two metal-water models always showed peak cladding temperatures without runaway to be at least 100°F higher with Cathcart-Pawel, which is consistent with the temperature difference in the rate equations. Thus in these calculations, the margin between 2300°F and the calculational instability using Cathcart-Pawel was always equal to or greater than the margin between 2200°F and the calculational instability using Baker-Just.[73]

It is significant that the Baker-Just equation calculated autocatalytic (runaway) oxidation to occur when cladding temperatures increased above approximately 2600°F

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

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

a. The Power Burst Facility Severe Fuel Damage 1-1, 1-3, and 1-4 Tests

The Power Burst Facility ("PBF") Severe Fuel Damage ("SFD") 1-1, 1-3, and 1-4 tests each used a PWR 17 by 17 assembly comprised of 32 fuel rods that were 0.9 meter in length.[74] Or according to a different account, the PBF SFD 1-1 test had 32 fuel rods that were 0.9 meter in length and the PBF SFD 1-3 and 1-4 tests had 28 fuel rods that were 1.0 meter in length.[75]

"Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code" states that "[t]he...SFD 1-1, 1-3, and 1-4 [tests] were conducted in a thermal-hydraulic condition similar to that expected to have occurred at TMI-2, which is characterized by slow heating up to 1600°K and rapid heating rate above 1600°K, driven by zirconium-water reaction."[76]

The same paper also states that "[i]n the [SFD 1-1] test, the rapid temperature rise in the bundle began near the center at the 0.5 to 0.7 [meter] elevation, and then spread radially outward and axially downward in a manner similar to a flame front

[74] Ken Muramatsu, Fumiya Tanabe, Tohru Suda, "Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code," Journal of Nuclear Science and Technology, 23[11], November 1986, p. 959.

[75] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC, "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 3.

[76] Ken Muramatsu, Fumiya Tanabe, Tohru Suda, "Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code," p. 959; this paper cites P. E. MacDonald, *et al.*, Proceedings from the 5th International Meeting on Thermal Reactor Safety, Karlsruhe, 1984, p. 876, as the source of this information.

propagation."[77] Additionally, three graphs of the cladding-temperature values (at the 35 cm, 50 cm, and 70 cm elevations) during the SFD 1-1 test indicate that that test's autocatalytic oxidation reaction began when cladding temperatures were approximately 1600°K.[78]

Offering a different account of the heatup rates during the PBF-SFD tests, "Review of Experimental Results on LWR Core Melt Progression" states that "[h]eatup rates in the moderately high-pressure PBF-SFD tests began in the neighborhood of 0.1 to 0.5°K /sec., but increased to about 1 to 2°K /sec. above 1300°K and >10°K /sec. above 1700°K."[79]

In the SFD 1-1, 1-3, and 1-4 tests, it is significant that rapid temperature excursions occurred at either approximately 1600°K (2420°F) or approximately 1700°K (2600°F)—as a result of the exothermic Zircaloy-water reaction—because the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[80]

b. Materials Test 6B: The NRU Reactor Transition Test

Discussing materials test 6B ("MT-6B") "Full-Length High-Temperature Severe Fuel Damage Test 1" states:

> In 1984, a proof of princip[le] test [(or transition test)] (MT-6B) was performed to determine whether a test on a full-length fuel bundle could be safely performed to demonstrate the kind and extent of the damage that would result to fuel rods from a boilaway of reactor coolant. Emphasized were the severe damage conditions that would result in the core. In this

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[77] *Id.*, p. 960; this paper cites Proceedings from the 5th International Meeting on Thermal Reactor Safety and P. E. MacDonald, *et al.*, American Nuclear Society Transcript, 46, 478, 1984, as the source of this information.
[78] *Id.*, pp. 962-963.
[79] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC, "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites D. J. Osetek, "Results of the Four PBF Severe Fuel Damage Tests," NRC, "Proceedings of the Fifteenth Water Reactor Safety Information Meeting," NUREG/CP-0090, 1987, as the source of this information.
[80] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

> proof of princip[le] test, the LB-LOCA test geometry was used. Demonstrated during the test was that adequate thermal insulation can protect the reactor under severe conditions and that it is possible to control a boilaway transient; the conclusion was that it would be safe to conduct in-reactor tests that cause severe damage to reactor fuel rods from a loss of coolant.[81]

During the MT-6B test the PCT was either 2060°F (1400°K),[82] 2200°F (1477°K),[83] or 2336°F (1553°K)[84]: three different publications report these inconsistent PCT values. 276°F (153°K) is a substantial temperature difference. One of the goals of the MT-6B test was to achieve a PCT of 1600°K (2420°F).

"Compendium of ECCS Research for Realistic LOCA Analysis" states that "[t]he MT-6B test...showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant."[85] However, because other reports state that the MT-6B test had a PCT of 1400°K (2060°F) and 1280°C (2336°F) (1553°K), the MT-6B test may have actually demonstrated that the Zircaloy oxidation rate was easily controllable by adding more coolant at cladding temperatures of either 2060°F (1400°K) or 1280°C (2336°F) (1553°K).

c. NRU Reactor Full-Length High-Temperature 1 Test

The first full-length high-temperature severe fuel damage ("FLHT-1") test was conducted at the National Research Universal ("NRU") reactor at Chalk River, Ontario, Canada, by Pacific Northwest Laboratory ("PNL"), "to evaluate degraded core behavior and the progression of light water reactor ("LWR") fuel damage resulting from [a] loss-of-coolant accident."[86] The FLHT-1 test was part of the PNL Coolant Boilaway and Damage Progression program. The FLHT-1 test used an assembly comprised of 12 fuel

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[81] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, p. x.
[82] *Id.*, p. viii.
[83] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[84] G. M. Hesson, *et al.,* Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," 1993, p. 2.
[85] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[86] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. v.

rods that were 3.7-meters in length.[87] During the test the nominal fuel rod linear power was 0.524 kW/m (0.160 kW/ft.) and the nominal bundle power was 23 kW (22 Btu/sec.).[88]

The FLHT-1 test is reported on in "Full-Length High-Temperature Severe Fuel Damage Test 1" ("FLHT-1 Test Report"). The Summary of "FLHT-1 Test Report" states:

> This report presents a summary of the FLHT-1 test operations. The test was performed on March 2, 1985. In the report, the actual test operations and data are compared to the planned operations and predicted test behavior. ... The test plan called for a gradual temperature increase to approximately 2150°K (3400°F). However, during the test, the fuel cladding began to rapidly oxidize, causing local bundle temperatures to rapidly increase from about 1700°K (2600°F) to 2275°K (3635°F), at which time the test was terminated. Much of the Zircaloy cladding in the central region (axially) of the 3.7-m-long (12-ft) fuel bundle was heavily oxidized, and some Zircaloy cladding melted.[89]

"FLHT-1 Test Report" states that at approximately 1700°K (2600°F) the Zircaloy cladding in the FLHT-1 test began to rapidly oxidize, causing a rapid local bundle temperature excursion; however, it is far more likely that the Zircaloy cladding actually began to rapidly oxidize at a temperature of approximately 1520°K (~2275°F) or lower. "FLHT-1 Test Report" has inconsistent statements regarding the time that the Zircaloy cladding temperature excursion began—the autocatalytic (runaway) oxidation reaction.

"FLHT-1 Test Report" states that "[t]he reactor power was decreased at approximately 17:11:07, 85 seconds after the start of the [cladding temperature] excursion;"[90] *i.e.*, the cladding temperature excursion began at 17:09:42. However, "FLHT-1 Test Report" also states that the cladding temperature excursion began 18 seconds latter at 17:10:00—when the cladding temperature was 1700°K.[91] The difference of 18 seconds is highly significant, because it means that the cladding temperatures were much lower than 1700°K when the temperature excursion actually began.

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[87] *Id.*, p. 3.1.
[88] *Id.*, pp. 4.1-4.2.
[89] *Id.*, p. v.
[90] *Id.*, p. 4.6.
[91] *Id.*, p. 4.11

"Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that during the FLHT-1, -2, -4, and -5 tests that "[t]he heatup phase of the tests culminated near 1700°K in a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signaling the onset of an autocatalytic oxidation reaction."[92] So if peak cladding temperatures increased at a rate of greater than 10°K/sec. during the FLHT-1 test, it is highly probable that 18 seconds before 17:10:00—when the peak cladding temperature was 1700°K (2600°F)—the peak cladding temperature was approximately 1520°K (~2275°F) or lower.

This is reasonable to postulate; after all, another severe fuel damage experiment—LOFT LP-FP-2—demonstrated "that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400°K (2060°F)."[93] According to a different account, in the LOFT LP-FP-2 experiment, the onset of rapid oxidation occurred at approximately 1500°K (2240°F).[94] Additionally, "Degraded Core Quench: Summary of Progress 1996-1999," states that autocatalytic (runaway) oxidation of Zircaloy cladding by steam occurs at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or higher.[95]

Furthermore, although the graphs of "Typical Cladding Temperature Behavior"[96] and "Pseudo Sensor Readings for Fuel Peak Temperature Region"[97, 98] are not large enough to clearly delineate what the temperature values were at given times during the FLHT-1 test, the graphs' cladding-temperature values are consistent with the postulation that the temperature excursion began at a temperature far lower than 1700°K, at a

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[92] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[93] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 33.
[94] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.
[95] 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.
[96] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.7.
[97] *Id.*, p. 5.3.
[98] Pseudo sensor readings are the averages of the readings of two or more thermocouples.

temperature closer to 1520°K (see Appendix E Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region). The slopes of the lines of the cladding-temperature value plots in the graphs become nearly vertical, when the cladding-temperature values reach approximately 1520°K, indicating the onset of the temperature excursion, at a rate of 10°K/sec. or greater.

Additionally, the description of the procedure of the FLHT-1 test in "FLHT-1 Test Report," also indicates that the temperature excursion began at a temperature of approximately 1520°K (~2275°F) or lower. "FLHT-1 Test Report" states:

> Typical cladding temperature behavior at one position in the assembly during the test is shown in Figure 4.1. At about 60 to 70 min. along the abscissa, a temperature increase [commenced] when the [bundle coolant] flow rate was about 9 kg/hr. (20 lb/hr.). The [cladding] temperature increased until about 95 min. and [reached] 1450°K (2150°F), at which time the bundle coolant [flow] rate was increased to 18 kg/hr. (40 lb/hr.) to stabilize the temperature. However, the [cladding] temperature rapidly dropped to about 1060°K (1450°F). The bundle coolant flow rate was then decreased through a series of steps to a minimum of 9 kg/hr. (20 lb/hr.). This action stopped the temperature decrease and started another temperature rise. When the temperature reached about 1475°K (2200°F), the bundle coolant flow [rate] was again increased to stop the temperature ramp. This led to a stabilized condition. The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly until the test director requested that the reactor power be reduced to zero power.[99]

First, it is obvious from the above description and from Figures 4.1 and 5.4 that when cladding temperatures reached approximately 1475°K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was not achieved. Cladding temperatures continued to rise. This is clearly stated: "The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly…"[100]

Second, it is obvious that the rapid metal-water reaction began at cladding temperatures far lower than 1700°K (2600°F). It makes no sense that the autocatalytic oxidation reaction would have begun at 1700°K (2600°F). How can it be explained that

[99] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.6.
[100] *Id.*

after the coolant flow rate was increased—when cladding temperatures reached approximately 1475°K (2200°F)—that the cladding temperatures were able to increase by 225°K (400°F)? Why would the test conductors have not been able to terminate the cladding-temperature rise, as they did earlier in the test when cladding temperatures reached 1450°K (2150°F)? And how can it be explained that the test conductors did not have enough time to increase the coolant flow rate back up to 18 kg/hr. (40 lb/hr.), as they did when cladding temperatures reached 1450°K (2150°F), earlier in the test?

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

In "Compendium of ECCS Research for Realistic LOCA Analysis," discussing an earlier NRU reactor test, the NRC states that "[t]he MT-6B test...showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant."[101] Furthermore, the test conductors would have thought "the zircaloy oxidation rate was easily controllable" at cladding temperatures far above 2200°F (1477°K): "[t]he [FLHT-1] test plan called for a gradual [cladding] temperature increase [up] to approximately 2150°K (3400°F)."[102]

(It is noteworthy that other reports state that the MT-6B test had a PCT of 1400°K (2060°F)[103] and 1280°C (2336°F) (1553°K).[104] So the MT-6B test may have actually demonstrated that the Zircaloy oxidation rate was easily controllable by adding more coolant at cladding temperatures of either 2060°F (1400°K) or 1280°C (2336°F) (1553°K).)

Discussing the FLHT-1 test plan in more detail, "FLHT-1 Test Report" states:

> Once the power is set, the test will be started through its transient operation. *The term transient is somewhat of a misnomer*; operation will consist of a series of preplanned, discrete flow-reduction steps. The size

[101] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[102] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. v.
[103] *Id.*, p. viii.
[104] G. M. Hesson, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," p. 2.

> and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate.
> ...
>
> The bundle [coolant] flow rate will then be decreased in a series of precalculated flow steps... The duration of the time between steps is dictated by the time needed to reach near steady state and also by *the requirement that the Zircaloy-steam reaction be limited.* About 14 steps, each of about 1/2 hr. duration, are expected. *The last flow reduction step will be calculated to give a peak cladding temperature of about 2150°K (3400°F).* ...
>
> The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150°K (3400°F) [emphasis added].[105]

Indeed, the test conductors must have been taken by surprise when they could not control the zircaloy oxidation rate by increasing the coolant flow rate. They realized that there was no way to terminate the cladding-temperature increase—after peak cladding temperatures reached approximately 1475°K (2200°F)—short of reducing the reactor power to zero power, as they did "85 seconds after the start of the [cladding temperature] excursion."[106]

It is important to remember that the events described above occurred within a period of approximately 85 seconds: peak cladding temperatures increased from approximately 1520°K (~2275°F) or lower to approximately 2275°K (3635°F), within approximately 85 seconds. Additionally, as discussed above, in the graphs of "Typical Cladding Temperature Behavior"[107] and "Pseudo Sensor Readings for Fuel Peak Temperature Region,"[108] the slopes of the lines of the cladding-temperature value plots of the FLHT-1 test become nearly vertical, after the cladding-temperature values reach approximately 1520°K, indicating that only a short time period passed before temperatures reached approximately 2275°K (3635°F).

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105 W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," pp. 4.3-4.5.
106 *Id.*, p. 4.6.
107 *Id.*, p. 4.7.
108 *Id.*, p. 5.3.

It is noteworthy that even after the reactor power was reduced to zero power, that the autocatalytic oxidation reaction may have continued; "FLHT-1 Test Report" states:

> The reactor power was decreased at approximately 17:11:07, 85 sec. after the start of the excursion (approximately 131 minutes in Figure 4.1). The reactor reached 10% of the initial power approximately 35 sec. later and reached low neutron level in another 30 sec.
>
> There were two Indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.[109]

Discussing the alternative possibility that the temperature excursions were, in fact, effectively terminated, "FLHT-1 Test Report" states:

> A review of the thermocouple data led to the conclusion that the temperatures were not rising after the reactor shutdown. Typical cladding, coolant, and liner temperatures immediately after the reactor shutdown are shown in Figures 4.2, 4.3, and 4.4, starting at 17:12:00. The temperatures shown are somewhat erratic and show noise (probably associated with some thermocouple damage), but the general trend is downward, indicating an effective shutdown.
>
> Additional Indications of an effective test shutdown are shown by the saddle temperature, MMPD [(molten material penetration detector)] response, and bypass coolant power (radial heat loss) after the reactor power shutdown. Typical data from these sources are shown in Figures 4.5 through 4.7. All three of these indicators show steadily decreasing temperatures.[110]

It is also noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis" states that "[i]n the [FLHT-1] test, completed in March 1985, 12 ruptured zircaloy-clad rods were subjected to an autocatalytic temperature excursion. From the measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500-2600°F (1371-1427°C) temperature region."[111]

The FLHT-1 test is highly significant precisely because, once cladding temperatures reached as high as approximately 1475°K (2200°F), the test conductors

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[109] *Id.*, pp. 4.6-4.7.
[110] *Id.*, p. 4.7.
[111] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

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

In the FLHT-1 test, it is significant that autocatalytic oxidation occurred at approximately 1520°K (~2275°F) or lower, because the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[112]

d. The LOFT LP-FP-2 Experiment

Petitioner will now discus the LOFT LP-FP-2 experiment that was conducted in the Loss-of-Fluid Test ("LOFT") facility at Idaho National Engineering Laboratory, on July 9, 1985. The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[113] The LOFT LP-FP-2 experiment—the second and final fission product test conducted at the LOFT facility—had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy 1.67 meter fuel rods; it was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud. The LOFT LP-FP-2 experiment combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[114]

The LOFT LP-FP-2 experiment had an initial heatup rate of ~1°K/sec.[115] It is significant that "heatup rates [of 1°K/s or greater] are typical of severe accidents initiated from full power."[116] And regarding the significance of the initial heatup rate in the LOFT

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[112] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."
[113] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 13.
[114] *Id.*
[115] *Id.*
[116] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2; this paper cites Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, as the source of this information.

LP-FP-2 experiment, "Review of Experimental Results on LWR Core Melt Progression" states:

> The higher initial heating rate [of 1°K/sec.] in the LOFT [LP-]FP-2 experiment is related to the higher fraction of decay heat available following rapid blowdown of the coolant inventory in the reactor vessel. This higher heating rate leads to smaller oxide thickness on the cladding for a particular temperature and, therefore, more rapid oxidation. The increase in heating rate at the higher temperatures is the result of rapid oxidation of zircaloy and the strongly exothermic nature of the reaction (6.45 kJ/g Zr oxidized).[117]

And regarding the value of the data from the LOFT LP-FP-2 experiment, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

> Data from [the LOFT LP-FP-2] experiment provide a wealth of information on severe accident phenomenology. The results provide important data on early phase in-vessel behavior relevant to core melt progression, hydrogen generation, fission product behavior, the composition of melts that might participate in core-concrete interactions, and the effects of reflood on a severely damaged core. The experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[118]

Discussing the metal-water reaction measured-temperature data of the LOFT LP-FP-2 experiment, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states:

> The first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400°K on a guide tube at the 0.69-m (27-in.) elevation. This temperature is shown in Figure 3.7. A cladding thermocouple at the same elevation (see Figure 3.7) reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase. Note

[117] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7.

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

> that, due to the limited number of measured cladding temperature locations, the precise location of the initiation of [the] metal-water reaction on any given fuel rod or guide tube is not likely to coincide with the location of a thermocouple. Thus, the temperature rises are probably associated with precursory heating as the metal-water reaction propagates away from the initiation point. Care must be taken in determining the temperature at which the metal-water reaction initiates, since the precursory heating can occur at a much lower temperature. It can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400°K (2060°F).[119, 120]

Additionally, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" also states that the hottest measured cladding temperature reached 2100°K (3320°F) by 1504 ± 1 seconds;[121] and states that it was difficult to determine the PCT reached during the entire experiment—because of thermocouple failure—but that the PCT exceeded 2400°K (3860°F).[122]

Therefore, after the onset of rapid oxidation—after a heating rate of ~1°K/sec.[123]—peak cladding temperatures increased from approximately 1400°K (2060°F) to 2100°K (3320°F) within a range of approximately 35 seconds; in other words, after the onset of rapid oxidation, cladding temperatures increased at an average rate of approximately 20°K/sec. (36°F/sec.). In general agreement with this postulation, "Review of Experimental Results on LWR Core Melt Progression" states that "[i]n the LOFT [LP-]FP-2 experiment, which was driven by decay heat, the heating rate started out at about 1°K/sec. and increased to about 10-20°K/sec. above 1500°K [(2240°F)]."[124]

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

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

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

122 *Id.*, p. 33.

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

124 R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment

It is significant that "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[125] So at the point when peak cladding temperatures increased at a rate of greater than 10°K/sec. during the LOFT LP-FP-2 experiment, an autocatalytic oxidation reaction commenced; and that occurred when the temperature of a Zircaloy fuel rod or guide tube reached approximately 1400°K (2060°F), or when cladding temperatures reached approximately 1500°K (2240°F).

In a different account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Degraded Core Quench: A Status Report" states that "[t]he initial heating rate in the central assembly was ~1°K/sec. with an onset to rapid oxidation at a temperature near 1500°K [(2240°F)]."[126] In a similar account, as already mentioned, "Review of Experimental Results on LWR Core Melt Progression" states that the initial heatup rate was 1°K/sec., and that the heatup rate increased to approximately 10-20°K/sec. at a cladding temperature greater than 1500°K (2240°F).[127]

And offering yet another account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2" states that in the LOFT LP-FP-2 experiment that the metal-water reaction was initiated at 1450.0 ± 30 sec. after the beginning of the experiment and that at 1500 ± 1 sec, after the beginning of the

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Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[125] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

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

[127] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

experiment, the maximum cladding temperatures reached 2100°K;[128] elsewhere the same paper states that the "[m]etal-water reaction began at about 1450 seconds and [that the] hottest measured cladding temperature reached 2100°K [(3320°F)] by 1504 seconds."[129]

It is important to clarify that "rapid oxidation" is not necessarily autocatalytic oxidation. It is also important to consider questions such as, "At what point does rapid oxidation become autocatalytic oxidation?" As mentioned above, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[130]

As also mentioned above, "Review of Experimental Results on LWR Core Melt Progression" states that "[i]n the LOFT [LP-]FP-2 experiment, which was driven by decay heat, the heating rate started out at about 1°K/sec. and increased to about 10-20°K/sec. above 1500°K [(2240°F)]."[131] For this reason, it is reasonable to conclude that when "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" uses the term "rapid oxidation," it is discussing autocatalytic oxidation or at least a phenomenon that occurs shortly before the onset of autocatalytic oxidation.

As quoted above, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states that "[t]he first recorded and qualified rapid-temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430

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[128] D. W. Akers, C. M. Allison, M. L. Carboneau, R. R. Hobbins, J. K. Hohorst, S. M. Jensen, S. M. Modro, NUREG/CR-6160, "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2," April 1994, p. 12.

[129] *Id.*, p. xii.

[130] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

[131] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[seconds] and 1400°K…"[132] So it is reasonable to conclude that at some point when peak cladding temperatures were 1500°K (2240°F) or lower, cladding temperatures began increasing at a rate of greater than 10°K/sec., signaling the onset of an autocatalytic oxidation reaction.

In the LOFT LP-FP-2 experiment, it is significant that rapid oxidation occurred at a temperature of 2240°F or lower, because the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[133] Data from the LOFT LP-FP-2 experiment also indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

e. The CORA Experiments

The CORA experiments were conducted to study severe accident sequences, with electrically heated bundles of 2-meter long fuel rod simulators, held in place by three spacer grids (two Zircaloy, one Inconel), and surrounded by a shroud. The Electric heating was done with tungsten heating elements, installed in the center of annular UO_2 pellets, which, in turn, were sheathed by PWR Zircaloy-4 cladding. The total available heating power was 96kW, which had the capability of being distributed among three bundles of the fuel rod simulators. There were also unheated rods, filled with solid UO_2 pellets to correspond to LWR fuel rods.[134] In the CORA experiments the initial heatup rate of the fuel rod simulators was approximately 1°K /sec., in the presence of steam.

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

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially

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

[133] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

[134] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230460, p. 77.

depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec.[, after an initial heatup rate of about 1°K /sec.] The maximum temperatures attained are about 2000°C; the oxide layers formed and the consumption of the available steam set limits on the temperature escalation due to rate-controlled diffusion processes. The temperature escalation starts in the hotter upper half of the bundle and the oxidation front subsequently migrates from there both upwards and downwards."[135]

"CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures" also states that temperature escalations "continued even after shut-off of the electric power, as long as steam was available."[136]

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

It is also significant that the CORA experiments demonstrated that "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation."[139] So with good fuel assembly insulation—like what the core

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[135] *Id.*, p. 83.
[136] *Id.*, p. 87.
[137] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."
[138] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[139] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.

of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between approximately 1100°C and 1200°C (2012°F to 2192°F), and cladding temperatures start increasing at a maximum rate of 15°C/sec. (27°F/sec.).

It is noteworthy that the LOFT LP-FP-2 experiment was conducted with good fuel assembly insulation; it had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy 1.67 meter fuel rods; the test assembly was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud. In the LOFT LP-FP-2 experiment, autocatalytic oxidation occurred at cladding temperatures of approximately 2240°F or lower.

f. The PHEBUS B9R Test

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[140]

Discussing the PHEBUS B9R-2 test, "Status of ICARE Code Development and Assessment" states:

> During the B9R-2 test, an *unexpected strong escalation of the Zr-water reaction occurred* at mid-bundle elevation during the steam injection. Considerable heatup rates of 20 to 30°K/sec. were measured in this zone with steam starved conditions at upper levels. Post Irradiation Examinations (PIE) show cladding failures and considerable deformations (about 70%) [emphasis added].[141]

And offering a different account of the elevation at which the temperature excursion occurred during the PHEBUS B9R-2 test, "Degraded Core Quench: A Status Report" states that the B9R-2 test had "an unexpected high oxidation escalation in the

[140] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230126, p. 311.
[141] *Id.*

upper bundle zone (20 to 30°K/sec.)"[142] "Degraded Core Quench: A Status Report" states that the temperature excursion occurred in steam-rich conditions, after an initial heatup phase in pure helium (up to 1000°C), and that the PCT was approximately 1900°K, during the first oxidation phase. The PHEBUS B9R-2 test had a second oxidation phase and temperature escalation.[143]

Neither paper states what peak cladding temperatures were at the outset of the autocatalytic oxidation reaction; however, a graph of the cladding-temperature values at the 0.6 meter "hot-level" indicates that the autocatalytic oxidation reaction began when cladding temperatures were below 1477°K (2200°F)[144] (see Appendix G Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF)).

g. The QUENCH-04 Test

"Degraded Core Quench: Summary of Progress 1996-1999" states that the QUENCH-04 test, conducted at the QUENCH facility at Forschungszentrum Karlsruhe ("FZKA"), with a bundle of 21 electrically-heated, Zircaloy-clad rods, had a temperature excursion at 1560°K (~2350°F), due to rapid oxidation of the Zircaloy cladding. The bundle was heated at an increasing rate of 0.5°K/sec. to 1.5°K/sec. and when peak cladding temperatures reached 1560°K (~2350°F) the temperature excursion occurred.[145]

"Degraded Core Quench: Summary of Progress 1996-1999" also states that runaway (autocatalytic) oxidation of Zircaloy cladding by steam occurs at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or higher.[146]

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[142] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 14.
[143] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," p. 311.
[144] *Id.*, p. 312.
[145] 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.
[146] *Id.*

Discussing low temperature excursions that occurred in the unheated region at the top of the shroud (with Zircaloy content) in the QUENCH experiments, "Degraded Core Quench: Summary of Progress 1996-1999" states:

> 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. This would suppose that the oxide layer which normally protects against hydrogen uptake is either absent (possibly absorbed into the metal under very steam-starved conditions as the excursion proceeds), or is defective in some way which allows hydrogen to diffuse in readily. This point has not yet been resolved. CEA/ISPN report that similar low temperature excursions have been seen in the Phebus C3 experiment, but no detailed data are yet readily available.[147]

This passage is significant for two reasons: 1) because (as mentioned above) it states that "excursion temperatures associated with [the] runaway oxidation [of Zircaloy] by steam" are 1050°C to 1100°C (1922°F to 2012°F) or higher;[148] and 2) because it states that it is postulated that an exothermic hydriding reaction of Zircaloy content in the shroud caused temperature excursions—beginning at 750-800°C (1382-1472°F)—during the QUENCH experiments.

Clearly, data from severe fuel damage experiments demonstrates that autocatalytic oxidation occurs at temperatures far below what the Baker-Just and the Cathcart-Pawel equations predict. Therefore, the Baker-Just and the Cathcart-Pawel equations are both non-conservative for predicting the metal-water reaction rates that would occur in the event of a LOCA. Additionally, experimental data from severe fuel damage indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

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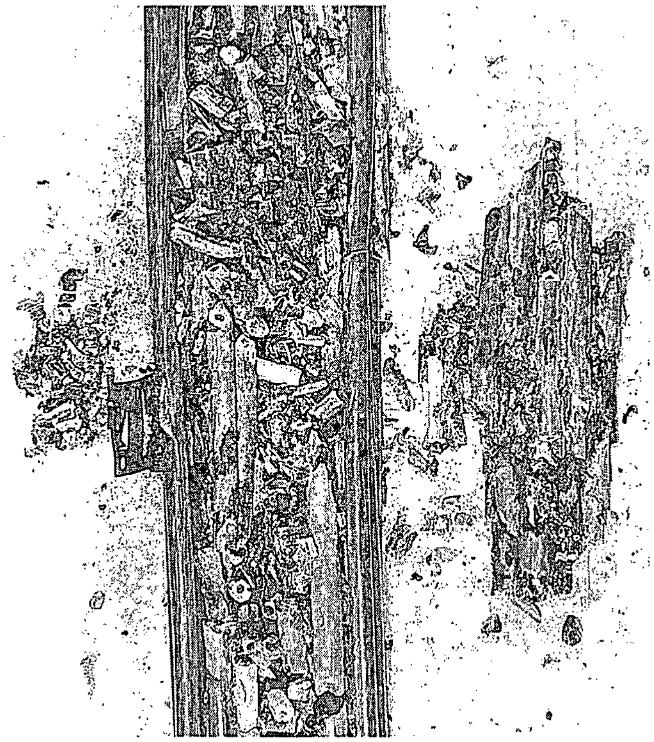
[147] *Id.*, pp. 9-10.
[148] *Id.*, p. 9.

h. Examining the Autocatalytic Metal-Water Reaction that Occurred during FLECHT RUN 9573



A section of the assembly from FLECHT run 9573[149]

"PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" ("PWR FLECHT Final Report") states that "[t]he objective of the PWR FLECHT...test program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[150]

FLECHT run 9573 was a thermal hydraulic test; however, in some respects it resembled a severe fuel damage test. During FLECHT run 9573, the Zircaloy assembly incurred autocatalytic oxidation.

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149 See Appendix A for more photographs of the assembly from FLECHT Run 9573; see also Appendix B for a photograph of the assembly from FLECHT Run 8874.

150 F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 1-1.

Discussing the extensive oxidation of the assembly of FLECHT run 9573, in its comments regarding petition for rulemaking 50-76 ("PRM-50-76"), Westinghouse stated:

> 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.[151]

First, it is important to point out that when Westinghouse performed the metallurgical analyses for FLECHT Run 9573, Westinghouse did not measure the oxide thicknesses in locations of the assembly that incurred runaway (autocatalytic) oxidation—at a temperature "beyond 2300°F."

Second, 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[152]—"the expected range."

Third, an occurrence of runaway (autocatalytic) oxidation at a temperature greater than 2300°F (assuming that means at a temperature below 2400°F) is not within "the expected range" of what the Baker-Just correlation would predict: the Baker-Just correlation predicts that autocatalytic oxidation of Zircaloy occurs at cladding temperatures of approximately 2600°F.[153]

It is significant that 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

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[151] 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.

[152] 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.

[153] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

> these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients.[154]

First, 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 ZrO2 thickness equations to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation.

In fact, there is no metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation, because Westinghouse did not obtain such data.

Second, the NRC did not consider that FLECHT run 9573 incurred autocatalytic (runaway) oxidation at a temperature "beyond 2300°F,"[155] as Westinghouse's comments regarding PRM-50-84 stated. An occurrence of autocatalytic oxidation at a temperature greater than 2300°F (assuming that means at a temperature below 2400°F) is not within the temperature range of where the Baker-Just correlation would predict autocatalytic oxidation of Zircaloy to occur.

So the NRC performed a technical safety analysis on issues raised in a petition for rulemaking that argued that the Baker-Just and Cathcart-Pawel equations were non-conservative for accurately calculating the extent of the Zircaloy-water reaction that would occur in the event of a LOCA, and the NRC did not consider that, with high probability, run 9573 incurred autocatalytic oxidation at a temperature below 2400°F.[156]

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[154] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 21-22.
[155] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[156] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157."

As mentioned above, at some point before the NRC conducted its technical safety analysis of PRM-50-76, it performed 50 LOCA calculations with RELAP5/Mod3 that found that:

> The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F.[157]

So when the NRC conducted its technical safety analysis of PRM-50-76, it failed to consider that according to its own RELAP5/Mod3 calculations that the Baker-Just and Cathcart-Pawel equations predict that the autocatalytic oxidation of Zircaloy cladding begins at approximately 2600°F and 2700°F, respectively.

Furthermore, the NRC failed to consider that data from multi-rod (assembly) severe fuel damage experiments indicates that the Baker-Just and Cathcart-Pawel equations are non-conservative for predicting the metal-water reaction rates that would occur in the event of a LOCA.

2. The Fact that the Baker-Just and Cathcart-Pawel Equations were not Developed to Consider how Heat Transfer would Affect Zirconium-Water Reaction Kinetics in the Event of a LOCA

It is significant that in the NRC's report on its denial of a petition for rulemaking—PRM-50-76—that argued that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA, the NRC states:

> [The petitioner] states that the Baker-Just equation does not include any allowance for the complex thermal-hydraulic conditions during a LOCA. The NRC disagrees with the petitioner's assertions. ...

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

> The petitioner is also concerned about the large water volume compared to the zirconium sample size with respect to the quench capability of zirconium-clad fuel rods. As noted, these experiments were atypical in that respect, but barely used in the formulation of the Baker-Just correlation. Further, it should be noted that *the Baker-Just report was not intended to be a heat transfer study*, but rather an investigation of zirconium-water reaction kinetics at very high temperatures [emphasis added].[158]

And in the same report on its denial of PRM-50-76, the NRC states:

> The petitioner stated that RG 1.157, which allows use of the data and the Cathcart-Pawel equation presented in NUREG-17, results in flawed evaluations of ECCS performance. The NRC disagrees with the petitioner's assertions on this issue. ...the petitioner states that the limited test conditions described in NUREG-17 preclude the use of the results for LOCA calculations. He further states that Zircaloy-4 specimens were not exposed to LOCA fluid conditions and that only steam was applied at very low velocities for the main test series. The petitioner states that there was no documented heat transfer from the Zircaloy surface to the slow-flowing steam and that as a result the conditions of the small-scale laboratory tests were not typical of the complex thermal-hydraulic conditions that prevail during a LOCA.
>
> The petitioner suggests that without liquid water, the tests are invalid. The NRC disagrees. The presence of liquid water would invalidate the tests. Accurate steady-flow measurement would be extremely difficult. The droplets or liquid film would make it difficult to achieve the relatively constant sample temperatures that are necessary in these reaction kinetics tests. However, adequate steam flow is a concern. If the flow is too low, the reaction becomes steam starved. Otherwise, it is unnecessary to have steam flow typical of LOCA/ECCS conditions. *These are not heat transfer tests*. Once a reaction rate model is developed using data from experiments like these, the model should be validated against transient tests under LOCA conditions, as in the four Zircaloy tests described in WCAP-7665 and the transient tests described in the Cathcart-Pawel report [emphasis added].[159]

So in the first passage above the NRC states that the "the Baker-Just report was not intended to be a heat transfer study, but rather an investigation of zirconium-water reaction kinetics at very high temperatures;"[160] and in the second passage above the NRC states that the zirconium-water reaction kinetics tests that were conducted to develop the

[158] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 11, 12.
[159] *Id.*, pp. 13-14.
[160] *Id.*, p. 12.

Cathcart-Pawel equation were not heat transfer tests. What the NRC does not consider is that under the complex thermal-hydraulic conditions that would occur in the event of a LOCA, heat transfer would affect zirconium-water reaction kinetics.

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

> There doesn't seem to be any magic temperature at which you get some autocatalytic reaction that runs away. *It's simply a matter of heat balances: how much heat from the chemical process and how much can you pull away* [emphasis added].[161]

And regarding Zircaloy oxidation tests where heat loss from the test samples to relatively cold surroundings prevented autocatalytic oxidation from occurring—in "Petitioner's Responses to Comments by Westinghouse and NEI [Regarding PRM—50-76]"—Robert H. Leyse states:

> The high temperature oxidation tests [reported on in WCAP-12610, Appendix E[162]] were performed by Nuclear Electric, plc in the United Kingdom. Twenty four ZIRLO alloy and six Zr-4 samples were tested at temperatures ranging from 1832°F to 2372°F. The cylindrical tubing specimens were approximately 0.6 inches long and were from production grade 17x17 tubing.
>
> Appendix E candidly discloses: "Since, particularly at high temperatures, the self heating of the specimen results in its being at a higher temperature than its surroundings, any temperature measured will be equal to or lower than that of the test specimen."[163] In other words, in order for the investigators at Nuclear Electric to prevent runaway [oxidation from occurring as a result of] the heat of reaction at high temperatures (self heating), it was necessary to maintain the surroundings at a substantially lower temperature than the specimen. In this manner, the heat loss by radiation to the relatively cold surroundings compensated for the heat produced by chemical reaction with the pure oxygen. This then leads to the question: What if Nuclear Electric had conducted the investigation with a 17x17 arrangement of ZIRLO or Zr-2 tubes captured within a

161 Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Committee, Meeting, April 4, 2001. In the transcript the second sentence was transcribed as a question; however, the second sentence was clearly not phrased as a question.
162 Westinghouse, "ZIRLO High Temperature Oxidation Tests," WCAP-12610, Appendix E, August, 1990. Only a limited portion of the report is available to the public; it is classified by Westinghouse as a proprietary report.
163 *Id.*, p. 2.

> Zircaloy-4 structural grid with ZIRLO thimbles? The answer is that the assembly would have rapidly been destroyed [by] runaway [oxidation] if a sufficient flow of oxygen had been maintained.[164]

And regarding how Zircaloy cladding incurs autocatalytic oxidation at approximately 1500°K (2240°F) under conditions of poor heat transfer, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

> The [Zircaloy-steam] reaction is highly exothermic (586 kJ/mol), and this may lead to uncontrolled temperature escalation *under conditions of poor heat transfer*; typically at temperatures above about 1500°K" [emphasis added].[165]

Furthermore, regarding how heat transfer affects the temperature at which the autocatalytic oxidation of Zircaloy bundles occurs, the paper, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," states:

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction *crucially depends on the heat loss from the bundle; i.e., on bundle insulation.* With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec.[, after an initial heatup rate of about 1°K /sec.] The maximum temperatures attained are about 2000°C; the oxide layers formed and the consumption of the available steam set limits on the temperature escalation due to rate-controlled diffusion processes. The temperature escalation starts in the hotter upper half of the bundle and the oxidation front subsequently migrates from there both upwards and downwards [emphasis added].[166]

So with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between approximately 1100°C and 1200°C (2012°F to 2192°F), and cladding temperatures start increasing at a maximum rate of

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[164] Robert H. Leyse, "Petitioner's Responses to Comments by Westinghouse and NEI [Regarding PRM—50-76]," December 14, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML023310144, p. 1.

[165] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 4.3.

[166] 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.

15°C/sec. (27°F/sec.). It is certainly evident that in the event of a LOCA, heat transfer would affect the temperature at which the autocatalytic oxidation of Zircaloy cladding would occur. Therefore, it seems obvious 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, precisely because they were not developed to consider how heat transfer would affect zirconium-water reaction kinetics.

3. Conclusion of the Metal-Water Reaction Rate Section

It has been demonstrated in the Metal-Water Reaction Rate Section that data from multi-rod (assembly) severe fuel damage experiments indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic 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. It has also been demonstrated that data from multi-rod (assembly) severe fuel damage experiments indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

In the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[167] and 2240°F,[168] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[169] Within a period of less than 60 seconds peak

[167] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[168] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.
[169] *Id.*

cladding temperatures would increase to above 3000°F;[170] the melting point of Zircaloy is approximately 3308°F.[171]

Regarding core-melt phenomena—some of which would occur in the relatively low temperature range from 1473°K to 1673°K (2192°F to 2552°F)—that would, with high probability, occur in the event of a event of a LOCA, if peak cladding temperatures were to increase to between approximately 2060°F and 2240°F, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

> The composition of an LWR core is such that melting could occur in a variety of ways involving complex chemical reactions. The major components of a PWR core are UO_2 and Zircaloy, which make up about 78 Wt% and 16 Wt%, respectively. The remaining materials are primarily stainless steel, Inconel, Ag-In-Cd control rod material and Al_2O_3 used in the burnable poison rods. For a BWR core, the major components are UO_2 (68 Wt%) and Zircaloy (24 Wt%). Stainless steel and B_4C control rod material comprise the remaining 8 Wt%. Hofmann *et al.*[172] identified three distinct temperature regimes for melting and liquid phase formation during a severe accident for the heatup rates of 1°K/s or greater. These heatup rates are typical of severe accidents initiated from full power. Each temperature regime is characterized by different processes (Figure 2.1).[173]
> The first temperature regime considered by Hofmann is between 1473°K and 1673°K. Within this temperature regime, control rods, burnable poison rods, and structural material can form low-temperature liquid phases.[174] These liquefied materials may relocate and form local blockages which could restrict flow and cause accelerated heatup of the core.

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

171 NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

172 Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, is cited as the source of this information.

173 See Appendix H Figure 2.1. Temperature Regimes for Extensive Liquid Phase Formation and Relocation, which depicts that the onset of temperature escalations of 10°K/s or greater occur when cladding temperatures reach 1200°C (2192°F).

174 Hofmann, P., Markiewicz, M., "Chemical Behavior of (Ag, In, Cd) Absorber Rods in Severe LWR Accidents," KfK Report 4670 (1990); Hofmann, P., Markiewicz, M., Spino, J., "Reaction Behavior of B_4C Absorber Material with Stainless Steel and Zircaloy in Severe LWR Accidents," Nuclear Technology, Vol. 90 (1990) 226-244; and Hofmann, P., Markiewicz, M., Spino, J., "Chemical Interactions Between A1203, which is Used in Burnable Poison Rods, and Zircaloy-4 up to 1500°C," J. Nuclear Mater. 166 (1989) pp. 287-299, are cited as the source of this information.

In a PWR core, the main reactions are those between the Ag-In-Cd alloy (control rods), Zircaloy (guide tubes), and Inconel (spacer grids). The Ag-In-Cd alloy has a very low melting temperature (1073°K) and it is likely to be the first component to melt after core uncovery. Any failure of the control rod cladding will allow the molten Ag-In-Cd alloy to contact the Zircaloy guide tubes and even some of the Zircaloy cladding around the fuel rods. The Zircaloy can be chemically dissolved by the molten Ag-In-Cd alloy which could cause local damage in the core region well below the melting temperature of Zircaloy (approximately 2033°K).[175] In addition, there could be contact between stainless steel cladding (control rods) and Zircaloy (guide tubes) and between the Inconel (space grids) and Zircaloy (fuel rods). The localized Zircaloy/stainless steel (or Inconel) contact results in chemical interactions with the formation of liquid phases at relatively low temperatures. This early-melt formation at about 1473°K could initiate the melt progression within the fuel assembly at low temperatures as recognized in the CORA tests, where fuel rod bundles were heated up to complete meltdown.[176] Carlson and Cook[177] have shown that the stainless steel/Zircaloy interaction is one of the major material reactions that occurred during the TMI-2 accident.

In a BWR core, the control rods consist of boron carbide (B_4C) pellets in stainless steel cladding; the control rods are located in a four-bladed stainless steel assembly. The major reaction in a BWR is between B_4C and stainless steel. Hofmann *et al.*,[178] showed that the B_4C control rods exhibit a strong reaction above 1523°K, which resulted in rapid liquefaction of the control rod material. Liquefaction occurred below the melting point of B_4C (2623°K) and stainless steel (1723°K) due to eutectic interactions. The liquid B_4C/stainless steel reaction products can also interact with the Zircaloy channel box. Interaction and liquefaction between B_4C and Zircaloy occurs at about 1923°K, which is about 400°K higher than that of the B_4C/stainless steel interaction.

The second temperature regime considered by Hofmann is between 2033°K and 2273°K. If the Zircaloy clad has not been oxidized, then it will melt at about 2033°K and relocate downward along the fuel rod. If a sufficient oxide layer has formed on the outside surface of the clad, then

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175 Hofmann, P., Markiewicz, M., "Chemical Behavior of (Ag, In, Cd) Absorber Rods in Severe LWR Accidents," KfK Report 4670 (1990), is cited as the source of this information.

176 Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, is cited as the source of this information.

177 Carlson, E. R., and Cook B. A., "Chemical Interactions Between Core and Structural Materials," Proceedings of the First International Information Meeting on the TMI-2 Accident, p. 191, 1985, Rev. May 91, is cited as the source of this information.

178 Hofmann, P., Markiewicz, M., Spino, J., "Reaction Behavior of B_4C Absorber Material with Stainless Steel and Zircaloy in Severe LWR Accidents," Nuclear Technology, Vol. 90 (1990) 226-244, is cited as the source of this information.

> relocation of any molten Zircaloy on the inside will be prevented because the oxide layer will remain solid until the core reaches much higher temperatures (the melting point of ZrO_2 is 2973°K), or until the oxide layer fails mechanically, or until the layer is dissolved by molten Zircaloy. Under these conditions, the molten Zircaloy will chemically dissolve part of the solid UO_2 pellet and ZrO_2 shell.[179] The result is chemical dissolution (*i.e.*, liquefaction) of UO_2 and ZrO_2 by the molten Zircaloy at about 1000°K below the melting points of UO_2 and ZrO_2. The molten (Zr, U, O) mixture and molten metallic Zircaloy flow downward in a "candling process" from the higher temperature regions of the core into the lower temperature region where they solidify. The relocation and solidification of the metallic and ceramic melts could form a blockage in the flow channels, which would inhibit flow and accelerate core damage. Since the mixture contains decay heat, remelting and solidification can occur repetitively as water boils-off and core meltdown proceeds.
>
> The third temperature regime is between 2873°K and 3123°K. If a reactor core ever reaches this high temperature regime, the remaining UO_2, ZrO_2, and the (U, Zr) O_2 solid solution will start to melt. This will lead to complete meltdown of all remaining core materials.[180, 181]

Summarizing its description of core-melt phenomena, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

> In summary, *for a heatup rate of 1°K/s or larger*, core meltdown processes have been characterized for three temperature regimes. Initial melting and relocation in a reactor core starts with the failure of control rods, guide tubes, and Inconel spacer grids *at relatively low temperatures*. Local damage caused by interactions with Zircaloy could also occur during this time period. Larger scale fuel damage occurs at higher temperatures after the metallic Zircaloy melts and dissolution of UO_2 pellets and ZrO_2 occurs. At even higher temperatures, the ZrO_2 and UO_2 melt, which leads to total core meltdown [emphasis added].[182]

The descriptions above are for severe accidents; however, the phenomena described, would also, with high probability, apply to LOCAs, if peak cladding

[179] Hofmann, P., Uetsuka, H., Wilhelm, A. N., Garcia, E. A., "Dissolution of Solid UO_2 by Molten Zircaloy and its Modeling," Int. Symposium. on Severe Accidents in Nuclear Power Plants, Sorrento, Italy, 21-25 March 1988 (IAEA-SM-2986/1), is cited as the source of this information.

[180] Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, is cited as the source of this information.

[181] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, pp. 2.2-2.4.

[182] *Id.*, p. 2.4.

temperatures were to reach between approximately 2060°F and 2240°F. It is significant that "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that heatup rates of 1°K/s or greater—typical of severe accidents initiated from full power—lead to the onset of autocatalytic oxidation and temperature excursions of 10°K/s or greater, when peak cladding temperatures reach approximately 1200°C (2192°F) (see Appendix H Figure 2.1. Temperature Regimes for Extensive Liquid Phase Formation and Relocation).

It is clear that the NRC has ignored data from multi-rod (assembly) severe fuel damage experiments that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. The NRC has also ignored data from such experiments that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

D. FLECHT Run 9573

1. Westinghouse's Analysis of the Experimental Data from FLECHT Run 9573

It is significant that "PWR FLECHT Final Report" has an inconsistent analysis of the experimental data from FLECHT run 9573.

Regarding run 9573, "PWR FLECHT Final Report" states:

> The final PWR-FLECHT Zircaloy bundle test, Run 9573, was conducted with a nominal initial cladding temperature of 2000°F and a flooding rate of 1 in./sec. For this test, the stainless steel guide tubes were replaced with Zircaloy guide tubes and the freedom of the heater rods for vertical expansion was increased. Cladding temperatures were predicted to reach 2400°F after about 30 seconds, at which time heater element failures were expected to occur.
>
> During the test, heater element failures started at 18.2 seconds; sixteen elements failed by 30 seconds and all but nine of the forty-two heater elements had failed when power was shut off at 55.5 seconds. At the time of the initial 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).
>
> Post-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation. The heater rod failures were apparently caused by localized temperatures in

excess of 2500°F. Possible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components. The remainder of the bundle was in excellent condition, however, and there was very little rod bowing compared to Run 8874.

Analysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test. However, the data from this period is suspect and has therefore not been considered in comparing stainless steel and Zircaloy heat transfer behavior. In addition to the short time involved, anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. These may have been related to the high steam probe temperatures measured at the 7 ft elevation. Data beyond the first eighteen seconds was not valid due to the large number of heater rod failures.

It should be noted that the heater element failures which occurred in Runs 8874 and 9573 were not related to the behavior of reactor fuel during a loss-of-coolant accident. The failures referred to were failures of the heater rod internal electrical resistance element. Failure of this element resulted in either a loss of power to the heater rod or, more commonly, arcing from the resistance element to the clad. Aside from the regions in which heater rod failures took place, the clad was generally in excellent condition throughout the remainder of the bundles, including the peak temperature midplane regions.[183]

First, as mentioned above, "PWR FLECHT Final Report" does not mention that run 9573 incurred autocatalytic oxidation. So Westinghouse omitted very significant information in its report of run 9573. However, Westinghouse does state that "[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[184, 185]

Second, the passage above has inconsistent conclusions: 1) it essentially states that the heater element failures which occurred in run 9573 were related to the behavior of reactor fuel during a LOCA: "[t]he heater rod failures were apparently caused by

[183] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 3-97, 3-98.
[184] *Id.*, p. 3-97.
[185] See Appendix A for photographs of the assembly from FLECHT Run 9573; see also Appendix B for a photograph of the assembly from FLECHT Run 8874.

localized temperatures in excess of 2500°F. Possible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components;"[186] and 2) then it states that "[i]t should be noted that the heater element failures which occurred in Run...9573 were not related to the behavior of reactor fuel during a loss-of-coolant accident. The failures referred to were failures of the heater rod internal electrical resistance element. Failure of this element resulted in either a loss of power to the heater rod or, more commonly, arcing from the resistance element to the clad."[187]

(It is noteworthy that "PWR FLECHT Final Report" has other similar inconsistencies: 1) it states that during run 9573, "several heaters failed during flooding while the mid-plane temperatures were only 2200-2300°F. The heaters apparently failed because of higher temperatures that developed above the mid-plane region which were most likely caused by steam reaction with a Zircaloy grid;"[188] and 2) elsewhere it states that "[e]ven though the specimens examined reached temperatures as high as 2545°F, there was no evidence of clad shattering or failure as a result of being exposed to typical loss-of-coolant accident environments."[189])

Third, the heater element failures that did occur—approximately 12 seconds before they were expected to occur—were caused by heat generated from the autocatalytic oxidation reaction that run 9573 incurred. Heater element failures were expected to occur when cladding temperatures reached 2400°F, after about 30 seconds. However, the heater element failures were not expected to be caused by heat generated from an autocatalytic oxidation reaction. An autocatalytic oxidation reaction was not predicted or expected to occur at any time during run 9573.

It is significant that, more than two years after FLECHT run 9573 was completed, at the 1973 ECCS hearing, Westinghouse argued that the regulated limit of the peak cladding temperature ("PCT") in the event of a LOCA should be "at least 2700°F;"[190] in

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[186] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[187] *Id.*, p. 3-98.
[188] *Id.*, p. A-14.
[189] *Id.*, p. 5-5.
[190] 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

1973, the limit was 2300°F.[191] So when run 9573 was conducted in December 1970, Westinghouse certainly did not believe that autocatalytic oxidation of Zircaloy cladding would occur at temperatures below 2700°F.

Fourth, the passage above states that "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test;"[192] and concludes that "the data from [the first 18 seconds] is suspect and has therefore not been considered in comparing stainless steel and Zircaloy heat transfer behavior."[193] Elsewhere "PWR FLECHT Final Report" "recommend[s] that stainless steel clad heat transfer coefficients be used as a conservative representation of Zircaloy behavior."[194]

This is highly significant because the data reported in "PWR FLECHT Final Report" is important for ECCS evaluation calculations, required for all holders of operating licenses for nuclear power plants. Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models, Post-Blowdown Phenomena, Refill and Reflood Heat Transfer for Pressurized Water Reactors*, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]."

Fifth, the passage above concludes that the negative heat transfer coefficients, found in the analysis of the test results of run 9573—indicating "heat transfer into (rather than out of) the rod"[195]—were, in fact, "anomalous."

The passage states that "anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period"[196] (*i.e.*, more heat was transferred into the bundle midplane than was removed from that

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Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1097. 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.

[191] *Id.*

[192] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.

[193] *Id.*, pp. 3-97, 3-98.

[194] *Id.*, p. 5-3.

[195] *Id.*, p. 3-40.

[196] *Id.*, p. 3-98.

location), and posits that the "anomalous" negative heat transfer coefficients "may have been related to the high steam probe temperatures measured at the 7 ft elevation."[197] The passage also posits that "[p]ossible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components."[198]

(It is noteworthy that in 2002, regarding this phenomenon, in Westinghouse's comments on PRM-50-76, Westinghouse stated that "[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."[199])

So "PWR FLECHT Final Report" concludes that the negative heat transfer coefficients were anomalous, yet it also posits that the phenomenon of "heat transfer into (rather than out of) the rod"[200] was caused by heat generated from the exothermic Zircaloy-steam reaction. "PWR FLECHT Final Report" offers no credible explanation for concluding that the negative heat transfer coefficients were anomalous.

"PWR FLECHT Final Report" does not even consider the possibility that the experimental data from the first 18.2 seconds of run 9573 is valid, even though it states that "[t]he heater rod failures were...caused by localized temperatures in excess of 2500°F;"[201] and that "the 7 ft steam probe [measured temperatures], which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure)."[202] Indeed, Westinghouse's decision to not consider the data from the first 18.2 seconds of FLECHT run 9573, for comparing stainless steel and Zircaloy heat transfer behavior, seems unscientific.

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[197] *Id.*
[198] *Id.*, p. 3-97.
[199] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[200] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-40.
[201] *Id.*, p. 3-97.
[202] *Id.*

2. Background Information from Two Westinghouse Memorandums that Indicates that the Data from FLECHT Run 9573 is Valid

Petitioner will now provide background information from two Westinghouse memorandums regarding FLECHT run 9573 that indicates that the data from the first 18.2 seconds of FLECHT run 9573 is valid.

First, three days after FLECHT run 9573 was conducted, on December 14, 1970, Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, wrote a memorandum regarding run 9573 that states:

> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.[203, 204]

Second, seven days after FLECHT run 9573 was conducted, on December 18, 1970, F. F. Cadek, Manager, Westinghouse, Thermal-Hydraulic Development, PWR Systems Division, wrote a memorandum that states:

> Preliminary results of...Zirc[aloy] Run 9573 are summarized in the attachment. The run is considered valid up to the point of the first heater failure at 18.2 sec.[205, 206]

So Leyse's, December 14, 1970, memorandum states that the temperature-measuring system used for FLECHT run 9573 was subjected to a thorough audit by Idaho Nuclear Corporation that found that the system was highly reliable. And Cadek's, December 18, 1970, memorandum explicitly states that FLECHT run 9573 "is considered valid up to the point of the first heater failure at 18.2 sec."[207]

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203 Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.

204 See Appendix I Memorandum RD-TE-70-616, FLECHT Monthly Report, December 14, 1970.

205 F. F. Cadek, Manager, Westinghouse, Thermal-Hydraulic Development, PWR Systems Division, Memorandum RD-THD-17, "FLECHT Technical Review Meeting Minutes No. 58," December 18, 1970, p. 1.

206 See Appendix J Memorandum RD-THD-17, FLECHT Technical Review Meeting Minutes No. 58, December 18, 1970.

207 F. F. Cadek, Manager, Westinghouse, Thermal-Hydraulic Development, PWR Systems Division, Memorandum RD-THD-17, "FLECHT Technical Review Meeting Minutes No. 58," December 18, 1970, p. 1.

3. The "Uncertain and Conflicting Evidence" of the FLECHT Zircaloy Runs

It was because of the "uncertain and conflicting evidence"[208] of the four Zircaloy runs that, 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 [FLECHT] run 9573."[209]

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

> The [four] FLECHT runs made with zircaloy clad rods provide uncertain and conflicting evidence. Westinghouse pointed out that all of the zircaloy runs except one (run 9573) yield higher heat transfer coefficients than were obtained with [stainless] steel... 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 rate was about one inch per second.[210]

In PRM-50-84, Robert H. Leyse, the principal engineer in charge of directing the Zircaloy FLECHT tests, states:

> The stainless steel heat transfer behavior is certainly not a conservative representation of Zircaloy behavior. The data for the first 18 seconds of Run 9573 are real and certifiable. There is no basis for rejecting the negative heat transfer coefficients in run 9573. The higher values of the heat transfer coefficients of Run 8874 are also valid. The differences in the behavior between these runs are explained by the differences in the thermal hydraulic conditions that led to a different combination of heat transfer and mass transfer factors; the differences are not explained on the basis of inconsistency of the data.[211]

In PRM-50-84, Robert H. Leyse also states:

> The 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

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[208] 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."

[209] *Id.*

[210] *Id.*

[211] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 8.

> 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.[212]

And below are two Westinghouse memorandums that help explain the data—the "uncertain and conflicting evidence"—from run 8874 (during the first 10 seconds) and run 9573 (during the first 18.2 seconds).

On July 24, 1970, Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, wrote a memorandum regarding run 8874 that states:

> The initial heat transfer coefficient[213] is at least 1.7 times higher in a Zircaloy bundle (run 8874) than in a stainless [steel] bundle (run 6155) for the same flooding rate (6 in./sec.[214]) and start-of-flood temperatures of 2300°F and 2200°F, respectively. The higher coefficients for the Zircaloy bundle may be explained by high hydrogen concentrations (20% or more) in the film at the surface of the heater. At 2000°F, the thermal conductivity of hydrogen is approximately five times that of superheated steam. Although hydrogen production rates are probably not sufficient to lead to significant concentrations in the bulk coolant (the mixture of superheated steam and water droplets), the hydrogen concentrations within the film at the surface of the heater can easily reach significant values.[215, 216]

And on December 14, 1970, Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, wrote a memorandum regarding run 9573 that states:

> The final FLECHT test (Bundle Z-10) was completed on December 11, 1970. The test was run with flooding of 1 in./sec. beginning at 2000°F. Several heaters failed approximately 18 seconds after flooding when the peak indicated midplane temperature was 2325°F. Heater failure at this temperature is unlikely, particularly under conditions of decay heat and increasing temperature. The steam probe thermocouple located one foot above midplane in close proximity to a Zircaloy grid indicated an

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212 *Id.*, p. 6.

213 "The initial heat transfer coefficient," refers to the heat transfer coefficient during the first 10 seconds of run 8874, after flooding; see "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-96.

214 The flood rate of run 8874 was 6.0 in./sec. for 8 seconds, followed by a step reduction to 1 in./sec.; the flood rate of run 6155 was a constant 5.9 in./sec.; see "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," pp. 3-6, 3-8, 3-96, B-2.

215 Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum PA-TE-70-419, "Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874)," July 24, 1970.

216 See Appendix K Memorandum PA-TE-70-419, Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874), July 24, 1970.

extremely rapid rate of temperature rise (over 300°F/sec.) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding. It appears likely that ignition of the Zircaloy grids led to high rates of heat input* at the elevation one foot above (and below) midplane and this caused over-temperature and failure of the heaters. Test results are currently being studied.

> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.
>
> *The ratio of surface area to heat capacity for a Zircaloy grid is approximately 15 times that of a heater rod; hence, Zircaloy-steam reactions can lead [to] steeper temperature ramps in the vicinity of a Zircaloy grid.[217, 218]

So the differences in the behavior of run 8874 (during the first 10 seconds) and run 9573 (during the first 18.2 seconds) are explained by the differences in the thermal hydraulic conditions and by the different quantities of heat generated from the Zircaloy-water reactions, not on the basis of inconsistency of the data. And the differences in the thermal hydraulic conditions and, in turn, the different quantities of heat generated from the Zircaloy-water reactions were a consequence of the different flood rates.

Regarding the phenomena of low flood rates and the superheated-steam heating of stainless steel cladding during the FLECHT tests, "PWR FLECHT Final Report," states:

> The negative heat transfer coefficient for the 10-foot elevation at early times indicates heat transfer into (rather than out of) the rod. This was caused by the presence of superheated steam having temperatures above the [stainless steel] clad temperature at the 10-foot elevation. ... Negative heat transfer coefficients were generally found at the 10-foot elevation for low flooding rate runs (2 in./sec. or less) at early times (from around 5 up to a maximum of about 120 sec. after flood).[219]

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[217] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.
[218] See Appendix I Memorandum RD-TE-70-616, FLECHT Monthly Report, December 14, 1970.
[219] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-40.

So FLECHT run 9573 was not the only FLECHT run where an analysis of the test results found negative heat transfer coefficients, indicating "heat transfer into (rather than out of) the rod."[220] In the case of run 9573, the presence of superheated steam caused the Zircaloy cladding to rapidly oxidize—an exothermic reaction that, in turn, generated yet more heat. At such temperatures the reaction became autocatalytic.

Indeed, there is no scientific basis for rejecting the data from the first 18.2 seconds of run 9573. The fact that the oxidation reaction of run 9573 became autocatalytic and stainless steel tests exposed to similar temperatures did not, has to do with the differing amounts of heat generated from the oxidation of Zircaloy and stainless steel, within the temperature range.

4. A Comparison of the High Temperature Oxidation Behavior of Zircaloy and Stainless Steel Assemblies

First, it is noteworthy that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400°K. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[221] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425°K. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain*" [emphasis added].[222]

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.[223] 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,

[220] *Id.*
[221] 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.
[222] *Id.*, p. 4.4.
[223] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-6.

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.

And regarding the differences in the oxidation behavior of Zircaloy and stainless steel heater-rod assemblies, 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:

> There is no reason to believe that the temperature measuring system was not reliable for the first 18 seconds of run 9573. The negative heat transfer coefficients were real values[; *i.e.*, a phenomenon where more heat was transferred into the bundle midplane than was removed from that location]. This is because there is an extremely significant and real difference between the behavior of Zircaloy and stainless steel heat transfer assemblies. In the case of stainless [steel], there is relatively little heat of reaction from oxidation in the temperature range. In contrast, the heat of reaction [from] oxidation of...Zircaloy is substantial in the temperature range. The intense heat of reaction yielded high enough temperatures of the Zircaloy cladding to force heart flow back into the heater. The thermocouples did not yield false signals. There is no justification for classifying the negative heat transfer coefficients as anomalous.
>
> The following FLECHT experience provides a very direct comparison of the high temperature behavior of Zircaloy and stainless steel bundles: Although there is no discussion in any of the FLECHT reports, on April 18, 1969, a stainless steel bundle was substantially overheated due to installation and operating errors. The event is discussed in...Westinghouse memo RD-ED-THE-33. [The memo states], "The maximum temperature of the [stainless steel] bundle was in excess of 2500°F (Chromel-Alumel thermocouple conversion tables terminate at 2500°F)."[224, 225] The bundle remained totally intact without any destruction of the stainless steel cladding, although most of the heating elements had burned out. This is in marked contrast to the experience with FLECHT Run 9573.[226]

Indeed, stainless steel cladding heat transfer coefficients are not a conservative representation of Zircaloy heat transfer coefficients, for some of the conditions that

[224] R. F. Farman, Westinghouse, Thermal and Hydraulic Experimentation, Memorandum RD-ED-THE-33, "Report of Events Leading to FLECHT 10 x 10 Bundle Test," April 23, 1969, p. 2.
[225] See Appendix L Memorandum RD-ED-THE-33, Report of Events Leading to FLECHT 10 x 10 Bundle Test, April 23, 1969.
[226] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.

would occur in the event of a LOCA. It is significant that for run 9573 the "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test."[227] Yet the data from run 9573 is not considered valid. And "PWR FLECHT Final Report" states:

> Properly used, PWR FLECHT test results can improve the accuracy of reactor LOCA analysis. The heat transfer correlations which were developed are *conservative* in that they do not take any credit for the effects of "fallback" or borated coolant and are *based on stainless steel clad data* [emphasis added].[228]

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

5. Conclusion of the FLECHT Run 9573 Section

It is significant that FLECHT run 9573 incurred autocatalytic oxidation and had a lower initial cladding temperature than, and the same power level as, other FLECHT Zircaloy tests that did not incur autocatalytic oxidation. The primary difference between run 9573 and the other FLECHT Zircaloy tests was that run 9573 had the lowest flood rate (see Appendix C Table B-1. Group III Test Results). "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 rate was about one inch per second."[229]

It would be reasonable to postulate that if run 9573 were repeated—with the same or a lower coolant flood rate, yet with lower initial cladding temperatures (that in the

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[227] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[228] *Id.*, p. 5-4.
[229] 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."

event of a LOCA, would occur at the beginning of reflood at current and/or proposed PWRs) and a lower power level (within the operational range of current and/or proposed PWRs)—that the fuel assembly would still incur autocatalytic oxidation, because FLECHT run 9573 had the lowest flood rate of the four Zircaloy tests.

FLECHT run 9573 demonstrates that the metal-water reaction becomes autocatalytic at temperatures lower than what the Baker-Just and Cathcart-Pawel equations predict. Westinghouse stated that run 9573 incurred autocatalytic oxidation at a temperature greater than 2300°F[230] (most likely, meaning at a temperature below 2400°F); the Baker-Just and Cathcart-Pawel equations predict that autocatalytic oxidation of Zircaloy cladding occurs at approximately 2600°F and 2700°F, respectively.[231]

The results from FLECHT run 9573 also demonstrate that stainless steel cladding heat transfer coefficients are not always a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

IV. PROPOSED ACTIONS

Petitioner requests that the NRC revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments.

Petitioner also requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models, Sources of Heat during the LOCA, Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments. 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.[232]

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[230] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.

[231] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

[232] Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.

Additionally, Petitioner requests that the NRC make a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA.

V. RATIONALE FOR THE NEEDED CHANGES

Petitioner requests that the NRC revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments, because such data indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

During the LOFT LP-FP-2 experiment, when peak cladding temperatures reached between approximately 2060°F[233] and 2240°F,[234] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec;[235] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[236]

And the CORA experiments demonstrated that with good fuel assembly insulation—like what the core of a nuclear power plant has—that cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, starts when the cladding reaches between 2012°F and 2192°F; cladding temperatures then start increasing at a maximum rate of 27°F/sec.[237]

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[233] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML062840091, pp. 30, 33.

[234] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[235] *Id.*

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

[237] 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

Petitioner requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5) to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments, because such data indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic 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.

As mentioned above, the LOFT LP-FP-2 experiment and CORA experiments demonstrated that the autocatalytic oxidation reaction of Zircaloy cladding occurs at temperatures far below what the Baker-Just and Cathcart-Pawel equations predict.

Petitioner requests that the NRC make a new regulation stipulating minimum allowable core reflood rates, in the event of a LOCA, because it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

The NRU Thermal-Hydraulic Experiment 1 ("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) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230460, p. 83.

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 reached temperatures exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

It is significant that, in the event of a LOCA, if reflood rates of approximately 1 in./sec. or lower did not prevent peak cladding temperatures from increasing to between approximately 2060°F[238] and 2240°F,[239] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[240]

Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[241] the melting point of Zircaloy is approximately 3308°F.[242]

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

[239] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[240] *Id.*

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

[242] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

VI. CONCLUSION

Discussing an estimate—in 1988 dollars—of the total amount of money spent on ECCS performance research between 1974 and 1988, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> In the years following the rulemaking [issued in January 1974], over $700 [million] has been spent by the NRC on research investigating ECCS performance. It is estimated that a similar amount has been spent by DOE (including AEC and ERDA), the U.S. industry, and foreign researchers, resulting in a total estimated expenditure of over $1.5 billion. The majority of this LOCA research is complete and has greatly improved the understanding of ECCS performance during a LOCA.[243]

Clearly, since 1988, substantial additional amounts of money have been spent on continuing LOCA research. So—in 2009 dollars—billions of dollars have been spent on LOCA research, yet the NRC has ignored the data from LOCA research experiments that indicates that some of its regulations are not conservative enough to help ensure public safety.

First, the NRC has ignored 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. For example, in NRU Thermal-Hydraulic Experiment 1, test no. 127, with a reflood rate of 1.0 in./sec., had a peak clad temperature at the start of reflood of 966°F and an overall peak clad temperature of 1991°F (an increase of 1025°F) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

It is noteworthy that 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)."[244]

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[243] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 8-1.

[244] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, p. 19.

Second, the NRC has ignored data from multi-rod (assembly) severe fuel damage experiments that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. The NRC has also ignored data from such experiments that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

"Compendium of ECCS Research for Realistic LOCA Analysis," states that "[a]ssessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature;"[245] and that "even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists."[246] However, "Compendium of ECCS Research for Realistic LOCA Analysis" does not mention that, during the LOFT LP-FP-2 experiment, autocatalytic oxidation occurred at cladding temperatures greater than either 2060°F[247] or 2240°F.[248]

The LOFT LP-FP-2 experiment was the most realistic severe fuel damage experiment that was conducted, so its temperature excursion data is very important for illustrating what, with high probability, would occur in the event of a LOCA at a PWR, if cladding temperatures were to reach between approximately 2060°F and 2240°F. The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[249] The LOFT LP-FP-2 experiment had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy

[245] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[246] *Id.*
[247] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[248] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.
[249] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.

1.67 meter fuel rods; it was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud.[250]

So the LOFT LP-FP-2 experiment was conducted with a well-insulated test assembly. This is significant, because the CORA experiments demonstrated that "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation;"[251] and that with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between 2012°F and 2192°F, and that then cladding temperatures start increasing at a maximum rate of 27°F/sec; "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[252]

The LOFT LP-FP-2 experiment was the only experiment that combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[253]

And regarding the value of the data from the LOFT LP-FP-2 experiment, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states:

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

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250 *Id.*

251 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, p. 83.

252 F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

253 T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, "Degraded Core Quench: A Status Report," p. 13.

> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[254]

It is noteworthy that, in 1985, the same year the LOFT LP-FP-2 experiment demonstrated that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures within the range of approximately 140°F below to 40°F above the 10 C.F.R. § 50.46(b)(1) PCT limit, "the [NRC] ruled by fiat in its Severe Accident Policy Statement that 'existing plants pose no undue risk to health and safety' and that no regulatory changes were required to reduce severe accident risk."[255]

It is also noteworthy that in 1983—five years before the NRC issued the regulations in Regulatory Guide 1.157, the best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50—the NRC, "[i]n recognition of the known conservatisms in Appendix K, ...adopted an interim approach..., described in SECY-83-472, *to accommodate industry requests* for improved evaluation models *for the purpose of reducing reactor operating restrictions*. This interim approach was a step in the direction of basing licensing decisions on realistic calculations of plant behavior" [emphasis added].[256]

So in 1983, the same year that the PBF Severe Fuel Damage 1-1 Test, according to some reports, had an onset of autocatalytic oxidation of Zircaloy cladding at approximately 2420°F[257] (the Baker-Just equation predicts that it occurs at approximately 2600°F[258]), and had results where a "rapid temperature rise in the bundle began near the center at the 0.5 to 0.7 [meter] elevation, and then spread radially outward and axially

[254] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3. 23.
[255] Edwin S. Lyman, Union of Concerned Scientists, "Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack at the Indian Point Nuclear Plant," 2004, p. 20.
[256] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, p. 2.
[257] Ken Muramatsu, Fumiya Tanabe, Tohru Suda, "Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code," p. 959; this paper cites P. E. MacDonald, *et al.*, Proceedings from the 5th International Meeting on Thermal Reactor Safety, Karlsruhe, 1984, p. 876, as the source of this information.
[258] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

downward in a manner similar to a flame front propagation,"[259] the NRC adopted new ECCS evaluation models, described in SECY-83-472, "to accommodate industry requests [to reduce] reactor operating restrictions."[260]

Additionally, 1983 was three years after the NRU Thermal-Hydraulic Experiment 1 tests indicated 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.

In 1988, the NRC continued to ignore data from severe fuel damage experiments that indicates that the PCT limit of 2200°F is non-conservative; it issued the regulations in Regulatory Guide 1.157 authorizing that, for postulated LOCAs, "[t]he rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam [could] be calculated in a best-estimate manner;"[261] *i.e.*, with the Cathcart-Pawel equation.[262] The Cathcart-Pawel equation is even more non-conservative, for calculating the metal-water reaction rates that would occur in the event of a LOCA, than the Baker-Just equation (required by Appendix K to Part 50 I(A)(5)); *e.g.*, the Cathcart-Pawel equation predicts that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures of approximately 2700°F; the Baker-Just equation predicts that it occurs at cladding temperatures of approximately 2600°F.[263]

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[259] Ken Muramatsu, Fumiya Tanabe, Tohru Suda, "Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code," p. 960; this paper cites Proceedings from the 5th International Meeting on Thermal Reactor Safety and P. E. MacDonald, *et al.*, American Nuclear Society Transcript, 46, 478, 1984, as the source of this information.

[260] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, p. 2.

[261] *Id.*, p. 6.

[262] NRC, Regulatory Guide 1.157, p. 6, states that "[t]he data of ["Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies"] are considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F;" J. V. Cathcart et al., "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977.

[263] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

Regulatory Guide 1.157 states that "the terms 'best-estimate' and 'realistic' have the same meaning."[264] And regarding best-estimate calculations, Regulatory Guide 1.157 states:

> A best-estimate calculation uses modeling that attempts to realistically describe the physical processes occurring in a nuclear reactor. There is no unique approach to the extremely complex modeling of these processes. The NRC has developed and assessed several best-estimate advanced thermal-hydraulic transient codes. These include TRAC-PWR, TRAC-BWR, RELAP-5, COBRA, and the FRAP series of codes... These codes reasonably predict the major phenomena observed over a broad range of thermal-hydraulic and fuel tests. ...
>
> A best-estimate model should provide a realistic calculation of the important parameters associated with a particular phenomenon to the degree practical with the currently available data and knowledge of the phenomenon. *The model should be compared with applicable experimental data* and should predict the mean of the data, rather than providing a bound to the data. ...
>
> A best-estimate code contains all the models necessary to predict the important phenomena that might occur during a loss-of-coolant accident. *Best-estimate code calculations should be compared with applicable experimental data* (*e.g.*, separate-effects tests and integral simulations of loss-of-coolant accidents) to determine the overall uncertainty and biases of the calculation. In addition to providing input to the uncertainty evaluation, *integral simulation data comparisons should be used to ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted* [emphasis added].[265]

So a best-estimate ECCS evaluation calculation is supposed to "be compared with applicable experimental data"[266] and "ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted,"[267] yet the Cathcart-Pawel equation—used in best-estimate ECCS evaluation calculations—is non-conservative—as indicated by data from multi-rod (assembly) severe fuel damage experiments—for calculating the metal-water reaction rates that would occur in the event of a LOCA.

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[264] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," p. 1, footnote 1.
[265] *Id.*, p. 3.
[266] *Id.*
[267] *Id.*

It is significant that Regulatory Guide 1.157 states:

> On September 16, 1988, the NRC staff amended the requirements of § 50.46 and Appendix K, "ECCS Evaluation Models" (53 FR 35996), so that these regulations *reflect the improved understanding of ECCS performance during reactor transients that was obtained through the extensive research performed since the promulgation of the original requirements in January 1974.* Paragraph 50.46(a)(1) now permits licensees or applicants to use either Appendix K features or a realistic evaluation model. *These realistic evaluation models must include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during a postulated loss-of-coolant accident.* 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits in paragraph 50.46(b) so that there is *a high probability that the criteria will not be exceeded* [emphasis added].[268]

First, the NRC may indeed have an "improved understanding of ECCS performance during reactor transients[,] obtained through...extensive research,"[269] yet it continues to ignore data from multi-rod (assembly) severe fuel damage experiments that, among other things, indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

Regulatory Guide 1.157 also states that " 'Compendium of ECCS Research for Realistic LOCA Analysis' (NUREG-1230), provides a summary of the large experimental database available, upon which best-estimate models may be based."[270] Indeed, "Compendium of ECCS Research for Realistic LOCA Analysis" does provide "a summary of the large experimental database available;"[271] however, among other things, it omits important experimental data regarding the cladding temperatures at which autocatalytic oxidation occurred during severe fuel damage experiments, like the LOFT LP-FP-2 experiment.

Second, the NRC certainly does not have "evaluation models [that] include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during...postulated loss-of-

268 *Id.*, p. 1.
269 *Id.*
270 *Id.*, p. 4.
271 *Id.*

coolant accident[s]."[272] For example, the Cathcart-Pawel equation predicts that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures of approximately 2700°F,[273] yet the LOFT LP-FP-2 experiment demonstrated that autocatalytic oxidation occurs at cladding temperatures of approximately 2060°F or 2240°F.

Third, there is not "a high probability that the criteria [of 10 C.F.R. § 50.46(b) would] not be exceeded,"[274] in the event of a LOCA.

For example, in the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[275] and 2240°F,[276] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[277] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[278] the melting point of Zircaloy is approximately 3308°F.[279]

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

[273] According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K."

[274] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," p. 1.

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

[276] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, V. T. Berta, and M. S. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989, as the source of this information.

[277] *Id.*

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

[279] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

It is noteworthy that when the AEC enacted its ECCS acceptance criteria for LWRs it did not consider that, in the event of a LOCA, the autocatalytic oxidation of Zircaloy cladding could occur at temperatures below 2498°F. Regarding this issue, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35" states:

> In the 1966-1967 time frame, research results indicated that zircaloy cladding exposed to LOCA-like conditions with peak temperatures in the vicinity of 1370°C (well below the zircaloy melting point of 1820°C) embrittled and ruptured, or even shattered upon cooldown. This threatened the integrity of the core geometry, which, in turn, was perceived to threaten core coolability. Therefore, *instead of the criterion of no (or very little) clad melt, which was based in part on the concern over the autocatalytic effect on zirconium oxidation*, and which had been proposed by the nuclear steam supply system (NSSS) vendors and accepted for some months, *a much lower limit on the highest acceptable clad temperature during a LOCA was indicated, somewhere between 1204°C and 1370°C (2200°F to 2498°F).* In 1971, the AEC issued a policy statement containing interim acceptance criteria for ECCS for light water reactors [emphasis added].[280, 281]

(The AEC's interim ECCS acceptance criteria for LWRs stipulated that in the event of a LOCA, the maximum allowable cladding temperature would be 2300°F; after the rulemaking hearings that began in January 1972, the AEC changed this temperature limit to 2200°F.[282])

So the AEC based its regulation for the maximum allowable cladding temperature, in the event of a LOCA, on the premise of preventing severe cladding embrittlement and/or preventing the cladding from shattering upon cooldown. "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[o]ur selection of the 2200°F limit results primarily from our belief that retention of ductility in the zircaloy is the best guarantee of its remaining intact during the hypothetical

[280] The AEC's interim ECCS acceptance criteria for LWRs is "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors—Interim Policy Statement," U.S. Federal Register, Vol. 36, No. 125, June 29, 1971 and No. 244, December 18, 1971.
[281] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," p. 3-1.
[282] *Id.*

LOCA;"[283] and that "[t]he limits specified in these criteria will assure that some ductility would remain in the zircaloy cladding as it goes through the quenching process, and therefore that the core would remain essentially intact, in a condition amenable to long-term cooling."[284]

Regarding the maximum allowable cladding temperature limit in the event of a LOCA, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" also states:

> None of the reactor manufactures agreed with the Staff's proposed stipulation of a 2200°F maximum calculated temperature... Westinghouse proposed a maximum calculated temperature limit of at least 2700°F; Combustion Engineering and the Utility Group agreed on 2500°F as the peak allowable calculated temperature on the basis that much of the data on oxidation and its effects stops at 2500°F. Babcock and Wilcox suggested a more conservative 2400°F as the peak calculated temperature to be allowed, presumably because *"significant eutectic reaction and an excessive metal-to-water reaction rate would be precluded below 2400°F."* ... General Electric argued strongly that the limit should not be reduced to 2200°F; that 2700°F is really all right as far as embrittlement is concerned, but that the Interim Acceptance Criterion value of 2300°F should be retained. In addition to being consistent with their expressed desire not to change any of the criteria, the GE recommendation of retaining the 2300°F limit is intended *to ensure that the core never "gets into regions where the metal-water reaction becomes a serious concern"* [emphasis added].[285]

It is interesting that Babcock and Wilcox suggested a PCT limit of 2400°F, based on the premise of avoiding temperatures where the metal-water reaction would become excessive and that General Electric thought the interim PCT limit of 2300°F should be retained "to ensure that the core never 'gets into regions where the metal-water reaction becomes a serious concern.' "[286]

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[283] 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. 1098. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."
[284] *Id.*, p. 1096.
[285] *Id.*, p. 1097.
[286] *Id.*

It is also noteworthy that in 2002, the NRC postulated that "with regard to runaway temperature escalation, the [10 C.F.R. § 50.46(b)(1) PCT limit] could be raised to 2300°F;"[287] regarding this issue the NRC stated:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F... Thus, *with regard to runaway temperature escalation, the peak cladding temperature could be raised to 2300°F* without affecting this sensitivity and without reducing the margin that the Commission would have perceived in 1973 [emphasis added].[288]

So the NRC has continued to ignore data from multi-rod (assembly) severe fuel damage experiments that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. In other words, the NRC has ignored experimental data that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit should be based on the premise of preventing the autocatalytic oxidation of Zircaloy cladding, at a limit below 2200°F.

Regarding best-estimate ECCS evaluation calculations and safety issues, Regulatory Guide 1.157 states:

> It was also found that some plants were being restricted in operating flexibility by limits resulting from conservative Appendix K requirements. Based on the research performed, it was determined that these restrictions could be relaxed through the use of more realistic calculations *without adversely affecting safety*. ...
>
> *Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations* [emphasis added].[289]

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

[288] *Id.*

[289] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," p. 2.

Indeed, safety would be best served if decisions concerning the limits within which nuclear reactors are permitted to operate were actually based on realistic calculations. For example, realistic ECCS evaluation calculations of the metal-water reaction rates would be based on data from multi-rod (assembly) severe fuel damage experiments that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

It is significant that in 2005, in the NRC's report on its denial of a petition for rulemaking—PRM-50-76—that argued that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA, the NRC stated:

> No data or evidence was...found in NRC records to suggest that the research, calculation methods, or data used to support ECCS performance evaluations were sufficiently flawed so as to create significant safety problems. NRC's technical safety analysis demonstrates that current procedures for evaluating performance of ECCS are based on sound science and that no amendments to the NRC's regulations and guidance documents are necessary. ...the NRC [has not] found, the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material.[290]

So the NRC was unable to locate data in NRC records from multi-rod (assembly) severe fuel damage experiments that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. And the NRC was unable to perceive "the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material."[291] For example, the NRC was unable to locate data in NRC records from the LOFT LP-FP-2 experiment that indicates that an autocatalytic oxidation reaction of Zircaloy cladding occurred at a temperature hundreds of degrees Fahrenheit below what either the Baker-Just or Cathcart-Pawel equations would predict.

Clearly, the NRC has failed to uphold its congressional mandate to protect the lives, property, and environment of the people of the United States of America.

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[290] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 23.
[291] *Id.*

If implemented, the regulations proposed in this petition for rulemaking would help improve public and plant-worker safety.

Respectfully submitted,

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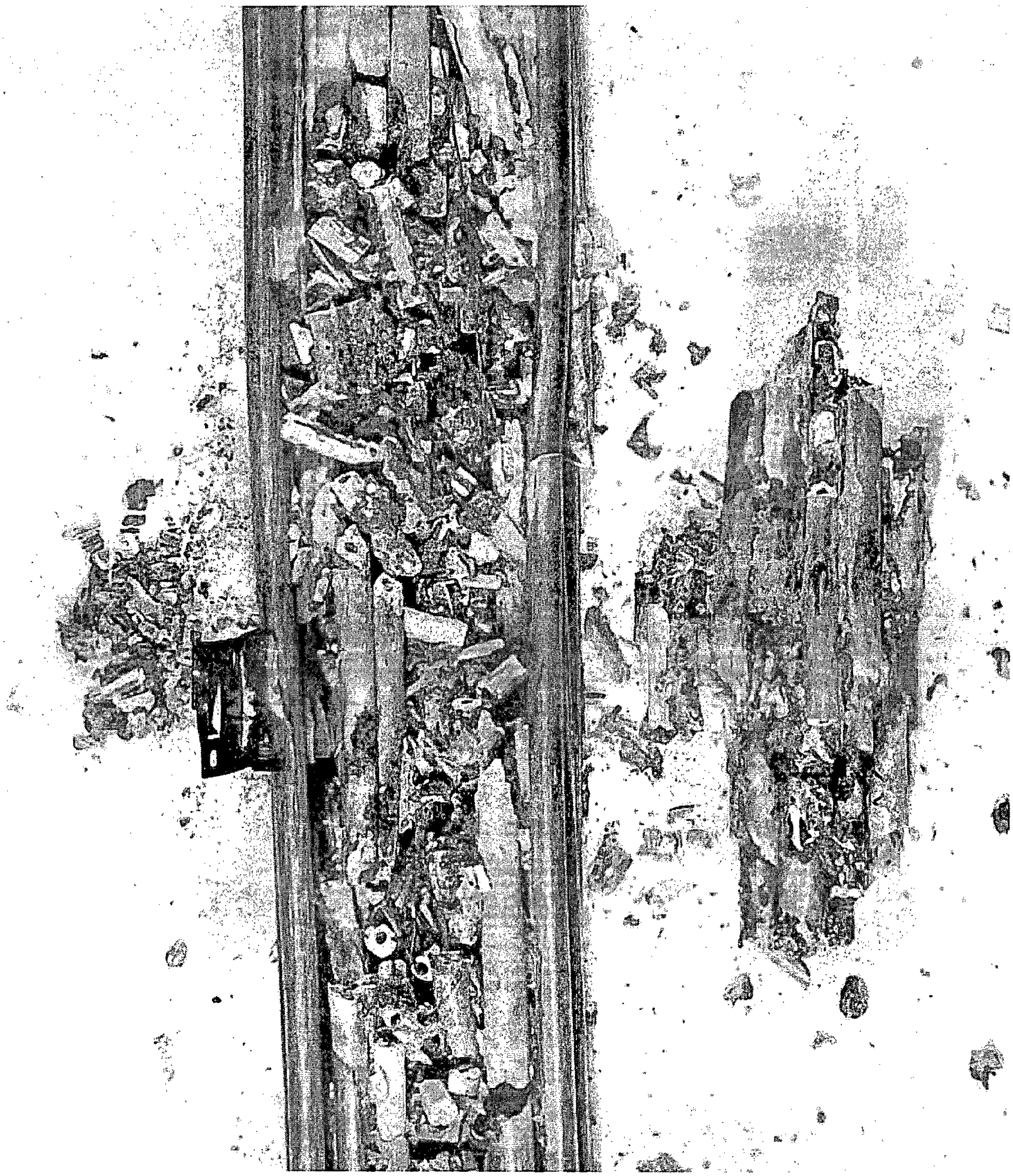
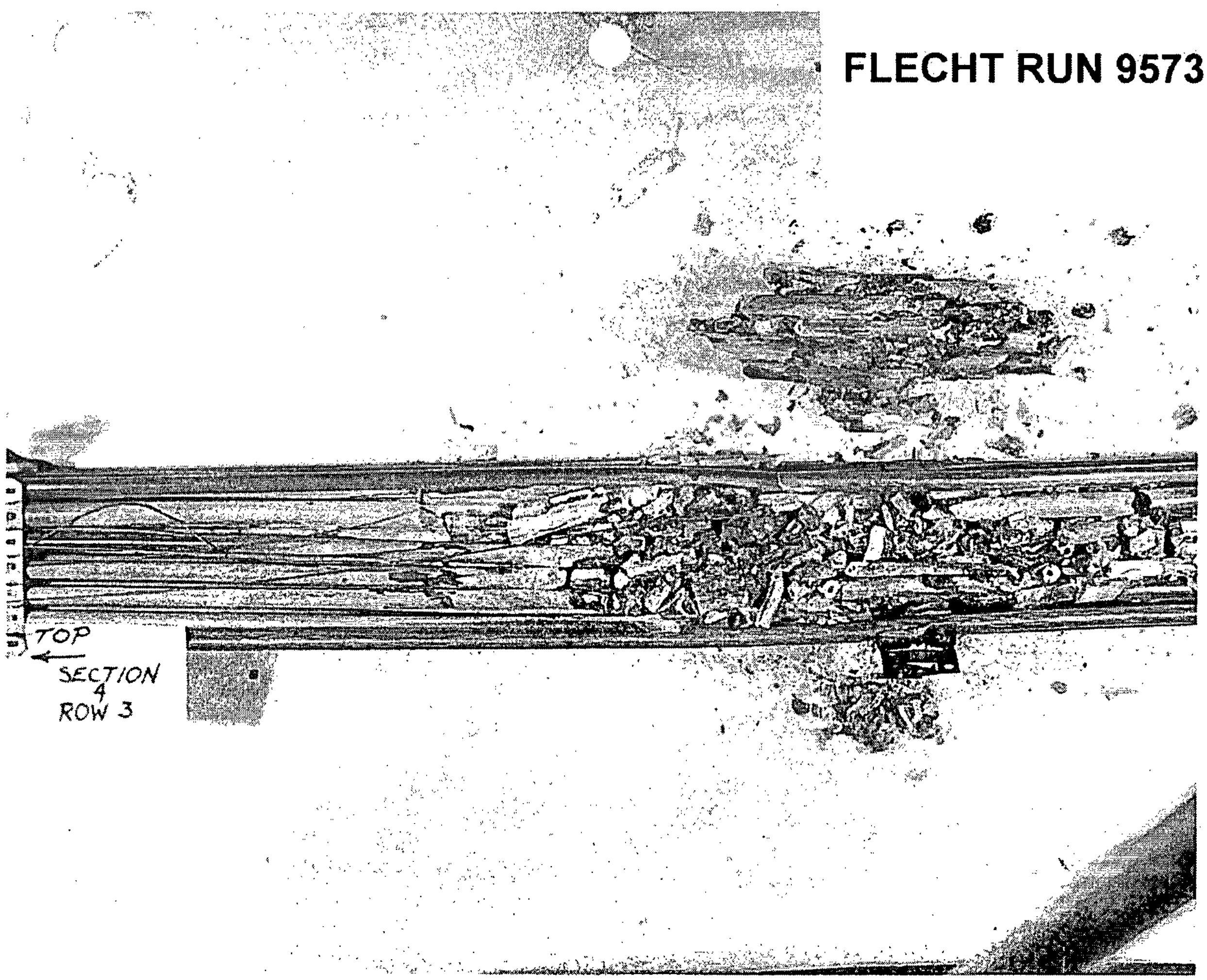
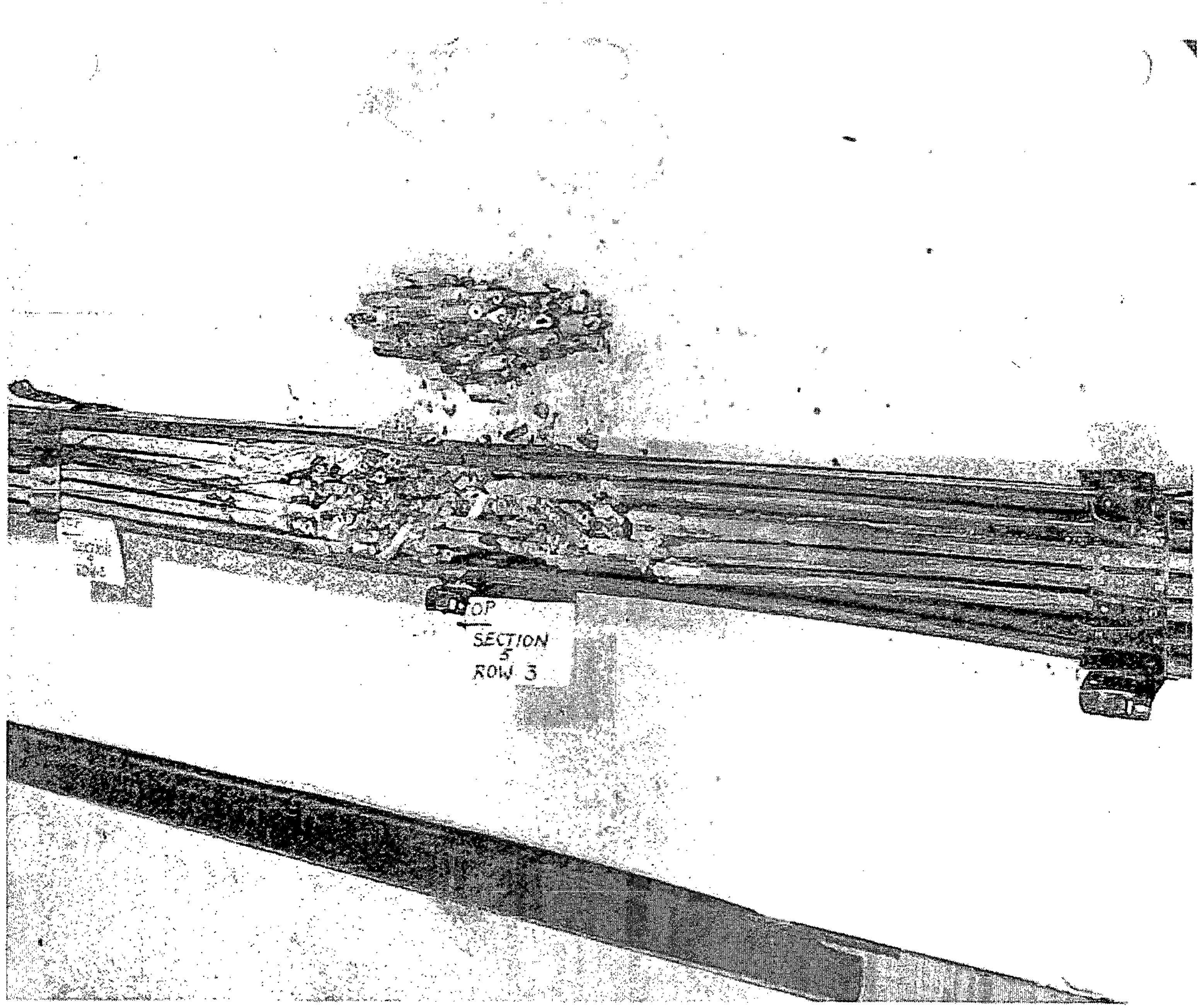
Dated: November 17, 2009

Appendix A Photographs of the Assembly from FLECHT Run 9573

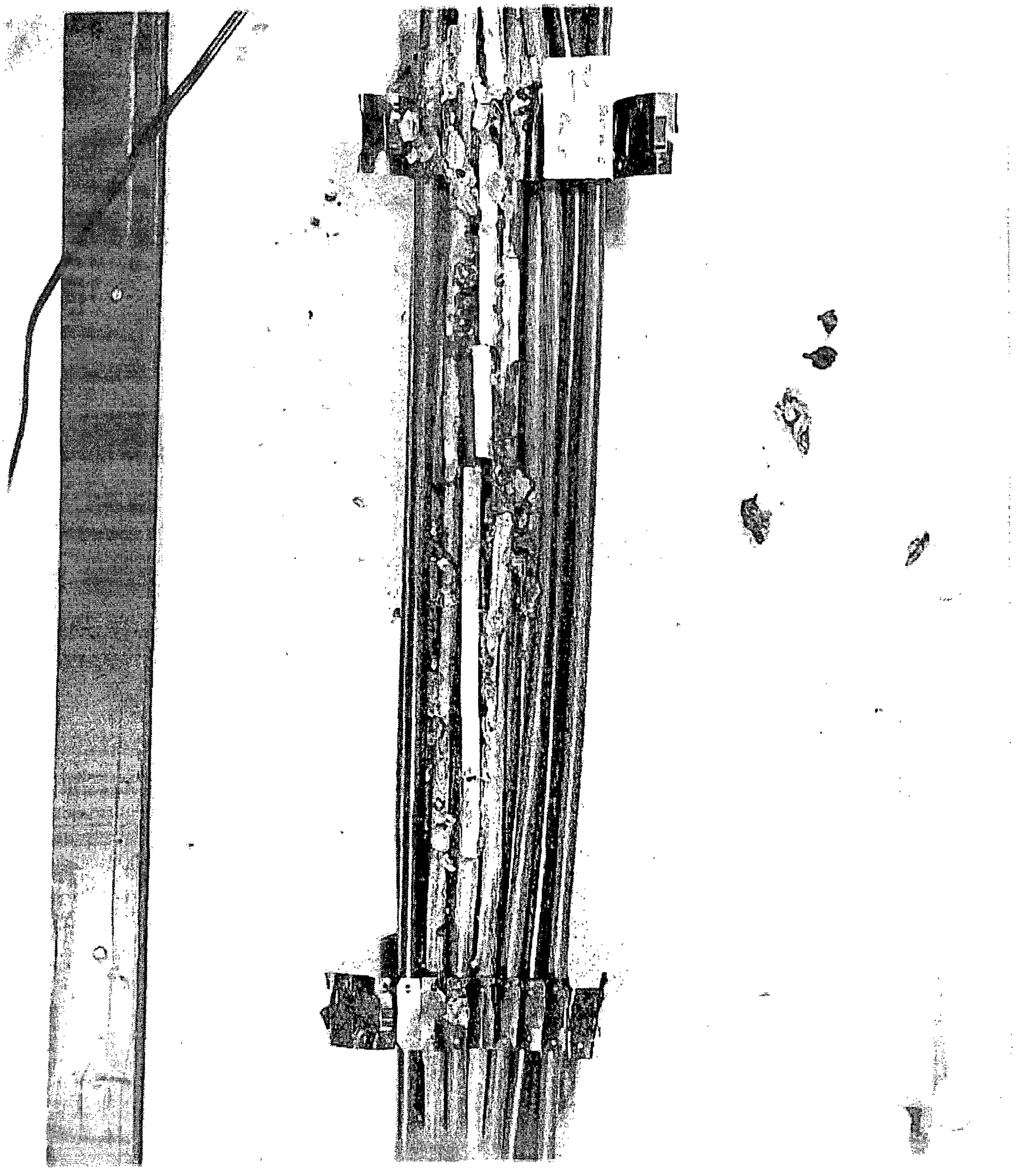


FLECHT RUN 9573




OP
SECTION
5
ROW 3

Appendix B Photograph of the Assembly from FLECHT Run 8874



Appendix C Table B-1. Group III Test Results (The Four FLECHT Zircaloy Tests)[1]

[1] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. B-2.

B.2 SPECIMEN SELECTION, PREPARATION, & EXAMINATION

Actual test conditions and transient temperature data for the Four Group III tests are presented in Table B-1. The transient temperature data reported was obtained from the midplane thermocouple (six foot elevation) on the hottest rod. The turnaround time reported (t_{turn}) represents the elapsed time from the start of flooding to the time the peak heater rod temperature (T_{peak}) is reached.

TABLE B-1

GROUP III TEST RESULTS

	"Early" Group III			
Run Number	2443	2544	8874[a]	9573
Initial temperature (°F)	2035	2017	2297	1970
Flow rate (in./sec)	10.0	4.0	6.0 (for 8 sec)-1.0	1.1
Peak Power (kw/ft)	1.24	1.24	1.24	1.24
Inlet temperature (°F)	150	150	141	140
Pressure (psia)	56	58	64	61
Peak heater rod temperature	2102	2144	2361	2320[b]
Turnaround time (sec)	6	12	4	--

[a] with fallback

[b] at 18 seconds

As can be seen from Table B-1, the peak temperatures for the two "early" Group III tests were only 42°F apart. Due to the similarity in peak temperatures for these two runs it was decided to concentrate the metallographic examination on the bundle used for Run 2443 (Zircaloy Bundle No. 1) and to take only a limited number of samples from the bundle used for Run 2544 (Zircaloy Bundle No. 2). Thirteen specimens were therefore taken from Bundle No. 1 and two from Bundle No. 2.

Appendix D Table 1. Experimental Heat Cladding Temperatures (The 28 Tests from Thermal-Hydraulic Experiment 1)[2]

[2] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119, p. 13.

TABLE 1. Experimental Heat Cladding Temperatures

TEST NO.	REFLOOD RATE IN/SEC	DELAY TIME SEC	PEAK CLAD TEMP AT START OF		PEAK CLAD TEMP AT TURNAROUND		
					MEASURED	PREDICTED	
			TRANSIENT DEG F	REFLOOD DEG F	DEG F	FLECHT-TRUMP DEG F	THERM DEG F
101	3.8	28 (1)	871	881	1403	1350	1365
104	3.8	37	853	1336	1487	1400	1445
105	1.9	7	858	907	1364	1400	1370
106	10.5 (2)	19	873	1101	1223	1100	1150
107	1.9	19	891	1154	1578	1500	1420
108	1.4	11	891	1010	1676	1700	1500
109	1.3	22	865	1158	1881	1800	1580
110	1.9	30	895	1314	1665	1600	1525
111	1.4 (3)	11	817	962	1696	1700	1500
112	3.8	37	843	1330	1589	1400	1425
113	7.6	37	845	1408	1526	1400	1395
114	7.6	32	858	1368	1477	1300	1300
115	9.5	66	795	1666	1758	1800	1720
116	3.8	51	836	1500	1707	1600	1605
117	3.8	66	817	1599	1788	1800	1800
118	2.9	52	844	1480	1756	1700	1675
119	2.9	46	862	1451	1673	1600	1620
120	5.9	51	847	1460	1611	1600	1580
121	3.8	36	833	1304	1579	1400	1425
122	7.6	52	866	1486	1611	1600	1575
123	2.9	51	848	1532	1788	1700	1675
124	5.9	52	861	1556	1688	1600	1580
125	1.4	20	872	1138	1802	1800	1565
126	1.2	3	797	800	1644	1700	1530
127	1.0	3	943	966	1991	1900	1650
128	2.0	50	911	1604	1991	1800	1735
129	1.4	32	940	1371	1898	1900	1670
130	0.7 (4)	5	929	998	2040		

(1) Unplanned delay caused by problems in prefill
(2) Malfunctioning equipment caused greater reflood rate than planned
(3) 1st two seconds of data missing
(4) Reactor tripped at ~1850 ^{o}F

Appendix E Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region[3] (Graphs of Cladding Temperature Values During the FLHT-1 Test)[4]

[3] Pseudo sensor readings are the averages of the readings of two or more thermocouples.

[4] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, pp. 4.7, 5.3.

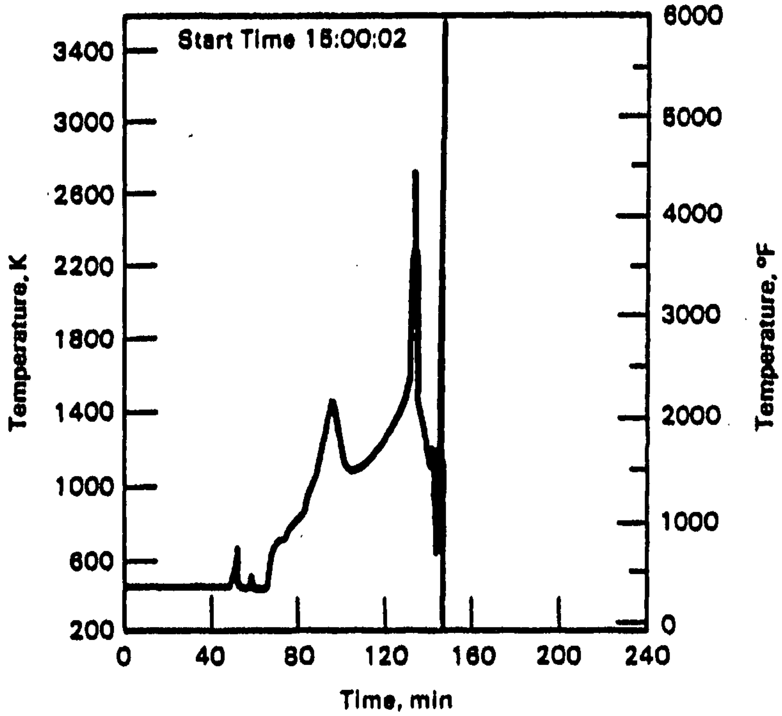


FIGURE 4.1. Typical Cladding Temperature Behavior

reached 10% of the initial power approximately 35 s later and reached low neutron level in another 30 s.

There were two indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.

A review of the thermocouple data led to the conclusion that the temperatures were not rising after the reactor shutdown. Typical cladding, coolant, and liner temperatures immediately after the reactor shutdown are shown in Figures 4.2, 4.3, and 4.4, starting at 17:12:00. The temperatures shown are somewhat erratic and show noise (probably associated with some thermocouple damage), but the general trend is downward, indicating an effective shutdown.

Additional indications of an effective test shutdown are shown by the saddle temperature, MMPD response, and bypass coolant power (radial heat loss) after the reactor power shutdown. Typical data from these sources are shown in Figures 4.5 through 4.7. All three of these indicators show steadily decreasing temperatures. Table 4.3 is a summary of the events of the FLHT-1 test.

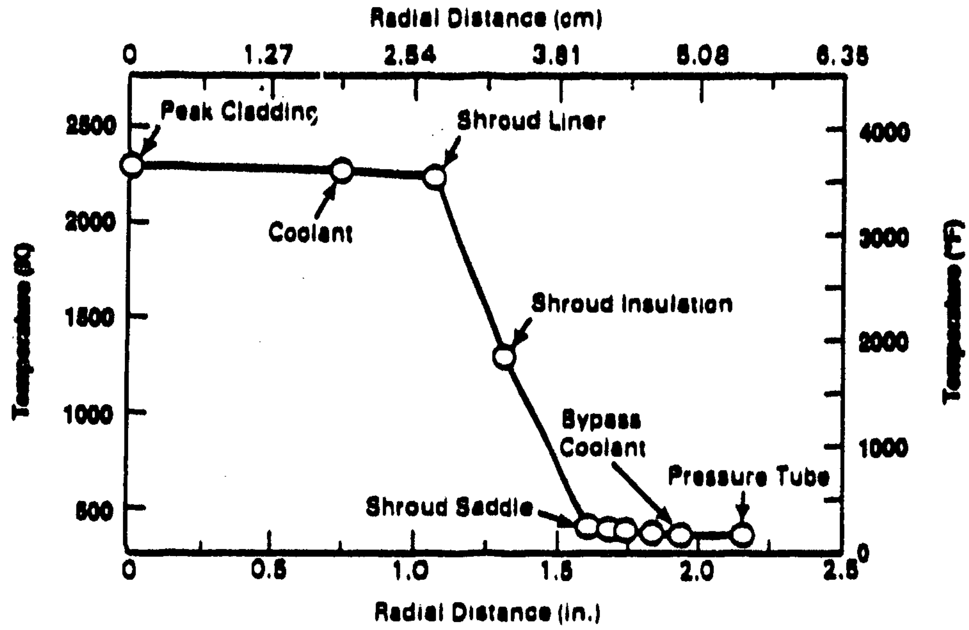


FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

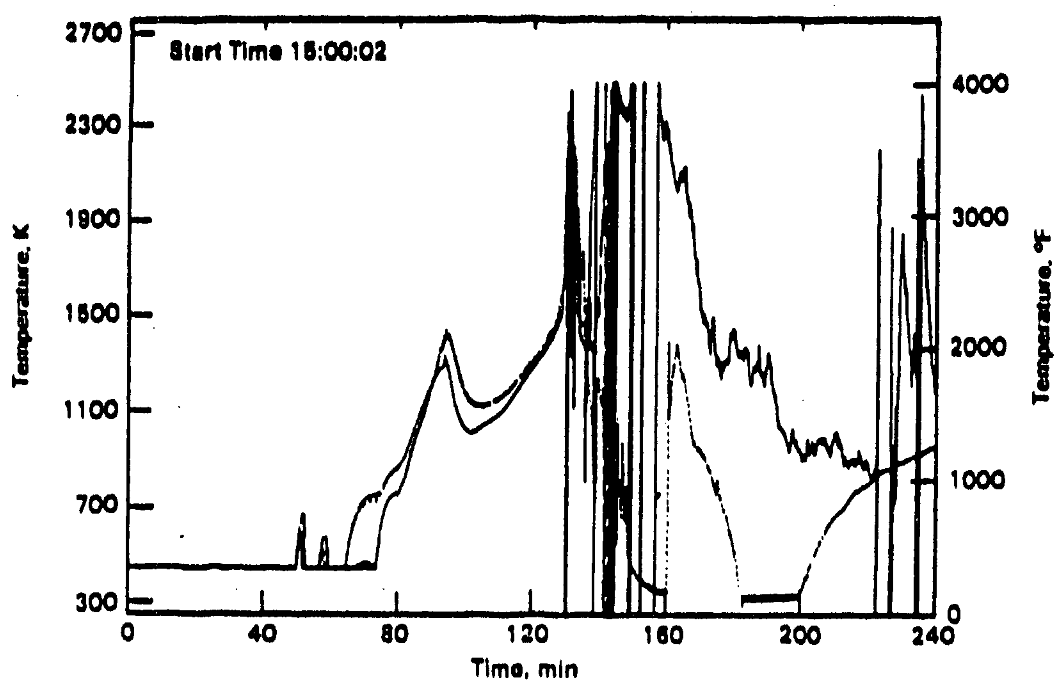


FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

Appendix F Figure 3.7. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 and Figure 3.10. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 with Saturation Temperature (Graphs of Cladding Temperature Values During the LOFT LP-FP-2 Experiment)[5]

[5] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML062840091, pp. 34, 35.

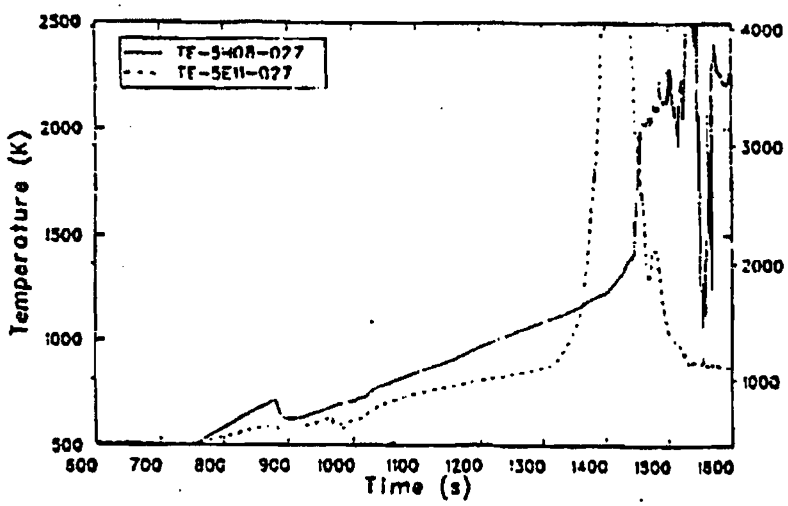


Figure 3.7 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5.

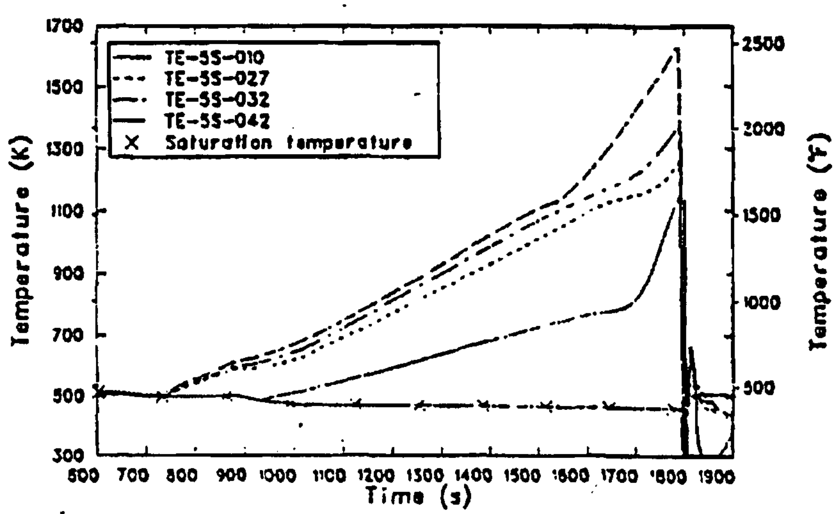


Figure 3.8 Comparison of four external wall temperatures at the 1.07-, 0.81-, 0.69-, and 0.25-m (42-, 32-, 27-, and 10-in.) elevations on the south side of the flow shroud.

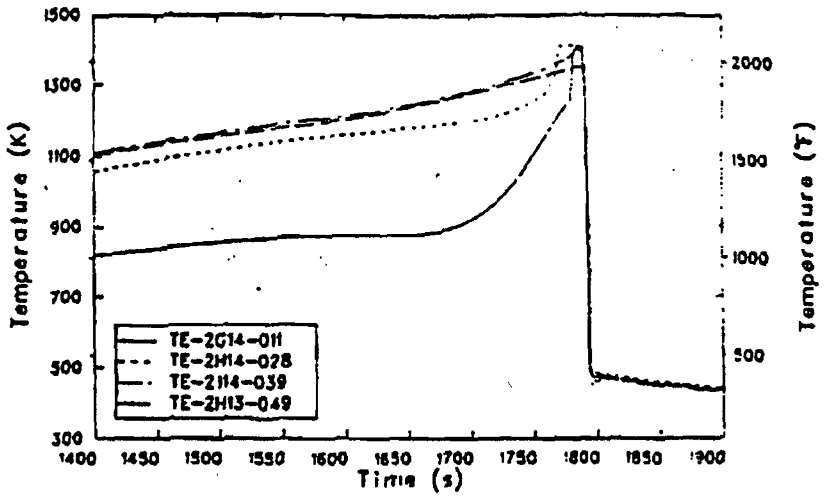


Figure 3.9 Comparison of cladding temperatures at the 1.24-, 0.99-, 0.71-, and 0.28-m (49-, 39-, 28-, and 11-in.) elevations in Fuel Assembly 2.

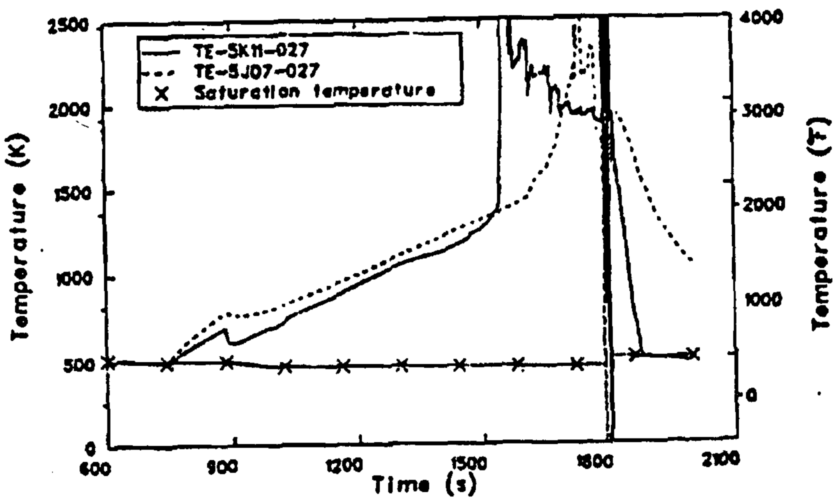


Figure 3.10 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5 with saturation temperature.

Appendix G Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF)[6]

[6] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230126, p. 312.

allow prediction of such an escalation. A solid debris bed was formed due to the rapid cooldown (10 K/s). These data are valuable to define general criteria for a loose rubble bed formation.

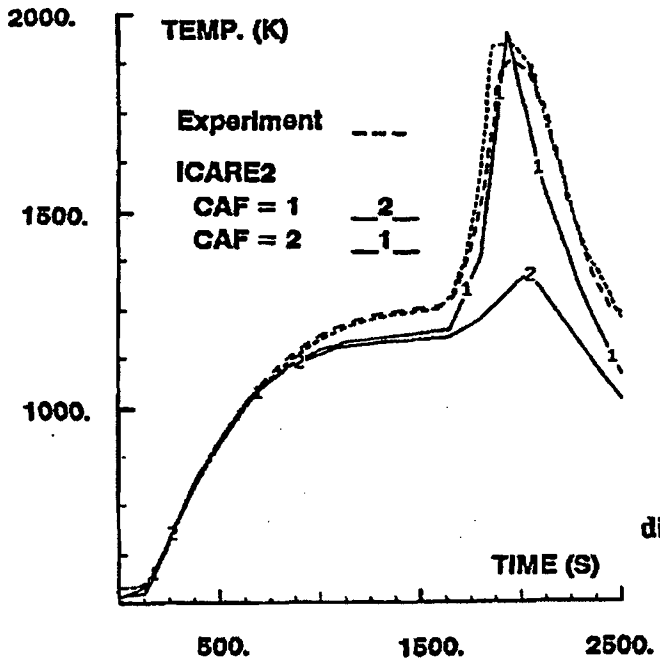


Fig.1 :
Sensitivity calculation on the B9R test. Temperature escalation at the hot level (0.6 m) with different Contact Area Factors (CAF)

3.2.2 PHEBUS C3 + test

The main objective of this test was to study UO_2 dissolution by chemical interaction with solid Zr in a first stage and with liquid Zr in a second stage in the case of limited cladding oxidation. The first low temperature oxidation phase was performed during 3000 s with pure steam at 0.6 MPa so as to reach a low cladding oxidation level. The second 11000 s phase long was performed in pure He at 3.5 MPa so as to obtain good UO_2-Zr contact inside the non-pressurized rods. The heat-up of the bundle was driven by several power step increases.

After adjusting the shroud heat losses in the first steam phase (see next section), the calculated and measured inner fuel rod temperatures at the 0.10, 0.40 and 0.60 m elevations agree well, until the thermocouple failures shown in Fig. 2 by arrows. Above 2200 K the calculation agrees with the fuel thermal behaviour estimated from the shroud measurements and PIEs. The calculated oxidation profile is shown in Fig. 3. A maximum of 18 % mean oxidation is predicted at the hot point (0.6 m from the bottom of the active length). The PIEs confirm a low level of oxidation but no significant measurement was performed due to the complete disappearance and relocation of the cladding between 0.05 and 0.60 m.

Fig. 4 shows two calculations of the UO_2 dissolution. In the two cases the first stage of the UO_2 dissolution by "Solid" Zr is calculated with the Hofmann (S) model but the second stage of UO_2 dissolution by "Molten" Zr is calculated in one case with the Kim model and in the other with the Hofmann (M) model. In these two cases the same UO_2 solubility limit

Appendix H Figure 2.1. Temperature Regimes for Extensive Liquid Phase Formation and Relocation[7]

[7] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, Figure. 2.1.

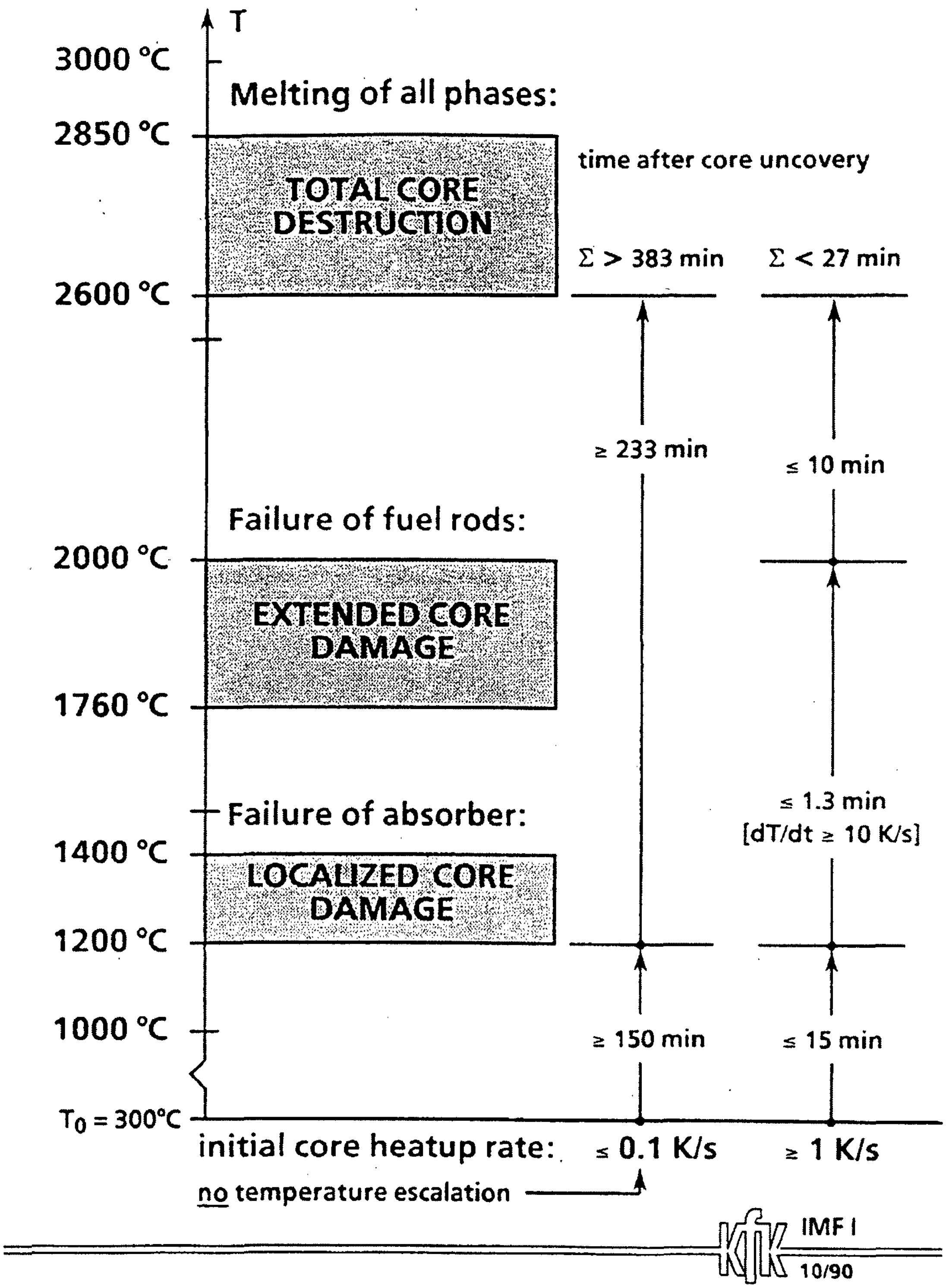


FIGURE 2.1: TEMPERATURE REGIMES FOR EXTENSIVE LIQUID PHASE FORMATION AND RELOCATION

Appendix I Memorandum RD-TE-70-616, FLECHT Monthly Report, December 14, 1970

RD-TE-70-616

From Nuclear Energy Systems
WIN
Date December 14, 1970
Subject FLECHT Monthly Report

PENN CENTER

H. A. Sindt
Development Projects

cc: L. S. Tong
F. F. Cadek
W. W. Spencer
R. L. Mason
A. S. Kitzes
J. F. Mellor

FLEC-500 - Facility Operation

The final FLECHT test (Bundle Z-10) was completed on December 11, 1970. The test was run with flooding of 1 in/sec. beginning at 2000°F. Several heaters failed approximately 18 seconds after flooding when the peak indicated midplane temperature was 2325°F. Heater failure at this temperature is unlikely, particularly under conditions of decay heat and increasing temperature. The steam probe thermocouple located one foot above midplane in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec.) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding. It appears likely that ignition of the Zircaloy grids led to high rates of heat input* at the elevation one foot above (and below) midplane and this caused overtemperature and failure of the heaters. Test results are currently being studied.

The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.

*The ratio of surface area to heat capacity for a Zircaloy grid is approximately 15 times that of a heater rod, hence Zircaloy-steam reactions can lead steeper temperature ramps in the vicinity of a Zircaloy grid.

R. H. Leyse

NUCLEAR ENERGY SYSTEMS
/bjl

Test Engineering

R. H. Leyse

Appendix J Memorandum RD-THD-17, FLECHT Technical Review Meeting Minutes No. 58, December 18, 1970

RD-THD-17

From: PWR SYSTEMS DIVISION
WIN: X-4720
Date: December 18, 1970
Subject: FLECHT Technical Review Meeting Minutes No. 58

NUCLEAR ENERGY SYSTEMS

H. A. Sindt
F. D. Kingsbury
A. S. Kitzes/R. H. Leyse
J. F. Mellor

cc: W. H. Arnold, Jr.
L. S. Tong
W. Rockenhauser
L. Chajson
J. O. Cermak
J. W. Dorrycott
K. R. Jordan
J. S. Moore
J. D. McAdoo

I. Results of Group III Zirc Run 9573

Preliminary results of the last Group III Zirc Run 9573 are summarized in the attachment. The run is considered valid up to the point of first heater failure at 18.2 sec. At least 12 heaters failed in a 5 second time span starting at 18.2 sec after flood. Typical 6 ft temperatures at which heaters failed were in the 2200 to 2300°F range. These are lower failure temperatures than anticipated (above 2400°F) and causes are being investigated. An indicated steam temperature greater than 2400°F at the 7 ft elevation prior to start of failures may be related to the phenomenon. The test conditions for this run were specified by INC and were not in agreement with W recommendations. We predicted failures would occur, but at approximately 10 to 20 sec later in the run.

II. Final Report Status and Plans

An outline has been prepared and effort has been initiated. Target is to publish my mid-April. A rough draft should be completed by the end of February. Materials evaluation input is scheduled to be received by the end of December. Heater rod development input is due the end of January.

III. Facility Inventory and Disassembly Plans

Facility inventory is planned for January. INC has been advised that subject to other W requirements the facility will be dismantled in May.

F. F. Cadek

F. F. Cadek, Manager
Thermal-Hydraulic Development

/jw

Attachment

FORM 15031

SUMMARY OF FINAL GROUP III ZIRC TEST

FLECHT RUN 9573

(PRELIMINARY RESULTS)

1. Run Conditions: Initial clad temperature - 1970°F

Flooding Rate	1.1 in/sec (constant)
Pressure	61 psia
Peak Power	1.24 kw/ft
Coolant Temperature	140°F
Clad Material	Zircaloy

2. Rod Failures - First rod failure occurred at 18.2 seconds after flood. Multiple failures occurred in the next 5 seconds. Post test inspection indicated all but 7 heater elements failed. Run is considered valid up to 18.2 seconds.

3. Bundle Power - Power trace indicates arcing started at time of first rod failure (18.2 sec). Power input to bundle due to arcing after 18.2 sec was about 10% greater than normal until power was cut off at 55.5 sec.

4. Typical Rod Temperatures

Elevation (ft)	T/C No.	$T_{initial}$ (°F)	Temp. at Time of Failure (°F)	Time of Failure (sec)
6 (Pen Recorder)	3D3	1970	2320	18.3
6 (VIDAR)	3C3	1892	2233	18.5
8 (VIDAR)	3C2	1679	1955	18.5

5. Steam Temperature Data - 7 ft steam temperature exceeded 2500°F at 16 sec (2 seconds prior to heater failure). 10 ft, 12 ft and outlet plenum temperatures were similar to earlier 1 in/sec stainless clad run.

6. External Thermocouples - (installed at INC's insistance) agreed very well with internal T/C's up to about 2000°F. Above this temperature all 5 external T/C's failed.

Appendix K Memorandum PA-TE-70-419, Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874), July 24, 1970

PA-TE-70-419

From Nuclear Energy Systems
WIN
Date July 24, 1970
Subject Higher Initial Heat Transfer Coefficients Zircaloy Bundle - Run 8874

J. O. Cermak
L. S. Tong
F. F. Cadek
H. A. Sindt
W. W. Spencer
A. S. Kitzes
R. L. Mason

The initial heat transfer coefficient is at least 1.7 times higher in a Zircaloy bundle (run 8874) than in a stainless bundle (run 6155) for the same flooding rate (6"/sec.) and start-of-flood temperatures of 2300°F and 2200°F respectively. The higher coefficients for the Zircaloy bundle may be explained by high hydrogen concentrations (20% or more) in the film at the surface of the heater. At 2000°F, the thermal conductivity of hydrogen is approximately five times that of superheated steam. Although hydrogen production rates are probably not sufficient to lead to significant concentrations in the bulk coolant (the mixture of superheated steam and water droplets), the hydrogen concentrations within the film at the surface of the heater can easily reach significant values.

R Leyse

NUCLEAR ENERGY SYSTEMS /bjl

Test Engineering

R. H. Leyse

Appendix L Memorandum RD-ED-THE-33, Report of Events Leading to FLECHT 10 x 10 Bundle Test, April 23, 1969

RD-ED-THE-33

From : PWR SYSTEMS DIVISION
WIN : Extension 4713
Date : April 23, 1969
Subject : Report of Events Leading to FLECHT 10 x 10 Bundle Test

FLEC-200

PWR SYSTEMS DIVISION

L. S. Tong, Manager
Engineering Development

cc: A. S. Kitzes
H. A. Sindt

The following is a summary of events relative to the FLECHT 10 x 10 bundle test on April 18, 1969.

I. Schedule for Testing

A. A meeting to review operational procedures was held at 1315. This meeting lasted until approximately 1600 because of discrepencies between the written procedure and past practices. Several alterations were made in the written procedure. The disagreements on test procedure were substantially more fundamental than one would expect to encounter during a final review meeting. In addition, it was not apparent to the observer that explicit assignment of operating responsibilities had been made prior to this meeting.

B. The test was originally scheduled for 1400 hrs. The test was performed at about 2300 hrs.

II. Operational Difficulties

The following operational difficulties were observed between 1800 and 2100 hrs. These factors caused the delay in running during this period.

A. The housing became overheated. It was visibly red (probably in the neighborhood of 1200°F).

B. Steam leakage occurred in the viewing ports during pressurization.

3511 B

C. The steam generator ran out of water and had to be refilled.

Subsequent to the leakage at the viewing ports the author suggested to H. Skreppen that continuance of the test be postponed until the following day. Preparation for the test continued and the test was performed at approximately 2300 hrs. Failure of the bundle occurred at this time.

III. Post Test Examination of Results

A. Open circuits were found in 60 heater rods.

B. The direct cause of bundle overheating was determined to be an incorrect thermocouple connection. The thermocouple which was supposed to be monitoring the midplane temperature was actually located at the lower end of its heater rod.

C. An examination of the heater rod temperature data reveals that at least 13 heater rods had misconnected thermocouples.

D. The maximum temperature of the bundle was in excess of 2500°F (chromel-Alumel thermocouple conversion tables terminate at 2500°F).

R. F. Farman
Thermal and Hydraulic Experimentation

APPROVED BY: J. O. Cermak, Manager
Thermal and Hydraulic Experimentation

/l

Rulemaking Comments

From: mel2005@columbia.edu
Sent: Tuesday, November 17, 2009 5:48 PM
To: Rulemaking Comments
Subject: Attn: Rulemakings and Adjudications Staff
Attachments: Rulemaking Petition.pdf

Dear Ms. Annette Vietti-Cook:

Attached to this e-mail in a PDF file is a cover letter and a petition for rulemaking, dated November 17, 2009, submitted pursuant to 10 C.F.R. § 2.802.

Sincerely,

Mark Leyse

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Tue, 17 Nov 2009
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