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From: Mark Leyse [markleyse@gmail.com]
Sent: Tuesday, November 23, 2010 11:18 PM
To: Rulemaking Comments; PDR Resource
Cc: Dave Lochbaum; necnp@necnp.org; Raymond Shadis; Powers, Dana A
Subject: NRC-2009-0554 (First)
Attachments: Comments November 2010.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is my first response, dated November 23, 2010, to the NRC's notice of solicitation of public comments on PRM-50-93 and PRM-50-95, NRC-2009-0554, published in the Federal Register on October 27, 2010.

Sincerely,

Mark Leyse

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November 23, 2010

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

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 AND PRM-50-95; NRC-2009-0554

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

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 and PRM-50-95; NRC-2009-0554

I. Statement of Petitioner's Interest

On November 17, 2009, Mark Edward Leyse, Petitioner (in these comments "Petitioner" means Petitioner for PRM-50-93 and sole author of PRM-50-95), submitted a petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the Nuclear Regulatory Commission ("NRC") make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments;[1] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[2, 3]

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation

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

[2] It can be extrapolated from experimental data that, in the event a large break ("LB") LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater and an average fuel rod power of approximately 0.37 kW/ft or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LB LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

[3] It is noteworthy that in 1975, Fred C. Finlayson stated, "[r]ecommendations are made for improvements in criteria conservatism, especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates);" see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, Abstract, p. iii.

calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[5]

On June 7, 2010, Petitioner, submitted an enforcement action 10 C.F.R. § 2.206 petition on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

On October 27, 2010, NRC published in the Federal Register a notice stating that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner submitted on behalf of NEC, meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[6]

When Petitioner wrote the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner did not foresee that NRC would docket it as PRM-50-95. PRM-50-95 was written and framed as a 10 C.F.R. § 2.206 petition, not as a 10 C.F.R. § 2.802 petition; however, it is laudable that NRC is reviewing the issues Petitioner raised in PRM-50-95.

II. Supplementary Information to PRM-50-93 and PRM-50-95

> Daniel Ford: I am concerned with one of the many gaps in the Interim Policy Statement and the computer code. I am concerned with a variety of chemical-metal-water reactions that are not considered at all in these codes, metal-water reactions which various recent experimental data indicate can prove [to] very significantly [impact] local temperature

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[4] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would commence in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA.

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

[6] Federal Register, Vol. 75, No. 207, Notice of consolidation of petitions for rulemaking and re-opening of comment period, October 27, 2010, pp. 66007-66008.

> during an accident, and [cause] extensive cladding damage. The specific metal-water reaction I am concerned with at the moment is the reaction between the Zircaloy-Inconel eutectic and steam, I am concerned to find out how the Applicant's analysis contained in the computer code, which does not consider this, how it would be different if it did.
>
> Leonard M. Trosten: I thank you for the explanation. I recognize this as being one of the principal points of concern in the critique by the Union of Concerned Scientists...[7]—IP-2 licensing hearing, November 1971

Experimental data discussed in PRM-50-93 (partly), PRM-50-95, and in these comments on PRM-50-93 and PRM-50-95—among other things—indicates that "low temperature" eutectic reactions could affect the progression of damage during a LOCA. For example, Inconel grid spacers would effect the progression of damage in a reactor core during a LOCA if their temperatures were to reach approximately 2012°F;[8] and experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.[9]

A. Some Parts of PRM-50-95 can be Interpreted as a Commentary on the Safety Issues Raised in PRM-50-93

When Petitioner wrote the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner did not foresee that NRC would docket it as PRM-50-95. PRM-50-95 was written and framed as a 10 C.F.R. § 2.206 petition, not as a 10 C.F.R. § 2.802 petition; however, it is laudable that NRC is reviewing the issues Petitioner raised in PRM-50-95.

Some parts of PRM-50-95 can be interpreted as a commentary on the safety issues raised in PRM-50-93.

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[7] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2520-2522.

[8] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 202.

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

In particular, when PRM-50-95 states that experimental data indicates that VYNPS's LBPCT of 1960°F[10] does not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA, and that such data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety,"[11] it means that the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit needs to be decreased to a temperature lower than 1832°F.

So NRC needs to determine how far below 1832°F the 10 C.F.R. § 50.46(b)(1) PCT limit needs to be decreased in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

(In PRM-50-93, Petitioner requests that NRC revise 10 C.F.R. § 50.46(b)(1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments.)

B. Clarifications of Some of Petitioner's Statements in PRM-50-93 and Comments on PRM-50-93

1. Clarifications of Statements on Reflood Rates

Petitioner wants to clarify that Petitioner was referring to experimental data from Thermal-Hydraulic Experiment 1 ("TH-1"), conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, when Petitioner stated in PRM-50-93 that:

> 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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[10] Entergy, "VYNPS 10 C.F.R. § 50.46(a)(3)(ii) Annual Report for 2009," January 14, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100260386, p. 2.

[11] Mark Edward Leyse, PRM-50-95, June 7, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101610121, pp. 5-6.

Additionally, in TH-1, a total of 28 tests were conducted to simulate large break ("LB") LOCAs, so Petitioner was referring to LB LOCAs, when Petitioner stated "[i]t can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F."

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

2. Clarifications of Statements on the Baker-Just and Cathcart-Pawel Correlations

In PRM-50-93, Petitioner's phrasing was imprecise when Petitioner stated that "[d]ata from multi-rod...severe fuel damage experiments...indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative *for calculating* the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA." Petitioner should have more clearly stated that "data from multi-rod...severe fuel damage experiments indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative *for use in analyses that would predict* the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would commence in the event of a LOCA."

Also, in PRM-50-93, Petitioner's phrasing was imprecise when Petitioner stated that "[t]his...indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative *for calculating* the metal-water reaction rates that would occur in the event of a LOCA." Petitioner should have more clearly stated that "[t]his...indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative *for use in analyses*

that would predict the metal-water reaction rates that would occur in the event of a LOCA."

Additionally, in other sentences in PRM-50-93, Petitioner stated imprecise phrases similar to "the Baker-Just and Cathcart-Pawel equations are both non-conservative *for calculating*." Petitioner should have more clearly stated that "the Baker-Just and Cathcart-Pawel correlations are both non-conservative *for use in analyses that would predict*."

Petitioner will now provide a brief explanation of Petitioner's claim that:

> Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would commence in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA.

First, it was reported in the early 1990s that for the CORA-16 experiment (a multi-rod severe fuel damage experiment), "[c]ladding oxidation was not accurately predicted by available correlations"[12] and "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted."[13] This indicates that the Baker-Just and Cathcart-Pawel correlations are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

Second, it is significant that in AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above

[12] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," CONF-9105173-3-Extd.Abst., Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

[13] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," ORNL/FTR-3780, October 16, 1990, p. 3.

> 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[14]

Clearly, the Zircaloy-steam reaction is not negligible below 1900°F, as experimental data from multi-rod experiments demonstrates. For example, a Karlsruhe paper states:

> As already observed in previous tests, the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above [1832°F]. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles.[15]

It is clear that ECCS evaluation calculations using the Baker-Just correlation under-predict the Zircaloy-steam reaction that would occur in a LOCA environment; this also applies to ECCS evaluation calculations using the Cathcart-Pawel correlation.

3. Additional Clarifications of Statements on the Baker-Just and Cathcart-Pawel Correlations

In PRM-50-93, in a number of places, petitioner stated that "the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F."[16]

First (as stated above for a similar phrase), Petitioner's phrasing was imprecise when Petitioner stated in PRM-50-93 that that "the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F." Petitioner

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[14] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

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

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

should have more clearly stated that "*analyses that use* the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and *analyses that use* the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA."

Second, in PRM-50-93, when Petitioner stated that (now rephrased) "analyses that use the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and analyses that use the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA," Petitioner qualified such statements with footnotes that stated:

> According to the NRC's more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K.[17]

So, it is important to clarify that Petitioner's statement that "analyses that use the Baker-Just equation calculate that autocatalytic oxidation occurs at approximately 2600°F, in the event of a LOCA, and analyses that use the Cathcart-Pawel equation calculate that autocatalytic oxidation occurs at approximately 2700°F, in the event of a LOCA," was qualified in PRM-50-93, as specifically referring to the more than 50 LOCA calculations with RELAP5/Mod3, discussed in "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K.

C. A Postulation that Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at 1000°C in the Three Mile Island Unit 2 Accident

First, Petitioner does not intend to present Dr. Robert E. Henry's postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C (1832°F) in the Three Mile Island Unit 2 ("TMI-2") accident as evidence that an autocatalytic reaction did in fact commence at 1000°C in the TMI-2 accident: there is no thermocouple data from the hot spots of the fuel assemblies to confirm if Dr. Henry is correct.

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

(It is acknowledged that runaway oxidation occurred in the TMI-2 accident; Petitioner's point, is to draw attention to the fact that Dr. Henry of Fauske & Associates postulated runaway oxidation commenced at 1832°F—368°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. It is noteworthy that, in 1981, Fauske & Associates developed the Modular Accident Analysis Program (MAAP) code in response to the TMI-2 accident—under sponsorship from Electric Power Research Institute and MAAP Users Group.)

Second, Petitioner does not intend to use Dr. Henry's postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident to support Petitioner's argument that the 10 C.F.R. § 50.46(b)(1) PCT limit should be decreased to a temperature lower than 1832°F.

Third, Petitioner is discussing what Dr. Henry postulated, because Petitioner finds it compelling that Dr. Henry postulated that an autocatalytic reaction commenced at 1000°C in the TMI-2 accident. In Dr. Henry's presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007,[18] Dr. Henry states that "[a]t about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal runaway state."[19]

Fourth, information presented in "TMI-2: A Textbook in Severe Accident Management," regarding the Zircaloy-steam reaction and core damage phenomena, does pertain to PRM-50-93 and PRM-50-95.

Fifth, it is significant that in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in PRM-50-93 and PRM-50-95—including the CORA experiments and LOFT LP-FP-2 experiment.

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[18] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 ANS/ENS International Meeting, November 11, 2007, seven of these presentation slides are in attachment 2 of the transcript from "10 C.F.R. 2.206 Petition Review Board Re: Vermont Yankee Nuclear Power Station", July 26, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML102140405, Attachment 2.

[19] *Id.*

(It is significant that Dr. Robert E. Henry is clearly very knowledgeable about severe accident phenomena. It is also significant that, in the acknowledgements for "TMI-2: A Textbook in Severe Accident Management," one of the presentation slides states that Dr. Dana Powers sent Dr. Henry the slides Dr. Powers had used in lectures on the TMI-2 accident and that Hans Fauske, D.Sc., reviewed all of the slides presented in "TMI-2: A Textbook in Severe Accident Management": Dr. Powers and Fauske, D.Sc., are also clearly very knowledgeable about severe accident phenomena.)

It is compelling that one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

> Fuel Cladding Oxidation
>
> - As the boil-off of the water in the core continued, the uncovered region continued to heatup with the highest cladding/fuel temperatures being at about the 3/4-core height location.
>
> - Increasing temperatures caused the Zircaloy oxidation rate to increase which was accompanied by an increased release rate of chemical energy.
> - *At about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal-runaway state.* During this interval the Zircaloy reaction was limited by the rate of steam generated in the covered part of the core which decreased as the water level decreased [emphasis added].[20]

So Dr. Henry postulated that runaway oxidation commenced at approximately 1000°C. And another one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states that "[t]he chemical energy release [from the oxidation of the Zircaloy fuel cladding by steam] caused the core to overheat faster and eventually melt or liquefy the individual constituents."[21]

It is significant that one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

> Fuel Cladding Oxidation
>
> - The Zr in the Zircaloy cladding will oxidize in a high temperature steam environment: hydrogen and energy (heat) are released by this reaction:
>
> $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \Delta H_R$

[20] *Id.*
[21] *Id.*

- The heat of reaction, ΔH_R, is about 6.5 MJ/kg.

- At about 1000°C, the rate of chemical energy release approximately equals the decay power.

- The oxidation rate increases with increasing temperature, which leads to an escalating core heatup rate.

- Therefore, the core damage was generally caused by the cladding oxidation.[22]

It is also significant that another one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states:

Example: Core Heatup Rate Escalation Due to Cladding Oxidation

- Important Tests:

- Out-of-Reactor: CORA

- In-Reactor: [PBF] SFD, FLHT, LOFT LP-FP-2, and PHEBUS[23]

So in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in PRM-50-93 and PRM-50-95—including the CORA experiments and LOFT LP-FP-2 experiment. And it is compelling that Dr. Henry postulated that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident—368°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

D. National Research Universal Thermal-Hydraulic Experiment 1

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

[22] *Id.*
[23] *Id.*

TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[24]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[25] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[26] "low enough that superheated steam conditions [would] exist at the loop inlet instrument location. The superheat requirement [was] imposed so that meaningful steam temperatures [could] be measured."[27])

In TH-1 test no. 130, there was a reflood rate of 0.7 in./sec. At the start of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[28]

In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[29] So because of the heat generated from the metal-water reaction, the peak cladding temperature increased by 190°F, after the reactor shutdown.

It is clear that, in TH-1 test no. 130, if the reactor had not shutdown when the PCT was approximately 1850°F, that the overall PCT would have been greater than 2040°F. In fact, it is highly probable that the multi-rod bundle in the TH-1 test no. 130, would have incurred autocatalytic oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

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[24] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119.
[25] *Id.*, p. 10.
[26] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024, p. 6-5.
[27] *Id.*
[28] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, p. 13.
[29] *Id.*

(It is significant that TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 would have been 0.37 kW/ft,[30] in the pre transient phase of the test.)

Of course, in the event of an actual LOCA, the energy from decay heating would not suddenly terminate if cladding temperatures were to reach approximately 1850°F.

The data of TH-1 test no. 130 indicates that, in the event of a LOCA, at a nuclear power plant, if peak cladding temperatures reached temperatures of approximately 1850°F, the Zircaloy cladding would begin to rapidly oxidize, and that—with the combination of heat generated by the metal-water reaction and decay heat—the oxidation would, with high probability, become autocatalytic and cladding temperatures would start increasing at a rate of tens of degrees Fahrenheit per second. Within a period of approximately 60 seconds peak cladding temperatures would increase to 3000°F or greater; the melting point of Zircaloy is approximately 3308°F.[31]

(Of course, as stated above, there would have been a small amount of actual decay heat in the bundle of TH-1 test no. 130, after the reactor shutdown; however, it would have been substantially lower than the amount of decay heat in a counterpart bundle, in the event of a LOCA.)

E. The Damage PWR Fuel Assembly Components would Incur at "Low Temperatures"

"Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents" states that "[e]xperiments were performed at several laboratories to investigate the behavior of (Ag,In,Cd) control rods during severe reactor accidents"[32, 33] and that the

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[30] *Id.*, p. 10.

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

[32] D.A. Powers, "Behavior of Control Rods during Core Degradation," NUREG/CR-4401, SAND85-0469, 1985; B.R. Bowsher, R.A. Jenkins, A.L. Nichols, N.A. Rowe, J.A.H. Simpson, "Silver-Indium-Cadmium Control Rod Behavior during a Severe Reactor Accident," AEEWR-R 1991, 1986; David A. Petti, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," NUREG/CR-4876, EG + E-2501, 1987; and F. Nagase, H. Uetsuka, "Some Topics from the Basic Experiments on High-Temperature Core Materials Behavior at JAERI," JAERI, Tokai Research Establishment, Japan.

essential results of the experiments are summarized in a paper titled "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents."[34]

Regarding the results of the experiments, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents." states:

> *The (Ag,In,Cd) alloy melts at about 800°C, but will not affect core degradation as long as the molten material is contained within the stainless steel cladding.* As the temperature increases, some of the control rod constituents will vaporize within the cladding until failure occurs, either from internal pressurization or from melting of the cladding. At low pressures of the primary system and when no Zircaloy is present, the control rod fails between 1350 and 1450°C. *Failure of the control rods with the Zircaloy guide tubes occurs at about 1200°C as a result of thermal expansion, physical contact, and eutectic chemical interactions between the stainless steel cladding and the Zircaloy guide tube.* The high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the cladding fails. The ejected material results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy. Due to the high vapor pressure of Cd it vaporizes. Liquid control rod material continues to vaporize if it remains at high temperatures. The control rod material which will flow out of the hot regions of the core freezes and may inhibit steam and/or water flow. At high system pressure, over-pressurization of the rod does not occur. Instead, upon failure, the alloy flows to cooler regions of the reactor core. In all cases the resulting reaction products melt at low temperatures and enhance by this the degradation of the reactor core [emphasis added].[35, 36]

So "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," states that "[f]ailure of the control rods with the Zircaloy guide tubes occurs at about 1200°C"[37] and that "[t]he high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the

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[33] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," Kernforschungszentrum Karlsruhe, KfK 4670, 1989, p. 1.
[34] David A. Petti, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," NUREG/CR-4876, EG + E-2501, 1987.
[35] *Id.*
[36] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, pp. 1-2.
[37] *Id.*, p. 1.

cladding fails"[38] that "results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy."[39]

And regarding eutectic interactions of the absorber rod's steel cladding tube and the Zircaloy guide tube that can cause liquefaction to occur locally at approximately 1200°C, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> The absorber rod should fail; *i.e.*, melt down, upon attainment of the melting point of the steel cladding tube (~1400°C) at the latest. *On account of eutectic interactions of the steel cladding tube and the zircaloy guide tube, liquefaction can take place locally as early as from 1200°C on.* The (Ag,In,Cd) absorber melt contributes essentially to the propagation of damage in the bundle which is an unambiguous finding of chemical-analytical studies of the reaction products by means of the scanning electron microscope [emphasis added].[40]

It is significant that "when no Zircaloy is present, the control rod fails between 1350°C and 1450°C"[41] or that the control rod fails at ~1400°C, at the latest.[42] So when Zircaloy is present, the control rod fails at a temperature—approximately 1200°C—that is between 150°C and 250°C lower—a substantial temperature difference.

Describing the damage PWR fuel assembly components would incur at relatively low temperatures, in more detail, the conclusion of "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents" states:

> - *The (Ag,In,Cd) absorber alloy starts to melt at about 800°C, but this will not affect core degradation as long as the molten material is contained within the stainless steel (AISI 316) cladding.* The chemical interaction between the absorber alloy and stainless steel is negligible.
>
> - Failure of the stainless steel absorber rod cladding takes place as a result of either internal pressurization (high Cd vapor pressure) or eutectic interactions with the Zircaloy guide tube (bowing of the rods at high

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[38] *Id.*

[39] *Id.*, pp. 1-2.

[40] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7448, 2008, p. 14.

[41] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, p. 1.

[42] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, p. 14.

temperatures). The released (Ag,In,Cd) melt can then interact with the Zircaloy guide tube.

- The Zircaloy will be chemically dissolved by the absorber alloy. The dissolution of the Zircaloy can be described by a parabolic rate law. *The dissolution rate is very fast; at 1200°C, it takes only about 50 [seconds] to dissolve 1 mm Zircaloy and about 4 minutes to destroy the entire 2.25 mm thick Zircaloy crucible wall.*

- As soon as solid state contact occurs between the stainless steel cladding and the Zircaloy guide tube, eutectic interactions take place which can be described by parabolic rate laws. *Liquid phases form at around 1000°C, and a fast and complete liquefaction of both components takes place above 1250°C. Only small amounts of stainless steel are necessary to dissolve great amounts of Zircaloy, and it takes only a little more than 2 minutes to destroy the 2.25 mm thick Zircaloy crucible wall at 1200°C.*

- Thin ZrO_2 layers (~10 μm) on the Zircaloy surface delay the chemical interactions of Zircaloy with the (Ag,In,Cd) alloy or the stainless steel, but cannot prevent them. The ZrO_2 layer must be dissolved by the Zry before chemical interactions can take place. The required incubation period depends on temperature and time. Dissolved oxygen in the Zircaloy, forming oxygen-stabilized α-Zr(O), reduces the reaction rates and shifts the liquefaction temperature to slightly higher levels.

- With respect to the chemical behavior of (Ag,In,Cd) absorber rods during severe reactor accidents, meltdown and relocation must be assumed to occur at temperatures around 1250°C. The resulting melt destroys the Zircaloy cladding of the fuel rods and dissolves a part of the UO_2, contributing substantially to fuel element degradation. *Since UO_2 fuel can be liquefied at temperatures as low as 1250°C, this process has a strong impact on the release of volatile fission products.*

- The premature low-temperature failure of the PWR absorber rods and the localized relocation of (Ag,In,Cd) alloy within the reactor core may cause criticality problems during flooding of the destroyed core [emphasis added].[43]

[43] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, pp. 13-14.

And describing chemical interactions between the (Ag, In, Cd) absorber rod alloy and Zircaloy, in detail, "Current Knowledge on Core Degradation Phenomena, a Review" states:

> The absorber rod alloy (80 wt% silver, 15% indium, 5% cadmium) is thermodynamically stable with its stainless steel cladding, even in the liquid state (>800°C). *However, the absorber rod guide tube is made from Zircaloy, which will chemically interact with the stainless steel cladding of the absorber rod. During a severe reactor accident, localized contact between stainless steel and Zircaloy exists at many places. This solid-state contact results in chemical interactions with the formation of liquid phases around 1150°C.* After failure of the absorber rod cladding, the molten Ag-In-Cd alloy (melting point ~800°C) comes into contact with the Zircaloy guide tube and chemically destroys it. Then, the molten Ag-In-Cd can even attack and chemically dissolve the Zircaloy cladding of the fuel rods well below the melting point of Zircaloy (~1760°C). The relocating Ag-In-Cd alloy is therefore able to propagate and accelerate the core-melt progression at rather low temperatures.
>
> The chemical interactions between Ag-In-Cd and Zircaloy were studied in separate-effects tests which are described in [reference 19].[44] The reaction zone growth rate (decrease in Zircaloy wall thickness) is plotted in an Arrhenius diagram against the reciprocal temperature in Fig. 10. *At temperatures >1200°C, the chemical interactions result in a sudden and complete liquefaction of the compatibility specimens.* As a consequence, the Zircaloy cladding can be chemically dissolved ~600 K below its melting point and may even result in a low-temperature UO_2 fuel dissolution. For phase considerations of melting reactions, the quaternary U-Zr-Fe-O system may be regarded as a model system for the complicated multi-component system of a beginning core melt; iron represents the stainless steel. A detailed description of the phase relations is given in [reference 4].[45]
>
> The chemical interaction between the Ag-In-Cd alloy and Zircaloy is theoretically described by a model under conditions of convective mixing in the (Zr, Ag, In) liquid phase in [reference 20].[46] Homogeneous bulk saturation of the liquid phase with Zr takes place in the course of the Zircaloy dissolution by the absorber melt resulting in a gradual decrease of the interaction process. Two main parameters of the model are calculated: Zr concentration in the saturated melt and convective mass transfer coefficient in the liquid phase[47] [emphasis added].[48]

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[44] P. Hofmann, M. Markiewicz, Journal of Nuclear Materials, 209, 1994, p. 92.
[45] P. Hofmann, *et al.*, Nuclear Technology 87, 1989, p. 146.
[46] M.S. Veshchunov, P. Hofmann, Journal of Nuclear Materials, 228, 1996, p. 318.
[47] *Id.*

And regarding the fact that control rod material (Ag-In-Cd) may influence the chemical reaction between Inconel grid spacers and Zircaloy fuel cladding, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core" states:

> The CORA-7 test[49] indicated that the reaction between [Inconel] grid spacer and [Zircaloy] cladding was not symmetrical and that control rod material (Ag-In-Cd) may influence the interaction between grid spacer and cladding.[50]

So clearly, in the event of a LOCA, PWR core component damage could commence at relatively low temperatures.

F. Chemical Interactions Between Zircaloy and Inconel and Between Zircaloy and Stainless Steel at "Low Temperatures"

It is significant that "[t]he chemical reaction between Inconel and Zircaloy influences the meltdown of the reactor core in the vicinity of Inconel grid spacers."[51]

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

> *Grid spacers can have a significant impact on the progression of damage in a reactor core during a severe accident.* ... The impact of grid spacers on damage progression has been revealed by out-of-pile experiments in Germany[52] and Japan,[53] in-pile experiments at the PBF facility in Idaho,[54]

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[48] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, pp. 201-202.

[49] P. Hofmann, *et al.*, "Material Behavior in the Large PWR Bundle Experiment CORA-7," International CORA Workshop 1991, September 23-26, 1991, Karlsruhe, Germany.

[50] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," Nuclear Engineering and Design 146, 1994, p. 436.

[51] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," Abstract, p. 427.

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

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

> and by examinations of the damaged Three Mile Island (TMI-2) core.[55] *The experiments in Germany and Japan have revealed the existence of chemical interactions between Inconel and Zircaloy that take place at temperatures as low as 1273 K [(1832°F)]*, more than 200 K lower than the melting temperature of Inconel. *Thus in a reactor core with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers* [emphasis added].[56]

It is significant that Inconel grid spacers would effect the progression of damage in a reactor core during a LOCA if their temperatures were to reach approximately 2012°F;[57] and significant that experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.

And discussing the fact that a first melting process started at approximately 1250°C at the central Inconel grid spacer in the CORA-2 and CORA-3 experiments, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> A first melting process starts already at about 1250°C at the central grid spacer of Inconel, due to diffusive interaction in contact with Zry cladding material, by which the melting temperatures of the interaction partners (ca. 1760°C for Zry, ca. 1450°C for Inconel) are dramatically lowered towards the eutectic temperature, where a range of molten mixtures solidifies. (This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[58]

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

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

[56] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.

[57] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.

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

It is also significant that "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> Only small amounts of Inconel are necessary to dissolve great amounts of [Zircaloy].[59]

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

> The Zircaloy/stainless steel (1.4919; corresponds to [stainless steel] Type 316 with 18 wt% Ni and 8 wt% Cr) interactions are important with respect to the contact between the absorber rod cladding and the Zircaloy guide tube and between the Inconel spacer grid and the Zircaloy fuel rod cladding. In both cases, the iron-zirconium and the nickel-zirconium phase diagrams show that due to eutectic interactions, early melt formation has to be expected, which initiates the melt progression within the fuel assembly at low temperatures. *Liquid phases form at temperatures <1000°C; however, the reaction kinetics become significant only above 1100°C.* This was seen in the CORA tests, where fuel rod bundles were heated up to complete meltdown. In all cases, the damage of the bundle was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions. *Localized liquefaction of these components started around 1200°C.*[60]
>
> The reaction kinetics between Zircaloy and stainless steel can be divided into a reaction zone growth rate in Zircaloy and one in stainless steel, as shown in Fig. 11. One can see that the Zircaloy is attacked more strongly than the stainless steel. Oxide layers on the Zircaloy cladding outside diameter delay the chemical interactions between Zircaloy and steel, but they cannot prevent them. *The influence of oxide layers becomes less important at temperatures >1100°C, since the dissolution of the protecting ZrO_2 layers occurs rather fast and the stainless steel is then in contact with metallic Zircaloy or oxygen-stabilized α-Zr(O).*[61]

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[59] *Id.*, p. 40.
[60] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[61] P. Hofmann, M. Markiewicz,, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Stainless Steel at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 5106, 1994.

> In a first approach, the reaction behavior of Zircaloy with Inconel 718 is comparable to that with Type 316 stainless steel.[62] *At temperatures <1100°C, Inconel attacks the Zircaloy faster than stainless steel; above 1100°C, the situation is the reverse.* In both cases, the melting of a relatively large quantity of Zircaloy with limited melting of the adjacent stainless steel or Inconel takes place. *During heat-up of the stainless steel/Zircaloy and Inconel/Zircaloy reaction systems, a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C. This may be the reason that melt progression in a fuel rod bundle initiates at absorber rod cladding (stainless steel)/Zircaloy guide tube contact areas and Inconel spacer grid/Zircaloy fuel rod contact locations*[63] [emphasis added].[64]

It is significant that in the CORA tests, in which fuel rod bundles were heated up to complete meltdowns, that "the damage of the [bundles] was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions"[65] and that "[l]ocalized liquefaction of these components started around 1200°C [(2192°F)]."[66] It was also observed in the CORA tests that "[l]iquid phases form at temperatures <1000°C [(1832°F)]" and that "the reaction kinetics become significant only above 1100°C [(2012°F)]."[67]

1. Additional Information—from the 1970s—on the Chemical Interaction Between Zircaloy and Inconel at "Low Temperatures"

The chemical interaction between Zircaloy and Inconel was a subject of the IP-2 licensing hearing, in 1971, as the selection below from the transcript of the IP-2 licensing hearing demonstrates:

> Daniel Ford: I am concerned with one of the many gaps in the Interim Policy Statement and the computer code. I am concerned with a variety of chemical-metal-water reactions that are not considered at all in these codes, metal-water reactions which various recent experimental data indicate can prove [to] very significantly [impact] local temperature during an accident, and [cause] extensive cladding damage. The specific

[62] P. Hofmann, M. Markiewicz, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Inconel 718 at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 4729, 1994.
[63] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[64] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[65] *Id.*
[66] *Id.*
[67] *Id.*

> metal-water reaction I am concerned with at the moment is the reaction between the Zircaloy-Inconel eutectic and steam, I am concerned to find out how the Applicant's analysis contained in the computer code, which does not consider this, how it would be different if it did.
>
> Leonard M. Trosten: I thank you for the explanation. I recognize this as being one of the principal points of concern in the critique by the Union of Concerned Scientists...[68]

So in 1971, Daniel Ford of UCS stated that he was concerned with one of the many gaps in the Interim Policy Statement: the Zircaloy-Inconel eutectic reaction. Unfortunately, to this day, nearly 40 years latter, NRC's regulations still do not consider the Zircaloy-Inconel eutectic reaction that would, with high probability, occur at temperatures lower than 2200°F.

Discussing chemical interactions between Zircaloy and Inconel X-750, a paper published in 1975, "Incompatibility between Zircaloy-2 and Inconel X-750 during Temperature Transients," states:

> All current designs of water reactors contain various components made from high-nickel and/or high-iron content alloys. In certain specific cases these components are either in contact with, or in close proximity to, other components constructed from high-zirconium content alloys. Typical examples of such high-nickel or high-iron content alloy components are, without mentioning the reactor type specifically, plenum springs, fuel element spacer grids, control rod claddings and slides, and wear pads.
>
> All of these components could potentially react with the adjacent zirconium alloy component, since, although the alloys themselves have relatively high melting points (*e.g.*, Inconel X-750 melts at 1395°C), their major components nickel and iron form eutectics with zirconium at the much lower temperatures of 960°C [(1760°F)] and 940°C [(1724°F)], respectively.[69]
>
> The problems which could arise because of the existence of these low melting point eutectics have been demonstrated during heat-treatments, carried out as part of the BWR-FLECHT test under emergency core

[68] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2520-2522.

[69] M. Hansen, "Construction of Binary Alloys" McGraw-Hill, 1958.

> cooling conditions,[70] in which Inconel springs became inseparably fused to zircaloy tubing because of melting at the points of contact. Similar results have been observed occasionally during vacuum brazing of Inconel in which zirconium has been present as a getter.[71] Also, it has long been recognized that high temperature mechanical testing of zirconium and its alloys cannot be accomplished using nickel or iron based alloys for the gripping devices.[72, 73]

And providing additional information on views from the early 1970s of the chemical interaction between Zircaloy and Inconel, the selection below from the transcript of the IP-2 licensing hearing states:

> James S. Moore: Well, we had the Inconel grid in a reactor contacting a Zircaloy rod as it does in a reactor at a local point with a spring, and then we heated up the total assemblage of the rod and the grid and to the point of temperatures, I forget the exact numbers, but they were up well into the zirc-water reaction temperatures approaching 2300 degrees [Fahrenheit], and then observed what happened at the local contact point between the Inconel and the zirc rod, and as I indicated and as other people have observed there is a eutectic formed between the Zircaloy and the nickel of the Inconel grid, which has a melting point of about 1760 degrees [Fahrenheit] and we did get in the test very local melting at this contact point. But this did not create any difficulty in that the heat was carried away from that local point sufficiently so that there was no blockage, any additional blockage, or effects on the Zircaloy rod itself. So that this is based on holding this rod in the Inconel grid at about 2300 degrees [Fahrenheit] for several minutes, which is well beyond what you would expect in a loss of coolant situation.
>
> Anthony Z. Roisman: Let me see if I understand this. The eutectic is formed at about 1760 degrees [Fahrenheit], is that right?
>
> James S. Moore: It melts at 1760 [degrees Fahrenheit].
>
> Anthony Z. Roisman: You are saying that no reaction occurs with the water until it reaches almost 2300 degrees Fahrenheit?

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[70] M.J. Grater, W.F. Zelenzny, R.E. Schmunk, Idaho Nuclear Corp., IN-1453, Idaho Falls, March 1971.

[71] J. Christensen: Danish Atomic Energy Commission, Research Establishment Risø, private communication, 1974.

[72] B. Weiler Madsen: Danish Atomic Energy Commission, Research Establishment Risø, private communication, 1975.

[73] M.R.Warren, K. Rørbo, E. Adolf, Danish Atomic Energy Commission, Research Establishment Risø, "Incompatibility between Zircaloy-2 and Inconel X-750 during Temperature Transients," Journal of Nuclear Materials, 58, 1975, p. 185.

James S. Moore: No.

Anthony Z. Roisman: Exactly how does the melting occur? Are droplets formed that would tend to drop down between the rods?

James S. Moore: Yes. There was very local melting, but we didn't observe any droplets or any sputtering which you might postulate under those conditions. It was a very localized effect and that is what gave us the assurance that this was not a problem for us.[74]

Clearly, James S. Moore's claim that the Inconel-Zircaloy reaction would not be a problem at 2300°F (1260°C) is erroneous.

Regarding the Inconel-Zircaloy reaction at 2282°F (1250°C), "Current Knowledge on Core Degradation Phenomena, a Review" states:

> During heat-up of the...Inconel/Zircaloy reaction systems, a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C [(2282°F)]. This may be the reason that melt progression in a fuel rod bundle initiates at...Inconel spacer grid/Zircaloy fuel rod contact locations.[75, 76]

Furthermore, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core" states:

> Grid spacers can have a significant impact on the progression of damage in a reactor core during a severe accident. ... The impact of grid spacers on damage progression has been revealed by out-of-pile experiments in Germany[77] and Japan,[78] in-pile experiments at the PBF facility in Idaho,[79]

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[74] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, pp. 2170-2171.

[75] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.

[76] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.

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

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

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

> and by examinations of the damaged Three Mile Island (TMI-2) core.[80] The experiments in Germany and Japan have revealed the existence of chemical interactions between Inconel and Zircaloy that take place at temperatures as low as 1273 K [(1832°F)], more than 200 K lower than the melting temperature of Inconel. *Thus in a reactor core with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers* [emphasis added].[81]

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by erroneous notions of the Inconel-Zircaloy reaction.

G. The Zircaloy-Steam Reaction could be Affected by the Boron Carbide (B_4C) Absorber

It is significant that in PRM-50-95, Petitioner discussed the fact that in a BWR LOCA, the Zircaloy-steam reaction could be affected by the boron carbide (B_4C) absorber.

In PRM-50-95, Petitioner quotes "Degraded Core Quench: A Status Report," which compares the BWR CORA-17 experiment with the PWR CORA-12 and CORA-13 experiments (which used typical PWR bundles and Ag-In-Cd absorber).

Regarding this issue, "Degraded Core Quench: A Status Report" states:

> The earlier starting and stronger reaction in the [CORA-17] BWR test can be interpreted as being due to the additional influence of the boron carbide [(B_4C)] absorber. *This material has an exothermic reaction rate three times larger than that of Zircaloy* and produces [four] to [eight] times more hydrogen [emphasis added].[82]

So according to "Degraded Core Quench: A Status Report," boron carbide (B_4C) has an exothermic reaction rate approximately three times greater than that of Zircaloy.

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

[81] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.

[82] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," OCDE/GD(97)5, August 1996, p. 16.

Additionally, comparing the BWR CORA-17 experiment with the PWR CORA-12 and CORA-13 experiments "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17" states:

> Immediately after quenching BWR test bundle CORA-17 experiences a modest increase for 20 sec. and changed then in a steep increase resulting in the highest temperature and hydrogen peaks of the three tests [(CORA-12, CORA-13, CORA-17)]. CORA-17 also showed a temperature increase in the lower part of the bundle... We interpret this earlier starting and stronger reaction [as being] due to the influence of the boron carbide, the absorber material of the BWR test.
>
> *B_4C has an exothermic reaction energy [four] to [five] times larger than Zry* and produces about [six] times more hydrogen. Probably the hot remained columns of B_4C (seen in the non-quench test CORA-16) react early in the quench process with the increased upcoming steam. *The bundle temperature, raised by this reaction increases the reaction rate of the remained metallic Zry* (exponential dependence) [emphasis added].[83]

And according to "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," boron carbide (B_4C) has an exothermic reaction rate approximately four to five times greater than that of Zircaloy. Furthermore, the increased bundle temperature—a consequence of the B_4C exothermic reaction energy—in turn, increases the reaction rate of the remaining Zircaloy.

H. A Portion of the IP-2 Licensing Hearing Transcript: Superheated Steam in a LOCA Environment

It is significant that in 1971, in the IP-2 licensing hearing, Daniel Ford of UCS was concerned about the role that superheated steam would play in a LOCA environment.

Regarding this issue, a portion of IP-2 licensing hearing transcript states:

> Daniel Ford: Mr. Moore, is it correct that in the [PWR] FLECHT tests[84] negative heat transfer coefficients [calculated as a result of heat transfer

[83] S. Hagen, P. Hofmann, V. Noack, L. Sepold, G. Schanz, G. Schumacher, "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," Forschungszentrum Karlsruhe, FZKA 5679, 1996, Abstract, pp. ii.

[84] The transcript states "BWR FLECHT tests, not "PWR FLECHT tests";" however, it is most likely that Daniel Ford was actually asking a question about the PWR FLECHT tests. First, James S. Moore was an employee of Westinghouse Electric, which conducted the PWR FLECHT tests. Second, negative heat transfer coefficients—calculated as a result of heat transfer from the coolant to the fuel cladding—occurred in the PWR FLECHT tests. Third, results from the PWR

from the coolant to the fuel cladding] were observed at axial levels in a number of different instances?

James S. Moore [of Westinghouse Electric]: They were recorded as negative heat transfer coefficients. What they actually indicate is reverse heat transfer from the coolant to the [fuel] cladding.

Daniel Ford: For the purpose of this discussion and since they are plotted as heat transfer coefficients, would you just accept the definition of terms, that is a negative heat transfer coefficient?

James S. Moore: I guess I'd prefer reverse heat transfer, which is more descriptive.

Daniel Ford: I see. It is correct, though, that the reverse heat transfer coefficients are represented in your data as negative heat transfer coefficients, is that correct?

James S. Moore: Yes, yes.

Daniel Ford: Thank you.

Do you agree that if you passed a saturated vapor, saturated steam through a furnace that you'd create superheated steam?

James S. Moore: If I pass saturated steam through a furnace I create superheated steam?

Daniel Ford: Yes.

James S. Moore: Yes.

Daniel Ford: Do the codes that you use for analyzing the loss-of-coolant accidents explicitly consider the formation of superheated steam or do they regard the coolant at different axial levels being simply liquid entrained in steam, period?

James S. Moore: It depends on which calculations you are talking about.

Daniel Ford: In the calculations that you have used for Indian Point 2 to calculate the maximum [fuel] clad temperature, have you separately considered the role of superheated steam in precipitating or yield[ing] the maximum clad temperature?

FLECHT tests would have been used for ECCS evaluation calculations for IP-2, because IP-2 is a PWR.

James S. Moore: In terms of reflooding, yes.

Daniel Ford: In terms of the code analysis that you have done, do you use negative heat transfer coefficients under any assumptions of flooding rate or pressure?

James S. Moore: If they would exist, yes. For the hot spot calculation, such a condition never does exist.

Daniel Ford: I see. In terms of the negative heat transfer coefficients that were observed, can you tell me at what axial levels these were observed?

James S. Moore: They were well above the hot spot. That is specifically the point. They were where the temperature was quite low [on] the [fuel] cladding.

Daniel Ford: Have you done any calculations which guarantee that the superheated steam, a negative heat transfer coefficient would always occur above the mid point?

James S. Moore: Yes.

Daniel Ford: Where are those calculations presented?

James S. Moore: Any one of these core cooling analyses were computed with the hot spot temperature. You can see the temperature itself is much greater than any saturated or even superheated condition that could exist.

Daniel Ford: Those are the calculations that you have presented. What I am asking is whether you have performed parametric calculations that indicate under no circumstances, that is, under no combination of parameters, which you get superheated steam at lower than the ten-foot elevations that it was observed at in the FLECHT test?

James S. Moore: Yes.[85]

It is rather odd that James S. Moore would answer, "yes," to Daniel Ford's last question in the portion of the IP-2 licensing hearing transcript quoted above, given the results of PWR FLECHT run 9573. Ford had asked Moore, "whether [Westinghouse had] performed parametric calculations that indicate under no circumstances, that is,

[85] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 8, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350639, pp. 2921-2924.

under no combination of parameters, [would there be] superheated steam at lower than the ten-foot elevations[, where] it was observed at in the FLECHT test[s],"[86] in the event of a LOCA?

Additionally, it is rather odd that James S. Moore would claim that "[f]or the hot spot calculation, [negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding would] never...exist."[87] And that "[negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding occurred] well above the hot spot. ... [That t]hey were where the temperature was quite low [on] the [fuel] cladding,"[88] given the results of PWR FLECHT run 9573.

It is significant that in FLECHT run 9573—a test conducted with a Zircaloy bundle—negative heat transfer coefficients were observed *at the bundle midplane* for 5 of 14 thermocouples and that steam temperatures exceeded 2500°F *at the seven-foot steam probe*, during a portion of the test.

This was reported in a Westinghouse document, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," in April 1971, months before James S. Moore's testimony.

Regarding the superheated steam, which exceeded 2500°F, and negative heat transfer coefficients observed at the bundle midplane in FLECHT run 9573, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states:

> At the time of the initial [heater element: fuel-cladding simulator] failures [in FLECHT run 9573], midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure). ...anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. These may have been related to the high steam probe temperatures measured at the 7 ft elevation.[89]

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[86] *Id.*, p. 2924.
[87] *Id.*, p. 2923.
[88] *Id.*
[89] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 3-97, 3-98.

It is also significant that, in 2002, regarding superheated steam being located at the hot spots of the fuel rod simulators in FLECHT run 9573, Westinghouse stated:

> The high fluid [superheated steam] temperature [that occurred in FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[90]

And discussing, in more detail, the superheated steam that was observed one foot above the midplane in FLECHT run 9573, a Westinghouse memorandum, written by Robert H. Leyse states:

> The final FLECHT test (Bundle Z-10) was completed on December 11, 1970. The test was run with flooding of 1 in./sec. beginning at 2000°F. Several heaters failed approximately 18 seconds after flooding when the peak indicated midplane temperature was 2325°F. Heater failure at this temperature is unlikely, particularly under conditions of decay heat and increasing temperature. *The steam probe thermocouple located one foot above midplane in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec.) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding.* It appears likely that *ignition of the Zircaloy grids led to high rates of heat input* at the elevation one foot above (and below) midplane* and this caused over-temperature and failure of the heaters. Test results are currently being studied.
>
> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.
>
> *The ratio of surface area to heat capacity for a Zircaloy grid is approximately 15 times that of a heater rod; hence, Zircaloy-steam reactions can lead [to] steeper temperature ramps in the vicinity of a Zircaloy grid [emphasis added].[91]

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[90] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

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

It is significant that "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states that the negative heat transfer coefficients that were observed at the bundle midplane in FLECHT run 9573 were "anomalous."[92] Perhaps that is why James S. Moore claimed that negative heat transfer coefficients would never occur at the hot spot of the fuel cladding, in the event of a LOCA. However, Westinghouse's conclusion that the negative heat transfer coefficients observed in FLECHT run 9573 were anomalous had no scientific basis. For example, Westinghouse did not conduct any subsequent tests with Zircaloy bundles after FLECHT run 9573 (with similar test parameters) to confirm that the negative heat transfer coefficients were in fact anomalous. This is unfortunate, given the importance of the safety issues involved.

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by unconfirmed notions that negative heat transfer coefficients—calculated as a result of heat transfer from the coolant to the fuel cladding—would never occur at the hot spot of the fuel cladding, in the event of a LOCA.

I. A Portion of the IP-2 Licensing Hearing Transcript: Integral Experiments Versus Separate Effects Experiments

It is significant that in 1971, in the IP-2 licensing hearing, UCS was concerned that the metal-water reaction rates predicted to occur in IP-2's core, in the event of a LOCA, had not been confirmed by data from large-scale integral experiments. In the portion of the IP-2 licensing hearing transcript quoted below, Daniel Ford of UCS questions the validity of the Baker-Just correlation for use in ECCS evaluation calculations and points out that the Baker-Just correlation was "derived from experimental data that is completely outside of the context of nuclear systems;" *i.e.*, from single-rod separate-effects experiments.

The 1971 IP-2 licensing hearing transcript states:

> James S. Moore: No. There are no large-scale tests for the core. You are talking about a very complex chain of events. You are ending up with a zirc-water reaction. And you have to start with the loss of coolant and go through the blowdown, the reflood, the heat-up, the time and temperature, and then the zirc-water reaction.

[92] F. F. Cadek, D. P. Dominicis, R. H. Leyse, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-98.

Daniel Ford: Right. Now in terms of simply your experimental philosophy do you see the necessity, since there are as you note so many complicated factors behind any independent phenomenon, do you see the necessity, the experimental necessity for the kind of integral test that I am talking about or do you think that you can just test individual small components of the problem, you know, assuming all the input from other phenomena?

James S. Moore: I believe it is my opinion that we can properly bound the calculation without a total completely integrated test.

...

Daniel Ford: I am talking about the water reactor safety program which has a variety of experiments on different... Using a variety of different equipment to simulate loss-of-coolant accident[s]. And I am talking about some of the large-scale experiments that are planned to take place [in] 1975 or so in which we will actually have a live reactor and have it subjected to loss-of-coolant transients and see what happens. I am talking about whether or not that is necessary in Mr. Moore's opinion, whether that would make a substantive contribution to the confirmation of these results on metal-water reactions inasmuch as they depend on all the other phenomena of the transient. I am asking him whether that is necessary or whether you can simply take Baker-Just's correlation, which is derived from experimental data that is completely outside of the context of nuclear systems? I am asking him whether we should have these kinds of integral experiments or whether we can just take empirical correlations and just use them with no hesitation...

James S. Moore: I count at least four or five questions in it. Do I think it necessary, do I think it would contribute?

Daniel Ford: I am purposely trying to find out what your philosophy is, what you regard as convincing experimental confirmation of, in this particular case, the metal-water reaction rates that you compute.

...

James S. Moore: In my opinion the totally integrated test is not necessarily a prerequisite to describe a physical phenomenon and in the case of the loss of coolant I don't think it is a requirement. I think you can get very good indications of what phenomena do occur with these separate effects kinds of experiments that have been performed. With respect to zirc-water reaction I would point out that we have come very close to simulating this through the FLECHT test[s].

Daniel Ford: Now in terms of the water reactor safety research program would you tend not to think that the integral tests were ever really worth their expenditure?

James S. Moore: I didn't say that. Are you asking that question?

Daniel Ford: Yes.

James S. Moore: It's my opinion we will get useful information out of that test, yes.

Daniel Ford: Are there any specific uncertainties that in relation to which the output of these tests will provide useful information?

James S. Moore: None specifically that I am aware of.

Daniel Ford: In terms of the experiments pertaining to accumulator water, are there any that have confirmed in any kind of integral way your own metal-water [reaction] prediction for Indian Point 2?

James S. Moore: I am again having trouble relating between [the] metal-water reaction and accumulators. Could we repeat the question again? That's a long train, from the accumulator to the metal-water reaction.

Daniel Ford: I see. Well your prediction of metal-water reactions as a function of accumulator water, the total reaction rate, has that prediction of yours been confirmed by any experiments?

James S. Moore: No specific experiment, complete integrated experiment.[93]

Unfortunately, to this day, nearly 40 years after the original IP-2 licensing hearing, the metal-water reaction rates predicted to occur in the event of a LOCA at IP-2 still have *not* been confirmed by data from large-scale integral experiments. In fact, now in 2010, there is a preponderance of metal-water-reaction-rate data from multi-rod severe fuel damage experiments like the LOFT LP-FP-2 experiment; nevertheless, IP-2's ECCS evaluation calculations still use metal-water reaction rate correlations that were derived from the data of single-rod experiments.

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[93] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2550-2553.

The LOFT LP-FP-2 experiment, conducted in 1985, is considered "particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[94] In the LOFT LP-FP-2 experiment, "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about...[2060°F]"[95]—approximately 140°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

J. The First Transient Experiment of a Zircaloy Fuel Rod Cluster in TREAT and the Baker-Just Correlation

1. Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT

In this section Petitioner discusses "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, from March 1971 and the Baker-Just correlation. The First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") was conducted in the Transient Reactor Test Facility ("TREAT").

Describing the FRF-1 experiment, the abstract of "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states:

> The first fuel rod failure experiment in Transient Reactor Test Facility ("TREAT") was performed with a seven-rod bundle of 27 in. long Zircaloy-clad UO_2 fuel rods in flowing steam atmosphere. A water reactor loss-of-coolant accident was simulated by operating TREAT reactor at constant power for 20 sec so that fission heat in the UO_2 pellets caused the Zircaloy cladding temperature to rise 72°F/sec to a maximum of approximately 1800°F. The fuel rods were initially pressurized with helium between 115 and 215 psia (77°F) to simulate accumulated fission gas.

[94] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 3.23.

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

> The Zircaloy cladding swelled and ruptured resulting in 48% blockage of the bundle coolant channel area at the location of maximum swelling. The average rod maximum circumferential swelling was 36%. Calculations related the hoop stress and ultimate strength at the onset of rapid expansion. The ideal gas law was used to calculate the rate of cladding expansion from measured rod temperature and internal pressure. Metallographic examination revealed ductile ruptures and significant oxygen pickup. Zirconium-steam reaction was 0.2%.[96]

And describing the Zircaloy-steam reaction that occurred in the FRF-1 experiment, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states:

> [W]e estimate the volume of hydrogen generated by metal-water reaction to be 1.2 ± 0.6 liters (STP). This is equivalent to about 0.2% metal-water reaction based on total cladding volume.[97]

It is significant that "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" states that the maximum Zircaloy cladding temperature was approximately 1800°F[98] and that the volume of hydrogen generated by the metal-water reaction was estimated to be 1.2 ± 0.6 liters, which, in turn, was estimated to have been caused by approximately a 0.2% metal-water reaction of the total cladding volume. Because the volume of hydrogen generated by the metal-water reaction was estimated to be 1.2 ± 0.6 liters, it would have made sense for "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT" to have also reported that the metal-water reaction was estimated to be 0.2% ± 0.1%.

2. Discussions of the Results of FRF-1 and the Baker-Just Correlation in the 1971 Indian Point Unit 2 Licensing Hearing

In 1971, in the IP-2 licensing hearing, the validity of the Baker-Just correlation for use in LOCA analyses was called into question, because data from the FRF-1 experiment indicated that in the experiment, at approximately cladding temperatures of 1800°F, the metal-water reaction had generated approximately 1.2 ± 0.6 liters of

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[96] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, Abstract.
[97] *Id.*, p. 16.
[98] See Appendix A Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1.

hydrogen. In the IP-2 licensing hearing, the Baker-Just correlation was criticized, because AEC had stated that at 1800°F, LOCA analyses using the Baker-Just correlation predicted that the metal-water reaction is "negligible."[99]

In AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[100]

And in two selections from the transcript of the IP-2 licensing hearing, from November 1, 1971, below, Anthony Z. Roisman—on behalf of Citizens' Committee for the Protection of the Environment and Environmental Defense Fund—addresses the fact that LOCA analyses using the Baker-Just correlation predict that the metal-water reaction is negligible at 1800°F and that that result does not agree with the results of the FRF-1 experiment (reported in ORNL-4635).

On this topic, the transcript states:

> Anthony Z. Roisman: In ORNL-4635, the 0.2 per cent was determined to be the amount of metal-water reaction that had occurred in rods at the 1800 degree Fahrenheit level. You said that the report pointed out there could be an error of plus or minus fifty per cent. In short, [the metal-water reaction] could have been 0.3 per cent or 0.1 per cent.
>
> You also mentioned that two per cent of the cladding in the analysis that Westinghouse does is assumed to reach 1800 degrees Fahrenheit or more temperature.
>
> What percentage of metal-water reaction do you predict will occur for that, for those rods at 1800 degrees Fahrenheit?
>
> James S. Moore [of Westinghouse Electric]: We would predict zero.
>
> Anthony Z. Roisman: You would predict no metal-water reaction at 1800 degrees Fahrenheit?
>
> James S. Moore: Yes.

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[99] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.
[100] *Id.*

Anthony Z. Roisman: What about at 1900 degrees Fahrenheit?

James S. Moore: Well, now it is a function of how long you are at that temperature.[101]

Then, continuing on the same topic, the transcript states:

Anthony Z. Roisman: ... I'd like to go back to the metal-water reaction. I guess I am still a little unclear about this, ORNL-4635 had predicted 0.2 percent of metal-water reaction for rods at 1800 degrees Fahrenheit.

You said that you predict for this plant no metal-water reaction at 1800 degrees Fahrenheit.

James S. Moore: I don't think the Oak Ridge report predicted that. They assumed they measured it.

Anthony Z. Roisman: Can you tell me, is your basis for not predicting 0.2 percent metal-water reaction at 1800 degrees Fahrenheit based upon some experiments which Westinghouse has run or with which you are familiar, that demonstrate that there won't be any metal-water reaction at that temperature?

James S. Moore: These are based on experiments that have been performed by others. I'm not aware of any specific Westinghouse experiments in this area. But these are the experiments which were added to and summarized in the reference by Baker and Just that I believe you have from Argonne. That's the basis for the parabolic rate assumption.

Anthony Z. Roisman: Can you explain in a little more detail that the most recent Oak Ridge National Laboratory report doesn't require modification of that?

James S. Moore: I'm not an expert on Zirc-water reaction per se. But just looking at the report, it seemed to me, number one, this is one data point only. Also, it looked like it was pretty susceptible to interpretation and measurement in the way they derived the amount of hydrogen, and related this back to Zirc-water reaction.

So I think I really don't know the validity of that or any conclusions with respect to Zirc-water. It was not an experiment, as I understood, to specifically work on Zirc-water reaction aspects.

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[101] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350644, pp. 2152-2153.

Anthony Z. Roisman: Mr. Wiesemann, is this your area, the [Zirc-]water reaction? Mr. Moore testified that he is not really expert in that. Are you?

Mr. Wiesemann: No. However, I am knowledgeable in general. I don't think my expertise in Zirc-water reaction area is any greater than Mr. Moore's. With regard to the question he just answered, about tests performed by Westinghouse, I am personally aware of some exploratory type tests which were done a long time ago when we first went into the use of Zircaloy cladding in reactors, exploring just exactly the thing you were discussing. That is the temperature range of 1800 to 2000 degrees [Fahrenheit], and the effect of this type of condition on Zircaloy rods to confirm for ourselves that there was no significant metal-water reaction in that range of temperatures in order to confirm for ourselves. The results of this, as far as I know, were never published. In order to get further details on that, we would probably have to consult some of the people who actually performed those tests. I observed those tests but I was not actually performing the test myself.[102]

And in a selection from the transcript of the IP-2 licensing hearing, from November 2, 1971, below, Leonard M. Trosten—on behalf of Consolidated Edison Company of New York, Inc.—also addresses the Baker-Just correlation, FRF-1 experiment, and the Zircaloy-water reaction—in response to Anthony Z. Roisman's questions from the previous day, on the same subjects.

On this topic, the transcript states:

Leonard M. Trosten: ... Now with respect to the question which appears on the transcript page 1720 [a different page number than the current transcript], relating to the zirconium-water reaction, are you familiar with the question that was raised by Mr. Roisman yesterday concerning that matter?

Dr. Jack Roll [of Westinghouse Electric]: Yes, sir. I reviewed the transcript.

Leonard M. Trosten: Would you please comment with regard to the question raised by Mr. Roisman.

Dr. Jack Roll: I believe the context of the question was that based upon the results reported in the reference ORNL document [ORNL-4635] did we have any reason to re-evaluate our application, I believe, of the Baker-Just equation to a computation of degree of zirc-water reaction, and I believe that Mr. Moore provided essentially the answer that I would have

[102] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 1, 1971, pp. 2166-2168.

provided, that that is no we could not use that single data point to re-evaluate or reapply the Baker-Just equations.

As pointed out by Mr. Moore in yesterday's proceedings, the measurement of the extent of zirc-water reaction was in fact by an inferred route, and there were no direct measurements taken. There was a large uncertainty in the measurement of total hydrogen evolution during the experiment.

The subtraction of other known effects resulted in a fifty per cent uncertainty in the amount of hydrogen which can be associated or applied with the zirc-water reaction, and from this they inferred the two-tenths per cent raw metal-water reaction. This was then compared, presumably by Mr. Roisman, to indicate that perhaps there was more zirc-water reaction here than one would expect based on reported temperatures.

But however, I pointed out in the Oak Ridge report there was not a direct measurement of temperature and they point out that the effects of thermocouple effects themselves and the power distribution with the bundle it enters result in an uncertainty in the temperatures of the fuel during the experiment.

Therefore, one cannot make a direct inference on reported temperatures and lead yourself to the conclusion that the extent of zirc-water reaction was higher or much higher than would have been predicted by Baker-Just.

I'd like to add further that we have, as a part of our work, in particular under the FLECHT program, reviewed the extent of zirc-water reaction, under what we considered to be much more representative conditions, that is zircaloy clad fuel rods with our particular time and temperature histories and our particular coolant content, that is our particular water conditions, and I believe as reported in the documentation summarized in the FLECHT reports we find very good agreement with the Baker-Just equation, and so we believe in summary that the Oak Ridge report presents a single data point to germaneness to our specific application must be questioned inasmuch as the data point was not, the test was not run to substantiate the Baker-Just equation.

And secondly, in summary, the work that we have done under the FLECHT program and reported in the FLECHT reports we believe reaffirms our use of the Baker-Just equations in evaluating zirc-water reaction under our conditions of loss-of-coolant accident.[103]

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[103] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350642, pp. 2297-2299.

In the transcript of the IP-2 licensing hearing, from November 2, 1971, right after, Leonard M. Trosten questioned Dr. Jack Roll, Daniel Ford stated "[t]he authors of that Oak Ridge report, ORNL-4635,[104] contend that [the FRF-1 experiment] is the most realistic simulation of loss-of-coolant accident conditions to date;" then Daniel Ford asks Dr. Roll, "[d]o you dispute that claim?"[105]

It is significant that, discussing the FRF-1 experiment, in *A Distant Light: Scientists and Public Policy*, Henry W. Kendall states, "[h]ydrogen generated by zirconium-steam reactions was identified. In the words of the report:

> The Zircaloy cladding swelled and ruptured resulting in 48% blockage of the bundle coolant channel area at the location of maximum swelling. ...examination revealed ductile ruptures and significant oxygen pickup.[106]

The relevance of these results derives from the fact that the test 'was conducted under the most realistic loss-of-coolant accident conditions of any experiment to date.' "[107]

(It is noteworthy that "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," states that "[t]he transient test program [for Zircaloy-clad fuel rod clusters in the TREAT facility] is presently inactive because funding is not available"[108] and that "[s]upport of ORNL work on fuel rod failure is now scheduled to be terminated at the end of FY-71."[109]

So the experimental program that conducted "the most realistic loss-of-coolant accident conditions of any experiment to date"[110]—up to 1971—was not provided with funding so investigators could continue researching important safety issues.)

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[104] The authors of ORNL-4635 are R. A. Lorenz, D. O. Hobson, and G. W. Parker.
[105] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2300.
[106] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, Abstract.
[107] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer , New York, 2000, p. 43.
[108] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.
[109] *Id.*, p. ix.
[110] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43:

It is also significant that, in one of the selections from the transcript of the IP-2 licensing hearing above, from November 2, 1971, Dr. Roll opines that the FLECHT program provided a more realistic representation of the Zircaloy-steam reaction in a LOCA environment, than the FRF-1 experiment; and that the FLECHT results were in "very good agreement with the Baker-Just equation."[111]

And on this topic—to repeat a section of the IP-2 licensing hearing transcript quoted above, from November 2, 1971—Dr. Jack Roll of Westinghouse Electric states:

> I'd like to add further that [Westinghouse Electric has], as a part of our work, in particular under the FLECHT program, reviewed the extent of zirc-water reaction, under what we considered to be much more representative conditions, that is zircaloy clad fuel rods with our particular time and temperature histories and our particular coolant content, that is our particular water conditions, and I believe as reported in the documentation summarized in the FLECHT reports we find very good agreement with the Baker-Just equation, and so we believe in summary that the Oak Ridge report [ORNL-4635] presents a single data point [that at cladding temperatures of approximately 1800°F, the metal-water reaction generated approximately 1.2 ± 0.6 liters of hydrogen and that the metal-water reaction was estimated to be 0.2% ± 0.1%] to germaneness to our specific application must be questioned inasmuch as the data point was not, the test was not run to substantiate the Baker-Just equation.
>
> And secondly, in summary, the work that we have done under the FLECHT program and reported in the FLECHT reports we believe reaffirms our use of the Baker-Just equations in evaluating zirc-water reaction under our conditions of loss-of-coolant accident.[112]

Then, soon afterwards in the transcript, describing metalographic cross-sections that were taken from rods from the four Zircaloy PWR FLECHT tests, Dr. Jack Roll states:

> The measurement that [Westinghouse Electric] took in evaluating the result of our FLECHT test with regard to the extent of [the] zirc-water reaction were in fact metalographic cross-sections at various enlargements from which the experienced metalographers can infer [the] nature of the phases in the cross-section. That is they can determine the portion of the original Zircaloy which remains as original Zircaloy. That portion which is oxygen saturated, that portion which is in fact converted to zirconium

[111] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2299.
[112] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, p. 2299.

> oxide. With these direct measurements at a number of cross-sections, one can then calculate explicitly the quantity of zirconium which has been converted to zirconium dioxide and the quantity of zirconium which is oxygen saturated from which you can then determine the total quantity of zirconium which has in fact reacted in some way with the oxygen.
>
> ...
>
> I believe the technique of looking at zirconium and zirconium oxide is in itself a primary source of data and need not be substantiated somewhere else. The question is, how do we know what is the extent of zirconium and oxygen reaction. The answer is, you know this by looking at the quantity of zirconium which has been converted to zirconium oxide.[113]

So Dr. Jack Roll explains that it was through examinations of metalographic cross-sections that were taken from rods from the four Zircaloy PWR FLECHT tests that "the work that [Westinghouse Electric did] under the FLECHT program...reaffirms [the] use of the Baker-Just equations in evaluating [the] zirc-water reaction under [the] conditions of [a] loss-of-coolant accident.[114]

However, this is problematic—as Petitioner explained in PRM50-93 and in Petitioner's comments on PRM-50-93, dated March 15, 2010—because there is no metallurgical data from the locations of run 9573 that incurred runaway (autocatalytic) oxidation: Westinghouse did not obtain such data. To explain this problem more completely, in the next section, Petitioner will replicate the text from Petitioner's comments on PRM-50-93, dated March 15, 2010 in the section titled "Supplementary Information to PRM-50-93 Section III.C.1.h. Examining the Autocatalytic Metal-Water Reaction that Occurred during FLECHT RUN 9573."

3. There is No Metallurgical Data from the Locations of FLECHT Run 9573 that Incurred Runaway (Autocatalytic) Oxidation

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

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[113] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 2, 1971, pp. 2302-2303.
[114] *Id.*, p. 2299.

It is significant that, regarding local steam starvation conditions postulated to have occurred in the CORA-2 and CORA-3 experiments, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> [T]he temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward. It is reasonable to assume, that *the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred* [emphasis added].[115]

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

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

> Despite the severity of the conditions [of FLECHT Run 9573] and the observed extensive zirconium-water reaction, the oxidation was within the expected range and runaway oxidation [occurred] beyond 2300°F. ...
>
> Westinghouse notes that the metallurgical analyses performed for FLECHT Run 9573 indicated that the measured oxide thickness was still within the expected range for specimens heated as high as 2500°F.[116]

(When Westinghouse performed the metallurgical analyses for the assemblies from the four FLECHT Zircaloy tests, it compared the measured oxide layer thicknesses to Baker-Just correlation predictions[117]—"the expected range.")

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

[116] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, pp. 3-4.

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

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

> The petitioner did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. The petitioner also ignored the Westinghouse application of the Baker-Just correlation to these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients.[118]

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

Therefore, Dr. Jack Roll's conclusion that the metallurgical data from the four Zircaloy PWR FLECHT tests reaffirmed the use of the Baker-Just correlation for evaluating the Zircaloy-steam reaction in the conditions of a loss-of-coolant accident is incorrect.

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[118] *Id.*, pp. 21-22.

4. It is Incorrect that the Zircaloy-Steam Reaction is Negligible below 1900°F, as Computer Codes Using the Baker-Just Correlation Predict

In AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[119]

Indeed, computer codes using the Baker-Just correlation may calculate that the Zircaloy-steam reaction is negligible below 1900°F; however, experimental data from multi-rod experiments demonstrates that the Zircaloy-steam reaction is very substantial below 1900°F.

For example, discussing the fact that the Zircaloy-steam reaction was very substantial below 1900°F in the CORA-2 and CORA-3 experiments, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> As already observed in previous tests [(CORA Tests B and C)],[120] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C [1832°F]. This temperature escalation [several tens of degrees Kelvin per second[121]] is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles. An effectively moderated escalation would be observed for smaller initial heatup rates, because the growth of protective scale during steam exposure counteracts by decreasing the oxidation rate of the material.
>
> This explains the observation that the temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward

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[119] AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100130976, Question: Page 12.

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

[121] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," KfK 4378, p. 1.

> and downward. It is reasonable to assume, that the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred.[122]

So "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states that autocatalytic oxidation commenced at 1832°F in the CORA-2 and CORA-3 experiments: peak cladding temperatures started increasing at several tens of degrees Fahrenheit per second.

Furthermore, it is also significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory" ("In-Vessel Phenomena—CORA"), presented in 1991, explicitly states that "[c]ladding oxidation [in the CORA-16 experiment] was not accurately predicted by available correlations."[123] (In 1991, the Baker-Just correlation was among the available correlations.)

Discussing "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[124] a BWR severe fuel damage experiment, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> The predicted and observed cladding thermal response are in excellent agreement *until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted.*
>
> ... Dr. Haste pointed out that he is chairing a committee (for the OECD) which is preparing a report on the state of the art with respect to Zircaloy oxidation kinetics. He will forward material addressing the low-temperature Zircaloy oxidation problems encountered in the CORA-16 analyses to ORNL [emphasis added].[125]

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

[123] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," CONF-9105173-3-Extd.Abst., Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

[124] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," ORNL/FTR-3780, October 16, 1990, p. 3.

[125] *Id.*

So, in the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations"[126] and "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted."[127] This indicates that available correlations—including the Baker-Just correlation—are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

Clearly, the Zircaloy-steam reaction is not negligible below 1900°F, as experimental data from multi-rod experiments demonstrates. And ECCS evaluation calculations using the Baker-Just correlation under-predict the Zircaloy-steam reaction that would occur in a LOCA environment.

Therefore, the AEC licensing of Indian Point Unit 2, in the early 1970s, was partly qualified by non-conservative ECCS evaluation calculations that used the Baker-Just correlation.

(It is noteworthy that the current power levels at Indian Point Unit 2 were qualified by non-conservative ECCS evaluation calculations that used the Cathcart-Pawel correlation. In 1991, the Cathcart-Pawel correlation was among the available correlations that when used in computer codes failed to predict cladding oxidation in the CORA-16 experiment.)

K. The Atomic Energy Commission Emergency Core Cooling Systems Rulemaking Hearing

> A.E.C. lawyers, at a meeting for A.E.C. staff witnesses a few days before the start of the hearing, also addressed the question of what the staff would say during cross-examination. The A.E.C. witnesses were given a one-page instruction sheet entitled "Hints at Being a Witness." It contained fifteen numbered instructions. ...although they were appearing as expert witnesses and were testifying under oath, item number ten on the list

[126] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.
[127] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.

admonished them: "Never disagree with established policy."[128]—Daniel Ford

Before the hearing Rittenhouse and other Oak Ridge staff members who would testify also spoke with Alvin Weinberg, the director of the Oak Ridge lab. Unlike the managers at Idaho, Weinberg told his researchers to "act responsibly and tell the truth."[129] —Daniel Ford

In this section, Petitioner discusses the Atomic Energy Commission's ("AEC") emergency core cooling system ("ECCS") rulemaking hearing. PRM-50-93 addresses issues that were debated in the ECCS rulemaking hearings: reflood rates, the Full Length Emergency Cooling Heat Transfer ("FLECHT") tests, the metal-water reaction, and what the PCT limit should be in the event of a LOCA. (Of these subjects, PRM-50-95 addresses the metal-water reaction and the PCT limit.)

In this section, Petitioner extensively quotes from "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"[130] the concluding statement of Henry. W. Kendall and Daniel F. Ford of Union of Concerned Scientists ("UCS"), on behalf of Consolidated National Intervenors ("CNI"), in the AEC ECCS rulemaking hearing. "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" provides a concise summary of reactor safety issues, debated in the AEC ECCS rulemaking hearing, including reactor safety issues that have not been resolved since 1973, when the hearing concluded.

Petitioner also quotes from "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," *Meltdown: The Secret Papers of the Atomic Energy Commission*, ORNL Review Vol. 25, Nos. 3 and 4, Chapter 6, "Responding to Social Needs," and *A Distant Light: Scientists and Public Policy.*

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[128] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, Simon & Schuster, New York, 1986, p. 119.
[129] *Id.*, p. 123.
[130] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," Concluding Statement—Safety Phase—Prepared by Union of Concerned Scientists on Behalf of Consolidated National Intervenors in the Matter of Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants, AEC Docket RM-50-l, April 1973.

1. Some of the Reactor Safety Issues Debated in the AEC ECCS Rulemaking Hearing, have Not been Resolved to this Day, Nearly Forty Years Latter

It is unfortunate that—despite extensive ECCS research, conducted after the rulemaking hearing concluded in 1973—some of the reactor safety issues, debated in the AEC ECCS rulemaking hearing, have not been resolved to this day, nearly forty years latter.

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

> In the years following the rulemaking [issued in January 1974], over $700 [million] has been spent by the NRC on research investigating ECCS performance. It is estimated that a similar amount has been spent by DOE (including AEC and ERDA), the U.S. industry, and foreign researchers, resulting in a total estimated expenditure of over $1.5 billion. The majority of this LOCA research is complete and has greatly improved the understanding of ECCS performance during a LOCA.[131]

Clearly, since 1988, substantial additional amounts of money have been spent on continuing LOCA research. So—in 2010 dollars—billions of dollars have been spent on LOCA research, yet NRC has ignored the data from LOCA research experiments that indicates that some of its regulations are not conservative enough to help ensure public safety.

For example, "Compendium of ECCS Research for Realistic LOCA Analysis," states that "[a]ssessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature;"[132] and that "even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists."[133] However, "Compendium of ECCS Research for Realistic LOCA Analysis" does not mention it is

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[131] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 8-1.
[132] *Id.*, p. 8-2.
[133] *Id.*

reported that in the LOFT LP-FP-2 experiment, autocatalytic oxidation commenced at cladding temperatures of approximately 2060°F[134].

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

> Data from [the LOFT LP-FP-2] experiment provide a wealth of information on severe accident phenomenology. The results provide important data on early phase in-vessel behavior relevant to core melt progression, hydrogen generation, fission product behavior... The experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[135]

2. A Brief Summary of the AEC ECCS Rulemaking Hearing

Regarding the AEC ECCS rulemaking hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The rulemaking hearing on reactor safety began in January 1972 and took place over a period of almost two years, until December 1973. The hearings...generated a record of more than 22,000 pages of transcript [of oral testimony] and thousands of pages of written direct testimony and exhibits.[136]

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

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

[136] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1086. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

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

> The [AEC's] Hearing Board consisted of Nathaniel H. Goodrich, Esq., presiding, Dr. Lawrence H. Quarles, and Dr. John H. Buck[, AEC employees[137]]. ... The primary participants included the Commission Regulatory Staff, four reactor manufactures, a consolidated group of electric utility companies, and the Consolidated National Intervenors ("CNI"), a group of about 60 organizations and individuals [UCS "served as the technical arm of CNI."[138]]. In addition, three states, the Lloyd Harbor Study Group, and several individuals participated to a lesser degree.[139]

Regarding the principal changes to AEC regulations, as a result of the hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The principal changes to the AEC's regulations were to lower the maximum allowed fuel cladding temperature, in the event of a loss-of-coolant accident ("LOCA"), from 2300°F to 2200°F, and to add a 17% local cladding oxidation limit.[140]

(It is noteworthy that "The History of LOCA Embrittlement Criteria" states that "the 17%-ECR[141] and 1204°C [PCT] criteria [of 10 C.F.R. § 50.46(b)] were primarily based on the results of post-quench ductility tests conducted by Hobson."[142] Furthermore, the experimental data that 50.46(b)(1) and (2) are primarily based on, is reported on in "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," ORNL-4758,[143] and "Ductile-Brittle Behavior of Zircaloy Fuel Cladding."[144]

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[137] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, Forward.
[138] *Id.*
[139] Dixy Lee Ray, *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1086.
[140] *Id.*, pp. 1130-1133.
[141] "ECR" is the initialism for "equivalent cladding reacted."
[142] G. Hache and H. M. Chung, "The History of LOCA Embrittlement Criteria," Proc. 28th Water Reactor Safety Information Meeting, Bethesda, USA, October 23-25, 2000, p. 10.
[143] D. O. Hobson and P. L. Rittenhouse, "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972.
[144] D. O. Hobson, "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, 26 March, 1973.

Additionally, it is noteworthy that "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," ORNL-4758, is currently (November 2010) "non-publicly available" in NRC's ADAMS Documents (Accession Number: ML082410413).[145] So one of the papers (from the early 1970s) that is one of the primary foundations of 50.46(b)(1) and (2) is non-publicly available in NRC's ADAMS Documents.)

And regarding AEC regulations that were not changed, as a result of the hearing, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The other three criteria [of the AEC's regulations were] retained, with some modification of the wording. These three criteria limit the hydrogen generation from metal-water reactions, require maintenance of a coolable core geometry, and provide for long-term cooling of the quenched core.[146]

(It is noteworthy that the AEC's ECCS rulemaking hearing generated a great deal of media attention; for example, on March 12, 1972, *The New York Times* reported: "A.E.C. EXPERTS SHARE DOUBTS OVER REACTOR SAFETY."[147])

Discussing the rulemaking hearing, the Oak Ridge National Laboratory ("ORNL") Review states:

> When protest greeted the AEC's interim criteria for emergency core cooling systems, [AEC Chairman James Schlesinger] convened...quasi-legal hearing[s] for comments from reactor manufactures, electric utility officials, nuclear scientists, environmentalists, and the public.[148]
>
> The hearings pitted the nuclear power industry against the opponents of nuclear power and seriously divided researchers at the AEC and its laboratories. Placed on the witness stand during heated adversarial legal proceedings, some scientists expressed confidence in the interim safety standards, and others did not.

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[145] This is stated in an e-mail to Petitioner from NRC Public Document Room, October 26, 2010.
[146] Dixy Lee Ray, *et al.*, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1130.
[147] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, pp. 126, 285 (Notes).
[148] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6.shtml

In a letter to Hans Bethe[149]...ORNL Director Alvin Weinberg,[150] pointed out that emergency cooling systems provided a final defense against [the] melting of [the] fuel in the case of a [LOCA] in the largest light water nuclear reactors. "And it makes me all the more unhappy," Weinberg concluded, "that certain quarters in the AEC have refused to take it seriously until forced by intervenors who are often intent on destroying nuclear energy!"[151]

Now that the AEC and nuclear industry had been called into account on this issue, Weinberg urged [ORNL] staff to offer their expertise fully and without reservation, regardless of whether they agreed with the existing criteria. Schlesinger agreed. Weinberg complained, however, that his staff should have been involved as fully in preparing the criteria as they would be in testifying at the hearings.

Among [ORNL] staff participating in [the] lengthy, sometimes contentious, sometimes tedious hearings were William Cottrell, Philip Rittenhouse, David Hobson, and George Lawson. They and other witnesses were grilled by attorneys for days. More than 20,000 pages of testimony were taken from scientists and engineers, who often expressed sharp dissent on technical matters concerning the adequacy of the safety program. Laboratory experts generally considered that existing criteria for reactor safety were based on inadequate research. ...

The [ORNL's] emphasis on reactor safety and environmental protection made it and Director Weinberg unpopular among some nuclear power advocates and members of the AEC staff—a strange turn of events for Laboratory scientists who had devoted their careers to inventing and advancing practical applications of nuclear energy. ...

Although other events and considerations also played a part, the ECCS hearings of 1972, no doubt influenced major management shifts in 1973 at [ORNL] and [the] AEC. More fundamentally, they influenced the federal government's subsequent decision to dissolve the AEC and to place its regulatory responsibilities and research- and development-related activities into two separate entities.[152]

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[149] Nobel laureate professor at Cornell University and former director of Los Alamos Scientific Laboratory's Theoretical Division.

[150] Theoretical physicist, Alvin Weinberg, "patented the first design for a water-cooled nuclear reactor;" see Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, p. 25.

[151] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6sb6.htm

[152] ORNL Review Vol. 25, Nos. 3 and 4, 2002, Chapter 6, "Responding to Social Needs;" text from web page located at: http://www.ornl.gov/info/ornlreview/rev25-34/chapter6.shtml

And discussing the rulemaking hearing, in *A Distant Light: Scientists and Public Policy*, Henry W. Kendall states:

> We discovered great vulnerabilities in the emergency systems required in all nuclear power plants. ...
>
> The safety issues we were documenting were quickly raised by intervenors in nuclear power plant construction and licensing hearings at a number of sites in the United States. So that the same safety matters would not be contested in numerous duplicate hearings, we and the AEC agreed that the issues would be pulled out of all local hearings and consolidated in a single [ECCS] rulemaking hearing in Washington. ...
>
> In preparing for the hearing, we very quickly discovered that we had uncovered a hornet's nest. The AEC had engaged in a far more extensive program of suppression of disconcerting safety information than anyone had ever imagined, had censored safety-related information, had pressured their own researchers to keep quiet on key issues, and was sitting on a mass of disquieting research results. In some cases, commission officials made public statements that were contrary to statements they had made in their internal records or reports. ...
>
> In the hearing, [Daniel F.] Ford, who had the instincts and skills of a fine lawyer, although without formal legal training, carried out extensive cross examination both of friendly and of hostile nuclear safety experts and managed, indeed stimulated, a flow of safety documents from whistleblowers in the AEC laboratories that had never been destined to see the light of day. ...
>
> My part was to digest the intricacies of the safety debate, help prepare our technical testimony, and defend it against attack by the 17 lawyers representing the electrical utilities, reactor manufactures, and the AEC who were participants in the hearing as well as having chosen the board who conducted the hearings and sat in judgment. ... I was on the witness stand five days a week for nearly a month, which must be close to a record for this sort of thing. With support from nuclear experts, some known only to us or wholly anonymous, we were able to withstand the numerous attempts to discredit us and our case. Nevertheless, the length and tension involved proved to be very wearing. A sadder consequence of the hearings was that the careers of a number of whistleblowers from the National Laboratories, who were identified during the hearings, were ruined by the AEC.
>
> While the direct result of the ECCS hearings was at best a minor improvement in reactor safety, Ford's work, combined with our written testimony, proved a major embarrassment for nuclear power. The

> testimony he elicited and the safety documents that were released were extraordinarily damaging to the AEC and contributed to the breakup of that agency by the Congress in January 1975.[153]

Discussing dissenting opinions regarding the effectiveness of ECCS and the AEC's and one of its contractor's attempts to intimidate witnesses prior to the rulemaking hearing, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> The testimony that the Hanauer task force[154] prepared for the hearing did not discuss...any of the...internal studies that conflicted with the official optimism about E.C.C.S. performance. It did not mention the fact that two of its members, Morris Rosen[155] and Robert Colmar, had strongly disagreed with its findings. ...
>
> The week before the hearing was to begin, Rosen had been told that the A.E.C. staff was being "reorganized," and he had found himself reorganized out of his job. When he explained to his superiors that this might look suspicious—that they might be accused of stifling dissent—he was given a job in another part of the staff, one where he no longer had any responsibility for E.C.C.S.
>
> The testimony presented by the Hanauer task force...made no reference to the dissenting opinion on the June 1971 policy statement[156] that had been expressed by the [Advisory Committee on Reactor Safeguards]. ...
>
> Shortly before the hearing the management at the Idaho lab [Aerojet Nuclear Company ("Aerojet")] met with staff researchers there and told them that they were free to say whatever they wanted at the hearing. But

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[153] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer , New York, 2000, pp. 14-15.

[154] The A.E.C.'s task force "was headed by Dr. Stephen Hanauer, who had served on the A.C.R.S. ... The other members of the task force...were Frank Schroeder, Edison Case, Marvin Mann, Victor Stello, Thomas Novak, Norman Lauben, Richard Tedesco, Warren Minners, Denwood Ross, Howard Richings, Paul Norian, Morris Rosen, and Robert Colmar. All of the members of the task force were engineers. Some of them had limited acquaintance with E.C.C.S. problems, but none was recognized as an expert in the field;" Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, 1986, pp. 102-103.

[155] "Rosen was noted by Dr. Hanauer as being far more knowledgeable in most areas of ECCS than Hanauer, who headed the Regulatory Staff ECCS task force; Milton Shaw, likewise, referred to Rosen and noted Rosen's synoptic understanding of the ECCS issue;" see Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.31.

[156] The AEC's interim ECCS acceptance criteria for LWRs: "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors—Interim Policy Statement," U.S. Federal Register, Vol. 36, No. 125, June 29, 1971 and No. 244, December 18, 1971.

> the management could not assure them that they would still have a job after the hearing if their testimony displeased the A.E.C.[157]

(It is noteworthy that regarding Oak Ridge staff researchers who would testify, *Meltdown: The Secret Papers of the Atomic Energy Commission* states:

> Before the hearing Rittenhouse and other Oak Ridge staff members who would testify also spoke with Alvin Weinberg, the director of the Oak Ridge lab. Unlike the managers at Idaho, Weinberg told his researchers to "act responsibly and tell the truth."[158])

And discussing the fact that AEC had attempted to intimidate ORNL witnesses prior to the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Despite the strenuous efforts on the part of the Commission to prevent [ORNL] witnesses from presenting their concern regarding ECCS effectiveness and the technical validity of the Interim Acceptance Criteria, and despite efforts at censorship and the suppression of data quite analogous to Commission efforts with respect to Aerojet views, cross-examination of ORNL witnesses by CNI was able to obtain valuable testimony that served to stimulate a number of major subsequent developments in the hearing. One of these developments includes, of course, the later revision in the Regulatory Staff's analysis of its embrittlement criterion.[159]

Discussing the testimony of Philip L. Rittenhouse of ORNL in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [A] safety researcher from Oak Ridge appearing in the hearing was P. L. Rittenhouse, a man who was willing to speak candidly about his own feelings regarding such important ECCS issues as embrittlement and flow blockage but also to share with the public, as an insider, his knowledge concerning the extensive reservations among the AEC's safety researchers regarding ECCS effectiveness. ...
>
> Rittenhouse testified that the AEC Regulatory Staff had presented what were, in his judgment, "arbitrary" and "unreasonable" interpretations of available PWR FLECHT blockage plate experiments and BWR FLECHT test ZR-2, which involved simulation of flow blockage as a result of

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[157] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, pp. 117, 122.
[158] *Id.*, p. 123.
[159] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.23.

> internal pressurization. Rittenhouse also provided extensive criticisms of the grievous misspecification of the embrittlement criterion in the Interim Policy Statement. Rittenhouse was able to provide important insight into the manner in which the AEC Regulatory Staff had avoided the substantive issues associated with flow blockage, had misunderstood the work that he had done, had merely "guessed" what the magnitude of flow blockage might be in a reactor, and had followed a procedure of making extrapolations that he did not believe constituted "good engineering practice." ...
>
> (It is an insight into the preparation of the Interim Policy Statement to note that Rittenhouse, who is identified by the Regulatory Staff as an expert on flow blockage and on embrittlement, the head of AEC research in those areas, was *never* consulted by the Regulatory Staff in connection with the preparation of the Interim Acceptance Criteria [emphasis not added].) ...
>
> [Rittenhouse] spoke with a clarity and candor that established an important precedent for all of the AEC safety research [personnel who] would [testify after him]. He made clear in his presentation, in the face of substantial institutional pressures exerted through written instructions to AEC witnesses to "never disagree with established policy,"[160] that this proceeding was the opportunity for those men to come forward with a full and honest technical evaluation of [ECCS] problems.[161]

(It is noteworthy that regarding written instructions to AEC witnesses, *Meltdown: The Secret Papers of the Atomic Energy Commission* states:

> A.E.C. lawyers, at a meeting for A.E.C. staff witnesses a few days before the start of the hearing, also addressed the question of what the staff would say during cross-examination. The A.E.C. witnesses were given a one-page instruction sheet entitled "Hints at Being a Witness." It contained fifteen numbered instructions. ...although they were appearing as expert witnesses and were testifying under oath, item number ten on the list admonished them: "Never disagree with established policy."[162])

And *Meltdown: The Secret Papers of the Atomic Energy Commission* states that at the end of his third day on the witness stand, Rittenhouse testified:

> I have worked in the fuel rod failure program [studying] the questions of fuel cladding and swelling, subsequent blockage, and the possible effects of this blockage on cooling effectiveness... As far as these points, in which I am an expert, there is not the information available to objectively

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[160] Exhibit 1013

[161] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 4.25, 4.26, 4.28.

[162] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, p. 119.

> confirm, by scientific or technical procedures, what exactly these materials-related phenomena... what effect they may have on the E.C.C.S. in the course of a loss-of-coolant accident.
>
> Beyond that, I can only say that I have talked to a number of [A.E.C. experts], people who work in the area of E.C.C.S., both the materials people, people who work primarily with [computer] codes, people who are experts, if you will, in heat transfer, fluid flow. And I get the genuine feeling from all of these people that they believe there are things we just do not know well enough yet. ... They have too many reservations—I believe shared too generally—for me to pass off. These reservations are primarily that certain phenomena, portions of the loss-of-coolant accident—maybe they're not quite sure what's going on. ...
>
> Certainly many of the things that we toss around in computer codes and use to predict maximum temperatures or to predict the course of the loss-of-coolant accident...have not been verified experimentally.[163]

"An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," states that Rittenhouse also testified:

> that he believed that there was a consensus that what might occur during a major LOCA is still open to question, and, [after he was asked for the names of colleagues who had expressed doubts about ECCS effectiveness,[164] he] read into the record the names of 28 [colleagues who] had influenced his own views concerning the serious unresolved problems [of ECCS].[165]

The colleagues Rittenhouse read into the record were George Brockett, Morris Rosen, Robert Colmar, George Lawson, Lawrence Ybarrando, Roger Griebe, Rex Shumway, and other nuclear power safety experts.[166] In the following weeks they also testified regarding "their own conclusions about the defects in the A.E.C.'s approval of current emergency-cooling-system designs.[167]

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[163] *Id.*, pp. 125-126.
[164] *Id.*, pp. 126-127.
[165] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 4.27.
[166] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, p. 127.
[167] *Id.*

Discussing the testimony of C. George Lawson of ORNL in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> One of the engineers from the Oak Ridge National Laboratory who testified was C. George Lawson, an expert in heat transfer. ...
>
> In response to CNI's question, Lawson affirmed: "There exists at this time [March 19, 1972] such a limited amount of information of the behavior during a loss-of-coolant accident of Zircaloy-clad fuel rods internally pressurized with fission gas that a conclusion of the adequacy of these emergency [core cooling] systems would be speculative."
>
> Lawson rejected "subjective" assessments of ECCS capabilities and stressed the need for experimental demonstrations of system performance. In the light of what Lawson regarded as "objective scientific confirmation" he said that such confirmation of ECCS effectiveness is not available and that ECCS effectiveness, he concludes, is "undemonstrated."
>
> At one point in his testimony Lawson was led to speculate on whether reactors were "safe," but his testimony clearly affirmed that if standards of "reasonable assurance" of ECCS effectiveness were applied then it must be concluded that such "reasonable assurance" does not exist with regard to presently designed emergency core cooling systems and that if one wished to rely on the Interim Acceptance Criteria they must be experimentally verified.[168]

Discussing a letter Alvin Weinberg, Director of Oak Ridge, wrote to the AEC Chairman at the beginning of the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> The concerns expressed by the witnesses from Oak Ridge National Laboratories who appeared in the proceeding were also communicated to the AEC by the respected Director of Oak Ridge, Alvin Weinberg. He affirmed the fundamental doubt of his researchers concerning the lack of adequate experimental proof of ECCS effectiveness in a February 9, 1972 letter to AEC Chairman James Schlesinger.
>
> [Weinberg's letter states:] "...I have a basic distrust of very elaborate calculations of complex situations, especially where the calculations have not been checked by full-scale experiments. As you know, much of our trust in the ECCS depends on the reliability of complex codes. It seems to me—when the consequences of failure are serious—then the ability of the

[168] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 4.23, 4.24-4.25.

> codes to arrive at a conservative prediction must be verified in experiments of complexity and scale approaching those of the system being calculated. I therefore believe that serious consideration should be given first to cross-checking different codes and then to verifying ECCS computations by experiments on large scale and, if necessary, on full scale. This is expensive, but there is precedent for such experimentation—for example, in the full-scale tests on COMET and on nuclear weapons."[169]

Discussing the testimony of Dr. Morris Rosen of the AEC, regarding the ECCS expertise and statements of the employees of Aerojet, in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Dr. Rosen emphasized that the AEC Regulatory Staff does not, in-house, have sufficient technical expertise to do a professional evaluation of the vendor LOCA analysis models and that only with [Aerojet] involvement could a state of the art assessment of the evaluation models be performed.
>
> Dr. Rosen, who supervised [Aerojet] Technical Assistance work for the Regulatory Staff, stated that [Aerojet]: "represents the most significant source of information to the Atomic Energy Commission and the regulatory organization in the field of emergency core cooling.
>
> "I think that as a result of the fact of this hearing, the testimony presented at this hearing, there is an impressive array of the top talent in that organization in my opinion indicating some strong reservations as to the course of the evaluation of emergency core cooling with respect to the large cold leg break.
>
> "My opinion of, let's say, the testimony of George Brockett is, I believe—I don't know his exact title but I think it is manager of development, nuclear safety development at Aerojet, I think he came out strongly indicating that steam binding indeed was a problem.
>
> "I think he indicated perhaps that reductions in operating power levels were required.
>
> "Personal observation about Mr. Brockett: I think in my opinion one would classify him as perhaps one of the leading experts in this country in emergency core cooling, in my opinion, if not the leading expert.
>
> "I think when that man comes out and says there is a problem, I take note of it."[170]

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[169] *Id.*, pp. 4.28-4.29.
[170] *Id.*, pp. 4.7-4.8.

And discussing the testimony of witnesses from Aerojet collectively in the rulemaking hearing, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] has stated that its "fundamental concern" about ECCS is based on the "lack of fundamental data." Witnesses from that laboratory affirmed that reactor designs have gotten ahead of the Commission's understanding of reactor safety problems, because reactor safety research has not kept abreast with the requirements for it in the expanding U.S. nuclear power program.[171]

Discussing the biased manner in which the rulemaking hearing was conducted and its conclusion, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> The industry and A.E.C. staff filed testimony to rebut what Rittenhouse, Brockett, Rosen, Colmar, and the other scientists from the A.E.C. laboratories had said on the witness stand. All of the concerns expressed by these scientists, the Hanauer task force asserted, had been resolved. (The A.E.C. hearing board refused to allow further questioning of these A.E.C. experts, however, to determine whether they agreed with the resolution of their concerns.) At the conclusion of the hearing, the A.E.C. staff, led by Stephen Hanauer, recommended that the A.E.C. reaffirm its June 1971 approval of the existing E.C.C.S. designs—and the Commission did so. Its "final" policy statement on E.C.C.S. left its June 1971 "interim" policy statement essentially intact. ...
>
> While the hearing was still in progress...Edison Case, the Deputy Director of Licensing, told *The New York Times*, in response to an inquiry, that "no costly changes" would be imposed on the industry as a result of the hearings.[172] ...
>
> Case's *faux pas* in disclosing the A.E.C.'s intransigence on E.C.C.S. was the subject of a sardonic private note that Edward J. Bauser, the staff director of the Joint Committee on Atomic Energy, sent to A.E.C. Director of Regulation [L. Manning] Muntzing. "The recent article in *The New York Times* which is enclosed would, in my view, be very disturbing to those who still have faith in the integrity of the administrative process," he wrote. "It could also be useful to anyone who wishes to discredit the integrity of the A.E.C. licensing process."[173]

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[171] *Id.*, p. 4.8.
[172] *The New York Times,* July 16, 1972.
[173] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, pp. 127, 128.

And discussing personnel changes that occurred after the rulemaking hearing concluded, in *Meltdown: The Secret Papers of the Atomic Energy Commission*, Daniel F. Ford states:

> Philip Rittenhouse was removed as head of the fuel-rod-failure program at Oak Ridge under orders from Herbert Kouts, who was the senior A.E.C. official in charge of safety research. Alvin Weinberg...was replaced as director of the [Oak Ridge] laboratory. Rosen and Colmar had already been reassigned prior to the hearing, but Rosen decided subsequently that his chances for advancement were better outside the agency, and he left it. At the Idaho lab, senior personnel who had criticized the "established policy" found themselves, as one of them noted, switched from responsible positions to "nothing jobs." Some of them, like George Brockett, [looked outside of Aerojet for new employment].[174]

(It is noteworthy that before the AEC ECCS rulemaking hearing began, ORNL work on fuel rod failure was scheduled to be terminated at the end of 1971. Regarding this issue, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," states that "[t]he transient test program [for Zircaloy-clad fuel rod clusters in the TREAT facility] is presently inactive because funding is not available"[175] and that "[s]upport of ORNL work on fuel rod failure is now scheduled to be terminated at the end of FY-71."[176]

So the experimental program that conducted "the most realistic loss-of-coolant accident conditions of any experiment to date"[177]—up to 1971—was not provided with funding so investigators could continue researching important safety issues.)

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[174] *Id.*, pp. 128-129.
[175] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.
[176] *Id.*, p. ix.
[177] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, p. 43.

III. CONCLUSION

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 would help improve public and plant-worker safety.

Respectfully submitted,

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Dated: November 23, 2010

Appendix A Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1[1]

[1] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971, p. 14.

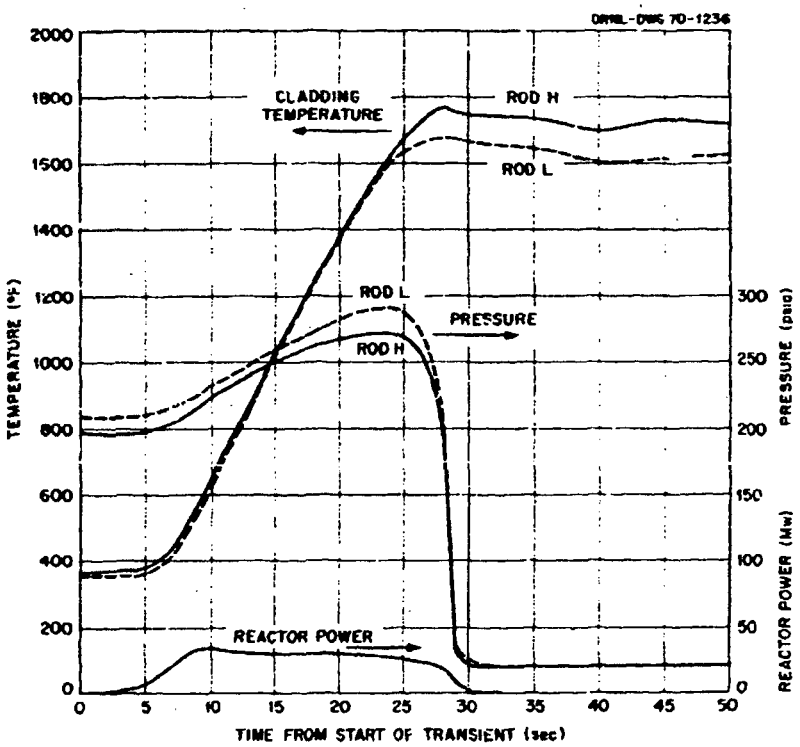

Fig. 4.3. Fuel Rod Temperatures and Pressures in TREAT Experiment FRF-1.

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<pdr.resource@nrc.gov>
CC: Dave Lochbaum <dlochbaum@ucsusa.org>, necnp@necnp.org,
Raymond Shadis <shadis@prexar.com>, "Powers, Dana A" <dapower@sandia.gov>
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