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January 20, 2010

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

Attention: Rulemakings and Adjudications Staff

Subject: Response to the Nuclear Regulatory Commission's ("NRC") notice of solicitation of public comments on risk-informed changes to loss-of-coolant accident technical requirements

Dear Ms. Vietti-Cook:

Enclosed is my response to the NRC's notice of solicitation of public comments on risk-informed changes to loss-of-coolant accident technical requirements, published in the Federal Register, August 10, 2009. I have responded to the three specific topics identified for public comment listed in Section VI on page 40038.

Respectfully submitted,

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January 20, 2010

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

MARK EDWARD LEYSE'S RESPONSES TO THE NUCLEAR REGULATORY COMMISSION'S NOTICE OF SOLICITATION OF PUBLIC COMMENTS ON RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS

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MARK EDWARD LEYSE'S RESPONSES TO THE NUCLEAR REGULATORY COMMISSION'S NOTICE OF SOLICITATION OF PUBLIC COMMENTS ON RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS

I. STATEMENT OF COMMENTATOR'S INTEREST

On March 15, 2007, Commentator, Mark Edward Leyse, submitted a petition for rulemaking, PRM-50-84 (ADAMS Accession No. ML070871368). PRM-50-84 was summarized briefly in the American Nuclear Society's *Nuclear News*'s June 2007 issue[1] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[2] by Union of Concerned Scientists. In 2008, the NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.

PRM-50-84 requests that the NRC make new regulations: 1) to require licensees to operate LWRs under conditions that effectively limit the thickness of crud (corrosion products) and/or oxide layers on fuel cladding, in order to help ensure compliance with 10 C.F.R. § 50.46(b) emergency core cooling system ("ECCS") acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.

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

On November 17, 2009, Commentator submitted a second petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests

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

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

that the 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;[3] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[4, 5]

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[6] 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.[7]

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

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[3] 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.

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

[5] 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.

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

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

Commentator is responding to "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," which the NRC published in the Federal Register on August 10, 2009, primarily because Commentator is aware of deficiencies in the NRC's and nuclear industry's ECCS evaluation models.

For example, data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F is non-conservative. During the LOFT LP-FP-2 experiment, when peak cladding temperatures reached between approximately 2060°F[8] and 2240°F,[9] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec.;[10] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[11]

So, in the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[12] and 2240°F,[13] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of

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[8] 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.

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

[10] *Id.*

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

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

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

approximately 18°F/sec. to 36°F/sec.[14] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[15] the melting point of Zircaloy is approximately 3308°F.[16]

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

> One of the bases for selecting 2200°F (1204°C) as the PCT [limit] was that it provided a safe margin, or conservatism, away from an area of zircaloy oxidation behavior known as the autocatalytic regime. The autocatalytic condition occurs when the heat released by the exothermic zircaloy-steam reaction (6.45 megajoules per kg zircaloy reacted) is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant. This reaction heat then further raises the zircaloy temperature, which in turn increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.[17]

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

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

[14] *Id.*

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

[16] 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.

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

> sensitivity and without reducing the margin that the Commission would have perceived in 1973.
>
> To explore this sensitivity further, we performed more than 50 LOCA calculations with RELAP5/Mod3. In about half of the cases, the Baker-Just equation was used for the metal-water heat generation rate, and in the other half, the Cathcart-Pawel equation was used. Reactor power just prior to the LOCA was varied parametrically to simulate incremental variations in decay heat. The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F. Each series of calculations done with the two metal-water models always showed peak cladding temperatures without runaway to be at least 100°F higher with Cathcart-Pawel, which is consistent with the temperature difference in the rate equations. Thus in these calculations, the margin between 2300°F and the calculational instability using Cathcart-Pawel was always equal to or greater than the margin between 2200°F and the calculational instability using Baker-Just.[18]

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

Clearly, there are deficiencies in the NRC's and nuclear industry's ECCS evaluation models that need to be reviewed and corrected before the NRC revises 10 C.F.R. § 50.46(a).

(For additional information on experimental data that indicates ECCS evaluation models are not realistic see Section III "Background.")

[18] "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.

Regarding the NRC's proposed revisions to 10 C.F.R. § 50.46(a) the NRC's Advisory Committee on Reactor Safeguards ("ACRS"), in SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" May 16, 2007, Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," states:

> It is likely that with this rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS [(transition break size)]. The rule currently contains acceptance criteria for fuel cladding performance under LOCA conditions based on the current 10 CFR 50.46. The Office of Nuclear Regulatory Research is now completing an examination of the adequacy of these criteria for high-burnup fuel. The adequacy of the acceptance criteria for cladding performance is important to maintain adequate safety margins. The rule should not be finalized until the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS are reviewed and/or revised to assure their adequacy for the higher burnup fuel and more demanding conditions of current reactor operating conditions. Alternatively, the acceptance criteria in the rule could be expressed in terms of general requirements, such as a high degree of confidence in maintaining a coolable geometry and retaining some ductility in the cladding. Specific cladding and core criteria could be placed in the associated regulatory guide.[19]

In response to the ACRS, the NRC staff, in SECY-07-0082, states:

> The staff agrees with the ACRS view that it is preferable to complete the review and revision of the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS before finalizing the § 50.46a rulemaking. Such an approach would assure that the issue of adequate safety margin with regard to cladding oxidation is addressed in a generic, structured rulemaking prior to any potential implementation under § 50.46a. This is a logical sequence because changes proposed by licensees adopting § 50.46a will likely result in more demanding reactor operating conditions that may further stress the fuel, or result in small break LOCAs becoming limiting. In addition, the trend toward higher fuel burnups where oxidation effects are most pronounced is expected to continue.

[19] NRC, SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," May 16, 2007, Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," pp. 4-5.

> Thus cladding safety margin considerations are likely to be important issues in § 50.46a applications.
>
> Although proceeding with the § 50.46a rulemaking by incorporating general cladding acceptance criteria could also be considered, resolution of safety margin questions would then be on a plant specific basis. Plant specific resolution is likely to complicate consistency in the regulatory process. In addition, incorporating general criteria in the near term would also result in the need for a subsequent rule change to § 50.46a when the cladding rulemaking is completed.
>
> Accordingly, the staff agrees with the ACRS that assuring the adequacy of the cladding oxidation criteria before implementing the § 50.46a rulemaking is a more appropriate approach for assuring that adequate safety margins are maintained and for assuring consistency in rule implementation.[20]

So, in 2007, both the ACRS and NRC staff agreed "that it is preferable to complete the review and revision of the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS before finalizing the § 50.46a rulemaking."[21] And, in SECY-07-0082, the NRC staff states that "[t]his is a logical sequence because changes proposed by licensees adopting § 50.46a will likely result in more demanding reactor operating conditions that may further stress the fuel, or result in small break LOCAs becoming limiting."[22]

Therefore, it would also be logical to review and correct the deficiencies in the NRC's and nuclear industry's current ECCS evaluation models, before finalizing the 10 C.F.R. § 50.46(a) rulemaking. For example, 10 C.F.R. § 50.46(b)(1) should be revised so that it is based on data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment); the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

It is also pertinent that in "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" the NRC states:

> As previously discussed in...this document, the NRC is working to revise the ECCS acceptance criteria in § 50.46(b) to account for new experimental data on cladding ductility and to allow for the use of

[20] *Id.*, p. 5.
[21] *Id.*
[22] *Id.*

> advanced cladding alloys. ... The NRC expects that this rulemaking...will establish new cladding embrittlement acceptance criteria in § 50.46(b) for design basis LOCAs. As these new acceptance criteria are established, the NRC will also make conforming changes to § 50.46a as necessary for both below and above TBS breaks.[23]

In this case, it would still be logical to review and correct the deficiencies in the NRC's and nuclear industry's current ECCS evaluation models and "make conforming changes to § 50.46a as necessary for both below and above TBS breaks."[24]

II. COMMENTS ON THE NRC'S THREE SPECIFIC TOPICS REGARDING RISK-INFORMED CHANGES TO LOCA TECHNICAL REQUIREMENTS

Commentator's responses to "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," published in the Federal Register on August 10, 2009, are below. Commentator has responded to the three specific topics identified for public comment listed in Section VI on page 40038.

Commentator is responding to the three specific topics identified for public comment, primarily because Commentator is aware of deficiencies in the NRC's and nuclear industry's ECCS evaluation models. As stated above, Commentator discusses additional experimental data that indicates ECCS evaluation models are not realistic in Section III "Background" of this document. The information provided in Section III "Background" is part of Commentator's responses to each of the three specific topics identified for public comment.

A. Topic One

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" states that "[t]he NRC seeks specific public comments on three topics."[25] Regarding the first topic, the NRC states:

> Although the revised proposed rule would permit licensees to make plant changes that result in very small risk increases, the NRC is requesting stakeholder comments on whether the rule should allow plant changes that

[23] NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Federal Register, August 10, 2009, p. 40030.
[24] *Id.*
[25] *Id.*, p. 40038.

> increase risk at all. Instead of the risk acceptance criteria allowing very small risk increases, should the risk acceptance criteria in final rule require that the net effect of plant changes made under § 50.46a be risk neutral or risk beneficial? The NRC requests stakeholders to provide comments on the use of risk acceptance criteria that would not allow a cumulative increase in risk for plant changes made under § 50.46a. (*See* Section V.E.4.b of this document.)[26]

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Section V.E.4.b states:

> In § 50.46a(f)(2)(ii), CDF and LERF are used as surrogates for early and latent health effects, which are used in the Commission's Policy Statement on Safety Goals (51 FR 30028; August 4, 1986). The NRC has used CDF and LERF in making regulatory decisions for over 20 years. The NRC endorsed the use of CDF and LERF as appropriate measures for evaluating risk and ensuring safety in nuclear power plants when it adopted RG 1.174 in 1997. After the adoption of RG 1.174, the NRC has had eleven years of experience in applying risk-informed regulation to support a variety of applications, including amending facility procedures and programs (*e.g.*, IST and ISI programs), amending facility operating licenses (*e.g.*, power up-rates, license renewals, and changes to the FSAR), and amending technical specifications. On the basis of this experience, for current operating reactors, the NRC has determined that CDF and LERF are acceptable measures for evaluating changes in risk as the result of changes to a facility, technical specifications, and procedures, with the exception of certain changes that affect containment performance but do not affect CDF or LERF. Changes that affect containment performance are considered as part of the defense-in-depth evaluation.
>
> For new reactors, CDF and LRF [(large release of radioactive material to the environment)] (instead of LERF) would apply as indicated in § 50.46a(f)(2)(iii). For new reactor licensing the Commission has established a goal based on LRF (*see* SRM on SECY–89–102—Implementation of the Safety Goals, June 15, 1990; and SRM on SECY–90–016—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, June 26, 1990).
>
> The Commission has concluded that changes under this rule should be restricted to very small risk increases. As discussed in RG 1.174, a very small risk increase is independent of a plant's overall risk as measured by the current CDF and LERF. Increases in CDF of 10^{-6} per reactor year or less, and increases in LERF of 10^{-7} per reactor year or less are very small risk increases for existing reactor facilities.

[26] *Id.*

For new reactors, the same CDF metric is used and the same definition of very small increase (*i.e.*, less than 10^{-6} per reactor year) would be used. The revised proposed rule uses LRF instead of LERF as a metric for new reactors. RG 1.174 provides no guidelines for LRF. The Commission has approved the overall mean frequency of a large release of radioactive material to the environment (LRF) to be less than 10^{-6} per reactor year. The revised proposed rule requires the total increase in LRF to be no more than very small. The NRC proposes that increases in LRF of 10^{-8} per reactor year or less are very small risk increases for new reactors. Because of the difference between the LERF acceptance criteria for existing reactors and the LRF acceptance criteria for new reactors, the NRC is seeking specific public comments on this topic. Additional background information on how the NRC is addressing this issue and how the NRC is soliciting public input on this topic in this revised proposed rule and in other regulatory areas is provided in Section J.2. of this document.

After adopting RG 1.174 in 1997, the NRC has applied the quantitative change in risk guidelines to individual plant changes and to sequences of plant changes implemented over time. The NRC has found these guidelines and the CDF and LERF values (when used together with the defense in depth, safety monitoring, and performance measurement criteria) are capable of differentiating between changes, and sequences of changes, that are not expected to endanger public health and safety from those that might. The NRC believes that applying the LRF guideline for determining very small risk increases would also be protective of public health and safety.

Section 50.46a(f)(1) would permit licensees to make changes under this provision without prior review and approval if the changes involve minimal increases in risk which also have no significant impact upon defense-in-depth capabilities. A minimal risk increase is one which, when considered qualitatively by itself or in combination with all other minimal increases, would never become significant. Logically, a minimal increase is less than the very small increase in CDF and in LERF, and was chosen as an increase of less than 10^{-7} per reactor year for CDF and an increase in LERF of less than 10^{-8} per reactor year. Similarly, for new reactor licensing, an increase in LRF less than 10^{-9} per reactor year is a minimal increase. Although ten of these changes could cause the combination of minimal increases to exceed the very small criteria, the NRC believes that most of these changes will have a much smaller (and, in some cases, an unmeasurable) increase in risk. Regardless of whether a licensee makes changes under § 50.46a(f)(1) instead of § 50.46a(f)(2), the total cumulative risk including all the individually minimal risk increases as well as any increases approved by the NRC under § 50.46a(f)(2), would have to be considered in the periodic reporting required by § 50.46a(g)(2). If a licensee implements an unexpectedly large number of minimal risk changes, the periodic reporting requirements in § 50.46a(g)(2) would provide adequate notice to ensure that the NRC is aware of potentially significant changes (or any collective impact), so that the NRC may undertake additional oversight actions as deemed necessary and appropriate.

Additionally, although the revised proposed rule would permit licensees to make plant changes that result in very small risk increases, the NRC is requesting stakeholder comments on whether the rule should allow plant changes that increase risk at all. Instead of the risk acceptance criteria allowing very small risk increases, should the risk acceptance criteria in final rule require that the net effect of plant changes made under § 50.46a be risk neutral or risk beneficial? The NRC requests stakeholders to provide comments on the use of risk acceptance criteria that would not allow a cumulative increase in risk for plant changes made under § 50.46a.[27]

1. Comment on Topic One

The rule should not allow plant changes that increase risk at all. The NRC should decrease the probabilities of core damage frequency ("CDF") and "the frequency of...accidents leading to significant, unmitigated releases from [the] containment[28] in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects"[29] ("LERF"), rather than increase them. Commentator makes these suggestions primarily because there are deficiencies in the NRC's and nuclear industry's ECCS evaluation models that among other things indicate the criteria of 10 C.F.R. § 50.46(b) are non-conservative, which, in turn, indicates that the

[27] *Id.*, pp. 40033-40034.
[28] NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, p. 8, footnote 3, states that "[s]uch accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines."
[29] NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, p. 8, footnote 3.

probabilities assigned to CDF, LERF, and an occurrence of a large release of radioactive material to the environment ("LRF")[30] are erroneous.

It is significant that data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. Data from such experiments also indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. Therefore, 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* (which uses the Baker-Just equation) and any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations (which use the Cathcart-Pawel equation) are deficient for calculating the metal-water reaction rates that would occur in the event of a LOCA.

(For additional information on severe fuel damage experiments and experimental data that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative see Section III.C "The Metal-Water Reaction Rate.")

Additionally, Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized Water Reactors*—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"]"—is erroneously based on the assumption that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions. (For additional information on this issue see Section III.D "FLECHT Run 9573.")

[30] For new reactors, LRF is used as the acronym for a large release of radioactive material to the environment, instead of LERF.

Deficient ECCS evaluation models cannot realistically model the phenomena that would occur in the event of a LOCA. Furthermore, deficient ECCS evaluation models are potentially dangerous because they provide erroneous simulations of the phenomena that would occur in the event of a LOCA. For example, the ECCS evaluation calculations that helped qualify Indian Point Unit 2's ("IP-2") 2004 stretch power uprate, calculated IP-2's PCT at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies during a postulated large break ("LB") LOCA.[31] This is highly problematic because, with high probability, if there were a LB LOCA at IP-2, there would be a partial or complete meltdown.

This is demonstrated by examining data from multi-rod (assembly) severe fuel damage experiments. During the LOFT LP-FP-2 experiment, when peak cladding temperatures reached between approximately 2060°F[32] and 2240°F,[33] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec.;[34] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[35]

And the CORA experiments demonstrated that with good fuel assembly insulation—like what the core of a nuclear power plant has—that cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, starts when the cladding reaches between 2012°F and 2192°F; cladding temperatures then start increasing at a maximum rate of 27°F/sec.[36]

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

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

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

[34] *Id.*

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

[36] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor

So, in the event of a LOCA at IP-2, if peak cladding temperatures increased to between approximately 2060°F[37] and 2240°F,[38] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[39] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[40] the melting point of Zircaloy is approximately 3308°F.[41]

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

(In the pretransient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[42] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[43] "low enough that superheated steam conditions [would] exist at the loop inlet

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

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

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

[39] *Id.*

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

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

[42] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, 1981, located in ADAMS Public Legacy, Accession Number: 8104300119, p. 10.

[43] 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.

instrument location. The superheat requirement [was] imposed so that meaningful steam temperatures [could] be measured."[44])

The NRU Thermal-Hydraulic Experiment 1 ("TH-1") tests illustrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126 (reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.7 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

Compare this to some of the TH-1 tests that had reflood rates of 5.9 in./sec. or greater: test no. 120 (reflood rate of 5.9 in./sec.) had a PCT at the start of reflood of 1460°F and an overall PCT of 1611°F (an increase of 151°F), test no. 113 (reflood rate of 7.6 in./sec.) had a PCT at the start of reflood of 1408°F and an overall PCT of 1526°F (an increase of 118°F), test no. 115 (reflood rate of 9.5 in./sec.) had a PCT at the start of reflood of 1666°F and an overall PCT of 1758°F (an increase of 92°F).

It seems obvious that if the three TH-1 tests with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 30 seconds or higher, or had PCTs at the start of reflood that were 1200°F or higher, that the fuel assemblies, with high probability, would have reached temperatures exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(For additional information on how low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases see Section III.B "Reflood Rates.")

It is noteworthy that in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test] program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[45]

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[44] *Id.*

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

It is also significant that there is little or no evidence that the thermal resistance of crud layers on fuel cladding have ever been properly factored into ECCS evaluation calculations for postulated LOCAs. (For information on this issue see Section III.E "The Thermal Resistance of Crud and/or Oxide Layers on Fuel Cladding and ECCS Evaluation Calculations for Postulated LOCAs.")

Clearly, the deficiencies of the NRC's and nuclear industry's ECCS evaluation models discussed above indicate that the probabilities assigned to CDF and LERF are erroneous.

It is significant that Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," states that "if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it;"[46] and states that "if there is an indication that the LERF may be considerably higher than 10^{-5} per reactor year, the focus should be on finding ways to decrease rather than increase it."[47]

It is highly probable that at nuclear power plants CDF and LERF are currently considerably higher than 10^{-4} per reactor year and 10^{-5} per reactor year, respectively, because ECCS evaluation models are deficient. Therefore, it is imperative that the NRC decrease the probabilities of CDF and LERF, rather than increase them.

It is significant that, regarding the NRC's proposed revisions to 10 C.F.R. § 50.46(a) the NRC's ACRS, in SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" May 16, 2007, Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," states:

> It is likely that with this rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS. The rule currently contains acceptance criteria for fuel cladding

[46] NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," p. 17.
[47] *Id.*

performance under LOCA conditions based on the current 10 CFR 50.46.[48]

And significant that, in response to the ACRS, the NRC staff, in SECY-07-0082, states that "changes proposed by licensees adopting § 50.46a will likely result in more demanding reactor operating conditions that may further stress the fuel, or result in small break LOCAs becoming limiting."[49]

So the NRC must not revise its regulations to allow for "design changes, such as increasing power [that] could cause increases in plant risk."[50] It is also imperative that the NRC not revise its regulations to "divide the current spectrum of LOCA break sizes into two regions"[51] and make "each break size region...subject to different ECCS requirements"[52] where "the smaller break size region [would] be analyzed by the methods, assumptions, and criteria currently used for LOCA analysis [and] accidents in the larger break size region [would] be analyzed by less conservative assumptions based on their lower likelihood."[53] Furthermore, "LOCAs for break sizes larger than the transition break [must not] become 'beyond design-basis accidents,' "[54] even if "the proposed rule would require licensees to maintain the ability to mitigate all LOCAs up to and including the [double-ended guillotine break ("DEGB")] of the largest [reactor coolant system ("RCS")] pipe during all operating configurations."[55]

(For additional information on experimental data that indicates ECCS evaluation models are not realistic see Section III "Background." The information provided in Section III "Background" is part of Commentator's responses to topic one.)

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[48] NRC, SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," p. 4.
[49] *Id.*, p. 5.
[50] NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," p. 40008.
[51] *Id.*
[52] *Id.*
[53] *Id.*
[54] *Id.*
[55] *Id.*

B. Topic Two

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" states that "[t]he NRC seeks specific public comments on three topics."[56] Regarding the second topic, the NRC states:

> Because of the difference in the risk acceptance criteria metrics used for currently operating reactors (LERF) and new reactors (LRF), the NRC is seeking public comments on whether LRF should be the metric of concern in lieu of LERF for new reactor applicants (or licensees) implementing the § 50.46a alternative ECCS requirements. Because the LRF goal for new reactors is a decade lower than the 10^{-5} per reactor year LERF reference value above which a facility would be limited to very small increases, should the definition of what constitutes "very small increase" and "minimal increase" for LRF (for new reactors) be a full decade lower than those defined for LERF (for existing reactors) or should the definition be based on *relative* change in LRF? (*See* Section V.J of this document.)[57]

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Section V.J states:

> As previously discussed under NRC Topic 1, the NRC has evaluated public comments and agrees with commenters who stated that there are no technical reasons which prevent the revised proposed § 50.46a regulations from being applied to new light water reactor designs that are similar in nature (with respect to design and expected LOCA pipe break frequency) to current operating reactors.
>
> 1. Similarity of New Reactor Designs to Existing Reactor Designs
>
> There are several new LWR designs for which the NRC expects that the frequency of large LOCAs could be as low as it is at current LWRs. Thus, it could be appropriate to allow applicants to apply the § 50.46a requirements to these future designs. Accordingly, the revised proposed rule has been modified to apply to new LWR reactor designs; *i.e.* facilities other than those which are currently licensed to operate. Applicants for design certification or combined licenses, holders of combined licenses under 10 CFR part 52, or future licensees of operating light-water reactors who wish to apply § 50.46a must submit an analysis for NRC approval demonstrating why it would be appropriate to apply the alternative ECCS requirements and what the appropriate transition break size (TBS) would be in order for the new design to meet the intent of the § 50.46a rule.

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[56] *Id.*, p. 40038.
[57] *Id.*

In its analysis, the applicant, holder, or licensee must demonstrate that the proposed reactor facility is similar to reactors licensed before the effective date of the rule. In addressing similarity of the proposed design to reactors licensed before the effective date of rule, the applicant, holder, or licensee would need to address design, construction and fabrication, and operational factors that include, but are not limited to:

(1) The similarity of the piping materials of construction and construction techniques for new reactors to those in the currently operating fleet;

(2) The similarity of service conditions and operational programs (*e.g.*, in-service inspection and testing, leak detection, quality assurance etc.) for new reactors to those for operating plants;

(3) The similarity of piping design, *e.g.* pipe sizes and pipe configuration, for new reactors to those found in operating plants;

(4) Adherence to existing regulatory requirements, regulatory guidance, and industry programs related to mitigation and control of age-related degradation (*e.g.*, aging management, fatigue monitoring, water chemistry, stress corrosion cracking mitigation *etc.*); and

(5) Any plant-specific attributes that may increase LOCA frequencies compared to the generic results in NUREG–1829 and NUREG–1903.

The analysis must also include a recommendation for an appropriate TBS and a justification that the recommended TBS is consistent with the technical basis for this proposed rule. For those new reactor designs that employ design features that effectively increase the break size via opening of specially designed valves to rapidly depressurize the reactor coolant system during any size loss of coolant accident, justification of the relevance of a TBS would also be necessary. The methodology used to determine the proposed TBS should be described in the justification.

Based on information currently available, new reactor designs may have similar piping materials, similar service conditions and operational programs, similar piping designs, and similar mitigation and control of age-related degradation programs to those found in currently operating plants. Therefore, the TBS defined in the proposed rule for currently operating reactors could potentially be applicable to some new reactor designs.

In addition, after obtaining an operating or combined license for a plant with a currently-approved standard design, a licensee could adopt § 50.46a if the design is demonstrated to be similar to the designs of plants licensed before the effective date of the rule (by evaluating the criteria above) and the TBS proposed by the licensee is found acceptable by the NRC.

2. NRC Request for Public Comments on the Use of Large Release Frequency
(LRF) as the Risk Acceptance Criteria Metric for New Reactors

Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis," was originally issued in July 1998. This RG provides guidance for a multitude of risk-informed applications and improves consistency in regulatory decisions in areas where the results of risk analyses are used to help justify regulatory action. The guide is the foundation for many other risk- informed programs (*e.g.*, inservice testing, inservice inspection of piping) at the agency.

Regulatory Guide 1.174 describes five key principles of the risk-informed, integrated decision making process. In Principle 4—When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement—the regulatory guide presents quantitative guidelines for acceptably small increases in CDF and LERF, as depicted in Figures 3 and 4 of the guide. The magnitude of acceptably small increases varies stepwise with the baseline CDF and LERF. A small increase up to 10^{-5} per reactor year for CDF and 10^{-6} per reactor year for LERF are normally acceptable until the baseline risk increases to reference values of approximately 10^{-4} per reactor year and 10^{-5} per reactor year for CDF and LERF, respectively. Plants with baseline CDF and LERF which exceed the reference values, or with baseline risks that are not known with precision, would normally be limited to very small risk increases of up to 10^{-6} per reactor year and 10^{-7} per reactor year for CDF and LERF, respectively. Before RG 1.174 was issued, the Commission's SRM dated June 26, 1990, prepared in response to SECY–90–016, "Evolutionary Light Water Reactor Certification Issues and their Relationships to Current Regulatory Requirements," established a goal for large release frequency (LRF) of less than 10^{-6} per reactor year for new reactor design certification and licensing. These goals are discussed further in Standard Review Plan (NUREG–0800) Chapter 19, and RG 1.206 "Combined License Applications for Nuclear Power Plants" Section C.I.19.

In light of this difference in the risk metrics used for currently operating reactors (LERF) and new reactors (LRF), the NRC is seeking public comments on whether LRF should be the metric of concern in lieu of LERF for new reactor applicants (or licensees) implementing the § 50.46a alternative ECCS requirements. Because the LRF goal for new reactors is a decade lower than the 10^{-5} per reactor year LERF reference value above which a facility would be limited to very small increases, should the definition of what constitutes "very small increase" and "minimal increase" for LRF (for new reactors) be a full decade lower than those defined for LERF (for existing reactors) or should the definition be based on *relative* change in LRF?

The NRC has previously sought stakeholder input on the issue of risk metrics for new light-water reactors. A memorandum dated February 12, 2009, from R. W. Borchardt, Executive Director for Operations, to the Commissioners, "Alternative Risk Metrics for New Light-Water Reactor Risk-Informed Applications" (Adams Accession No. ML090160008), provides a discussion of the issues. The white paper attached to that memorandum presents a full discussion of the issues and options for applying or modifying the current set of reactor risk metrics to new reactors. The paper discusses the issues posed by the lower risk estimates of new reactors in risk-informed applications, including changes to the licensing basis and the reactor oversight process, and describes the advantages and disadvantages of each option.

On February 18, 2009, the NRC held a public meeting with stakeholders on the topic of risk metrics for new light-water reactors (see meeting summary; Adams Accession No. ML090570356). Additionally, both the NRC and industry representatives provided a briefing on the topic at the April 3, 2009, meeting of the ACRS.

As discussed in these documents, the NRC is considering several options regarding risk metrics for new reactor risk-informed applications. The options include applying the existing operating reactor acceptance guidelines to new reactors, using new guidelines and thresholds for new reactors, or postponing any significant change to the process and evaluating new reactors on a case-by-case basis for an indeterminate period. As described in the NEI paper, "Risk Metrics for Operating New Reactors" (ML090900674; March 27, 2009), NEI has expressed its preference for applying the existing operating reactor acceptance guidelines to new reactors (which is referred to as Option 1 in the NRC white paper).

As part of the public comment process for this revised proposed rule, public stakeholders are invited to comment on the use of any of the alternative risk metric approaches for determining compliance with the risk acceptance criteria in § 50.46a.[58]

[58] *Id.*, pp. 40037-40038.

1. Comment on Topic Two

The definition of what constitutes a "very small increase" and "minimal increase" for LRF should be a full decade lower than those defined for LERF. However, it would be difficult to determine the values for LERF and LRF and ensure that the probabilities assigned to a "very small increase" and "minimal increase" for LRF would indeed be a full decade lower than those assigned to such an increase for LERF, because the NRC's and nuclear industry's ECCS evaluation models are deficient.

It is significant that data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. Data from such experiments also indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. Therefore, 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* (which uses the Baker-Just equation) and any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations (which use the Cathcart-Pawel equation) are deficient for calculating the metal-water reaction rates that would occur in the event of a LOCA.

(For additional information on severe fuel damage experiments and experimental data that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative see Section III.C "The Metal-Water Reaction Rate.")

Additionally, Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena, Refill and Reflood Heat Transfer for Pressurized Water Reactors*—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"]"—is erroneously based on the assumption that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions. (For additional information on this issue see Section III.D "FLECHT Run 9573.")

Deficient ECCS evaluation models cannot realistically model the phenomena that would occur in the event of a LOCA. Furthermore, deficient ECCS evaluation models are potentially dangerous because they provide erroneous simulations of the phenomena that would occur in the event of a LOCA. For example, the ECCS evaluation calculations that helped qualify IP-2's 2004 stretch power uprate, calculated IP-2's PCT at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies during a postulated LB LOCA.[59] This is highly problematic because, with high probability, if there were a LB LOCA at IP-2, there would be a partial or complete meltdown.

This is demonstrated by examining data from multi-rod (assembly) severe fuel damage experiments. During the LOFT LP-FP-2 experiment, when peak cladding temperatures reached between approximately 2060°F[60] and 2240°F,[61] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec.;[62] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[63]

And the CORA experiments demonstrated that with good fuel assembly insulation—like what the core of a nuclear power plant has—that cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, starts when the cladding

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[59] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.
[60] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[61] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
[62] *Id.*
[63] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

reaches between 2012°F and 2192°F; cladding temperatures then start increasing at a maximum rate of 27°F/sec.[64]

So, in the event of a LOCA at IP-2, if peak cladding temperatures increased to between approximately 2060°F[65] and 2240°F,[66] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[67] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[68] the melting point of Zircaloy is approximately 3308°F.[69]

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

(In the pretransient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[70] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40

[64] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.
[65] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[66] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
[67] *Id.*
[68] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p 23.
[69] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," p. 3-1.
[70] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," p. 10.

psia):[71] "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."[72])

The NRU Thermal-Hydraulic Experiment 1 ("TH-1") tests illustrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126 (reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.7 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

Compare this to some of the TH-1 tests that had reflood rates of 5.9 in./sec. or greater: test no. 120 (reflood rate of 5.9 in./sec.) had a PCT at the start of reflood of 1460°F and an overall PCT of 1611°F (an increase of 151°F), test no. 113 (reflood rate of 7.6 in./sec.) had a PCT at the start of reflood of 1408°F and an overall PCT of 1526°F (an increase of 118°F), test no. 115 (reflood rate of 9.5 in./sec.) had a PCT at the start of reflood of 1666°F and an overall PCT of 1758°F (an increase of 92°F).

It seems obvious that if the three TH-1 tests with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 30 seconds or higher, or had PCTs at the start of reflood that were 1200°F or higher, that the fuel assemblies, with high probability, would have reached temperatures exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(For additional information on how low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases see Section III.B "Reflood Rates.")

It is noteworthy that in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test]

[71] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 6-5.
[72] *Id.*

program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[73]

It is also significant that there is little or no evidence that the thermal resistance of crud layers on fuel cladding have ever been properly factored into ECCS evaluation calculations for postulated LOCAs. (For information on this issue see Section III.E "The Thermal Resistance of Crud and/or Oxide Layers on Fuel Cladding and ECCS Evaluation Calculations for Postulated LOCAs.")

Clearly, the deficiencies of the NRC's and nuclear industry's ECCS evaluation models discussed above indicate that the probabilities assigned to CDF and LERF are erroneous. Furthermore, it is highly probable that at nuclear power plants CDF and LERF are currently considerably higher than 10^{-4} per reactor year and 10^{-5} per reactor year, respectively, because of the deficiencies of ECCS evaluation models.

Therefore, it is imperative that the NRC correct the deficiencies of the NRC's and nuclear industry's ECCS evaluation models. If the deficiencies of the ECCS evaluation models were corrected, the NRC would be better able to ensure that the definition of what constitutes a "very small increase" and "minimal increase" for LRF would indeed be a full decade lower than those defined for LERF.

(For additional information on experimental data that indicates ECCS evaluation models are not realistic see Section III "Background." The information provided in Section III "Background" is part of Commentator's responses to topic two.)

C. Topic Three

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" states that "[t]he NRC seeks specific public comments on three topics."[74] Regarding the third topic, the NRC states:

> In § 50.46a(e)(4)(i) of the revised proposed rule the NRC proposes coolable core geometry as a high level performance-based ECCS analysis acceptance criterion for beyond-TBS LOCAs. Applicants would be allowed to justify appropriate metrics to demonstrate coolable geometry or use the current metrics (2200°F PCT [(peak cladding temperature)] and 17

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[73] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.
[74] NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," p. 40038.

> percent MLO [(maximum local cladding oxidation)]). However, the NRC acknowledges that it would be expensive and time-consuming for industry to develop the necessary experimental and analytical data to justify alternative acceptance criteria as a surrogate for demonstrating coolable geometry. Because of the difficulty in demonstrating alternative metrics, the NRC is requesting stakeholder comments on whether the final § 50.46a rule should retain the coolable geometry criterion for beyond-TBS breaks. Retaining coolable geometry would give licensees the option to demonstrate alternative coolable geometry metrics or use the current metric (2200°F PCT and 17 percent MLO). If the NRC removed the coolable geometry criterion, the beyond-TBS acceptance criteria would be the same as the acceptance criteria for TBS and smaller breaks (2200°F PCT and 17 percent MLO). The NRC will evaluate stakeholder comments on this question before deciding which beyond-TBS acceptance criteria to include in the final rule. (*See* Section V.D.2 of this document.)[75]

"Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Section V.D.2 states:

> ECCS acceptance criteria in § 50.46a(e)(3) for breaks at or below the TBS would be the same as those currently required in § 50.46. Therefore, licensees would be required to use an approved methodology to demonstrate that the following acceptance criteria
> are met for the limiting LOCA at or below the TBS:
>
> - PCT less than 2200°F;
> - Maximum local cladding oxidation (MLO) less than 17 percent;
> - Maximum hydrogen production—core wide cladding oxidation less than one percent;
> - Maintenance of coolable geometry; and
> - Maintenance of long-term cooling.
>
> Commensurate with the lower probability of occurrence, the acceptance criteria in § 50.46a(e)(4) for breaks larger than the TBS would be less prescriptive:
>
> - Maintenance of coolable geometry, and
> - Maintenance of long-term cooling.
>
> The revised proposed rule would allow licensees flexibility in establishing appropriate metrics and quantitative acceptance criteria for maintenance of coolable geometry. A licensee's metrics and acceptance criteria must realistically demonstrate that coolable core geometry and long-term cooling will be maintained. Unless data or other valid justification criteria are provided, licensees should use 2200°F and 17 percent for the limits on

[75] *Id.*

PCT and MLO, respectively, as metrics and quantitative acceptance criteria for meeting the rule. Other less conservative criteria would be acceptable if properly justified by licensees.

However, the NRC acknowledges that it would be expensive and time-consuming for industry to develop the necessary experimental and analytical data to justify alternative acceptance criteria as a surrogate for demonstrating coolable geometry. Because of the difficulty in demonstrating alternative metrics, the NRC is requesting stakeholder comments on whether the final § 50.46a rule should retain the coolable geometry criterion for beyond-TBS breaks. Retaining coolable geometry would give licensees the option to demonstrate alternative coolable geometry metrics or use the current metric (2200°F PCT and 17 percent MLO). If the NRC removed the coolable geometry criterion, the beyond-TBS acceptance criteria would be the same as the acceptance criteria for TBS and smaller breaks (2200°F PCT and 17 percent MLO). The NRC will evaluate stakeholder comments on this question before deciding which beyond-TBS acceptance criteria to include in the final rule.

As previously discussed in Section IV.C of this document, the NRC is working to revise the ECCS acceptance criteria in § 50.46(b) to account for new experimental data on cladding ductility and to allow for the use of advanced cladding alloys. The NRC will soon issue an ANPR seeking public comments on a planned regulatory approach. The NRC expects that this rulemaking (Docket ID NRC–2008–0332) will establish new cladding embrittlement acceptance criteria in § 50.46(b) for design basis LOCAs. As these new acceptance criteria are established, the NRC will also make conforming changes to § 50.46a as necessary for both below and above TBS breaks.[76]

1. Comment on Topic Three

Beyond-TBS acceptance criteria should be the same as the acceptance criteria for TBS and smaller breaks; *i.e.*, the criteria of 10 C.F.R. § 50.46(b). The criteria of maintenance of coolable core geometry and maintenance of long-term core cooling should not be used as a substitute for the criteria of 10 C.F.R. § 50.46(b) for beyond-TBS LOCAs. Commentator makes these suggestions primarily because there are deficiencies in the NRC's and nuclear industry's ECCS evaluation models that indicate the criteria of 10 C.F.R. § 50.46(b) are non-conservative: using the criteria of maintenance of coolable

[76] *Id.*, p. 40030.

core geometry and maintenance of long-term core cooling would be even more non-conservative than using the criteria of 10 C.F.R. § 50.46(b).

It is significant that data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative. Data from such experiments also indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. Therefore, 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* (which uses the Baker-Just equation) and any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations (which use the Cathcart-Pawel equation) are deficient for calculating the metal-water reaction rates that would occur in the event of a LOCA.

(For additional information on severe fuel damage experiments and experimental data that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative see Section III.C "The Metal-Water Reaction Rate.")

Additionally, Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized Water Reactors*—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"]"—is erroneously based on the assumption that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions. (For additional information on this issue see Section III.D "FLECHT Run 9573.")

Deficient ECCS evaluation models cannot realistically model the phenomena that would occur in the event of a LOCA. Furthermore, deficient ECCS evaluation models are potentially dangerous because they provide erroneous simulations of the phenomena that would occur in the event of a LOCA. For example, the ECCS evaluation calculations that helped qualify IP-2's 2004 stretch power uprate, calculated IP-2's PCT at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies during a postulated LB LOCA.[77] This is highly problematic because, with high probability, if there were a LB LOCA at IP-2, there would be a partial or complete meltdown.

This is demonstrated by examining data from multi-rod (assembly) severe fuel damage experiments. During the LOFT LP-FP-2 experiment, when peak cladding temperatures reached between approximately 2060°F[78] and 2240°F,[79] the Zircaloy cladding began to rapidly oxidize, and cladding temperatures started increasing at a rate of approximately 18°F/sec. to 36°F/sec.;[80] "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[81]

And the CORA experiments demonstrated that with good fuel assembly insulation—like what the core of a nuclear power plant has—that cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, starts when the cladding reaches between 2012°F and 2192°F; cladding temperatures then start increasing at a maximum rate of 27°F/sec.[82]

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[77] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.
[78] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[79] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
[80] *Id.*
[81] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[82] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.

So, in the event of a LOCA at IP-2, if peak cladding temperatures increased to between approximately 2060°F[83] and 2240°F,[84] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[85] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[86] the melting point of Zircaloy is approximately 3308°F.[87]

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

(In the pretransient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[88] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[89] "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."[90])

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

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

[85] *Id.*

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

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

[88] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," p. 10.

[89] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 6-5.

[90] *Id.*

The NRU Thermal-Hydraulic Experiment 1 ("TH-1") tests illustrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126 (reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.7 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

Compare this to some of the TH-1 tests that had reflood rates of 5.9 in./sec. or greater: test no. 120 (reflood rate of 5.9 in./sec.) had a PCT at the start of reflood of 1460°F and an overall PCT of 1611°F (an increase of 151°F), test no. 113 (reflood rate of 7.6 in./sec.) had a PCT at the start of reflood of 1408°F and an overall PCT of 1526°F (an increase of 118°F), test no. 115 (reflood rate of 9.5 in./sec.) had a PCT at the start of reflood of 1666°F and an overall PCT of 1758°F (an increase of 92°F).

It seems obvious that if the three TH-1 tests with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 30 seconds or higher, or had PCTs at the start of reflood that were 1200°F or higher, that the fuel assemblies, with high probability, would have reached temperatures exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(For additional information on how low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases see Section III.B "Reflood Rates.")

It is noteworthy that in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test] program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[91]

It is also significant that there is little or no evidence that the thermal resistance of crud layers on fuel cladding have ever been properly factored into ECCS evaluation calculations for postulated LOCAs. (For information on this issue see Section III.E "The

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

Thermal Resistance of Crud and/or Oxide Layers on Fuel Cladding and ECCS Evaluation Calculations for Postulated LOCAs.")

Clearly, the deficiencies of the NRC's and nuclear industry's ECCS evaluation models discussed above indicate that the probabilities assigned to CDF and LERF[92] are erroneous.

It is significant that Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," states that "if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it;"[93] and states that "if there is an indication that the LERF may be considerably higher than 10^{-5} per reactor year, the focus should be on finding ways to decrease rather than increase it."[94]

It is highly probable that at nuclear power plants CDF and LERF are currently considerably higher than 10^{-4} per reactor year and 10^{-5} per reactor year, respectively, because ECCS evaluation models are deficient. Therefore, it is imperative that the NRC decrease the probabilities of CDF and LERF, rather than increase them.

It is significant that, regarding the NRC's proposed revisions to 10 C.F.R. § 50.46(a) the NRC's ACRS, in SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" May 16, 2007, Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," states:

> It is likely that with this rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS. The rule currently contains acceptance criteria for fuel cladding

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92 NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," p. 8, footnote 3, states that "[s]uch accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines."

93 NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," p. 17.

94 *Id.*

> performance under LOCA conditions based on the current 10 CFR 50.46.[95]

And significant that, in response to the ACRS, the NRC staff, in SECY-07-0082, states that "changes proposed by licensees adopting § 50.46a will likely result in more demanding reactor operating conditions that may further stress the fuel, or result in small break LOCAs becoming limiting."[96]

So the NRC must not revise its regulations to allow for "design changes, such as increasing power [that] could cause increases in plant risk."[97] It is also imperative that the NRC not revise its regulations to "divide the current spectrum of LOCA break sizes into two regions"[98] and make "each break size region...subject to different ECCS requirements"[99] where "the smaller break size region [would] be analyzed by the methods, assumptions, and criteria currently used for LOCA analysis [and] accidents in the larger break size region [would] be analyzed by less conservative assumptions based on their lower likelihood."[100] Beyond-TBS acceptance criteria should be the same as the acceptance criteria for TBS and smaller breaks; *i.e.*, the criteria of 10 C.F.R. § 50.46(b). The criteria of maintenance of coolable core geometry and maintenance of long-term core cooling should not be used as a substitute for the criteria of 10 C.F.R. § 50.46(b) for beyond-TBS LOCAs.

Furthermore, "LOCAs for break sizes larger than the transition break [must not] become 'beyond design-basis accidents,' "[101] even if "the proposed rule would require licensees to maintain the ability to mitigate all LOCAs up to and including the DEGB of the largest RCS pipe during all operating configurations."[102]

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95 NRC, SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," p. 4.
96 *Id.*, p. 5.
97 NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," p. 40008.
98 *Id.*
99 *Id.*
100 *Id.*
101 *Id.*
102 *Id.*

(For additional information on experimental data that indicates ECCS evaluation models are not realistic see Section III "Background." The information provided in Section III "Background" is part of Commentator's responses to topic three.)

III. BACKGROUND

Commentator is responding to the three specific topics identified for public comment in "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," published in the Federal Register on August 10, 2009, primarily because Commentator is aware of deficiencies in the NRC's and nuclear industry's ECCS evaluation models. As stated above, Commentator discusses additional experimental data that indicates ECCS evaluation models are not realistic in Section III "Background" of this document. The information provided in Section III "Background" is part of Commentator's responses to each of the three specific topics identified for public comment.

A. Introduction

In 1973, the Commissioners of the Atomic Energy Commission ("AEC") stated, "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[103] Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.[104]

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

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[103] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1124. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.
[104] See Appendix A for photographs of the assembly from FLECHT Run 9573; see also Appendix B for a photograph of the assembly from FLECHT Run 8874.

emergency core cooling systems."[105] An autocatalytic oxidation reaction was not expected to occur in any of the FLECHT tests.[106]

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

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT Final Report"] when compared to the equivalent stainless steel tests."[107] The NRC also stated, regarding the results of FLECHT run 9573, that the AEC Commissioners were not "concern[ed] about the zirconium-water reaction models."[108]

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

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

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105 F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 1-1.
106 "PWR FLECHT Final Report" does not mention that an autocatalytic oxidation reaction occurred during FLECHT run 9573.
107 NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 16-17.
108 *Id.*, p. 17.

stainless steel tests."[109] The NRC also stated, regarding the results of FLECHT run 9573, that the AEC Commissioners were not "concern[ed] about the zirconium-water reaction models."[110]

Discussing the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The second reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating the [metal-water] reaction from the physical processes of cooling rods which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the assumption that the radiative and convective heat transfer processes for heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions.* The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.
>
> *The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms* [emphasis added].[111]

And opining on the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Commission Decision on Rulemaking for

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[109] *Id.*, pp. 16-17.
[110] *Id.*, p. 17.
[111] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-7.

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

> The reasonable conclusion was reached that the effect of the difference between Zircaloy and stainless steel, if any, would be small. There is a difference, of course, in the rate of heat generation from steam oxidation, but this heat is deposited within the metal under the surface of the oxide film. *The presence of this heat source should not affect the heat transfer coefficients, which depend on conditions in the coolant outside the rod.*[112]

So the AEC Commissioners concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients, maintaining that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside the rod.

It is significant that within the first 18.2 seconds of FLECHT run 9573,[113] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[114] *i.e.*, more heat was transferred into the bundle midplane than was removed from that location. In petition for rulemaking 50-76 ("PRM-50-76"), Robert H. Leyse, the principal engineer in charge of directing the Zircaloy FLECHT tests and one of the authors of "PWR FLECHT Final Report," states that "[t]he negative heat transfer coefficients [occurring within the first 18.2 seconds of run 9573] were calculated as a result of a heat transfer condition during which more heat was being transferred into the heater than was being removed from the heater[; used in the FLECHT tests to simulate fuel rods]. And the reason for that condition was that the heat generated from Zircaloy-water reactions at the surface of the heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[115]

So the heat generated from the exothermic oxidation reaction of the Zircaloy cladding (and Zircaloy spacer grids) was transferred from the cladding's reacting surface

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[112] 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," pp. 1123-1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."
[113] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[114] *Id.*, p. 3-98.
[115] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.

inward. Indeed, the Zircaloy-cladding heater rods were very hot internally, where the thermocouples were located; yet, nonetheless, the heater rods became a heat sink.[116]

Additionally, the exothermic oxidation reaction of the Zircaloy heated a mixture of steam and hydrogen, and entrained water droplets. Westinghouse agrees with this claim; in its comments regarding PRM-50-76, Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[117]

And regarding steam temperatures measured by the seven-foot steam probe, "PWR FLECHT Final Report" states:

> At the time of the initial [heater element] failures, midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure).[118]

Therefore, it is reasonable to conclude that a superheated mixture of steam and hydrogen, and entrained water droplets, caused heating of Zircaloy cladding in the midplane location of the fuel rod. It is also reasonable to conclude that the "negative heat transfer coefficients [that] were observed at the bundle midplane for 5...thermocouples"[119]—the occurrence of more heat being transferred into the bundle midplane than was removed from that location—within the first 18.2 seconds of FLECHT run 9573, were caused by an exothermic zirconium-water reaction. Additionally, it is reasonable to conclude that "the impression left from [FLECHT] run 9573" cannot be separated from concerns about zirconium-water reaction models.

Furthermore, because, as Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the

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

[117] 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.

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

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

zirconium and the steam,"[120] the AEC Commissioners' conclusion that "the presence of...heat [generated from the exothermic zirconium-water reaction] should not affect...heat transfer coefficients, which depend on conditions in the coolant outside the rod"[121] is erroneous. Clearly, the exothermic zirconium-water reaction affects the coolant outside the cladding by heating a mixture of steam and hydrogen, and entrained water droplets; therefore, the zirconium-water reaction cannot legitimately be separated from cladding heat transfer mechanisms.

2. Commentator's Argument

In this background section, Commentator will argue that data from severe fuel damage experiments conducted with Zircaloy fuel assemblies (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. In such tests Zircaloy cladding incurred autocatalytic (runaway) oxidation at temperatures far below where the Baker-Just and Cathcart-Pawel equations predict autocatalytic oxidation to occur. Commentator will also argue that data from such experiments indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

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

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

121 Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

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

Unfortunately, contrary to the claims of the NRC,[122] it has not been empirically established that "the impression left from run 9573" has ever been overcome by subsequent experiments with Zircaloy cladding.

B. Reflood Rates

1. The Low Flood Rate of Run 9573

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

> At this time [2004] we know that high temperature tests similar to run 9573 would require rod bundle powers well outside the range of operation of any current or proposed PWRs. Also, no realistic transient experiments or analyses have indicated cladding temperatures at the beginning of reflood anywhere near the 1970°F achieved in run 9573. If run 9573 were repeated the results would probably be the same, the high temperatures and high power would quickly catapult the cladding into the severe metal-water reaction regime, destroying the bundle and producing very little useful heat transfer information.[123]

Indeed, it is reasonable to postulate that if run 9573 were repeated that the fuel assembly would once again be destroyed by autocatalytic oxidation; however, this would be as a consequence of the low flood rate of the coolant (1.1 in./sec.) as well as the high initial cladding temperatures and high power of the assembly. In "Technical Safety Analysis of PRM-50-76," the NRC neglected to mention the fact that run 9573 had a low coolant flood rate. Regarding the significance that coolant flood rates played in the PWR FLECHT test program, "PWR FLECHT Final Report" states, "[i]n general, the effect on

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

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

heat transfer coefficient[s] of varying system parameters was clearly discernable, *with flooding rate being by far the most influential parameter investigated*" [emphasis added].[124] The NRC's "Compendium of ECCS Research for Realistic LOCA Analysis" reiterates that in the PWR FLECHT test program, flooding rates were the most influential parameter for affecting heat transfer coefficients.[125]

It is significant that run 9573 had a lower initial cladding temperature than, and the same power level as, other Zircaloy tests conducted in the PWR FLECHT test program that did not incur autocatalytic oxidation. It is also significant that run 9573 had the lowest flood rate of the four Zircaloy tests (see Appendix C Table B-1. Group III Test Results). Additionally, it is noteworthy that "Consolidated National Intervenors pointed out that most of [the Zircaloy] runs were made at unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding rate was about one inch per second."[126, 127]

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

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

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

> The petitioner [Robert H. Leyse] states that more experiments with Zircaloy cladding have not been conducted on the scale necessary to

[124] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 5-1.
[125] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.4-14.
[126] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."
[127] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists.

overcome the impression left from run 9573. The NRC disagrees. In fact, additional Zircaloy tests have been performed. In the early 1980s, the NRC contracted with National Research Universal (NRU) at Chalk River, Ontario, Canada to run a series of LOCA tests in the NRU reactor. More than 50 tests were conducted to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length 32-rod nuclear bundle during the heatup, reflood, and quench phases of a large-break LOCA. The NRC is reviewing the data from this program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE).[128]

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

The more than 50 NRU reactor thermal-hydraulic and mechanical deformation tests were conducted in a series of experiments: Thermal-Hydraulic Experiment 1 ("TH-1"), Thermal-Hydraulic Experiment 2 ("TH-2"), Thermal-Hydraulic Experiment 3 ("TH-3"), Materials Test 1 ("MT-1"), Materials Test 2 ("MT-2"), Materials Test 3 ("MT-3"), Materials Test 4 ("MT-4"), and Materials Test 6A ("MT-6A"). In "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor" ("LOCA Simulations in the NRU Reactor"), there is an overview of 50 tests that were planned at the NRU reactor—45 thermal hydraulic tests and five cladding materials tests.[129]

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[128] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 18-19.
[129] 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.

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

> [O]ne assembly will be used for thermal-hydraulic testing for a maximum of 45 test runs...
>
> All rods will be unpressurized; consequently, no severe cladding deformation will occur. ...
>
> The current design for the thermal-hydraulic tests is based on using one heatup rate to minimize reactor control problems and experimental perturbations. The reflood rate and reflood injection times will be used as the prime independent variables and will in various combinations be used to reverse the temperature transient at the desired peak cladding temperature limit.[130]

"LOCA Simulations in the NRU Reactor" also states that the planned heatup rate for all the tests was 15°F/sec.,[131] that the highest predicted PCTs were 1900°F,[132] for seven of the 45 tests, and that "for safety purposes," the maximum PCTs of the tests would be 1900°F.[133] So it is obvious that the NRU reactor tests were not planned to simulate LOCA conditions severe enough to overcome the impression left from FLECHT run 9573.

One may be sympathetic toward the test planners who "for safety purposes" did not want the maximum PCTs of the tests to exceed 1900°F; however, in reality, at a nuclear power plant, in the event of a LOCA, the PCT would not necessarily be limited to 1900°F. Furthermore, thermal hydraulic tests planned to have PCTs of only 1900°F, would not provide valid data for calculating heat transfer coefficients for cladding temperatures greater than 1900°F. Regarding this point, the NRC states that "[h]eat transfer coefficients are not directly measurable quantities. They must be calculated from *measured temperatures*, known heat sources, and known thermal properties" [emphasis added].[134]

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[130] *Id.*, p. 3-1.
[131] *Id.*, pp. 3-2, 3-3.
[132] *Id.*, p. 3-3.
[133] *Id.*
[134] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," p. 7.

a. Thermal-Hydraulic Experiment 1

In TH-1, a total of 28 tests were conducted. The TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[135] The TH-1 tests had the highest cladding temperatures of the more than 50 thermal-hydraulic and mechanical deformation behavior tests conducted in the NRU reactor—three of the tests had PCTs that exceeded 1900°F[136]—that the NRC claimed were conducted on the scale necessary to overcome the impression left from FLECHT run 9573.

Unfortunately, the TH-1 tests were conducted with parameters that would not be severe enough to overcome the impression left from run 9573. The PCTs reached in the TH-1 tests ranged from 1223°F to 2040°F (see Appendix D Table 1. Experimental Heat Cladding Temperatures). The TH-1 tests had reflood rates ranging from 0.7 in./sec. to 10.5 in./sec. and delay times to initiate reflood ranging from 3 sec. to 66 sec.[137] And the TH-1 tests had PCTs at the start of reflood ranging from 800°F to 1666°F.[138]

(In the pretransient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[139] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[140] "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."[141])

It is significant that the three TH-1 tests (no. 126, no. 127, and no. 130) with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 5 seconds or lower and PCTs at the start of reflood that were 998°F or lower. In other words, the TH-1 tests were conducted with parameters that would prevent the fuel assemblies' overall PCTs from rising much above 2000°F. In fact, the highest predicted

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[135] 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.
[136] *Id.*, p. 12.
[137] *Id.*, p. 13.
[138] *Id.*
[139] *Id.*, p. 10.
[140] C. L. Mohr, *et al.*, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," p. 6-5.
[141] *Id.*

PCTs for the TH-1 tests were 1900°F (no. 127 and no. 129); test no. 130 apparently did not have a predicted PCT. As discussed above, the test planners—"for safety purposes"—did not want the maximum PCTs of the tests to exceed 1900°F.

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

> A loss-of-coolant accident (LOCA) in a commercial light water reactor (LWR) consists of four distinct phases: blowdown, heatup, reflood, and quench. Each of these phases has a path-dependent process that is a function of 1) the type of event that initiated the accident and 2) the reactor's operating conditions at the time the LOCA was initiated. *No single set of conditions would exist at the time of a LOCA; rather, a broad range of operating parameters could exist in one of many possible combinations* [emphasis added].[142]

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

> In general, based on best-estimate or conservative assumptions during design-basis accidents, the boundary and initial conditions for reflooding tests can be established during the design-basis accidents. The variation of some of the main parameters can be summarized as: system pressure 0.1-1.0 MPa, flooding velocities 1.5-30 cm/sec. (including natural reflood velocities), mass fluxes 7-300 kg/m^{2}sec., heater rod peak power 0.7-3 kW/m.[143]

It is also noteworthy that "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data" states that "[c]ladding temperature increases during blowdown from normal operating conditions of approximately 325°C to approximately 550-800°C (roughly 1000-1500°F)."[144]

Indeed, "[n]o single set of conditions would exist at the time of a LOCA; rather, a broad range of operating parameters could exist in one of many possible

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

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

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

combinations."[145] For this reason, the TH-1 tests—conducted with strictly controlled parameters that prevented the fuel assemblies' PCTs from rising much higher than 2000°F—are not realistic tests for simulating a wide variety of possible LOCAs; *e.g.*, LOCAs with long reflood delay times and low reflood rates.

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

Compare this to some of the TH-1 tests that had reflood rates of 5.9 in./sec. or greater: test no. 120 (reflood rate of 5.9 in./sec.) had a PCT at the start of reflood of 1460°F and an overall PCT of 1611°F (an increase of 151°F), test no. 113 (reflood rate of 7.6 in./sec.) had a PCT at the start of reflood of 1408°F and an overall PCT of 1526°F (an increase of 118°F), test no. 115 (reflood rate of 9.5 in./sec.) had a PCT at the start of reflood of 1666°F and an overall PCT of 1758°F (an increase of 92°F).

It seems obvious that if the three TH-1 tests with reflood rates of 1.2 in./sec. or lower also had delay times to initiate reflood that were 30 seconds or higher, or had PCTs at the start of reflood that were 1200°F or higher, that the fuel assemblies, with high probability, would have incurred autocatalytic (runaway) oxidation, clad shattering, and failure—like FLECHT run 9573. It certainly seems obvious that if the parameters were the same for test no. 115 (PCT at the start of reflood of 1666°F), except it had a reflood rate of 1.2 in./sec. or lower, that its overall PCT would have increased above 2200°F and the fuel assembly, with high probability, would have incurred autocatalytic oxidation, clad shattering, and failure—like FLECHT run 9573.

Rather than "overcome the impression left from [FLECHT] run 9573," the TH-1 tests, with high probability, confirm Commentator's claim that if run 9573 were repeated—with the same or a lower coolant flood rate, yet with lower initial cladding

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

temperatures (that in the event of a LOCA, would occur at the beginning of reflood at current and/or proposed PWRs) and a lower power level (within the operational range of current and/or proposed PWRs)—that the fuel assembly would still incur autocatalytic oxidation.

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

b. Thermal-Hydraulic Experiments 2 and 3

In TH-2, a total of 14 tests were conducted. The TH-2 tests are reported on in "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)" ("Data Report for TH-2").[146]

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

> The primary objective of TH-2 was to develop reliable cladding temperature control of a simulated LOCA. Peak cladding temperatures were to range from 1033 to 1089°K (1400 to 1500°F) for at least 150 s, using variable rate reflood water coolant.[147]

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

> A full-length test bundle containing nonpressurized water reactor fuel rods was used to develop reflood control parameters and procedures that [would] produce a reduced heatup rate or a "flat top" transient for extended periods of time. Variable reflood rates were used, and experimentally determined control system logic parameters were developed. Using these concepts, fuel cladding temperatures from 1033 to 1274°K (1400 to 1834°F) were produced for 283 sec.[148]

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

In TH-3, a total of three tests were conducted. The TH-3 tests are reported on in "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)" ("Data Report for TH-3").[149]

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

> The objective of TH-3 was to further refine the feedback control parameters developed in the TH-2 experiment and to re-establish the operability of the loop prior to the subsequent materials deformation and rupture test (MT-3). The TH-3 and MT-3 experiments were planned for the same reactor window and were run within two days of each other. The TH-3 test results insured the success of MT-3 and provided the opportunity to demonstrate the reactor control improvements and to evaluate a new desuperheater concept that would allow the test to run for extended times at high temperatures. The control system improvements and the addition of the new desuperheater resulted in fuel cladding temperatures above 1033°K (1400°F) for 340 s.[150]

It is significant that in the TH-2 tests the highest PCT was 1834°F,[151] 366°F lower than the 10 C.F.R. § 50.46(b)(1) limit; in the TH-3 tests the highest PCT was 1912°F,[152] 288°F lower than the 10 C.F.R. § 50.46(b)(1) limit. Regardless of the achievements of the TH-2 tests and TH-3 tests in developing "reflood control parameters and procedures that [produced] a reduced heatup rate or a 'flat top' transient for extended periods of time,"[153] it is obvious that they did not simulate LOCA conditions that were severe enough to overcome the impression left from FLECHT run 9573.

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[149] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)," NUREG/CR-2527, 1983, located in ADAMS Public Legacy, Accession Number: 8304120660.

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

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

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

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

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

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

> The [four] fuel cladding performance tests will be considered for selected conditions based on the results obtained during the thermal-hydraulic tests. The objective of the tests will be to use a constant heatup rate and vary the reflood rate and reflood delay time to obtain peak cladding temperatures between 1033°K (1400°F) and 1255°K (1800°F).[154]

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

d. Materials Test 1

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

> The first materials experiment (MT-1), i.e., the test on the expansion of Zircaloy fuel cladding...was performed in April 1981, using a cruciform of 11 rods pressurized to 3.21 MPa (465 psia) and [one] water tube surrounded by 20 guard rods[155] sealed at atmospheric pressure. The objective of this test was to assess the rate at which the expanded cladding can be cooled, based on evaluations of the rates of heatup and quenching and the measurements of post-test cladding strain. The delay time and the rate of reflood were selected to duplicate one of the experiments at high temperatures, specifically TH-1.10, in which the fuel cladding reached a peak temperature of 1145°K (1600°F). These conditions were achieved: [six] of the 11 rods ruptured and all 11 pressurized test rods expanded significantly. The average peak rupture strain was 43%; the average time to rupture was 43 sec.; and the average temperature at rupture was 1145°K (1600°F).[156]

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

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

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

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

e. Materials Test 2

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

> In the second materials experiment (MT-2), ...performed in July 1981, the MT-1 guard rods and shroud assembly were reconstituted underwater and reused with a new cruciform test bundle. One of the objectives of the test was to perform a low-temperature, 1090°K (1500°F), test using variable rates of reflooding. The 12 test rods were pressurized to 3.21 MPa (465 psia). A malfunction of the reflood system, however, resulted in higher temperatures than desired and [eight] of the 11 rods ruptured. The average peak rupture strain was 43%, the average time to rupture was 65 sec., and the average temperature at rupture was 1160°K (1625°F).[157]

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

f. Materials Test 3

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

> The primary objective of the third materials experiment (MT-3)...was to determine the expansion and restrictions on the flow channel for a flat-top temperature transient using pressurized fuel rods. Peak temperatures of the cladding were maintained above 1035°K (1400°F) for 180 sec. The MT-3 experiment repeated the test conditions demonstrated by the TH-3.03 test using a completely new test train with 12 fuel rods pressurized to 3.9 MPa (565 psia) and 20 guard rods. All 12 test rods ruptured during the active two-phase cooling regime. The average peak rupture strain was 46%, the average time of rupture was 133 sec., and the average temperature at rupture was 1070°K (1460°F). The MT-3 experiment had a lower average temperature at rupture and a longer time until rupture than any of the other materials experiments because of the significant amount of reflood water that was introduced early in the transient (the delay time for reflooding was 7 sec.). The active strain region was spread over ~2-m (80-in.) length, and no loss of cooling because of coplanar blockage or liftoff[158] was observed.[159]

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

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[157] *Id.*

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

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

g. Materials Test 4

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

> The fourth materials experiment (MT-4)...was conducted in May 1982. Its primary objective was to evaluate the expansion and rupture of cladding during heatup in the temperature range from 1035 to 1200°K (1400 to 1700°F). The 12 test rods in the 32-rod bundle were initially pressurized to 4.62 MPa (670 psia) at 295°K (70°F) to assure rupture in the correct temperature range. The MT-4 experiment was most similar to the MT-2 experiment; three differences existed: 1) MT-4 rods were pressurized to 4.62 MPa (670 psia), whereas MT-2 rods were pressurized to 3.21 MPa (465 psia); 2) After the temperature turnaround following the heatup transient, the peak temperatures of the cladding were stabilized to measure the characteristics of the heat transfer of the expanded and ruptured fuel rods, whereas during MT-2 the peak temperatures of the cladding were not stabilized; and 3) self-powered neutron detectors (SPNDs) mounted on the shroud were moved to grid elevations to minimize distortion of axial fission power, whereas MT-2 had the SPNDs mounted away from the Inconel grids. During the test all 12 test rods ruptured with an average peak rod strain of 72.1%. The active strain region was spread over 0.189 m (7.42 in.), the average time of rupture was 55 sec.; and the average temperature at rupture was 1094°K (1511°F).
>
> The MT-4 experiment used a new cruciform bundle of 12 pressurized test fuel rods and the guard fuel rods and shroud previously used in MT-3. Test operations most closely followed the operating conditions of the TH-1.16, during which cooling by reflooding was used to terminate the transient temperature of the heatup at ~1200°K (1700°F). Stabilized operations at the post-transient stage closely followed the operating conditions used in the MT-3 experiment.[160]

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

h. Materials Test 6A

The MT-6A test is discussed in "Large-Break LOCA, In-Reactor Fuel Bundle Materials Test MT-6A" ("Materials Test MT-6A"). After a reflood delay of approximately 70 seconds—controlled by the data acquisition and control system ("DACS")—the MT-6A test had varying reflood rates: 8 in./sec. for 3 sec., 7 in./sec. for 3

[160] *Id.*, p. x.

sec., and 2 in./sec. for approximately 170 sec.[161] "Materials Test MT-6A" states that after the reflood rate was held at 2 in./sec. for 3 sec. "the DACS was supposed to take over reflood control to maintain fuel temperatures [that were] approximately constant. An anomaly in the reflood control prevented the DACS from taking control once the reflood rate reached [2 in./sec]. The continued reflood at this rate caused the fuel to cool and quench, ending the test."[162]

In the MT-6A test, the PCT was approximately 1750°F,[163] or 450°F lower than the 10 C.F.R. § 50.46(b)(1) limit. It is obvious that the MT-6A test, and the other NRU reactor materials tests, did not simulate LOCA conditions that were severe enough to overcome the impression left from FLECHT run 9573.

3. Conclusion of the Reflood Rates Section

It has been demonstrated in the Reflood Rates Section that the Zircaloy cladding tests, performed in the early 1980s, at the NRU reactor—"to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length 32-rod nuclear bundle during the heatup, reflood, and quench phases of a large-break LOCA"[164]—did not simulate LOCA conditions severe enough to "overcome the impression left from [FLECHT] run 9573."[165]

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

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[161] *Id.*, pp. 6, 21.
[162] *Id.*, p. 6.
[163] *Id.*, pp. B.10, B.11.
[164] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.
[165] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

reflood rate throughout the core; however, at times the reflood rate could be approximately one inch per second or lower at different locations throughout the core.

It is noteworthy that in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test] program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[166]

It is clear that the NRC has failed to analyze the data from the NRU thermal-hydraulic and mechanical deformation tests that indicates that, in the event a LOCA, a constant core reflood rate of approximately 1 in./sec. or lower would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

It will be demonstrated in the following Metal-Water Reaction Rate Section that, in the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[167] and 2240°F,[168] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[169]

Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[170] the melting point of Zircaloy is approximately 3308°F.[171]

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

[167] 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.

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

[169] *Id.*

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

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

C. The Metal-Water Reaction Rate

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

> One of the bases for selecting 2200°F (1204°C) as the PCT [limit] was that it provided a safe margin, or conservatism, away from an area of zircaloy oxidation behavior known as the autocatalytic regime. The autocatalytic condition occurs when the heat released by the exothermic zircaloy-steam reaction (6.45 megajoules per kg zircaloy reacted) is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant. This reaction heat then further raises the zircaloy temperature, which in turn increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.[172]

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

> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. This type of comparison implicitly includes...complex heat transfer mechanisms...and the effects of fuel rod ballooning and rupture on coolability... Analysis of experiments performed in the Power Burst Facility, in the Annular Core Research Reactor, and in the NEILS-CORA (facilities in West Germany) program have shown that temperatures above 2200°F are required before the zircaloy-steam reaction becomes sufficiently rapid to produce an autocatalytic temperature excursion. Another group of relevant experimental data were produced from the MT-6B and FLHT-LOCA and Coolant Boilaway and Damage Progression tests conducted in the NRU Reactor in Canada. ...even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists.[173]

It is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" states that assessing the conservatism of the 2200°F PCT limit, as a boundary that would prevent autocatalytic oxidation from occurring, can be accomplished by

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

analyzing data from multi-rod severe accident tests, because such data indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative.

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

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

A maximum heating rate of 15°K/sec. indicates that an autocatalytic oxidation reaction commenced. "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[174] So at the point when peak cladding temperatures increased at a rate of greater than 10°K/sec. during the CORA experiments, autocatalytic oxidation reactions commenced—at cladding temperatures between 2012°F and 2192°F.

It is noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis," published in 1988, does not mention that autocatalytic oxidation occurred during the LOFT LP-FP-2 experiment, conducted in 1985, at cladding temperatures greater than either 1400°K (2060°F)[175] or 1500°K (2240°F).[176]

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

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

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

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

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

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

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

> these calculations, the margin between 2300°F and the calculational instability using Cathcart-Pawel was always equal to or greater than the margin between 2200°F and the calculational instability using Baker-Just.[177]

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

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

The Power Burst Facility ("PBF") Severe Fuel Damage ("SFD") 1-1, 1-3, and 1-4 tests each used a PWR 17 by 17 assembly comprised of 32 fuel rods that were 0.9 meter in length.[178] Or according to a different account, the PBF SFD 1-1 test had 32 fuel rods that were 0.9 meter in length and the PBF SFD 1-3 and 1-4 tests had 28 fuel rods that were 1.0 meter in length.[179]

"Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code" states that "[t]he...SFD 1-1, 1-3, and 1-4 [tests] were conducted in a thermal-hydraulic condition similar to that expected to have occurred at TMI-2, which

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[177] "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.

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

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

is characterized by slow heating up to 1600°K and rapid heating rate above 1600°K, driven by zirconium-water reaction."[180]

The same paper also states that "[i]n the [SFD 1-1] test, the rapid temperature rise in the bundle began near the center at the 0.5 to 0.7 [meter] elevation, and then spread radially outward and axially downward in a manner similar to a flame front propagation."[181] Additionally, three graphs of the cladding-temperature values (at the 35 cm, 50 cm, and 70 cm elevations) during the SFD 1-1 test indicate that that test's autocatalytic oxidation reaction began when cladding temperatures were approximately 1600°K.[182]

Offering a different account of the heatup rates during the PBF-SFD tests, "Review of Experimental Results on LWR Core Melt Progression" states that "[h]eatup rates in the moderately high-pressure PBF-SFD tests began in the neighborhood of 0.1 to 0.5°K /sec., but increased to about 1 to 2°K /sec. above 1300°K and >10°K /sec. above 1700°K."[183]

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

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

181 *Id.*, p. 960; this paper cites Proceedings from the 5th International Meeting on Thermal Reactor Safety and P. E. MacDonald, *et al.*, American Nuclear Society Transcript, 46, 478, 1984, as the source of this information.

182 *Id.*, pp. 962-963.

183 R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC, "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites D. J. Osetek, "Results of the Four PBF Severe Fuel Damage Tests," NRC, "Proceedings of the Fifteenth Water Reactor Safety Information Meeting," NUREG/CP-0090, 1987, as the source of this information.

Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[184]

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

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

> In 1984, a proof of princip[le] test [(or transition test)] (MT-6B) was performed to determine whether a test on a full-length fuel bundle could be safely performed to demonstrate the kind and extent of the damage that would result to fuel rods from a boilaway of reactor coolant. Emphasized were the severe damage conditions that would result in the core. In this proof of princip[le] test, the LB-LOCA test geometry was used. Demonstrated during the test was that adequate thermal insulation can protect the reactor under severe conditions and that it is possible to control a boilaway transient; the conclusion was that it would be safe to conduct in-reactor tests that cause severe damage to reactor fuel rods from a loss of coolant.[185]

During the MT-6B test the PCT was either 2060°F (1400°K),[186] 2200°F (1477°K),[187] or 2336°F (1553°K)[188]: three different publications report these inconsistent PCT values. 276°F (153°K) is a substantial temperature difference. One of the goals of the MT-6B test was to achieve a PCT of 1600°K (2420°F).

"Compendium of ECCS Research for Realistic LOCA Analysis" states that "[t]he MT-6B test...showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant."[189] However, because other reports state that the MT-6B test had a PCT of 1400°K (2060°F) and 1280°C (2336°F) (1553°K), the MT-6B test may have actually demonstrated that the Zircaloy

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184 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."

185 W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, p. x.

186 *Id.*, p. viii.

187 NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

188 G. M. Hesson, *et al.*, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," 1993, p. 2.

189 NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

oxidation rate was easily controllable by adding more coolant at cladding temperatures of either 2060°F (1400°K) or 1280°C (2336°F) (1553°K).

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

The first full-length high-temperature severe fuel damage ("FLHT-1") test was conducted at the National Research Universal ("NRU") reactor at Chalk River, Ontario, Canada, by Pacific Northwest Laboratory ("PNL"), "to evaluate degraded core behavior and the progression of light water reactor ("LWR") fuel damage resulting from [a] loss-of-coolant accident."[190] The FLHT-1 test was part of the PNL Coolant Boilaway and Damage Progression program. The FLHT-1 test used an assembly comprised of 12 fuel rods that were 3.7-meters in length.[191] During the test the nominal fuel rod linear power was 0.524 kW/m (0.160 kW/ft.) and the nominal bundle power was 23 kW (22 Btu/sec.).[192]

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

> This report presents a summary of the FLHT-1 test operations. The test was performed on March 2, 1985. In the report, the actual test operations and data are compared to the planned operations and predicted test behavior. ... The test plan called for a gradual temperature increase to approximately 2150°K (3400°F). However, during the test, the fuel cladding began to rapidly oxidize, causing local bundle temperatures to rapidly increase from about 1700°K (2600°F) to 2275°K (3635°F), at which time the test was terminated. Much of the Zircaloy cladding in the central region (axially) of the 3.7-m-long (12-ft) fuel bundle was heavily oxidized, and some Zircaloy cladding melted.[193]

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

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190 W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. v.
191 *Id.*, p. 3.1.
192 *Id.*, pp. 4.1-4.2.
193 *Id.*, p. v.

"FLHT-1 Test Report" has inconsistent statements regarding the time that the Zircaloy cladding temperature excursion began—the autocatalytic (runaway) oxidation reaction.

"FLHT-1 Test Report" states that "[t]he reactor power was decreased at approximately 17:11:07, 85 seconds after the start of the [cladding temperature] excursion;"[194] *i.e.*, the cladding temperature excursion began at 17:09:42. However, "FLHT-1 Test Report" also states that the cladding temperature excursion began 18 seconds latter at 17:10:00—when the cladding temperature was 1700°K.[195] The difference of 18 seconds is highly significant, because it means that the cladding temperatures were much lower than 1700°K when the temperature excursion actually began.

"Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that during the FLHT-1, -2, -4, and -5 tests that "[t]he heatup phase of the tests culminated near 1700°K in a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signaling the onset of an autocatalytic oxidation reaction."[196] So if peak cladding temperatures increased at a rate of greater than 10°K/sec. during the FLHT-1 test, it is highly probable that 18 seconds before 17:10:00—when the peak cladding temperature was 1700°K (2600°F)—the peak cladding temperature was approximately 1520°K (~2275°F) or lower.

This is reasonable to postulate; after all, another severe fuel damage experiment—LOFT LP-FP-2—demonstrated "that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400°K (2060°F)."[197] According to a different account, in the LOFT LP-FP-2 experiment, the onset of rapid oxidation occurred at approximately

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[194] *Id.*, p. 4.6.
[195] *Id.*, p. 4.11
[196] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[197] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 33.

1500°K (2240°F).[198] Additionally, "Degraded Core Quench: Summary of Progress 1996-1999," states that autocatalytic (runaway) oxidation of Zircaloy cladding by steam occurs at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or higher.[199]

Furthermore, although the graphs of "Typical Cladding Temperature Behavior"[200] and "Pseudo Sensor Readings for Fuel Peak Temperature Region"[201, 202] are not large enough to clearly delineate what the temperature values were at given times during the FLHT-1 test, the graphs' cladding-temperature values are consistent with the postulation that the temperature excursion began at a temperature far lower than 1700°K, at a temperature closer to 1520°K (see Appendix E Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region). The slopes of the lines of the cladding-temperature value plots in the graphs become nearly vertical, when the cladding-temperature values reach approximately 1520°K, indicating the onset of the temperature excursion, at a rate of 10°K/sec. or greater.

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

> Typical cladding temperature behavior at one position in the assembly during the test is shown in Figure 4.1. At about 60 to 70 min. along the abscissa, a temperature increase [commenced] when the [bundle coolant] flow rate was about 9 kg/hr. (20 lb/hr.). The [cladding] temperature increased until about 95 min. and [reached] 1450°K (2150°F), at which time the bundle coolant [flow] rate was increased to 18 kg/hr. (40 lb/hr.) to stabilize the temperature. However, the [cladding] temperature rapidly dropped to about 1060°K (1450°F). The bundle coolant flow rate was then decreased through a series of steps to a minimum of 9 kg/hr. (20 lb/hr.). This action stopped the temperature decrease and started another temperature rise. When the temperature reached about 1475°K (2200°F), the bundle coolant flow [rate] was again increased to stop the temperature ramp. This led to a stabilized condition. The flow was increased in steps

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198 T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.

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

200 W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.7.

201 *Id.*, p. 5.3.

202 Pseudo sensor readings are the averages of the readings of two or more thermocouples.

> and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly until the test director requested that the reactor power be reduced to zero power.[203]

First, it is obvious from the above description and from Figures 4.1 and 5.4 that when cladding temperatures reached approximately 1475°K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was not achieved. Cladding temperatures continued to rise. This is clearly stated: "The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly…"[204]

Second, it is obvious that the rapid metal-water reaction began at cladding temperatures far lower than 1700°K (2600°F). It makes no sense that the autocatalytic oxidation reaction would have begun at 1700°K (2600°F). How can it be explained that after the coolant flow rate was increased—when cladding temperatures reached approximately 1475°K (2200°F)—that the cladding temperatures were able to increase by 225°K (400°F)? Why would the test conductors have not been able to terminate the cladding-temperature rise, as they did earlier in the test when cladding temperatures reached 1450°K (2150°F)? And how can it be explained that the test conductors did not have enough time to increase the coolant flow rate back up to 18 kg/hr. (40 lb/hr.), as they did when cladding temperatures reached 1450°K (2150°F), earlier in the test?

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

In "Compendium of ECCS Research for Realistic LOCA Analysis," discussing an earlier NRU reactor test, the NRC states that "[t]he MT-6B test…showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant."[205] Furthermore, the test conductors would have thought "the

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[203] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.6.
[204] *Id.*
[205] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

zircaloy oxidation rate was easily controllable" at cladding temperatures far above 2200°F (1477°K): "[t]he [FLHT-1] test plan called for a gradual [cladding] temperature increase [up] to approximately 2150°K (3400°F)."[206]

(It is noteworthy that other reports state that the MT-6B test had a PCT of 1400°K (2060°F)[207] and 1280°C (2336°F) (1553°K).[208] So the MT-6B test may have actually demonstrated that the Zircaloy oxidation rate was easily controllable by adding more coolant at cladding temperatures of either 2060°F (1400°K) or 1280°C (2336°F) (1553°K).)

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

> Once the power is set, the test will be started through its transient operation. *The term transient is somewhat of a misnomer*; operation will consist of a series of preplanned, discrete flow-reduction steps. The size and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate. ...
>
> The bundle [coolant] flow rate will then be decreased in a series of precalculated flow steps... The duration of the time between steps is dictated by the time needed to reach near steady state and also by *the requirement that the Zircaloy-steam reaction be limited.* About 14 steps, each of about 1/2 hr. duration, are expected. *The last flow reduction step will be calculated to give a peak cladding temperature of about 2150°K (3400°F).* ...
>
> The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150°K (3400°F) [emphasis added].[209]

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

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[206] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. v.
[207] *Id.*, p. viii.
[208] G. M. Hesson, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 2 Final Safety Analysis," p. 2.
[209] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," pp. 4.3-4.5.

power to zero power, as they did "85 seconds after the start of the [cladding temperature] excursion."[210]

It is important to remember that the events described above occurred within a period of approximately 85 seconds: peak cladding temperatures increased from approximately 1520°K (~2275°F) or lower to approximately 2275°K (3635°F), within approximately 85 seconds. Additionally, as discussed above, in the graphs of "Typical Cladding Temperature Behavior"[211] and "Pseudo Sensor Readings for Fuel Peak Temperature Region,"[212] the slopes of the lines of the cladding-temperature value plots of the FLHT-1 test become nearly vertical, after the cladding-temperature values reach approximately 1520°K, indicating that only a short time period passed before temperatures reached approximately 2275°K (3635°F).

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

> The reactor power was decreased at approximately 17:11:07, 85 sec. after the start of the excursion (approximately 131 minutes in Figure 4.1). The reactor reached 10% of the initial power approximately 35 sec. later and reached low neutron level in another 30 sec.
>
> There were two Indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.[213]

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

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

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[210] *Id.*, p. 4.6.
[211] *Id.*, p. 4.7.
[212] *Id.*, p. 5.3.
[213] *Id.*, pp. 4.6-4.7.

> Additional Indications of an effective test shutdown are shown by the saddle temperature, MMPD [(molten material penetration detector)] response, and bypass coolant power (radial heat loss) after the reactor power shutdown. Typical data from these sources are shown in Figures 4.5 through 4.7. All three of these indicators show steadily decreasing temperatures.[214]

It is also noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis" states that "[i]n the [FLHT-1] test, completed in March 1985, 12 ruptured zircaloy-clad rods were subjected to an autocatalytic temperature excursion. From the measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500-2600°F (1371-1427°C) temperature region."[215]

The FLHT-1 test is highly significant precisely because, once cladding temperatures reached as high as approximately 1475°K (2200°F), the test conductors could not prevent the cladding-temperature rise by increasing the coolant flow rate. Increasing the coolant flow rate did not prevent the onset of an autocatalytic oxidation reaction—which occurred at cladding temperatures of approximately 1520°K (~2275°F) or lower.

In the FLHT-1 test, it is significant that autocatalytic oxidation occurred at approximately 1520°K (~2275°F) or lower, because the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[216]

d. The LOFT LP-FP-2 Experiment

Commentator will now discus the LOFT LP-FP-2 experiment that was conducted in the Loss-of-Fluid Test ("LOFT") facility at Idaho National Engineering Laboratory, on July 9, 1985. The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[217] The

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[214] *Id.*, p. 4.7.

[215] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

[216] 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."

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

LOFT LP-FP-2 experiment—the second and final fission product test conducted at the LOFT facility—had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy 1.67 meter fuel rods; it was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud. The LOFT LP-FP-2 experiment combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[218]

The LOFT LP-FP-2 experiment had an initial heatup rate of ~1°K/sec.[219] It is significant that "heatup rates [of 1°K/s or greater] are typical of severe accidents initiated from full power."[220] And regarding the significance of the initial heatup rate in the LOFT LP-FP-2 experiment, "Review of Experimental Results on LWR Core Melt Progression" states:

> The higher initial heating rate [of 1°K/sec.] in the LOFT [LP-]FP-2 experiment is related to the higher fraction of decay heat available following rapid blowdown of the coolant inventory in the reactor vessel. This higher heating rate leads to smaller oxide thickness on the cladding for a particular temperature and, therefore, more rapid oxidation. The increase in heating rate at the higher temperatures is the result of rapid oxidation of zircaloy and the strongly exothermic nature of the reaction (6.45 kJ/g Zr oxidized).[221]

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

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

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218 *Id.*

219 *Id.*

220 S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2; this paper cites Hofmann, P., *et al.*, "Reactor Core Materials Interactions at Very High Temperatures," Nuclear Technology, Vol. 87, p. 146, 1990, as the source of this information.

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

> 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.[222]

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

> The first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400°K on a guide tube at the 0.69-m (27-in.) elevation. This temperature is shown in Figure 3.7. A cladding thermocouple at the same elevation (see Figure 3.7) reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase. Note that, due to the limited number of measured cladding temperature locations, the precise location of the initiation of [the] metal-water reaction on any given fuel rod or guide tube is not likely to coincide with the location of a thermocouple. Thus, the temperature rises are probably associated with precursory heating as the metal-water reaction propagates away from the initiation point. Care must be taken in determining the temperature at which the metal-water reaction initiates, since the precursory heating can occur at a much lower temperature. It can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400°K (2060°F).[223, 224]

Additionally, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" also states that the hottest measured cladding temperature reached 2100°K (3320°F) by 1504 ± 1 seconds;[225] and states that it was difficult to determine the PCT

[222] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3. 23.
[223] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[224] See Appendix F Figure 3.7. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 and Figure 3.10. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 with Saturation Temperature.
[225] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 23.

reached during the entire experiment—because of thermocouple failure—but that the PCT exceeded 2400°K (3860°F).[226]

Therefore, after the onset of rapid oxidation—after a heating rate of ~1°K/sec.[227]—peak cladding temperatures increased from approximately 1400°K (2060°F) to 2100°K (3320°F) within a range of approximately 35 seconds; in other words, after the onset of rapid oxidation, cladding temperatures increased at an average rate of approximately 20°K/sec. (36°F/sec.). In general agreement with this postulation, "Review of Experimental Results on LWR Core Melt Progression" states that "[i]n the LOFT [LP-]FP-2 experiment, which was driven by decay heat, the heating rate started out at about 1°K/sec. and increased to about 10-20°K/sec. above 1500°K [(2240°F)]."[228]

It is significant that "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[229] So at the point when peak cladding temperatures increased at a rate of greater than 10°K/sec. during the LOFT LP-FP-2 experiment, an autocatalytic oxidation reaction commenced; and that occurred when the temperature of a Zircaloy fuel rod or guide tube reached approximately 1400°K (2060°F), or when cladding temperatures reached approximately 1500°K (2240°F).

In a different account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Degraded Core Quench: A Status Report" states that "[t]he initial heating rate in the central assembly was ~1°K/sec. with an onset to rapid oxidation at a

[226] *Id.*, p. 33.
[227] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 13.
[228] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
[229] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

temperature near 1500°K [(2240°F)]."[230] In a similar account, as already mentioned, "Review of Experimental Results on LWR Core Melt Progression" states that the initial heatup rate was 1°K/sec., and that the heatup rate increased to approximately 10-20°K/sec. at a cladding temperature greater than 1500°K (2240°F).[231]

And offering yet another account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2" states that in the LOFT LP-FP-2 experiment that the metal-water reaction was initiated at 1450.0 ± 30 sec. after the beginning of the experiment and that at 1500 ± 1 sec, after the beginning of the experiment, the maximum cladding temperatures reached 2100°K;[232] elsewhere the same paper states that the "[m]etal-water reaction began at about 1450 seconds and [that the] hottest measured cladding temperature reached 2100°K [(3320°F)] by 1504 seconds."[233]

It is important to clarify that "rapid oxidation" is not necessarily autocatalytic oxidation. It is also important to consider questions such as, "At what point does rapid oxidation become autocatalytic oxidation?" As mentioned above, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[234]

As also mentioned above, "Review of Experimental Results on LWR Core Melt Progression" states that "[i]n the LOFT [LP-]FP-2 experiment, which was driven by

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[230] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 13.
[231] R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
[232] D. W. Akers, C. M. Allison, M. L. Carboneau, R. R. Hobbins, J. K. Hohorst, S. M. Jensen, S. M. Modro, NUREG/CR-6160, "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2," April 1994, p. 12.
[233] *Id.*, p. xii.
[234] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.

decay heat, the heating rate started out at about 1°K/sec. and increased to about 10-20°K/sec. above 1500°K [(2240°F)]."[235] For this reason, it is reasonable to conclude that when "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" uses the term "rapid oxidation," it is discussing autocatalytic oxidation or at least a phenomenon that occurs shortly before the onset of autocatalytic oxidation.

As quoted above, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states that "[t]he first recorded and qualified rapid-temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400°K..."[236] So it is reasonable to conclude that at some point when peak cladding temperatures were 1500°K (2240°F) or lower, cladding temperatures began increasing at a rate of greater than 10°K/sec., signaling the onset of an autocatalytic oxidation reaction.

In the LOFT LP-FP-2 experiment, it is significant that rapid oxidation occurred at a temperature of 2240°F or lower, because the Baker-Just equation calculates that autocatalytic oxidation occurs at approximately 2600°F and the Cathcart-Pawel equation calculates that autocatalytic oxidation occurs at approximately 2700°F.[237] Data from the LOFT LP-FP-2 experiment also indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

e. The CORA Experiments

Regarding the CORA experiments the abstract of "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" states:

> The CORA experiments carried out in an out-of-pile facility at the Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany, are part of the international "Severe Fuel Damage" (SFD) program.

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

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

[237] 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."

> The experimental program is to provide information on the failure mechanisms of Light Water Reactor (LWR) fuel elements in a temperature range from 1200°C to 2000°C and in a few cases up to 2400°C. [238]

The CORA experiments were conducted to study severe accident sequences, with electrically heated bundles of 2-meter long fuel rod simulators, held in place by three spacer grids (two Zircaloy, one Inconel), and surrounded by a shroud. The Electric heating was done with tungsten heating elements, installed in the center of annular UO_2 pellets, which, in turn, were sheathed by PWR Zircaloy-4 cladding. The total available heating power was 96kW, which had the capability of being distributed among three bundles of the fuel rod simulators. There were also unheated rods, filled with solid UO_2 pellets to correspond to LWR fuel rods.[239] In the CORA experiments the initial heatup rate of the fuel rod simulators was approximately 1°K /sec., in the presence of steam.

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

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec.[, after an initial heatup rate of about 1°K /sec.] The maximum temperatures attained are about 2000°C; the oxide layers formed and the consumption of the available steam set limits on the temperature escalation due to rate-controlled diffusion processes. The temperature escalation starts in the hotter upper half of the bundle and the oxidation front subsequently migrates from there both upwards and downwards."[240]

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

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

[240] *Id.*, p. 83.

"CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures" also states that temperature escalations "continued even after shut-off of the electric power, as long as steam was available."[241]

It is significant that in the CORA experiments, at cladding temperatures between 1100°C and 1200°C (2012°F to 2192°F), that the cladding began to rapidly oxidize and cladding temperatures started increasing at a maximum rate of 15°C/sec. (27°F/sec.), because the Baker-Just and Cathcart-Pawel equations calculate that autocatalytic oxidation occurs at approximately 2600°F and approximately 2700°F, respectively;[242] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[243] Data from the CORA experiments also indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

It is also significant that the CORA experiments demonstrated that "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation."[244] So with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between approximately 1100°C and 1200°C (2012°F to 2192°F), and cladding temperatures start increasing at a maximum rate of 15°C/sec. (27°F/sec.).

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

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[241] *Id.*, p. 87.

[242] 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."

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

[244] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.

LOFT LP-FP-2 experiment, autocatalytic oxidation occurred at cladding temperatures of approximately 2240°F or lower.

Two additional papers on the CORA experiments also provide information on cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding that occurred below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.[245]

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

> The transient phases of the tests were initiated with a temperature ramp rate of 1 K/sec. *The temperature escalation due to the exothermal zircaloy (Zry)-steam reaction started at about 1100°C*, leading the bundles to maximum temperatures of approximately 2000°C [emphasis added].[246]

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

> The transient of a SFD-type accident is initiated by a slow temperature rise in the order of 0.5 [to] 1.0 K/sec., followed by a *rapid temperature escalation (several tens of degrees Kelvin per second)* due to the exothermal heat produced by the cladding oxidation in steam environment [emphasis added].[247]

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

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

[247] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, p. 1.

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

> In the CORA experiments two different bundle configurations are tested: PWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor) bundles. The PWR-type assemblies usually consist of 25 rods with 16 electrically heated fuel rod simulators and nine unheated rods (full-pellet and absorber rods). Bundle CORA-13, a PWR-type assembly, contained two Ag/In/Cd-steel absorber rods. The test bundle was subjected to temperature transients of a slow heatup rate in a steam environment; *i.e.*, the transient phase of the test was initiated with a temperature ramp rate of 1 K/sec. *The temperature escalation due to the exothermal zircaloy(Zry}-steam reaction started at about 1100°C at an elevation of 850 mm (1000 sec. after [the] onset of the transient), leading to a temperature plateau of 1850°C* and after initiation of quenching to maximum temperatures of approximately 2000°C to 2300°C. CORA-13 was terminated by quenching with water from the bottom with a flooding rate of 1 cm/sec.
>
> Rod destruction started with the failure of the absorber rod cladding at about 1200°C; *i. e.*, about 250 K below the melting regime of steel. Penetration of the steel cladding was presumably caused by a eutectic interaction between steel and the zircaloy guide tube. As a consequence, the absorber-steel-zircaloy melt relocated radially outward and axially downward. Besides this melt relocation the test bundle experienced severe oxidation and partial melting of the cladding, fuel dissolution by Zry/UO_2 interaction, complete Inconel grid spacer destruction, and relocation of melts and fragments to lower elevations in the bundle. An extended flow blockage has formed at the axial midplane.
>
> Quenching of the hot test bundle by water resulted, besides additional fragmentation of fuel rods and shroud, in an additional temperature increase in the upper bundle region. Coinciding with the temperature response an additional hydrogen buildup was detected. During the flooding phase 48% of the total hydrogen [was] generated [emphasis added].[248]

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

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

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

> *contribution to the energy input, resulting in a temperature escalation.* The escalation starts at [the] 950 mm and 750 mm elevation. For the outer fuel rod simulator [number] 3.7 the escalation is delayed at 750 mm by about 150 sec. A possible reason for this delay could be the heat losses due to the window at 790 mm adjacent to this rod. The escalation at the 550 mm elevation follows 200 sec. later. The escalation at 1150 mm develops before that at the 350 mm elevation [emphasis added].[249]

So "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" and "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" both state that temperature escalations due to the exothermic Zircaloy-steam reaction began at approximately 1100°C (2012°F). "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" also states that "having reached about 1000°C [(1832°F)], the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation." [250] Additionally, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states that the "rapid temperature escalation[s were] several tens of degrees Kelvin per second...due to the exothermal heat produced by the cladding oxidation in [a] steam environment."[251]

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

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[249] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, Kernforschungszentrum Karlsruhe, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," 1993, p. 12.

[250] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, Kernforschungszentrum Karlsruhe, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," 1993, p. 12.

[251] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, p. 1.

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

> In the escalation phase; *i.e.*, starting from about 1100°C the slow temperature rise is followed by a rapid increase caused by the increased electric power input *and the additional energy from the exothermal zirconium-steam reaction. The contribution of this exothermal heat to the total energy input is generally between 30 and 40%* [emphasis added].[252]

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

> Based on the accumulated H_2 productions of tests CORA-15, CORA-9, and CORA-7 the oxidation energy is determined. Its percentage amounts to 30 - 45% of the total energy input (electric supply plus exothermal energy)...[253]

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

f. The PHEBUS B9R Test

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[254]

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

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

[254] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development

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

> During the B9R-2 test, an *unexpected strong escalation of the Zr-water reaction occurred* at mid-bundle elevation during the steam injection. Considerable heatup rates of 20 to 30°K/sec. were measured in this zone with steam starved conditions at upper levels. Post Irradiation Examinations (PIE) show cladding failures and considerable deformations (about 70%) [emphasis added].[255]

And offering a different account of the elevation at which the temperature excursion occurred during the PHEBUS B9R-2 test, "Degraded Core Quench: A Status Report" states that the B9R-2 test had "an unexpected high oxidation escalation in the upper bundle zone (20 to 30°K/sec.)"[256] "Degraded Core Quench: A Status Report" states that the temperature excursion occurred in steam-rich conditions, after an initial heatup phase in pure helium (up to 1000°C), and that the PCT was approximately 1900°K, during the first oxidation phase. The PHEBUS B9R-2 test had a second oxidation phase and temperature escalation.[257]

Neither paper states what peak cladding temperatures were at the outset of the autocatalytic oxidation reaction; however, a graph of the cladding-temperature values at the 0.6 meter "hot-level" indicates that the autocatalytic oxidation reaction began when cladding temperatures were below 1477°K (2200°F)[258] (see Appendix G Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF)).

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and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230126, p. 311.

[255] *Id.*

[256] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 14.

[257] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," p. 311.

[258] *Id.*, p. 312.

g. The QUENCH-04 Test

"Degraded Core Quench: Summary of Progress 1996-1999" states that the QUENCH-04 test, conducted at the QUENCH facility at Forschungszentrum Karlsruhe ("FZKA"), with a bundle of 21 electrically-heated, Zircaloy-clad rods, had a temperature excursion at 1560°K (~2350°F), due to rapid oxidation of the Zircaloy cladding. The bundle was heated at an increasing rate of 0.5°K/sec. to 1.5°K/sec. and when peak cladding temperatures reached 1560°K (~2350°F) the temperature excursion occurred.[259]

"Degraded Core Quench: Summary of Progress 1996-1999" also states that runaway (autocatalytic) oxidation of Zircaloy cladding by steam occurs at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or higher.[260]

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

"PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" ("PWR FLECHT Final Report") states that "[t]he objective of the PWR FLECHT...test program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[261]

FLECHT run 9573 was a thermal hydraulic test; however, in some respects it resembled a severe fuel damage test. During FLECHT run 9573, the Zircaloy assembly incurred autocatalytic oxidation.[262]

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[259] T. J. Haste, K. Trambauer, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Degraded Core Quench: Summary of Progress 1996-1999," Executive Summary, February 2000, p. 9.
[260] *Id.*
[261] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 1-1.
[262] See Appendix A for photographs of the assembly from FLECHT Run 9573; see also Appendix B for a photograph of the assembly from FLECHT Run 8874.

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

> Despite the severity of the conditions [of FLECHT Run 9573] and the observed extensive zirconium-water reaction, the oxidation was within the expected range and runaway oxidation [occurred] beyond 2300°F. ...
>
> Westinghouse notes that the metallurgical analyses performed for FLECHT Run 9573 indicated that the measured oxide thickness was still within the expected range for specimens heated as high as 2500°F.[263]

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

Second, when Westinghouse performed the metallurgical analyses for the assemblies from the four FLECHT Zircaloy tests, it compared the measured oxide layer thicknesses to Baker-Just correlation predictions[264]—"the expected range."

Third, an occurrence of runaway (autocatalytic) oxidation at a temperature greater than 2300°F (assuming that means at a temperature below 2400°F) is not within "the expected range" of what the Baker-Just correlation would predict: the Baker-Just correlation predicts that autocatalytic oxidation of Zircaloy occurs at cladding temperatures of approximately 2600°F.[265]

It is significant that in "Denial of Petition for Rulemaking (PRM-50-76)," discussing the metallurgical analyses performed for the Zircaloy FLECHT tests, the NRC states:

> The petitioner did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. The petitioner also ignored the Westinghouse application of the Baker-Just correlation to

[263] 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.

[264] 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.

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

> these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients.[266]

First, neither Westinghouse nor the NRC applied the Baker-Just correlation to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation; furthermore, the NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO2 thickness equations to metallurgical data from the locations of run 9573 that incurred autocatalytic oxidation.

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

Second, the NRC did not consider that FLECHT run 9573 incurred autocatalytic (runaway) oxidation at a temperature "beyond 2300°F,"[267] as Westinghouse's comments regarding PRM-50-84 stated. An occurrence of autocatalytic oxidation at a temperature greater than 2300°F (assuming that means at a temperature below 2400°F) is not within the temperature range of where the Baker-Just correlation would predict autocatalytic oxidation of Zircaloy to occur.

So the NRC performed a technical safety analysis on issues raised in a petition for rulemaking that argued that the Baker-Just and Cathcart-Pawel equations were non-conservative for accurately calculating the extent of the Zircaloy-water reaction that would occur in the event of a LOCA, and the NRC did not consider that, with high probability, run 9573 incurred autocatalytic oxidation at a temperature below 2400°F.[268]

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

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

> The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F.[269]

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

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

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

It is significant that during the AEC's ECCS rulemaking hearings, conducted in the early '70s, that Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[270] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[271] conducted with a

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[269] "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.

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

[271] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-17; this paper cites Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of

Zircaloy assembly. Among other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [metal-water] reactions, in spite of the 'failed' heater rods. They compared test results for SS2N [(conducted with a stainless steel assembly)] with Zr2K, showing satisfactory correlation during approximately the first five minutes of the test with substantial deviations (Zr2K temperatures greater than SS2N) during the subsequent periods of substantial heater failures."[272]

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

> The first complaint [of the BWR-FLECHT tests] was that although all BWR fuel rods are manufactured of a zirconium...alloy, Zircaloy, only 5 of the 143 FLECHT tests utilized [Zircaloy] rods. The remaining 138 tests were conducted with stainless steel...rods. *Since...[Zircaloy] reacts exothermically with water at elevated temperatures, contributing additional energy to that of the decaying fission products, the application of water to the core has the potential of increasing the heat input to the fuel rods rather than cooling them, as desired.* The small number of [Zircaloy] tests in comparison with the total test program was seriously faulted by the CNI [emphasis added].[273]

And discussing the use of stainless steel heater-rod assemblies in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The [stainless steel] rods were apparently chosen primarily for their durability. They could be used repeatedly in testing (for 30 or 40 individual tests) without substantial changes in response over the series.
> ...
>
> On the other hand, *as a result of metal-water reactions, [Zircaloy] rods could be used only once* and then had to be subjected to a destructive post-mortem examination after the test [emphasis added].[274]

General Electric ("GE") argued that the exothermic metal-water reactions were insignificant in the thermal response of the Zircaloy heater rods. Regarding this issue,

Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

[272] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-18.

[273] *Id.*, pp. A8-2, A8-6.

[274] *Id.*, p. A8-6.

"Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> Attempts by GE to show that [metal-water] reactions were insignificant in the thermal response of the rods were not overly convincing since they did not evaluate actual dynamic heat rate inputs but depended instead upon arbitrarily time averaged heat inputs over arbitrary time intervals...[275] Gross estimates were made of the total energy contributed to the thermal transient through the [metal-water] reaction of 1/4 B/inch of cladding length (based upon the maximum observed depth of ZrO_2 penetration for the Zr2K experiment of 1.8 mils). This was compared with a design total delivered decay power to the center of the maximum peaked rod over the 24 minute spray cooling transient of 29.7 B/inch (14.5 B/inch over the first 10 minutes). Thus, GE inferred the total [metal-water] reaction to be 5-10 percent of the decay energy depending upon which of the two time periods was used in the estimation. They acknowledge that the rate of [metal-water reaction] energy addition is more significant than the comparisons with [the] total energy shown above, but state that rate information cannot be obtained from the Zr2K data. Irrespective of the validity of this observation, it seems that comparisons with rod input energy increments taken over 10 to 24 minute intervals are too insensitive to be adequate indications of the significance of the [metal-water reaction] energy contribution. No feeling of confidence is gained that [metal-water] reactions were unimportant as a result of this GE analysis. However, the case for [metal-water reaction] induced thermal runaway in the Zr2K test is equally weak.[276]

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

[275] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, Appendix A.
[276] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-18, A8-19.

rate of 15°K/sec."[277] Furthermore, during the escalation phase of the CORA experiments, the percentage of oxidation energy from the exothermic Zircaloy-water reaction was generally between 30 and 40%, and in some cases was as high as 45%,[278] of the total energy input.[279]

So during the Zr2K test it is highly probable that—like the CORA experiments—the energy from the exothermic Zircaloy-water reaction was between 30 and 40% of the total energy input,[280] not between 5 and 10% as GE estimated. (It is noteworthy that GE "acknowledge[d] that the rate of [metal-water reaction] energy addition [was] more significant than the[ir] comparisons with [the] total energy...but state[d] that rate information [could not] be obtained from the Zr2K data."[281])

Second, when taking into account data from the CORA experiments and other severe fuel damage experiments, it is highly probable that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI...implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[282, 283] It is significant that "in the CORA test facility, [cladding] temperature escalation start[ed] between 1100 and 1200°C [(2012

[277] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.

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

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

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

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

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

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

to 2192°F)], giving rise to a maximum heating rate of 15°K/sec:"[284] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[285]

Furthermore, the graphs of "Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies"[286] and "Analysis of Zr2K Thermal Response"[287] depict thermocouple measurements taken during the Zr2K test that resemble thermocouple measurements taken during severe fuel damage experiments: the graphs depict temperature excursions that began when cladding temperatures reached between approximately 2100 and 2200°F. The graphs depict cladding-temperature values at separate points in approximately 20-second intervals; in some cases the temperature increases by several hundred degrees Fahrenheit within approximately 20 seconds, indicating the onset of temperature excursions, at rates greater than 10°K/sec (see Appendix P Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies and Figure A8.10 Analysis of Zr2K Thermal Response).

It is significant that GE concluded that the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test were not valid. GE stated "that the 'erratic thermocouple outputs do not represent actual cladding temperatures, but

[284] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.

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

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

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

are the result of equipment malfunctions'[288] associated with the Zr2K test."[289] However, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies it is highly probable that GE's claim that the thermocouple measurements did not represent actual cladding temperatures is erroneous; after all, the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test resemble thermocouple measurements of cladding-temperature excursions taken during severe fuel damage experiments.

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

> One of the more difficult aspects of evaluation of Zr2K test results is associated with the fundamental data for the tests, the recorded thermocouple...responses. *GE has been very liberal with their accreditation of observed [thermocouple] responses as erratic.* However, several proffered examples of erratic response seem to show well defined inter-rod correlations. Under such circumstances, "unexplained" might be a better description for the observed [thermocouple] behavior than "erratic" [emphasis added].[290]

Discussing the "well defined inter-rod correlations"[291] that occurred during "the extreme temperature excursion,"[292] "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> A rigorously thorough analysis of the Zr2K thermal response measurements is beyond the scope of this report. It should be noted, however, that the recorded temperatures of rod 16, which developed the first electrical anomaly after the official start of the test, were almost identical to those of rod 24, which was given credit for the maximum temperature measurement. The intra- and inter-rod temperature measurements for rod 16 and its neighbors show consistent correlations over the first two minutes of the transient, in spite of the current anomaly being experienced by the rod (which started essentially at the beginning of the thermal transient test period and lasted for nearly six minutes).

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[288] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," Appendix D, p. 107.
[289] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-24, A8-27.
[290] *Id.*, p. A8-19.
[291] *Id.*
[292] *Id.*, p. A8-21.

> Between 2 and 3 minutes after transient initiation, however, thermocouples...on rod 16 indicate an apparent sharp temperature rise. Because of the anomalous electrical activity of rod 16 at this time, experimental analysts have been inclined to discount this [thermocouple] response as anomalous also. *However, it is interesting to note that the extreme temperature excursion... (adjacent to rod 16) occurred at the same time the rod 16 [thermocouple] excursion occurred and is matched by [the] nearly identical temperature excursion in rod 9, the other rod diametrically adjacent to rod 16. Moreover, it seems entirely too coincidental that temperature turnaround should be achieved in rod 24 at essentially the same time that the actual failure (rod current going to zero) for both rods 16 and 24 occurred.* Under those circumstances, it does not seem surprising that rod 17, still being driven by "normal" electric current and in direct view of the three hottest rods in the test (rods 16, 23. and 24) should then become the highest temperature rod for most of [the] remaining significant portion of the temperature transient. During this period, rods 17 and 23 both underwent electrical anomalies in which excessive currents were delivered to them. It was not until the current to both of these rods actually went to zero, approximately 12 minutes after the thermal transient began, that rod 17 relinquished its role as the highest temperature rod for the test.
>
> The relationships described above seem to indicate a systematic correlation between the electrical anomalies of the "failed" rods and temperature extremes for the bundle [emphasis added].[293]

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

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

> Based upon analysis of the material presented, it appears unquestionable that the [thermocouple] response was badly affected by short circuits and equipment malfunction. The net result is that it is not possible to certify that [metal-water] reactions were insignificant in the measured thermal transient, but the case for near "thermal runaway" proposed by the CNI is also unconvincing. It is probable that most of the dramatic [thermocouple] slope changes, as well as several of the other [thermocouple] aberrations

[293] *Id.*, pp. A8-21, A8-23.

> associated with the test, were short-circuit induced rather than [metal-water] reactions. *However, more results seem to be systematically correlatable between rods [than] the GE test analysis is willing to concede. This leads to uncertainty over the proper interpretation of [the] results. A more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable* [emphasis added].[294]

Indeed, "a more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable."[295] However, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies more than a decade after the Zr2K test, it is clear that GE's claim that the metal-water reactions were insignificant during the Zr2K test is erroneous and that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI…implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[296, 297]

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

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294 *Id.*, p. A8-27.
295 *Id.*
296 Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, p. 5.63.
297 Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-24.

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

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

> [The petitioner] states that the Baker-Just equation does not include any allowance for the complex thermal-hydraulic conditions during a LOCA. The NRC disagrees with the petitioner's assertions. …
>
> The petitioner is also concerned about the large water volume compared to the zirconium sample size with respect to the quench capability of zirconium-clad fuel rods. As noted, these experiments were atypical in that respect, but barely used in the formulation of the Baker-Just correlation. Further, it should be noted that *the Baker-Just report was not intended to be a heat transfer study*, but rather an investigation of zirconium-water reaction kinetics at very high temperatures [emphasis added].[298]

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

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

[298] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 11, 12.

> steam flow typical of LOCA/ECCS conditions. *These are not heat transfer tests*. Once a reaction rate model is developed using data from experiments like these, the model should be validated against transient tests under LOCA conditions, as in the four Zircaloy tests described in WCAP-7665 and the transient tests described in the Cathcart-Pawel report [emphasis added].[299]

So in the first passage above the NRC states that the "the Baker-Just report was not intended to be a heat transfer study, but rather an investigation of zirconium-water reaction kinetics at very high temperatures;"[300] and in the second passage above the NRC states that the zirconium-water reaction kinetics tests that were conducted to develop the Cathcart-Pawel equation were not heat transfer tests. What the NRC does not consider is that under the complex thermal-hydraulic conditions that would occur in the event of a LOCA, heat transfer would affect zirconium-water reaction kinetics.

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

> There doesn't seem to be any magic temperature at which you get some autocatalytic reaction that runs away. *It's simply a matter of heat balances: how much heat from the chemical process and how much can you pull away* [emphasis added].[301]

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

> The high temperature oxidation tests [reported on in WCAP-12610, Appendix E[302]] were performed by Nuclear Electric, plc in the United Kingdom. Twenty four ZIRLO alloy and six Zr-4 samples were tested at temperatures ranging from 1832°F to 2372°F. The cylindrical tubing

[299] *Id.*, pp. 13-14.
[300] *Id.*, p. 12.
[301] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Committee, Meeting, April 4, 2001. In the transcript the second sentence was transcribed as a question; however, the second sentence was clearly not phrased as a question.
[302] Westinghouse, "ZIRLO High Temperature Oxidation Tests," WCAP-12610, Appendix E, August, 1990. Only a limited portion of the report is available to the public; it is classified by Westinghouse as a proprietary report.

specimens were approximately 0.6 inches long and were from production grade 17x17 tubing.

Appendix E candidly discloses: "Since, particularly at high temperatures, the self heating of the specimen results in its being at a higher temperature than its surroundings, any temperature measured will be equal to or lower than that of the test specimen."[303] In other words, in order for the investigators at Nuclear Electric to prevent runaway [oxidation from occurring as a result of] the heat of reaction at high temperatures (self heating), it was necessary to maintain the surroundings at a substantially lower temperature than the specimen. In this manner, the heat loss by radiation to the relatively cold surroundings compensated for the heat produced by chemical reaction with the pure oxygen. This then leads to the question: What if Nuclear Electric had conducted the investigation with a 17x17 arrangement of ZIRLO or Zr-2 tubes captured within a Zircaloy-4 structural grid with ZIRLO thimbles? The answer is that the assembly would have rapidly been destroyed [by] runaway [oxidation] if a sufficient flow of oxygen had been maintained.[304]

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

The [Zircaloy-steam] reaction is highly exothermic (586 kJ/mol), and this may lead to uncontrolled temperature escalation *under conditions of poor heat transfer*; typically at temperatures above about 1500°K" [emphasis added].[305]

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

The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction *crucially depends on the heat loss from the bundle; i.e., on bundle insulation.* With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec.[, after an initial heatup rate of about 1°K /sec.] The maximum temperatures attained are about 2000°C;

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

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

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

> the oxide layers formed and the consumption of the available steam set limits on the temperature escalation due to rate-controlled diffusion processes. The temperature escalation starts in the hotter upper half of the bundle and the oxidation front subsequently migrates from there both upwards and downwards [emphasis added].[306]

So with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between approximately 1100°C and 1200°C (2012°F to 2192°F), and cladding temperatures start increasing at a maximum rate of 15°C/sec. (27°F/sec.). It is certainly evident that in the event of a LOCA, heat transfer would affect the temperature at which the autocatalytic oxidation of Zircaloy cladding would occur. Therefore, it seems obvious that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA, precisely because they were not developed to consider how heat transfer would affect zirconium-water reaction kinetics.

3. Conclusion of the Metal-Water Reaction Rate Section

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

[306] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230460, p. 83.

In the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[307] and 2240°F,[308] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[309] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[310] the melting point of Zircaloy is approximately 3308°F.[311]

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

> The composition of an LWR core is such that melting could occur in a variety of ways involving complex chemical reactions. The major components of a PWR core are UO_2 and Zircaloy, which make up about 78 Wt% and 16 Wt%, respectively. The remaining materials are primarily stainless steel, Inconel, Ag-In-Cd control rod material and Al_2O_3 used in the burnable poison rods. For a BWR core, the major components are UO_2 (68 Wt%) and Zircaloy (24 Wt%). Stainless steel and B_4C control rod material comprise the remaining 8 Wt%. Hofmann *et al.*[312] identified three distinct temperature regimes for melting and liquid phase formation during a severe accident for the heatup rates of 1°K/s or greater. These heatup rates are typical of severe accidents initiated from full power. Each temperature regime is characterized by different processes (Figure 2.1).[313]

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

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

[309] *Id.*

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

[311] 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.

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

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

The first temperature regime considered by Hofmann is between 1473°K and 1673°K. Within this temperature regime, control rods, burnable poison rods, and structural material can form low-temperature liquid phases.[314] These liquefied materials may relocate and form local blockages which could restrict flow and cause accelerated heatup of the core.

In a PWR core, the main reactions are those between the Ag-In-Cd alloy (control rods), Zircaloy (guide tubes), and Inconel (spacer grids). The Ag-In-Cd alloy has a very low melting temperature (1073°K) and it is likely to be the first component to melt after core uncovery. Any failure of the control rod cladding will allow the molten Ag-In-Cd alloy to contact the Zircaloy guide tubes and even some of the Zircaloy cladding around the fuel rods. The Zircaloy can be chemically dissolved by the molten Ag-In-Cd alloy which could cause local damage in the core region well below the melting temperature of Zircaloy (approximately 2033°K).[315] In addition, there could be contact between stainless steel cladding (control rods) and Zircaloy (guide tubes) and between the Inconel (space grids) and Zircaloy (fuel rods). The localized Zircaloy/stainless steel (or Inconel) contact results in chemical interactions with the formation of liquid phases at relatively low temperatures. This early-melt formation at about 1473°K could initiate the melt progression within the fuel assembly at low temperatures as recognized in the CORA tests, where fuel rod bundles were heated up to complete meltdown.[316] Carlson and Cook[317] have shown that the stainless steel/Zircaloy interaction is one of the major material reactions that occurred during the TMI-2 accident.

In a BWR core, the control rods consist of boron carbide (B_4C) pellets in stainless steel cladding; the control rods are located in a four-bladed stainless steel assembly. The major reaction in a BWR is between B_4C

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

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

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

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

and stainless steel. Hofmann *et al.*,[318] showed that the B_4C control rods exhibit a strong reaction above 1523°K, which resulted in rapid liquefaction of the control rod material. Liquefaction occurred below the melting point of B_4C (2623°K) and stainless steel (1723°K) due to eutectic interactions. The liquid B_4C/stainless steel reaction products can also interact with the Zircaloy channel box. Interaction and liquefaction between B_4C and Zircaloy occurs at about 1923°K, which is about 400°K higher than that of the B_4C/stainless steel interaction.

The second temperature regime considered by Hofmann is between 2033°K and 2273°K. If the Zircaloy clad has not been oxidized, then it will melt at about 2033°K and relocate downward along the fuel rod. If a sufficient oxide layer has formed on the outside surface of the clad, then relocation of any molten Zircaloy on the inside will be prevented because the oxide layer will remain solid until the core reaches much higher temperatures (the melting point of ZrO_2 is 2973°K), or until the oxide layer fails mechanically, or until the layer is dissolved by molten Zircaloy. Under these conditions, the molten Zircaloy will chemically dissolve part of the solid UO_2 pellet and ZrO_2 shell.[319] The result is chemical dissolution (*i.e.*, liquefaction) of UO_2 and ZrO_2 by the molten Zircaloy at about 1000°K below the melting points of UO_2 and ZrO_2. The molten (Zr, U, O) mixture and molten metallic Zircaloy flow downward in a "candling process" from the higher temperature regions of the core into the lower temperature region where they solidify. The relocation and solidification of the metallic and ceramic melts could form a blockage in the flow channels, which would inhibit flow and accelerate core damage. Since the mixture contains decay heat, remelting and solidification can occur repetitively as water boils-off and core meltdown proceeds.

The third temperature regime is between 2873°K and 3123°K. If a reactor core ever reaches this high temperature regime, the remaining UO_2, ZrO_2, and the (U, Zr) O_2 solid solution will start to melt. This will lead to complete meltdown of all remaining core materials.[320, 321]

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

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

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

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

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

> In summary, *for a heatup rate of 1°K/s or larger*, core meltdown processes have been characterized for three temperature regimes. Initial melting and relocation in a reactor core starts with the failure of control rods, guide tubes, and Inconel spacer grids *at relatively low temperatures*. Local damage caused by interactions with Zircaloy could also occur during this time period. Larger scale fuel damage occurs at higher temperatures after the metallic Zircaloy melts and dissolution of UO_2 pellets and ZrO_2 occurs. At even higher temperatures, the ZrO_2 and UO_2 melt, which leads to total core meltdown [emphasis added].[322]

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

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

D. FLECHT Run 9573

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

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

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

> The final PWR-FLECHT Zircaloy bundle test, Run 9573, was conducted with a nominal initial cladding temperature of 2000°F and a flooding rate

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

of 1 in./sec. For this test, the stainless steel guide tubes were replaced with Zircaloy guide tubes and the freedom of the heater rods for vertical expansion was increased. Cladding temperatures were predicted to reach 2400°F after about 30 seconds, at which time heater element failures were expected to occur.

During the test, heater element failures started at 18.2 seconds; sixteen elements failed by 30 seconds and all but nine of the forty-two heater elements had failed when power was shut off at 55.5 seconds. At the time of the initial failures, midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure).

Post-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation. The heater rod failures were apparently caused by localized temperatures in excess of 2500°F. Possible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components. The remainder of the bundle was in excellent condition, however, and there was very little rod bowing compared to Run 8874.

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

It should be noted that the heater element failures which occurred in Runs 8874 and 9573 were not related to the behavior of reactor fuel during a loss-of-coolant accident. The failures referred to were failures of the heater rod internal electrical resistance element. Failure of this element resulted in either a loss of power to the heater rod or, more commonly, arcing from the resistance element to the clad. Aside from the regions in which heater rod failures took place, the clad was generally in excellent condition throughout the remainder of the bundles, including the peak temperature midplane regions.[323]

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[323] 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,

First, as mentioned above, "PWR FLECHT Final Report" does not mention that run 9573 incurred autocatalytic oxidation. So Westinghouse omitted very significant information in its report of run 9573. However, Westinghouse does state that "[p]ost-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation."[324, 325]

Second, the passage above has inconsistent conclusions: 1) it essentially states that the heater element failures which occurred in run 9573 were related to the behavior of reactor fuel during a LOCA: "[t]he heater rod failures were apparently caused by localized temperatures in excess of 2500°F. Possible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components;"[326] and 2) then it states that "[i]t should be noted that the heater element failures which occurred in Run...9573 were not related to the behavior of reactor fuel during a loss-of-coolant accident. The failures referred to were failures of the heater rod internal electrical resistance element. Failure of this element resulted in either a loss of power to the heater rod or, more commonly, arcing from the resistance element to the clad."[327]

(It is noteworthy that "PWR FLECHT Final Report" has other similar inconsistencies: 1) it states that during run 9573, "several heaters failed during flooding while the mid-plane temperatures were only 2200-2300°F. The heaters apparently failed because of higher temperatures that developed above the mid-plane region which were most likely caused by steam reaction with a Zircaloy grid;"[328] and 2) elsewhere it states that "[e]ven though the specimens examined reached temperatures as high as 2545°F, there was no evidence of clad shattering or failure as a result of being exposed to typical loss-of-coolant accident environments."[329])

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located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, pp. 3-97, 3-98.

[324] *Id.*, p. 3-97.

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

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

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

[328] *Id.*, p. A-14.

[329] *Id.*, p. 5-5.

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

It is significant that, more than two years after FLECHT run 9573 was completed, at the 1973 ECCS hearing, Westinghouse argued that the regulated limit of the peak cladding temperature ("PCT") in the event of a LOCA should be "at least 2700°F;"[330] in 1973, the limit was 2300°F.[331] So when run 9573 was conducted in December 1970, Westinghouse certainly did not believe that autocatalytic oxidation of Zircaloy cladding would occur at temperatures below 2700°F.

Fourth, the passage above states that "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test;"[332] and concludes that "the data from [the first 18 seconds] is suspect and has therefore not been considered in comparing stainless steel and Zircaloy heat transfer behavior."[333] Elsewhere "PWR FLECHT Final Report" "recommend[s] that stainless steel clad heat transfer coefficients be used as a conservative representation of Zircaloy behavior."[334]

This is highly significant because the data reported in "PWR FLECHT Final Report" is important for ECCS evaluation calculations, required for all holders of operating licenses for nuclear power plants. Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-*

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[330] 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. 1097. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.
[331] *Id.*
[332] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
[333] *Id.*, pp. 3-97, 3-98.
[334] *Id.*, p. 5-3.

Blowdown Phenomena, Refill and Reflood Heat Transfer for Pressurized Water Reactors, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]."

Fifth, the passage above concludes that the negative heat transfer coefficients, found in the analysis of the test results of run 9573—indicating "heat transfer into (rather than out of) the rod"[335]—were, in fact, "anomalous."

The passage states that "anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period"[336] (*i.e.*, more heat was transferred into the bundle midplane than was removed from that location), and posits that the "anomalous" negative heat transfer coefficients "may have been related to the high steam probe temperatures measured at the 7 ft elevation."[337] The passage also posits that "[p]ossible causes of the high temperatures include metal-water reaction of (a) the Zircaloy grid, (b) the Zircaloy steam probe or (c) a eutectic solution of the steam probe stainless steel and Zircaloy components."[338]

(It is noteworthy that in 2002, regarding this phenomenon, in Westinghouse's comments on PRM-50-76, Westinghouse stated that "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[339])

So "PWR FLECHT Final Report" concludes that the negative heat transfer coefficients were anomalous, yet it also posits that the phenomenon of "heat transfer into (rather than out of) the rod"[340] was caused by heat generated from the exothermic Zircaloy-steam reaction. "PWR FLECHT Final Report" offers no credible explanation for concluding that the negative heat transfer coefficients were anomalous.

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[335] *Id.*, p. 3-40.
[336] *Id.*, p. 3-98.
[337] *Id.*
[338] *Id.*, p. 3-97.
[339] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[340] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-40.

"PWR FLECHT Final Report" does not even consider the possibility that the experimental data from the first 18.2 seconds of run 9573 is valid, even though it states that "[t]he heater rod failures were...caused by localized temperatures in excess of 2500°F;"[341] and that "the 7 ft steam probe [measured temperatures], which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure)."[342] Indeed, Westinghouse's decision to not consider the data from the first 18.2 seconds of FLECHT run 9573, for comparing stainless steel and Zircaloy heat transfer behavior, seems unscientific.

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

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

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

> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.[343, 344]

[341] *Id.*, p. 3-97.
[342] *Id.*
[343] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.
[344] See Appendix I Memorandum RD-TE-70-616, FLECHT Monthly Report, December 14, 1970.

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

> Preliminary results of...Zirc[aloy] Run 9573 are summarized in the attachment. The run is considered valid up to the point of the first heater failure at 18.2 sec.[345, 346]

So Leyse's, December 14, 1970, memorandum states that the temperature-measuring system used for FLECHT run 9573 was subjected to a thorough audit by Idaho Nuclear Corporation that found that the system was highly reliable. And Cadek's, December 18, 1970, memorandum explicitly states that FLECHT run 9573 "is considered valid up to the point of the first heater failure at 18.2 sec."[347]

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

It was because of the "uncertain and conflicting evidence"[348] of the four Zircaloy runs that, in 1973, the Commissioners of the AEC stated, "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from [FLECHT] run 9573."[349]

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

> The [four] FLECHT runs made with zircaloy clad rods provide uncertain and conflicting evidence. Westinghouse pointed out that all of the zircaloy runs except one (run 9573) yield higher heat transfer coefficients than were obtained with [stainless] steel... Consolidated National Intervenors pointed out that most of [the Zircaloy] runs were made at

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

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

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

348 Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

349 *Id.*

unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding rate was about one inch per second.[350]

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

> The stainless steel heat transfer behavior is certainly not a conservative representation of Zircaloy behavior. The data for the first 18 seconds of Run 9573 are real and certifiable. There is no basis for rejecting the negative heat transfer coefficients in run 9573. The higher values of the heat transfer coefficients of Run 8874 are also valid. The differences in the behavior between these runs are explained by the differences in the thermal hydraulic conditions that led to a different combination of heat transfer and mass transfer factors; the differences are not explained on the basis of inconsistency of the data.[351]

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

> The negative heat transfer coefficients [occurring within the first 18.2 seconds of run 9573] were calculated as a result of a heat transfer condition during which more heat was being transferred into the heater than was being removed from the heater[; used in the FLECHT tests to simulate fuel rods]. And the reason for that condition was that the heat generated from Zircaloy-water reactions at the surface of the heater added significantly to the linear heat generation rate at the location of the midplane thermocouples.[352]

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

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

> The initial heat transfer coefficient[353] is at least 1.7 times higher in a Zircaloy bundle (run 8874) than in a stainless [steel] bundle (run 6155) for

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[350] *Id.*
[351] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 8.
[352] *Id.*, p. 6.
[353] "The initial heat transfer coefficient," refers to the heat transfer coefficient during the first 10 seconds of run 8874, after flooding; see "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-96.

> the same flooding rate (6 in./sec.[354]) and start-of-flood temperatures of 2300°F and 2200°F, respectively. The higher coefficients for the Zircaloy bundle may be explained by high hydrogen concentrations (20% or more) in the film at the surface of the heater. At 2000°F, the thermal conductivity of hydrogen is approximately five times that of superheated steam. Although hydrogen production rates are probably not sufficient to lead to significant concentrations in the bulk coolant (the mixture of superheated steam and water droplets), the hydrogen concentrations within the film at the surface of the heater can easily reach significant values.[355, 356]

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

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

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[354] The flood rate of run 8874 was 6.0 in./sec. for 8 seconds, followed by a step reduction to 1 in./sec.; the flood rate of run 6155 was a constant 5.9 in./sec.; see "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," pp. 3-6, 3-8, 3-96, B-2.
[355] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum PA-TE-70-419, "Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874)," July 24, 1970.
[356] See Appendix K Memorandum PA-TE-70-419, Higher Initial Heat Transfer Coefficients Zircaloy Bundle (Run 8874), July 24, 1970.

> reactions can lead [to] steeper temperature ramps in the vicinity of a Zircaloy grid.[357, 358]

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

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

> The negative heat transfer coefficient for the 10-foot elevation at early times indicates heat transfer into (rather than out of) the rod. This was caused by the presence of superheated steam having temperatures above the [stainless steel] clad temperature at the 10-foot elevation. ... Negative heat transfer coefficients were generally found at the 10-foot elevation for low flooding rate runs (2 in./sec. or less) at early times (from around 5 up to a maximum of about 120 sec. after flood).[359]

So FLECHT run 9573 was not the only FLECHT run where an analysis of the test results found negative heat transfer coefficients, indicating "heat transfer into (rather than out of) the rod."[360] In the case of run 9573, the presence of superheated steam caused the Zircaloy cladding to rapidly oxidize—an exothermic reaction that, in turn, generated yet more heat. At such temperatures the reaction became autocatalytic.

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

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

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

359 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-40.

360 *Id.*

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

First, it is noteworthy that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400°K. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[361] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425°K. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain*" [emphasis added].[362]

FLECHT stainless steel runs 6553 and 9278 (with the same peak power levels as Zircaloy run 9573), at the hot rod midplane elevation, at the onset of flood, had cladding temperatures of 2012°F and 2028°F, respectively, flood rates of 1 in./sec., and peak cladding temperatures of 2290°F and 2286°F, respectively.[363] In contrast to Zircaloy run 9573—with a slightly lower clad temperature at the onset of flood and a slightly higher flood rate—runs 6553 and 9278 did not incur autocatalytic oxidation reactions. In fact, runs 6553 and 9278 were conducted with the same stainless steel assembly, and after run 9278 was conducted, the assembly was reused for more tests, because it remained intact.

Discussing the durability of stainless steel heater-rod assemblies in the FLECHT program, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[s]tainless steel was used instead of Zircaloy as the cladding material for nearly all of the FLECHT tests because it is more durable under the test conditions."[364]

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361 S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2.
362 *Id.*, p. 4.4.
363 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-6.
364 Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1123. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

And also discussing the durability of stainless steel heater-rod assemblies in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The [stainless steel] rods were apparently chosen primarily for their durability. They could be used repeatedly in testing (for 30 or 40 individual tests) without substantial changes in response over the series. . . .
>
> On the other hand, *as a result of metal-water reactions, [Zircaloy] rods could be used only once* and then had to be subjected to a destructive post-mortem examination after the test [emphasis added].[365]

And regarding the differences in the oxidation behavior of Zircaloy and stainless steel heater-rod assemblies, Robert H. Leyse, the principal engineer in charge of directing the Zircaloy PWR FLECHT tests and one of the authors of "PWR FLECHT Final Report," states:

> There is no reason to believe that the temperature measuring system was not reliable for the first 18 seconds of run 9573. The negative heat transfer coefficients were real values[; *i.e.*, a phenomenon where more heat was transferred into the bundle midplane than was removed from that location]. This is because there is an extremely significant and real difference between the behavior of Zircaloy and stainless steel heat transfer assemblies. In the case of stainless [steel], there is relatively little heat of reaction from oxidation in the temperature range. In contrast, the heat of reaction [from] oxidation of...Zircaloy is substantial in the temperature range. The intense heat of reaction yielded high enough temperatures of the Zircaloy cladding to force heart flow back into the heater. The thermocouples did not yield false signals. There is no justification for classifying the negative heat transfer coefficients as anomalous.
>
> The following FLECHT experience provides a very direct comparison of the high temperature behavior of Zircaloy and stainless steel bundles: Although there is no discussion in any of the FLECHT reports, on April 18, 1969, a stainless steel bundle was substantially overheated due to installation and operating errors. The event is discussed in...Westinghouse memo RD-ED-THE-33. [The memo states], "The maximum temperature of the [stainless steel] bundle was in excess of 2500°F (Chromel-Alumel thermocouple conversion tables terminate at

[365] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-6.

> 2500°F)."[366, 367] The bundle remained totally intact without any destruction of the stainless steel cladding, although most of the heating elements had burned out. This is in marked contrast to the experience with FLECHT Run 9573.[368]

Indeed, stainless steel cladding heat transfer coefficients are not a conservative representation of Zircaloy heat transfer coefficients, for some of the conditions that would occur in the event of a LOCA. It is significant that for run 9573 the "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test."[369] Yet the data from run 9573 is not considered valid. And "PWR FLECHT Final Report" states:

> Properly used, PWR FLECHT test results can improve the accuracy of reactor LOCA analysis. The heat transfer correlations which were developed are *conservative* in that they do not take any credit for the effects of "fallback" or borated coolant and are *based on stainless steel clad data* [emphasis added].[370]

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

5. Conclusion of the FLECHT Run 9573 Section

It is significant that FLECHT run 9573 incurred autocatalytic oxidation and had a lower initial cladding temperature than, and the same power level as, other FLECHT Zircaloy tests that did not incur autocatalytic oxidation. The primary difference between run 9573 and the other FLECHT Zircaloy tests was that run 9573 had the lowest flood

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366 R. F. Farman, Westinghouse, Thermal and Hydraulic Experimentation, Memorandum RD-ED-THE-33, "Report of Events Leading to FLECHT 10 x 10 Bundle Test," April 23, 1969, p. 2.
367 See Appendix L Memorandum RD-ED-THE-33, Report of Events Leading to FLECHT 10 x 10 Bundle Test, April 23, 1969.
368 Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.
369 F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. 3-97.
370 *Id.*, p. 5-4.

rate (see Appendix C Table B-1. Group III Test Results). "Consolidated National Intervenors pointed out that most of [the Zircaloy] runs were made at unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding rate was about one inch per second."[371]

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

FLECHT run 9573 demonstrates that the metal-water reaction becomes autocatalytic at temperatures lower than what the Baker-Just and Cathcart-Pawel equations predict. Westinghouse stated that run 9573 incurred autocatalytic oxidation at a temperature greater than 2300°F[372] (most likely, meaning at a temperature below 2400°F); the Baker-Just and Cathcart-Pawel equations predict that autocatalytic oxidation of Zircaloy cladding occurs at approximately 2600°F and 2700°F, respectively.[373]

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

[371] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

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

[373] 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."

E. The Thermal Resistance of Crud and/or Oxide Layers on Fuel Cladding and ECCS Evaluation Calculations for Postulated LOCAs

1. The NRC's Proposed Revisions to the ECCS Acceptance Criteria of 10 C.F.R. § 50.46(b)

It is significant that, regarding the NRC's proposed revisions to the ECCS acceptance criteria of 10 C.F.R. § 50.46(b), "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" states:

> [T]he NRC is working to revise the ECCS acceptance criteria in § 50.46(b) to account for new experimental data on cladding ductility and to allow for the use of advanced cladding alloys. ... The NRC expects that this rulemaking (Docket ID NRC–2008–0332) will establish new cladding embrittlement acceptance criteria in § 50.46(b) for design basis LOCAs. As these new acceptance criteria are established, the NRC will also make conforming changes to § 50.46a as necessary for both below and above TBS breaks.[374]

The NRC's proposed revisions to the ECCS acceptance criteria of 10 C.F.R. § 50.46(b), also include addressing the issues raised in PRM-50-84.[375]

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

2. Appendix K to Part 50—ECCS Evaluation Models, and the Stored Energy in Fuel Sheathed within Crudded and Oxidized Cladding

Appendix K to Part 50—ECCS Evaluation Models I(A)(1), *The Initial Stored Energy in the Fuel*, requires that "[t]he steady-state temperature distribution and stored

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374 NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Federal Register, August 10, 2009, p. 40030.

375 NRC, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," Federal Register, November 25, 2008, pp. 71564-71568.

energy in the fuel before [a] hypothetical accident...be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally the highest calculated stored energy)."

Clearly, the primary purpose of Appendix K to Part 50, regarding the stored energy in the fuel, is to require that the stored energy in the fuel be calculated that "yields the highest calculated cladding temperature" or PCT. Therefore, because layers of crud and/or oxide increase the quantity of stored energy in the fuel, Appendix K to Part 50 must require that the thermal conductivity of layers of crud and/or oxide be factored into calculations of the stored energy in the fuel.

To calculate "the steady-state temperature distribution and stored energy in the fuel...for the burn-up that yields the highest calculated cladding temperature" Appendix K to Part 50 requires that:

> [T]he *thermal conductivity* of the UO2...be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the *thermal conductance* of the gap between the UO2 and the cladding...be evaluated as a function of the burnup, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances and cladding creep [emphasis added].

The "thermal conductivity of the UO2" and the "thermal conductance of the gap between the UO2 and the cladding" are obviously important for calculating "the steady-state temperature distribution and stored energy in the fuel...for the burn-up that yields the highest calculated cladding temperature;" therefore, it seems obvious that the effect of the thermal conductivity of layers of crud and/or oxide that increases the stored energy in the fuel must also be taken into account for this calculation.

Regarding how a heavy crud layer would increase the initial stored energy in the fuel during a LOCA, James F. Klapproth, Manager, Engineering and Technology at GE Nuclear Energy, states, "[one of the] primary effects of [a] heavy crud layer during a postulated LOCA would be an increase in the fuel stored energy at the onset of the event."[376]

[376] Letter from James F. Klapproth, Manager, Engineering and Technology, GE Nuclear Energy to Annette L. Vietti-Cook, Secretary of the Commission, NRC, April 8, 2002, located at:

The fact that a heavy crud layer would increase the quantity of stored energy in the fuel at the onset of a LOCA is significant; it means that the value of the PCT would also increase, above that of fuel with the same burnup, sheathed within clean cladding. (Of course, this does not hold for fresh, BOL fuel, because such fuel has clean cladding at the beginning of its use.) And heavily crudded one-cycle fuel has a higher quantity of stored energy in the fuel than BOL fuel. It has been documented that crud has caused cladding temperatures to increase by over 270[377] or 600°F[378] during operation. Furthermore, the effects of crud can be quick; *e.g.*, at TMI-1 Cycle 10, one-cycle fuel had a cladding perforation detected, caused by corrosion, only 121 days into the cycle.[379] It is also significant that most of the cladding that experienced crud-induced corrosion failures recently at PWRs was high-power, one-cycle cladding,[380] and that the cladding that experienced crud-induced corrosion failures at River Bend Cycles 8 and 11 was high-power, one-cycle cladding,[381] and that crud layers approximately 100 μm thick at Callaway Cycle 6 were on high-power, one-cycle cladding.[382]

3. Appendix K to Part 50 Already Requires Modeling the Thermal Resistance of Crud and/or Oxide Layers on Fuel and Fuel Cladding

Although it is not explicitly stated in Appendix K to Part 50, Appendix K to Part 50 already requires modeling the affects of crud and/or oxide layers on fuel and fuel

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www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021020383.

377 R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," American Nuclear Society, Proceedings of the *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida, September 19-22, 2004, p. 342.

378 NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," 02/28/06, Report Details, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML060600503, p.10.

379 R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.

380 NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 235.

381 See Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs;" see also Edward J. Ruzauskas and David L. Smith, "Fuel Failures During Cycle 11 at River Bend," American Nuclear Society, Proceedings of the *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida, September 19-22, 2004, pp. 221-222.

382 Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs."

cladding, because the thermal resistance of such layers on cladding increases the fuel rod internal pressure and affects the fuel-cladding gap width. Internal pressure and the status of the fuel-cladding gap width are phenomena that Appendix K to Part 50 currently requires to be factored into calculations of the stored energy in the fuel.

It is essential that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of crud and/or oxide layers on fuel cladding plays in increasing the stored energy in the fuel. In addition to increasing the stored energy in the fuel, the thermal resistance of crud and/or oxide layers on cladding also increases fuel rod internal pressure.

Regarding this phenomenon, NRC document, "Safety Evaluation by the Office of Nuclear Regulation, Topical Report WCAP-15604-NP. REV. 1, 'Limited Scope High Burnup Lead Test Assemblies' Westinghouse Owners Group, Project No. 694," states:

> Clad[ding] oxidation can lead to significantly increased fuel rod internal pressures. Above certain oxidation levels, the impacts on rod internal pressure and the significant impacts on the cladding pressure limit characteristics could result in the rod internal pressure criterion being exceeded. Therefore, if oxidation is kept to a minimum, the fuel rod internal pressure criterion is less limiting than simply the oxidation criterion by itself. ... In addition to oxidation causing increases in rod internal pressures, crud deposition has a similar effect since crud is a poor conductor of heat. Keeping crud deposition to a minimum also reduces the impact on rod internal pressures.[383]

The "fuel rod internal pressure criterion" referred to in the above citation is "a criterion requiring that the internal pressure of the fuel rod not exceed reactor coolant system pressure."[384] Concerning cladding sheathing high burnup fuel, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation" explains that fuel-cladding gap reopening may occur "when internal pressure in the [fuel] rod exceeds

[383] NRC, "Safety Evaluation by the Office of Nuclear Regulation, Topical Report WCAP-15604-NP. REV. 1, 'Limited Scope High Burnup Lead Test Assemblies' Westinghouse Owners Group, Project No. 694," 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070740225 (See Section A), p. 4.

[384] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998, located at: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1998/in98029.html (accessed on 01/21/07).

reactor coolant system pressure."[385] Concerning the possibility of gap reopening due to the low thermal conductivity of oxide layers on high burnup cladding, "NRC Information Notice 98-29" states:

> Using the corrected corrosion model, Westinghouse interpreted the PAD [computer code (Westinghouse Improved Performance Analysis and Design Model)] results to indicate that the degraded thermal conductivity of the cladding due to the higher oxidation levels produced an increase in fuel cladding temperatures and consequent higher clad creep rates. These higher creep rates could, in turn, lead to gap reopening, which would be contrary to a Westinghouse design criterion.[386]

It is significant that the thermal resistance of crud and/or oxide layers on cladding increases the fuel rod internal pressure and affects the fuel-cladding gap width, because internal pressure and the status of the fuel-cladding gap width are phenomena that Appendix K to Part 50 currently requires to be factored into calculations of the stored energy in the fuel. To calculate "the steady-state temperature distribution and stored energy in the fuel...for the burn-up that yields the highest calculated cladding temperature" Appendix K to Part 50 requires that:

> [T]he thermal conductivity of the UO2...be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO2 and the cladding...be evaluated as a function of the burnup, taking into consideration fuel densification and expansion, the composition and *pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances and cladding creep* [emphasis added].

Clearly, not realistically modeling crud and/or oxide layers in ECCS evaluation calculations would already be a violation of Appendix K to Part 50, because Appendix K to Part 50 requires that ECCS evaluation calculations "[take] into consideration...the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances and cladding creep," to determine "the thermal conductance of the gap between the UO2 and the cladding." If ECCS evaluation calculations do not factor in the thermal resistance of crud and/or oxide layers on cladding, such calculations will not properly determine "the thermal conductance of the gap between the UO2 and the cladding" or "the steady-state temperature distribution and stored energy in the fuel."

[385] Id.
[386] Id.

And improperly calculating "the steady-state temperature distribution and stored energy in the fuel" would undermine the primary purpose of Appendix K to Part 50, regarding the stored energy in the fuel: to calculate the stored energy in the fuel that "yields the highest calculated cladding temperature."

It is also significant that, in some cases, thick crud and oxide layers have quickly accumulated on one-cycle cladding sheathing high-duty fuel. (At Three Mile Island Unit 1 Cycle 10, such cladding was perforated by oxidation only 121 days into the cycle.[387]) It is highly probable—because of substantial increases in fuel rod internal pressure—that quickly accumulated layers of crud and oxide on one-cycle cladding sheathing high-duty fuel would slow down fuel-cladding gap closure from normal closure rates, during operation or prevent fuel-cladding gap closure, altogether. And prevent cladding from "creep[ing] down towards the fuel pellets, due to the system pressure exceeding the [fuel] rod internal pressure...relatively early in the first cycle of operation"[388] (as a recent Entergy document, describes clean-cladding behavior at pressurized water reactors). This effect would prevent the reduction of the average temperature "at the hot spot [of the fuel rod] by several hundred degrees [Fahrenheit] relatively early in the first cycle of operation"[389] (as the same Entergy document, describes fuel (sheathed in clean-cladding) behavior).

It is clear that crud and/or oxide layers on cladding affect the stored energy in the fuel in two ways: 1) their external thermal resistance increases the stored energy in the fuel and 2) their external thermal resistance increases the fuel rod internal pressure and affects the fuel-cladding gap width, which, in turn, affects the thermal conductance of the fuel-cladding gap and the quantity of the stored energy in the fuel. Therefore, it is imperative that the NRC amend Appendix K to Part 50 to require that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of crud and/or oxide

[387] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.
[388] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, p. 6.
[389] Id.

layers on cladding plays in increasing the stored energy in the fuel, and that Appendix K to Part 50 also provide instructions for how to carry out calculations that factor in the role that the thermal resistance of crud and/or oxide layers on cladding plays in determining the quantity of stored energy in the fuel at the onset of a postulated LOCA. Such requirements also must apply to any NRC approved best-estimate ECCS evaluation models used in lieu of Appendix K calculations.

4. There is Little or No Evidence that the Thermal Resistance of Crud has Ever been Properly Factored into ECCS Evaluation Calculations for Postulated LOCAs

As already discussed, the thermal resistance of crud layers on fuel cladding is very significant for how cladding would be affected during a LOCA; their thermal resistance would increase the PCT. However, there is little or no evidence that crud has ever been properly factored into ECCS evaluation calculations for postulated LOCAs for nuclear power plants.

An attachment to a letter dated June 17, 2003 from Gary W. Johnsen, RELAP5-3D Program Manager, Idaho National Engineering and Environmental Laboratory ("INEEL"), to Robert H. Leyse states:

> [W]e are not aware of any user who has modeled crud on fuel elements with SCDAP/RELAP5-3D. ... We suspect that none of the other [severe accident analysis] codes have been applied to consider [fuel crud buildup] (because it has not been demonstrated conclusively that this effect should be considered). ... SCDAP/RELAP5-3D *can* be used to consider this effect, it is simply that users have not chosen to consider this phenomen[on] [emphasis not added].[390]

An example of not properly factoring the thermal conductivity of crud into a PCT calculation for a postulated LOCA is in "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," dating from 2002. It states, "+4.0°F Cycle 6 crud deposition penalty has been deleted. A PCT penalty of 0°F has been assessed for 4 mils [(~100 μm)] of crud, provided BOL conditions remain limiting. In the event that the

[390] From an attachment of a letter from Gary W. Johnsen, RELAP5-3D Program Manager, INEEL to Robert H. Leyse, June 17, 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML032050508.

SBLOCA cumulative PCT becomes ≥ 1700°F, this issue must be reassessed."[391] Clearly, little attention was given to the thermal resistance of the heavy crud layer at Callaway Cycle 6 (1993), which affected high-duty, one-cycle cladding, at the upper spans 4, 5, and 6 of the fuel assembly.[392]

A recent paper, "The Chemistry of Fuel Crud Deposits and its Effect on AOA in PWR Plants," describing computer codes that model chemical conditions and heat transfer within crud deposits, helps clarify the magnitude of the error of the Callaway Cycle 6 ECCS evaluation: it states that a crud layer that is 59 μm thick is modeled so that "the rise in temperature [from the water side to the fuel side of the layer] is dramatic, reaching temperatures near 400°C [at the fuel side]," up from around 345°C at the water side of the layer.[393] This means, according to the calculations of these codes, that a 59 μm crud layer increases cladding surface temperatures by approximately 55°C or 100°F during operation. And also, according to the calculations of these codes, that a 100 μm crud layer would increase cladding temperatures by more than 100°F during operation. Therefore, according to these codes, at onset of a postulated LOCA, at Callaway Cycle 6, the temperature of the cladding, at some locations, would be over 100°F higher than it would be if the cladding were clean: this would result in a substantially higher than "+4.0°F...crud deposition penalty"[394] for the Cycle 6 calculated PCT.

It is significant that "The Chemistry of Fuel Crud Deposits and its Effect on AOA in PWR Plants" states that the "rise in temperature [across crud layers] was not accounted for in previous models [of crud layers]."[395] And significant that these computer codes that model chemical conditions and heat transfer within crud deposits do not seem to

[391] Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," October 14, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML023010263, Attachment 2, p. 6, note 3.

[392] Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs."

[393] Jim Henshaw, John C. McGuire, Howard E. Sims, Ann Tuson, Shirley Dickinson, Jeff Deshon "The Chemistry of Fuel Crud Deposits and Its Effect on AOA in PWR Plants," 2005/2006, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML063390145, p. 8.

[394] Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," Attachment 2, p. 6, note 3.

[395] Jim Henshaw, John C. McGuire, Howard E. Sims, Ann Tuson, Shirley Dickinson, Jeff Deshon "The Chemistry of Fuel Crud Deposits and Its Effect on AOA in PWR Plants," p. 8.

model morphologies of crud that have been documented to increase local cladding temperatures by over 180 or 270°F or greater during PWR operation. Therefore, it is possible that the actual thermal resistance of the crud at Callaway Cycle 6 was greater than what these computer codes would predict. In reality, the increase in temperature across the 100 μm crud layer might have been significantly greater than what these codes would have calculated in 2005/2006, when the paper was written.

5. Considering the Thermal Resistance of a Crud Layer on Fuel Cladding in an ECCS Evaluation Calculation for a Postulated LOCA

The paper "Considering the Thermal Resistance of Crud in LOCA Analysis" reports on an ECCS evaluation calculation for a postulated LB LOCA that factored in the thermal resistance of a crud layer on fuel cladding.

"Considering the Thermal Resistance of Crud in LOCA Analysis" states:

> For this work, we used a RELAP5-3D model of a reference Westinghouse four-loop PWR plant that MIT developed for a previous study.[396] RELAP5-3D simulated a LBLOCA—a double-ended guillotine break—at the modeled plant: surface temperatures of clean fuel cladding ("the reference case") were compared to those of fuel cladding with a 100 μm thick crud layer ("the crud case"). The reference case and the crud case both examined the surface temperatures of the hottest fuel rod of the hottest assembly. The crud layer was assigned a thermal conductivity of 0.8648 W/mK;[397, 398] the heat capacity of the crud layer was assigned the same value as that of the fuel cladding: these values would be close under the high-temperature conditions of the postulated LBLOCA.[399, 400]

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396 D. Feng, *et al.*, "Safety Analysis of High-Power-Density Annular Fuel for PWRs," Nuclear Technology, Vol. 160 Iss. 1, 2007, p. 45 – 62.

397 Pacific Northwest National Laboratory, NUREG/CR-6534, Volume 2, "Frapcon-3: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," 1997, p. 2.8.

398 NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.14-4.

399 Petrova et al., "Thermophysical Properties of Zirconium Alloy E110 (Zr-0.01Nb) After Oxidation in Air Atmosphere", International Journal of Thermophysics, Vol. 23, No. 5, 2002.

400 Rui Hu, Mujid S. Kazimi, Mark E. Leyse, "Considering the Thermal Resistance of Crud in LOCA Analysis," American Nuclear Society, 2009 Winter Meeting, Washington, D.C., November 15-19, 2009.

Regarding the affect of the thermal resistance of the crud layer, the conclusion of "Considering the Thermal Resistance of Crud in LOCA Analysis" states:

> The RELAP5-3D analysis demonstrated that the PCT of the crud case is 77°K higher than that of the reference case, for the postulated LBLOCA. Hence, the thermal resistance of crud deposits on fuel cladding should be considered in LOCA analysis for licensing and other related activities. It should be noted that, conservatively, a uniform crud layer was modeled with RELAP5-3D, not a varied crud layer, whose thinner portions would offer less thermal resistance. To better understand how crud would affect the severity of a LOCA, further investigations are required; for example, the full range of thermal conductivity of crud should be established and whether crud deposits of any thickness will continue to adhere to fuel cladding under LOCA conditions should be investigated. Finally, recent work at MIT shows considerable advantageous effect, during quenching of hot surfaces, of nano-particle deposits on the surfaces.[401] Implications of this work should be considered as well.[402]

So the RELAP5-3D analysis demonstrated that the PCT of the crud case was 77°K higher than that of the reference case and that the thermal resistance of crud deposits on fuel cladding should be considered in LOCA analysis. This is significant because after decades of operating experience, heavy crud and/or oxide layers on cladding remain within the realm of anticipated operational occurrences at nuclear power plants. Moreover, power uprates and longer fuel cycles increase the likelihood of heavy crud and/or oxide layers on cladding.

F. Conclusion of the Background Section

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,

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[401] H. Kim, T. McKrell, G. Dewitt, J. Buongiorno, L. W. Hu, "On the Quenching of Steel and Zircaloy Spheres in Water-Based Nanofluids with Alumina, Silica and Diamond Nanoparticles", Int J. Multiphase Flow, 35, 2009, p. 427–438.

[402] Rui Hu, Mujid S. Kazimi, Mark E. Leyse, "Considering the Thermal Resistance of Crud in LOCA Analysis," American Nuclear Society, 2009 Winter Meeting, Washington, D.C., November 15-19, 2009.

> 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.[403]

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

First, the NRC has ignored the data from the NRU thermal-hydraulic and mechanical deformation tests that indicates that, in the event a LOCA, a constant core reflood rate of approximately 1 in./sec. or lower would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. For example, in NRU Thermal-Hydraulic Experiment 1, test no. 127, with a reflood rate of 1.0 in./sec., had a peak clad temperature at the start of reflood of 966°F and an overall peak clad temperature of 1991°F (an increase of 1025°F) (see Appendix D Table 1. Experimental Heat Cladding Temperatures).

It is noteworthy that in 2005, the NRC stated that it was "reviewing...data from [the early '80s, from the NRU thermal-hydraulic and mechanical deformation test] program to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[404]

It is also noteworthy that in 1975, the paper, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" stated, "[r]ecommendations are made for improvements in criteria conservatism, especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates)."[405]

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

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

405 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.

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

"Compendium of ECCS Research for Realistic LOCA Analysis," states that "[a]ssessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature;"[406] 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."[407] However, "Compendium of ECCS Research for Realistic LOCA Analysis" does not mention that, during the LOFT LP-FP-2 experiment, autocatalytic oxidation occurred at cladding temperatures greater than either 2060°F[408] or 2240°F.[409]

The LOFT LP-FP-2 experiment was the most realistic severe fuel damage experiment that was conducted, so its temperature excursion data is very important for illustrating what, with high probability, would occur in the event of a LOCA at a PWR, if cladding temperatures were to reach between approximately 2060°F and 2240°F. The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[410] The LOFT LP-FP-2 experiment had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy 1.67 meter fuel rods; it was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud.[411]

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406 NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
407 *Id.*
408 J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
409 R. R. Hobbins, *et al.*, "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," p. 7; this paper cites M. L. Carboneau, *et al.*, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2" as the source of this information.
410 T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.
411 *Id.*

So the LOFT LP-FP-2 experiment was conducted with a well-insulated test assembly. This is significant, because the CORA experiments demonstrated that "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation;"[412] and that with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, begins when the cladding reaches between 2012°F and 2192°F, and that then cladding temperatures start increasing at a maximum rate of 27°F/sec; "a rapid [cladding] temperature escalation, [greater than 18°F/sec.], signal[s] the onset of an autocatalytic oxidation reaction."[413]

The LOFT LP-FP-2 experiment was the only experiment that combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[414]

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

> Data from [the LOFT LP-FP-2] experiment provide a wealth of information on severe accident phenomenology. The results provide important data on early phase in-vessel behavior relevant to core melt progression, hydrogen generation, fission product behavior, the composition of melts that might participate in core-concrete interactions, and the effects of reflood on a severely damaged core. The experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
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> 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.[415]

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[412] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, p. 83.
[413] F. E. Panisko, N. J. Lombardo, Pacific Northwest Laboratory, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230126, p. 282.
[414] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, "Degraded Core Quench: A Status Report," p. 13.

It is noteworthy that, in 1985, the same year the LOFT LP-FP-2 experiment demonstrated that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures within the range of approximately 140°F below to 40°F above the 10 C.F.R. § 50.46(b)(1) PCT limit, "the [NRC] ruled by fiat in its Severe Accident Policy Statement that 'existing plants pose no undue risk to health and safety' and that no regulatory changes were required to reduce severe accident risk."[416]

It is also noteworthy that in 1983—five years before the NRC issued the regulations in Regulatory Guide 1.157, the best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50—the NRC, "[i]n recognition of the known conservatisms in Appendix K, ...adopted an interim approach..., described in SECY-83-472, *to accommodate industry requests* for improved evaluation models *for the purpose of reducing reactor operating restrictions*. This interim approach was a step in the direction of basing licensing decisions on realistic calculations of plant behavior" [emphasis added].[417]

So in 1983, the same year that the PBF Severe Fuel Damage 1-1 Test, according to some reports, had an onset of autocatalytic oxidation of Zircaloy cladding at approximately 2420°F[418] (the Baker-Just equation predicts that it occurs at approximately 2600°F[419]), and had results where a "rapid temperature rise in the bundle began near the center at the 0.5 to 0.7 [meter] elevation, and then spread radially outward and axially downward in a manner similar to a flame front propagation,"[420] the NRC adopted new

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

416 Edwin S. Lyman, Union of Concerned Scientists, "Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack at the Indian Point Nuclear Plant," 2004, p. 20.

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

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

419 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."

420 Ken Muramatsu, Fumiya Tanabe, Tohru Suda, "Thermal-Hydraulics in Uncovered Core of Light Water Reactor in Severe Core Damage Accident, (III): Analysis of Power Burst Facility Severe Fuel Damage 1-1 Test with SEFDAN Code," p. 960; this paper cites Proceedings from the

ECCS evaluation models, described in SECY-83-472, "to accommodate industry requests [to reduce] reactor operating restrictions."[421]

Additionally, 1983 was three years after the NRU Thermal-Hydraulic Experiment 1 tests indicated that, in the event a LOCA, a constant core reflood rate of approximately 1 in./sec. or lower would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

In 1988, the NRC continued to ignore data from severe fuel damage experiments that indicates that the PCT limit of 2200°F is non-conservative; it issued the regulations in Regulatory Guide 1.157 authorizing that, for postulated LOCAs, "[t]he rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam [could] be calculated in a best-estimate manner;"[422] *i.e.*, with the Cathcart-Pawel equation.[423] The Cathcart-Pawel equation is even more non-conservative, for calculating the metal-water reaction rates that would occur in the event of a LOCA, than the Baker-Just equation (required by Appendix K to Part 50 I(A)(5)); *e.g.*, the Cathcart-Pawel equation predicts that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures of approximately 2700°F; the Baker-Just equation predicts that it occurs at cladding temperatures of approximately 2600°F.[424]

5th International Meeting on Thermal Reactor Safety and P. E. MacDonald, *et al.*, American Nuclear Society Transcript, 46, 478, 1984, as the source of this information.

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

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

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

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

Regulatory Guide 1.157 states that "the terms 'best-estimate' and 'realistic' have the same meaning."[425] And regarding best-estimate calculations, Regulatory Guide 1.157 states:

> A best-estimate calculation uses modeling that attempts to realistically describe the physical processes occurring in a nuclear reactor. There is no unique approach to the extremely complex modeling of these processes. The NRC has developed and assessed several best-estimate advanced thermal-hydraulic transient codes. These include TRAC-PWR, TRAC-BWR, RELAP-5, COBRA, and the FRAP series of codes... These codes reasonably predict the major phenomena observed over a broad range of thermal-hydraulic and fuel tests. ...
>
> A best-estimate model should provide a realistic calculation of the important parameters associated with a particular phenomenon to the degree practical with the currently available data and knowledge of the phenomenon. *The model should be compared with applicable experimental data* and should predict the mean of the data, rather than providing a bound to the data. ...
>
> A best-estimate code contains all the models necessary to predict the important phenomena that might occur during a loss-of-coolant accident. *Best-estimate code calculations should be compared with applicable experimental data* (*e.g.*, separate-effects tests and integral simulations of loss-of-coolant accidents) to determine the overall uncertainty and biases of the calculation. In addition to providing input to the uncertainty evaluation, *integral simulation data comparisons should be used to ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted* [emphasis added].[426]

So a best-estimate ECCS evaluation calculation is supposed to "be compared with applicable experimental data"[427] and "ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted,"[428] yet the Cathcart-Pawel equation—used in best-estimate ECCS evaluation calculations—is non-conservative—as indicated by data from multi-rod (assembly) severe fuel damage experiments—for calculating the metal-water reaction rates that would occur in the event of a LOCA.

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[425] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," p. 1, footnote 1.
[426] *Id.*, p. 3.
[427] *Id.*
[428] *Id.*

It is significant that Regulatory Guide 1.157 states:

> On September 16, 1988, the NRC staff amended the requirements of § 50.46 and Appendix K, "ECCS Evaluation Models" (53 FR 35996), so that these regulations *reflect the improved understanding of ECCS performance during reactor transients that was obtained through the extensive research performed since the promulgation of the original requirements in January 1974.* Paragraph 50.46(a)(1) now permits licensees or applicants to use either Appendix K features or a realistic evaluation model. *These realistic evaluation models must include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during a postulated loss-of-coolant accident.* 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits in paragraph 50.46(b) so that there is *a high probability that the criteria will not be exceeded* [emphasis added].[429]

First, the NRC may indeed have an "improved understanding of ECCS performance during reactor transients[,] obtained through...extensive research,"[430] yet it continues to ignore data from multi-rod (assembly) severe fuel damage experiments that, among other things, indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

Regulatory Guide 1.157 also states that " 'Compendium of ECCS Research for Realistic LOCA Analysis' (NUREG-1230), provides a summary of the large experimental database available, upon which best-estimate models may be based."[431] Indeed, "Compendium of ECCS Research for Realistic LOCA Analysis" does provide "a summary of the large experimental database available;"[432] however, among other things, it omits important experimental data regarding the cladding temperatures at which autocatalytic oxidation occurred during severe fuel damage experiments, like the LOFT LP-FP-2 experiment.

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

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[429] *Id.*, p. 1.
[430] *Id.*
[431] *Id.*, p. 4.
[432] *Id.*

coolant accident[s]."[433] For example, the Cathcart-Pawel equation predicts that the autocatalytic oxidation of Zircaloy cladding occurs at cladding temperatures of approximately 2700°F,[434] yet the LOFT LP-FP-2 experiment demonstrated that autocatalytic oxidation occurs at cladding temperatures of approximately 2060°F or 2240°F.

Third, there is not "a high probability that the criteria [of 10 C.F.R. § 50.46(b) would] not be exceeded,"[435] in the event of a LOCA.

For example, in the event of a LOCA, if peak cladding temperatures increased to between approximately 2060°F[436] and 2240°F,[437] with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of approximately 18°F/sec. to 36°F/sec.[438] Within a period of less than 60 seconds peak cladding temperatures would increase to above 3000°F;[439] the melting point of Zircaloy is approximately 3308°F.[440]

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

> In the 1966-1967 time frame, research results indicated that zircaloy cladding exposed to LOCA-like conditions with peak temperatures in the

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433 *Id.*, p. 1.

434 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."

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

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

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

438 *Id.*

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

440 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.

> vicinity of 1370°C (well below the zircaloy melting point of 1820°C) embrittled and ruptured, or even shattered upon cooldown. This threatened the integrity of the core geometry, which, in turn, was perceived to threaten core coolability. Therefore, *instead of the criterion of no (or very little) clad melt, which was based in part on the concern over the autocatalytic effect on zirconium oxidation*, and which had been proposed by the nuclear steam supply system (NSSS) vendors and accepted for some months, *a much lower limit on the highest acceptable clad temperature during a LOCA was indicated, somewhere between 1204°C and 1370°C (2200°F to 2498°F).* In 1971, the AEC issued a policy statement containing interim acceptance criteria for ECCS for light water reactors [emphasis added].[441, 442]

(The AEC's interim ECCS acceptance criteria for LWRs stipulated that in the event of a LOCA, the maximum allowable cladding temperature would be 2300°F; after the rulemaking hearings that began in January 1972, the AEC changed this temperature limit to 2200°F.[443])

So the AEC based its regulation for the maximum allowable cladding temperature, in the event of a LOCA, on the premise of preventing severe cladding embrittlement and/or preventing the cladding from shattering upon cooldown. "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[o]ur selection of the 2200°F limit results primarily from our belief that retention of ductility in the zircaloy is the best guarantee of its remaining intact during the hypothetical LOCA;"[444] and that "[t]he limits specified in these criteria will assure that some ductility would remain in the zircaloy cladding as it goes through the quenching process, and therefore that the core would remain essentially intact, in a condition amenable to long-term cooling."[445]

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

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

[443] *Id.*

[444] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1098. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

[445] *Id.*, p. 1096.

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

> None of the reactor manufactures agreed with the Staff's proposed stipulation of a 2200°F maximum calculated temperature... Westinghouse proposed a maximum calculated temperature limit of at least 2700°F; Combustion Engineering and the Utility Group agreed on 2500°F as the peak allowable calculated temperature on the basis that much of the data on oxidation and its effects stops at 2500°F. Babcock and Wilcox suggested a more conservative 2400°F as the peak calculated temperature to be allowed, presumably because *"significant eutectic reaction and an excessive metal-to-water reaction rate would be precluded below 2400°F."* ... General Electric argued strongly that the limit should not be reduced to 2200°F; that 2700°F is really all right as far as embrittlement is concerned, but that the Interim Acceptance Criterion value of 2300°F should be retained. In addition to being consistent with their expressed desire not to change any of the criteria, the GE recommendation of retaining the 2300°F limit is intended *to ensure that the core never "gets into regions where the metal-water reaction becomes a serious concern"* [emphasis added].[446]

It is interesting that Babcock and Wilcox suggested a PCT limit of 2400°F, based on the premise of avoiding temperatures where the metal-water reaction would become excessive and that General Electric thought the interim PCT limit of 2300°F should be retained "to ensure that the core never 'gets into regions where the metal-water reaction becomes a serious concern.' "[447]

It is also noteworthy that during the AEC's ECCS rulemaking hearings that Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors,[448] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[449] conducted with a Zircaloy assembly. Among

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[446] *Id.*, p. 1097.

[447] *Id.*

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

[449] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-17; this paper cites Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [metal-water] reactions, in spite of the 'failed' heater rods. They compared test results for SS2N [(conducted with a stainless steel assembly)] with Zr2K, showing satisfactory correlation during approximately the first five minutes of the test with substantial deviations (Zr2K temperatures greater than SS2N) during the subsequent periods of substantial heater failures."[450]

The Zr2K test had cladding-temperature increases of several hundred degrees Fahrenheit within approximately 20 seconds, at some locations of its assembly, after cladding temperatures reached between approximately 2100 and 2200°F (see Appendix P Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies and Figure A8.10 Analysis of Zr2K Thermal Response).

Additionally, it is noteworthy that in 2002, the NRC postulated that "with regard to runaway temperature escalation, the [10 C.F.R. § 50.46(b)(1) PCT limit] could be raised to 2300°F;"[451] regarding this issue the NRC stated:

> We now know with a high degree of confidence that the Baker-Just equation is substantially conservative at 2200°F, and recent data exhibit very little scatter. A good representation of Zircaloy oxidation at this temperature is given by the Cathcart-Pawel correlation. If one examines the heat generation rate predicted with these two correlations, it is found that one needs a significantly higher temperature to get a given heat generation rate with the Cathcart-Pawel correlation than with the Baker-Just correlation. In particular, Cathcart-Pawel would give the same metal-water heat generation rate at 2307°F as Baker-Just would give at 2200°F... Thus, *with regard to runaway temperature escalation, the peak cladding temperature could be raised to 2300°F* without affecting this sensitivity and without reducing the margin that the Commission would have perceived in 1973 [emphasis added].[452]

So the NRC has continued to ignore data from multi-rod (assembly) severe fuel damage experiments that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F

[450] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-18.
[451] "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 3; Attachment 2 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720709; the letter's Accession Number: ML021720690.
[452] *Id.*

is non-conservative. In other words, the NRC has ignored experimental data that indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit should be based on the premise of preventing the autocatalytic oxidation of Zircaloy cladding, at a limit below 2200°F.

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

> It was also found that some plants were being restricted in operating flexibility by limits resulting from conservative Appendix K requirements. Based on the research performed, it was determined that these restrictions could be relaxed through the use of more realistic calculations *without adversely affecting safety*. ...
>
> *Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations* [emphasis added].[453]

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

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

> No data or evidence was...found in NRC records to suggest that the research, calculation methods, or data used to support ECCS performance evaluations were sufficiently flawed so as to create significant safety problems. NRC's technical safety analysis demonstrates that current procedures for evaluating performance of ECCS are based on sound science and that no amendments to the NRC's regulations and guidance documents are necessary. ...the NRC [has not] found, the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material.[454]

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

[454] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 23.

So the NRC was unable to locate data in NRC records from multi-rod (assembly) severe fuel damage experiments that indicates that the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA. And the NRC was unable to perceive "the existence of any safety issues regarding calculation methods or data used to support ECCS performance evaluations that would compromise the secure use of licensed radioactive material."[455] For example, the NRC was unable to locate data in NRC records from the LOFT LP-FP-2 experiment that indicates that an autocatalytic oxidation reaction of Zircaloy cladding occurred at a temperature hundreds of degrees Fahrenheit below what either the Baker-Just or Cathcart-Pawel equations would predict.

Regarding the NRC's current proposed revisions to 10 C.F.R. § 50.46(a), in 2007, both the ACRS and NRC staff agreed "that it is preferable to complete the review and revision of the fuel cladding acceptance criteria for LOCAs involving breaks at or below the [transition break size ("TBS")] before finalizing the § 50.46a rulemaking."[456] And, in SECY-07-0082, the NRC staff states that "[t]his is a logical sequence because changes proposed by licensees adopting § 50.46a will likely result in more demanding reactor operating conditions that may further stress the fuel, or result in small break LOCAs becoming limiting."[457]

Therefore, it would also be logical to review and correct the deficiencies in the NRC's and nuclear industry's current ECCS evaluation models, before finalizing the 10 C.F.R. § 50.46(a) rulemaking. For example, 10 C.F.R. § 50.46(b)(1) should be revised so that it is based on data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment); the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

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[455] *Id.*

[456] NRC, SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements; 10 CFR 50.46(a), Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," May 16, 2007, Enclosure 1, "Rule Overview and Summary of ACRS Recommendations," p. 5.

[457] *Id.*

It is also pertinent that in "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" the NRC states:

> As previously discussed in...this document, the NRC is working to revise the ECCS acceptance criteria in § 50.46(b) to account for new experimental data on cladding ductility and to allow for the use of advanced cladding alloys. ... The NRC expects that this rulemaking...will establish new cladding embrittlement acceptance criteria in § 50.46(b) for design basis LOCAs. As these new acceptance criteria are established, the NRC will also make conforming changes to § 50.46a as necessary for both below and above TBS breaks.[458]

In this case, it would still be logical to review and correct the deficiencies in the NRC's and nuclear industry's current ECCS evaluation models and "make conforming changes to § 50.46a as necessary for both below and above TBS breaks."[459]

Clearly, the deficiencies of the NRC's and nuclear industry's ECCS evaluation models discussed above indicate that the probabilities assigned to core damage frequency ("CDF") and "the frequency of...accidents leading to significant, unmitigated releases from [the] containment[460] in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects"[461] ("LERF") are erroneous.

It is significant that Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," states that "if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it;"[462] and states that "if there is an indication that the LERF

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458 NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," Federal Register, August 10, 2009, p. 40030.

459 *Id.*

460 NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, p. 8, footnote 3, states that "[s]uch accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines."

461 NRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002, p. 8, footnote 3.

462 *Id.*, p. 17.

may be considerably higher than 10^{-5} per reactor year, the focus should be on finding ways to decrease rather than increase it."[463]

It is highly probable that at nuclear power plants CDF and LERF are currently considerably higher than 10^{-4} per reactor year and 10^{-5} per reactor year, respectively, because ECCS evaluation models are deficient. Therefore, it is imperative that the NRC decrease the probabilities of CDF and LERF, rather than increase them.

So the NRC must not revise its regulations to allow for "design changes, such as increasing power [that] could cause increases in plant risk."[464] It is also imperative that the NRC not revise its regulations to "divide the current spectrum of LOCA break sizes into two regions"[465] and make "each break size region...subject to different ECCS requirements"[466] where "the smaller break size region [would] be analyzed by the methods, assumptions, and criteria currently used for LOCA analysis [and] accidents in the larger break size region [would] be analyzed by less conservative assumptions based on their lower likelihood."[467] Beyond-TBS acceptance criteria should be the same as the acceptance criteria for TBS and smaller breaks; *i.e.*, the criteria of 10 C.F.R. § 50.46(b). The criteria of maintenance of coolable core geometry and maintenance of long-term core cooling should not be used as a substitute for the criteria of 10 C.F.R. § 50.46(b) for beyond-TBS LOCAs.

Furthermore, "LOCAs for break sizes larger than the transition break [must not] become 'beyond design-basis accidents,' "[468] even if "the proposed rule would require licensees to maintain the ability to mitigate all LOCAs up to and including the DEGB of the largest RCS pipe during all operating configurations."[469]

If implemented, the suggestions proposed in Commentator's responses to the three specific topics identified for public comment in "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" would help improve public and plant-worker safety.

---

[463] *Id.*
[464] NRC, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," p. 40008.
[465] *Id.*
[466] *Id.*
[467] *Id.*
[468] *Id.*
[469] *Id.*

Respectfully submitted,

Mark E Leyse

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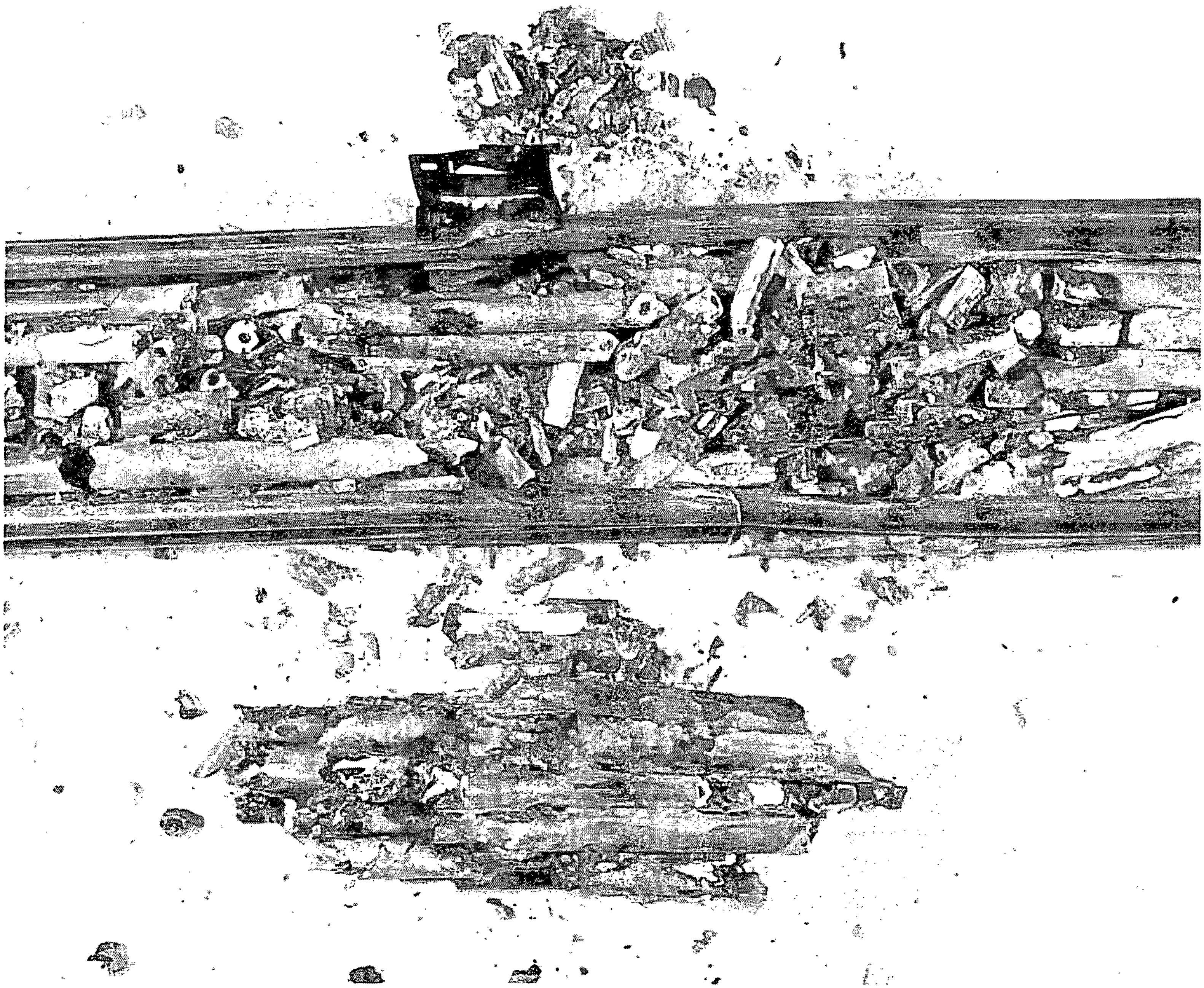
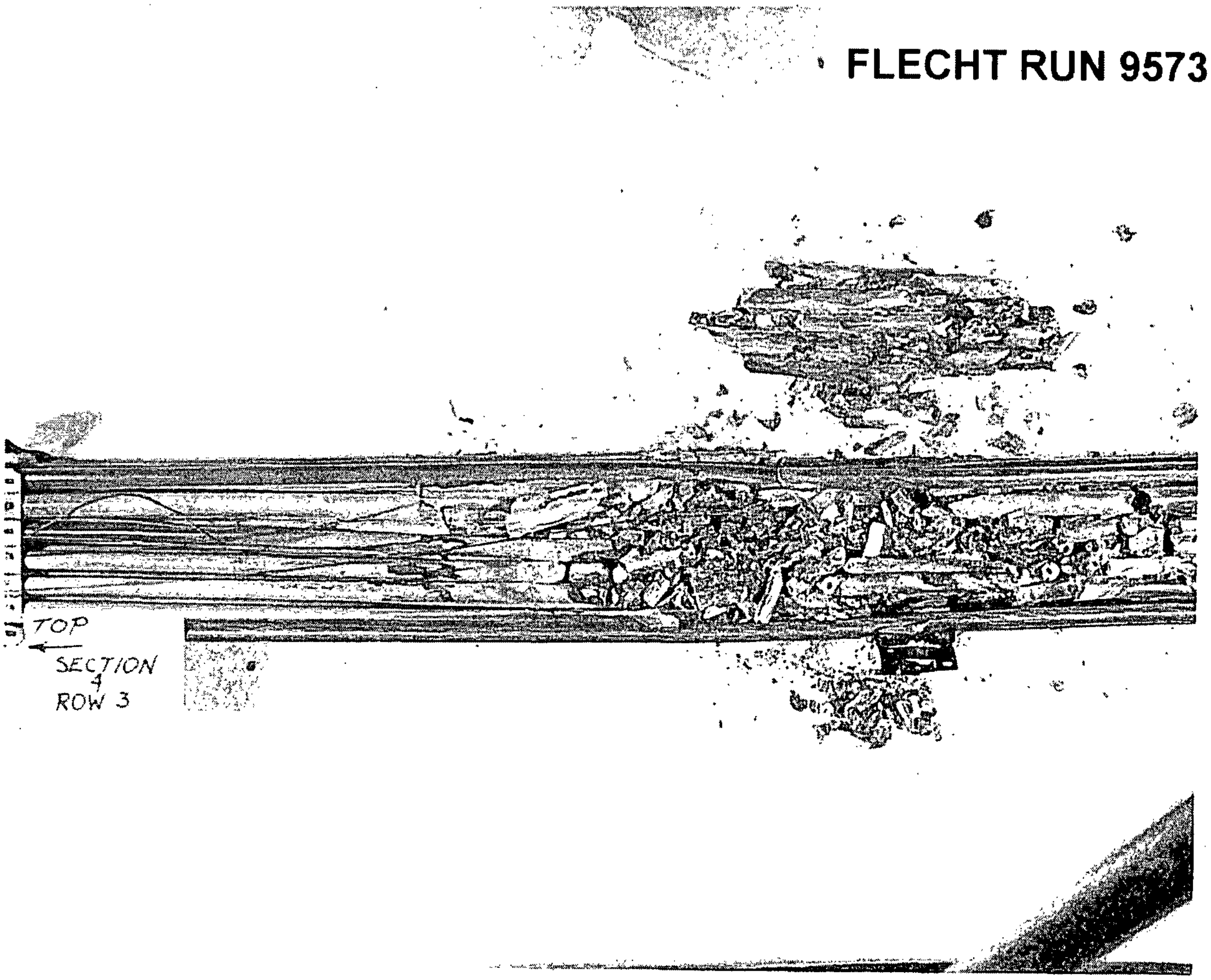
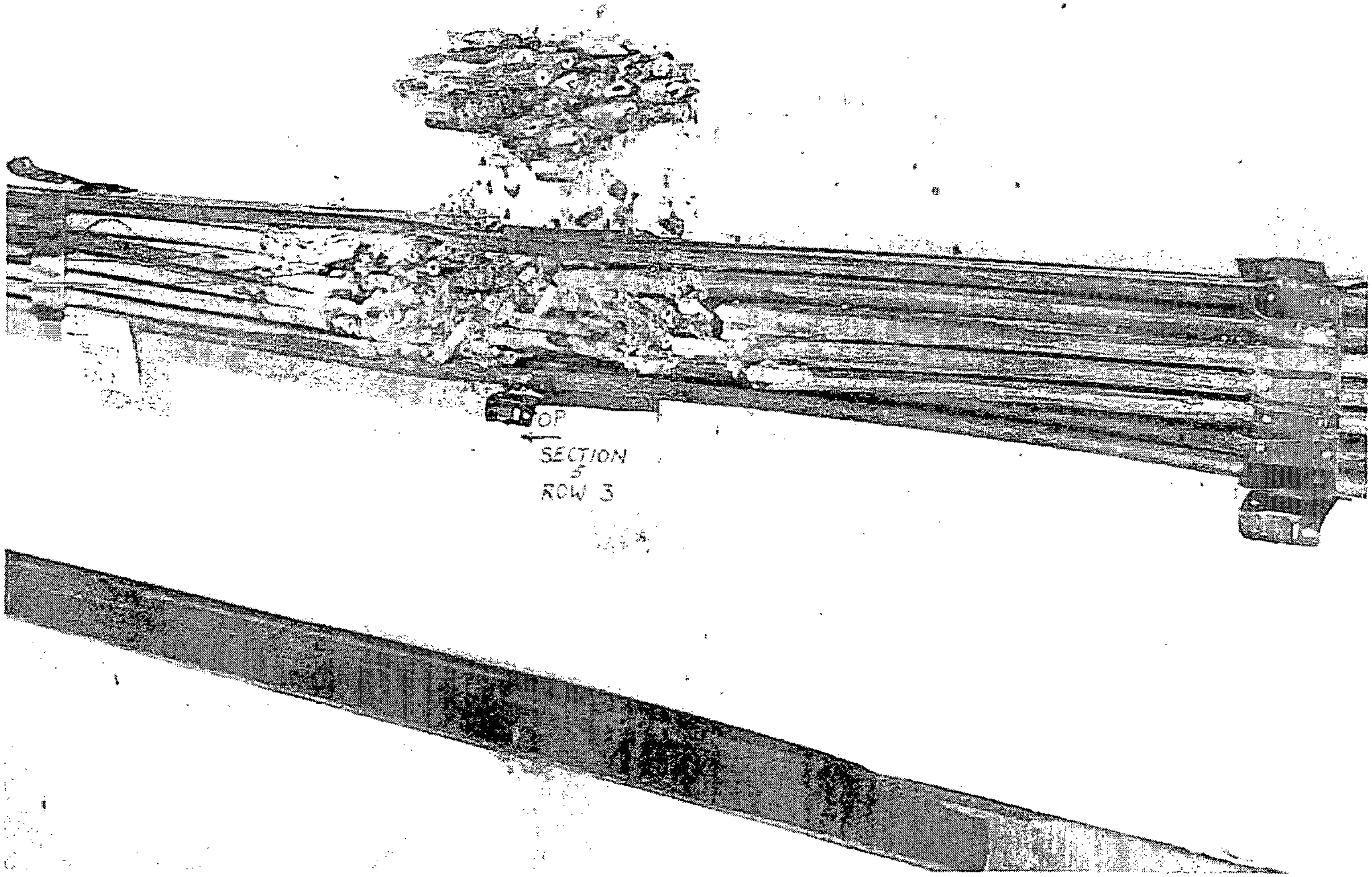
Dated: January 20, 2010

Appendix A Photographs of the Assembly from FLECHT Run 9573

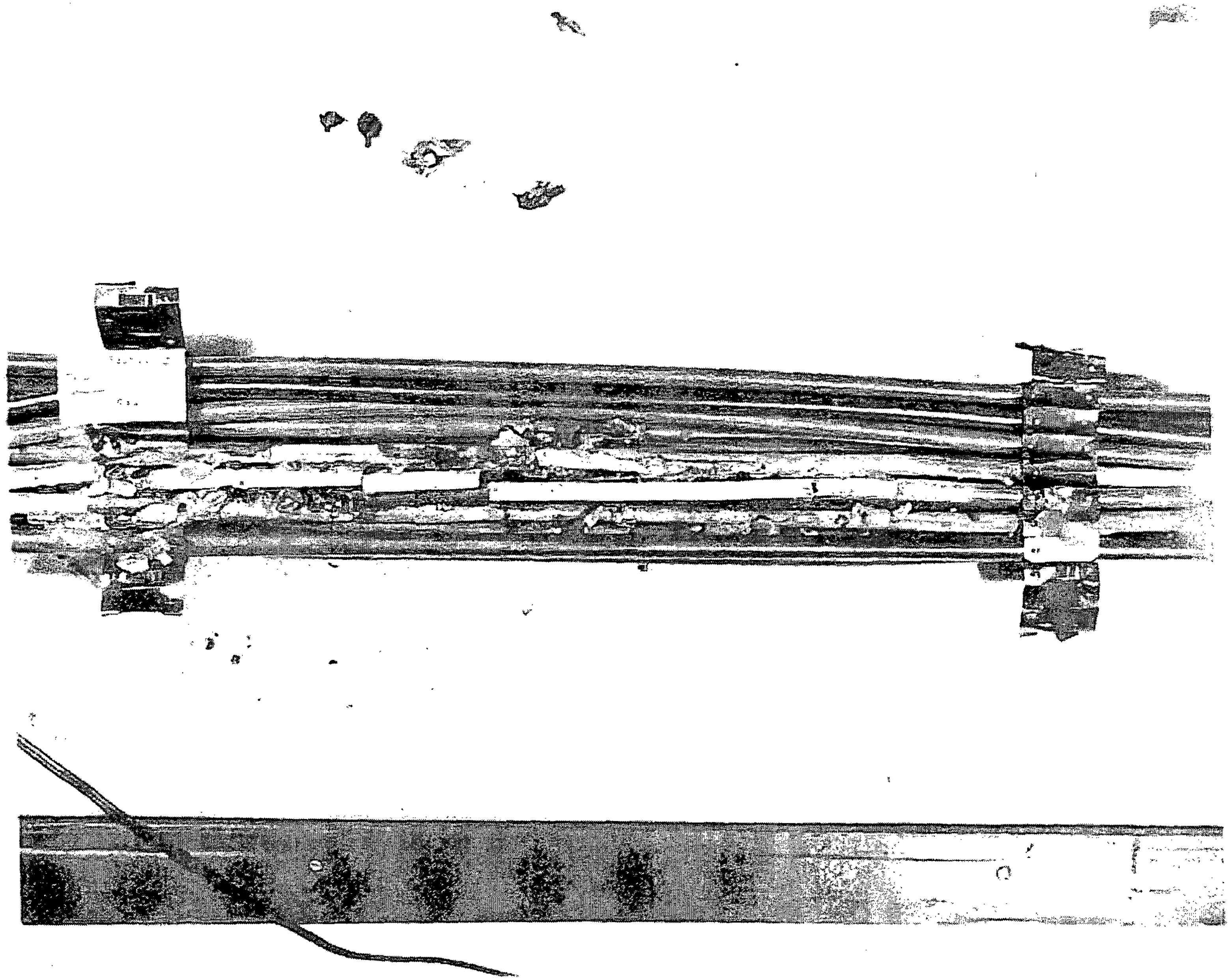


FLECHT RUN 9573




OP
SECTION
5
ROW 3

Appendix B Photograph of the Assembly from FLECHT Run 8874



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

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

B.2 SPECIMEN SELECTION, PREPARATION, & EXAMINATION

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

TABLE B-1

GROUP III TEST RESULTS

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

[a] with fallback

[b] at 18 seconds

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

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

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

TABLE 1. Experimental Heat Cladding Temperatures

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

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

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

[3] Pseudo sensor readings are the averages of the readings of two or more thermocouples.
[4] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, pp. 4.7, 5.3.

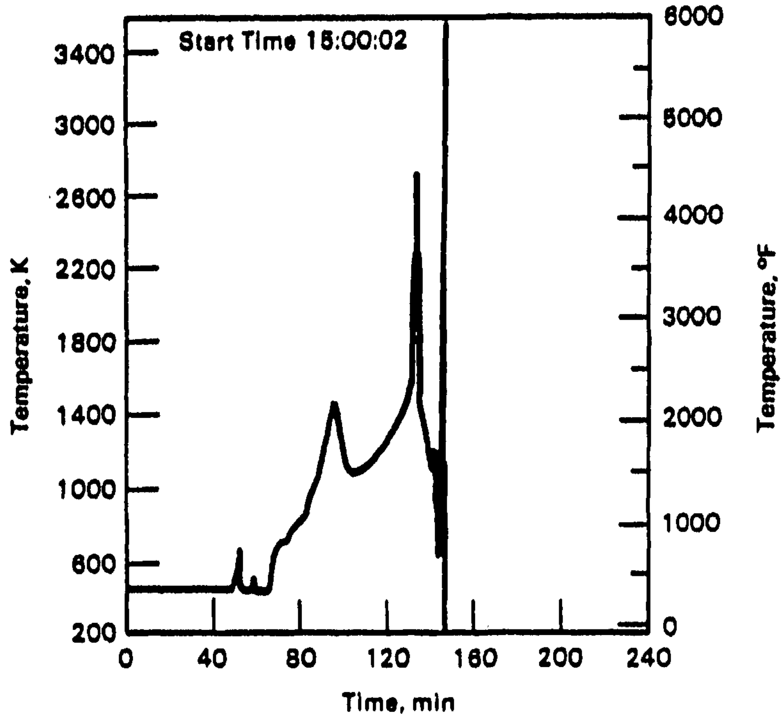


FIGURE 4.1. Typical Cladding Temperature Behavior

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

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

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

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

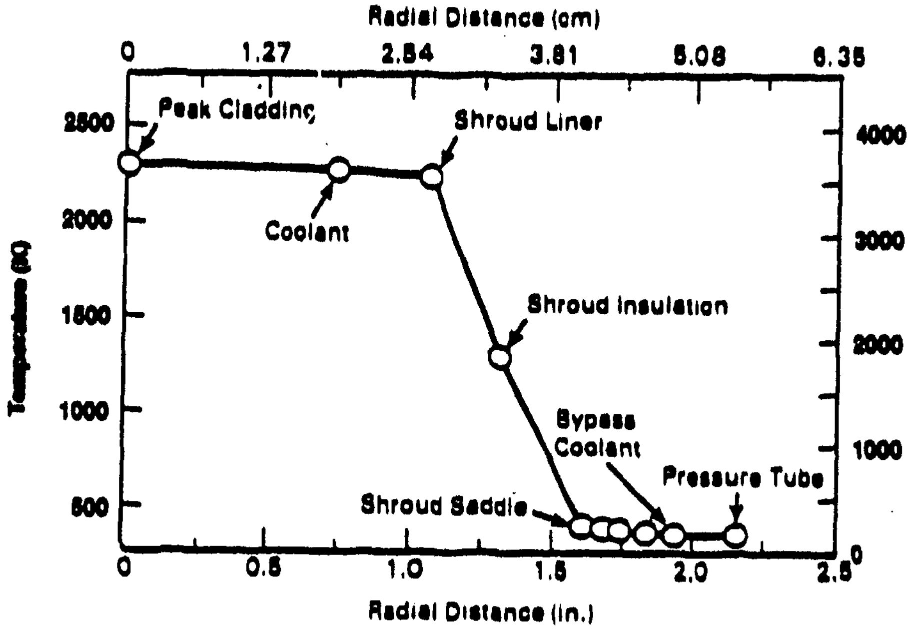


FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

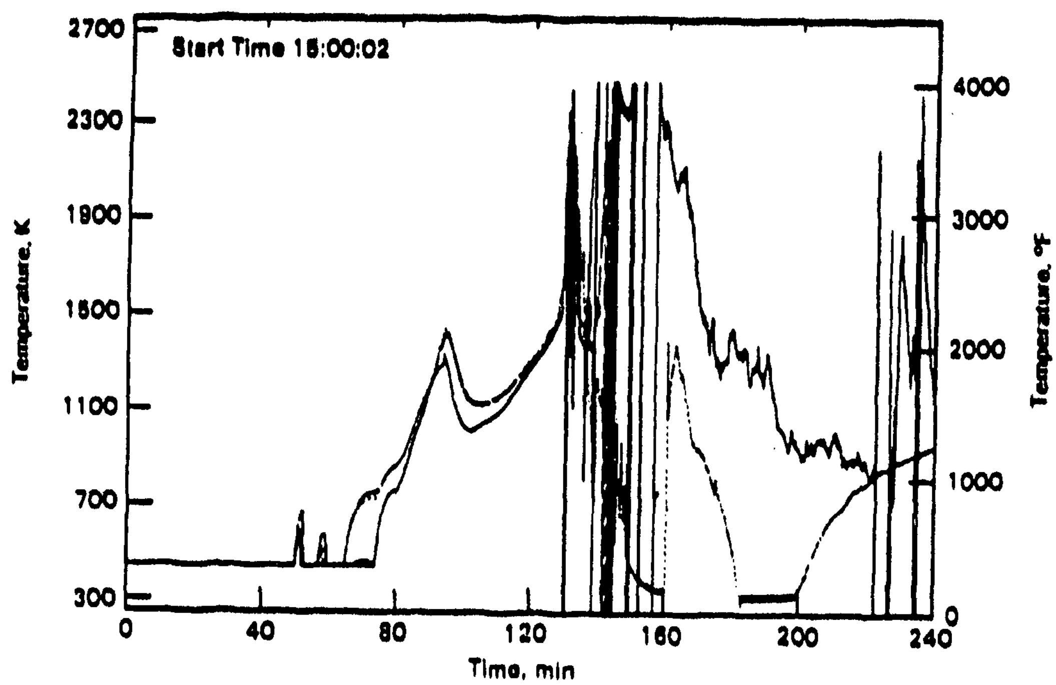


FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

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

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

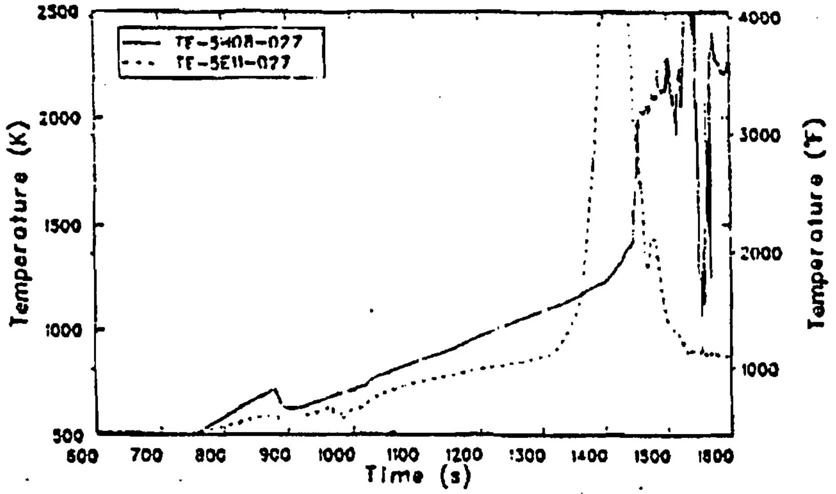


Figure 3.7 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5.

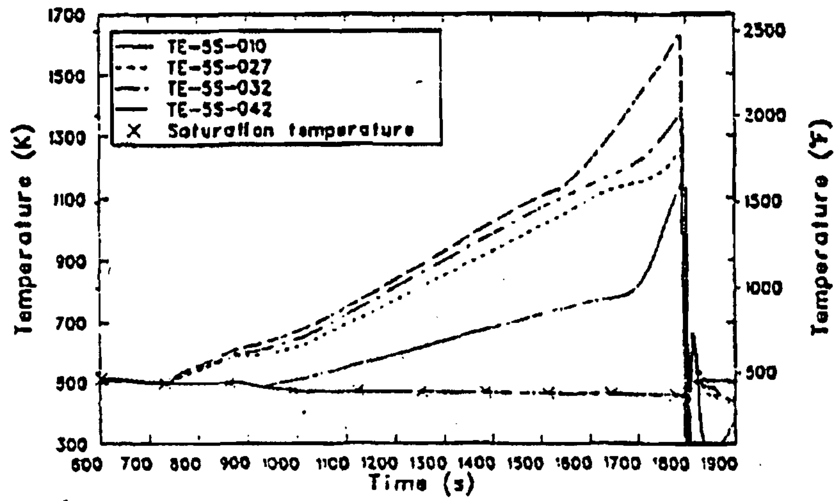


Figure 3.8 Comparison of four external wall temperatures at the 1.07-, 0.81-, 0.69-, and 0.25-m (42-, 32-, 27-, and 10-in.) elevations on the south side of the flow shroud.

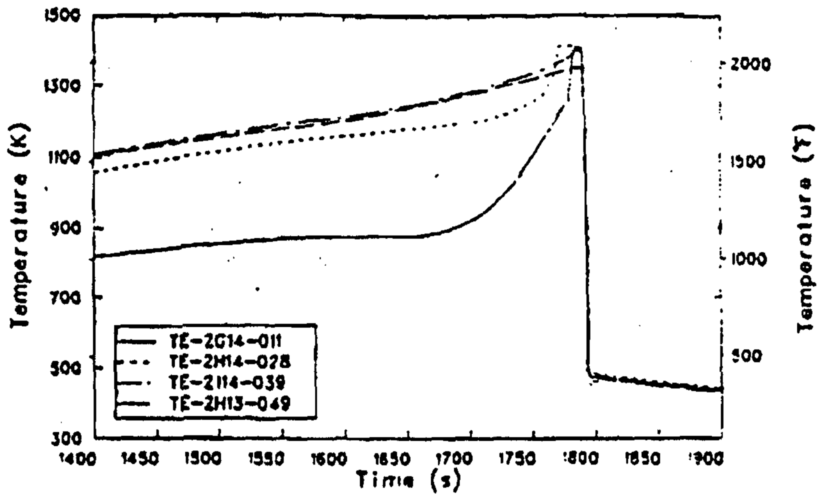


Figure 3.9 Comparison of cladding temperatures at the 1.24-, 0.99-, 0.71-, and 0.28-m (49-, 39-, 28-, and 11-in.) elevations in Fuel Assembly 2.

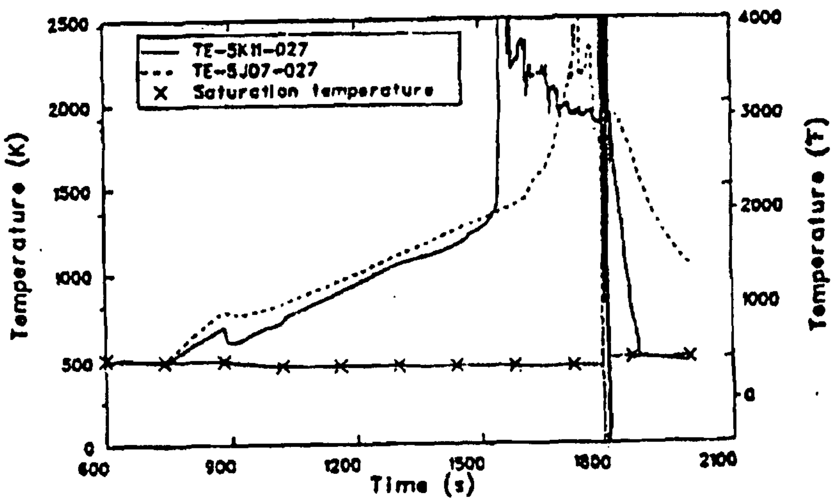


Figure 3.10 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5 with saturation temperature.

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

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

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

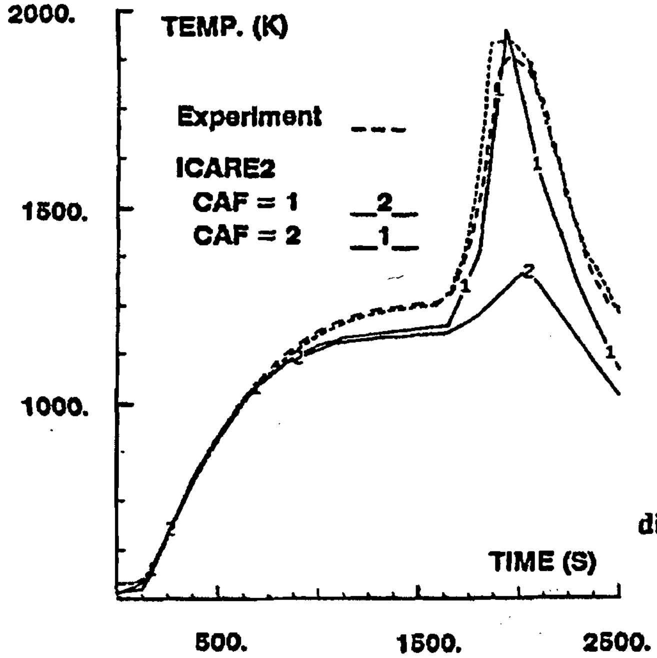


Fig.1 :
Sensitivity calculation on the B9R test. Temperature escalation at the hot level (0.6 m) with different Contact Area Factors (CAF)

3.2.2 PHEBUS C3 + test

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

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

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

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

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

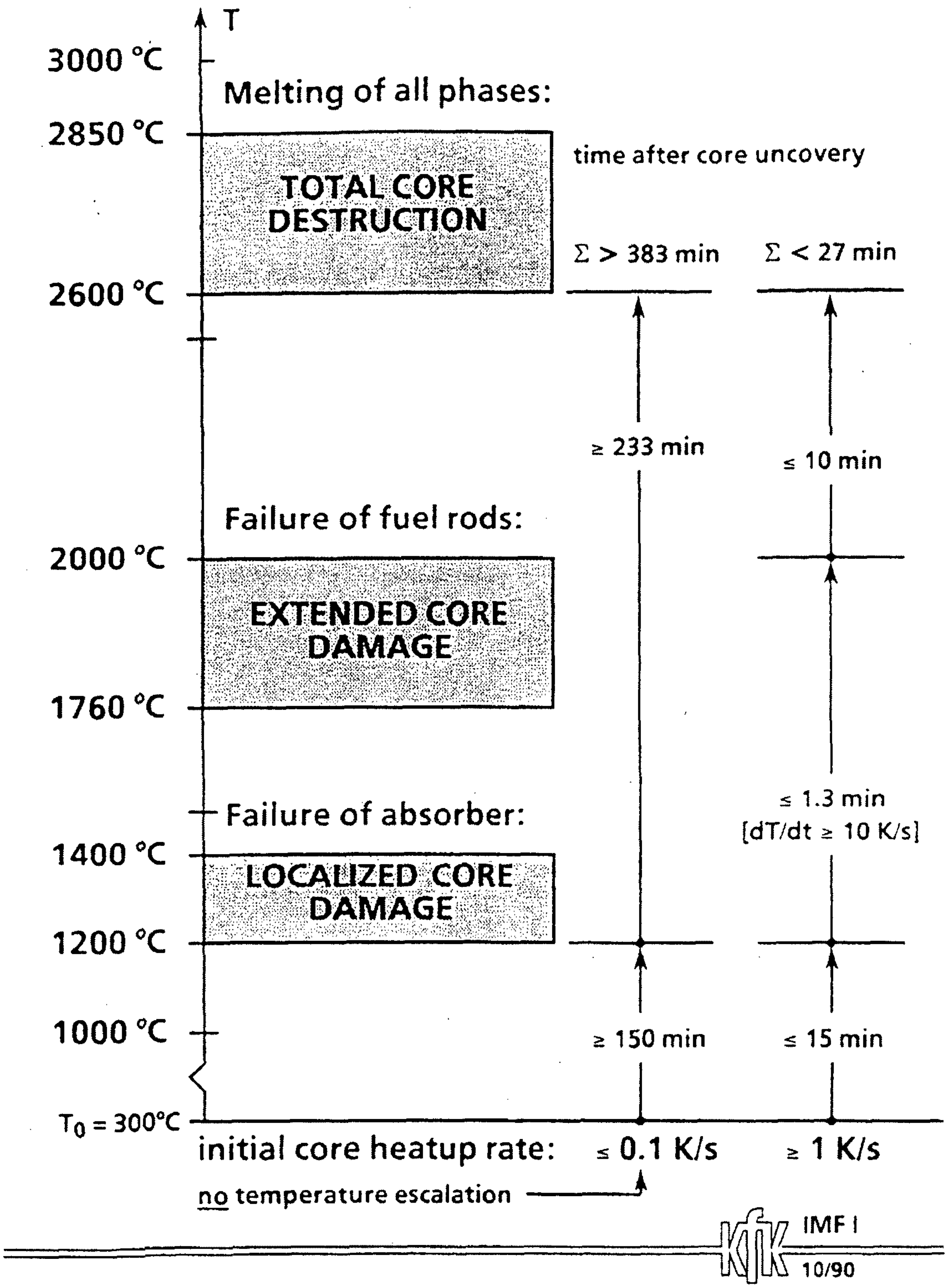


FIGURE 2.1: TEMPERATURE REGIMES FOR EXTENSIVE LIQUID PHASE FORMATION AND RELOCATION

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

RD-TE-70-616

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

PENN CENTER

H. A. Sindt
Development Projects

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

FLEC-500 - Facility Operation

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

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

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

NUCLEAR ENERGY SYSTEMS Test Engineering

R. H. Leyse

/bjl

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

RD-THD-17

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

NUCLEAR ENERGY SYSTEMS

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

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

I. Results of Group III Zirc Run 9573

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

II. Final Report Status and Plans

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

III. Facility Inventory and Disassembly Plans

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

F F Cadek
F. F. Cadek, Manager
Thermal-Hydraulic Development

/jw

Attachment

FORM

SUMMARY OF FINAL GROUP III ZIRC TEST

FLECHT RUN 9573

(PRELIMINARY RESULTS)

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

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

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

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

4. Typical Rod Temperatures

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

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

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

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

PA-TE-70-419

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

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

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

R. Leyse

NUCLEAR ENERGY SYSTEMS | Test Engineering | R. H. Leyse
/bjl

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

RD-ED-THE-33

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

FLEC-200

PWR SYSTEMS DIVISION

L. S. Tong, Manager
Engineering Development

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

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

I. Schedule for Testing

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

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

II. Operational Difficulties

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

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

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

3511 B

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

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

III. Post Test Examination of Results

A. Open circuits were found in 60 heater rods.

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

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

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

R. F. Farman
Thermal and Hydraulic Experimentation

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

/1

Appendix M Figure 15. Temperatures of Unheated Rods and Power History of CORA-5, Figure 16. Temperatures of Unheated Rods during CORA-12, Figure 17. Temperatures at Different Elevations during CORA-15, Figure 18. Temperatures of Unheated Rods during CORA-9, Figure 19. CORA-7; Temperatures at Elevations Given (750 mm), and Figure 20. Temperatures of Guide Tube and Absorber Rod during Test CORA-5[1]

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

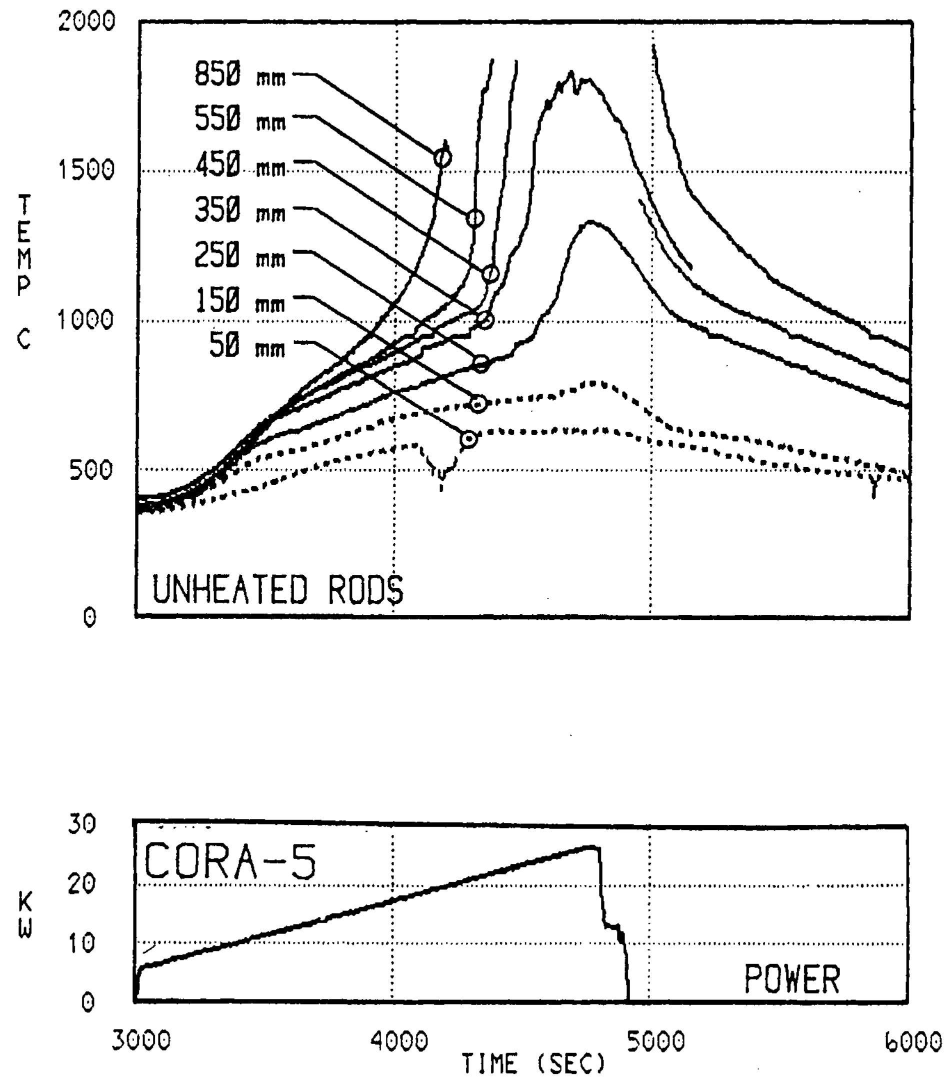


Fig.15: Temperatures of unheated rods and power history of CORA-5

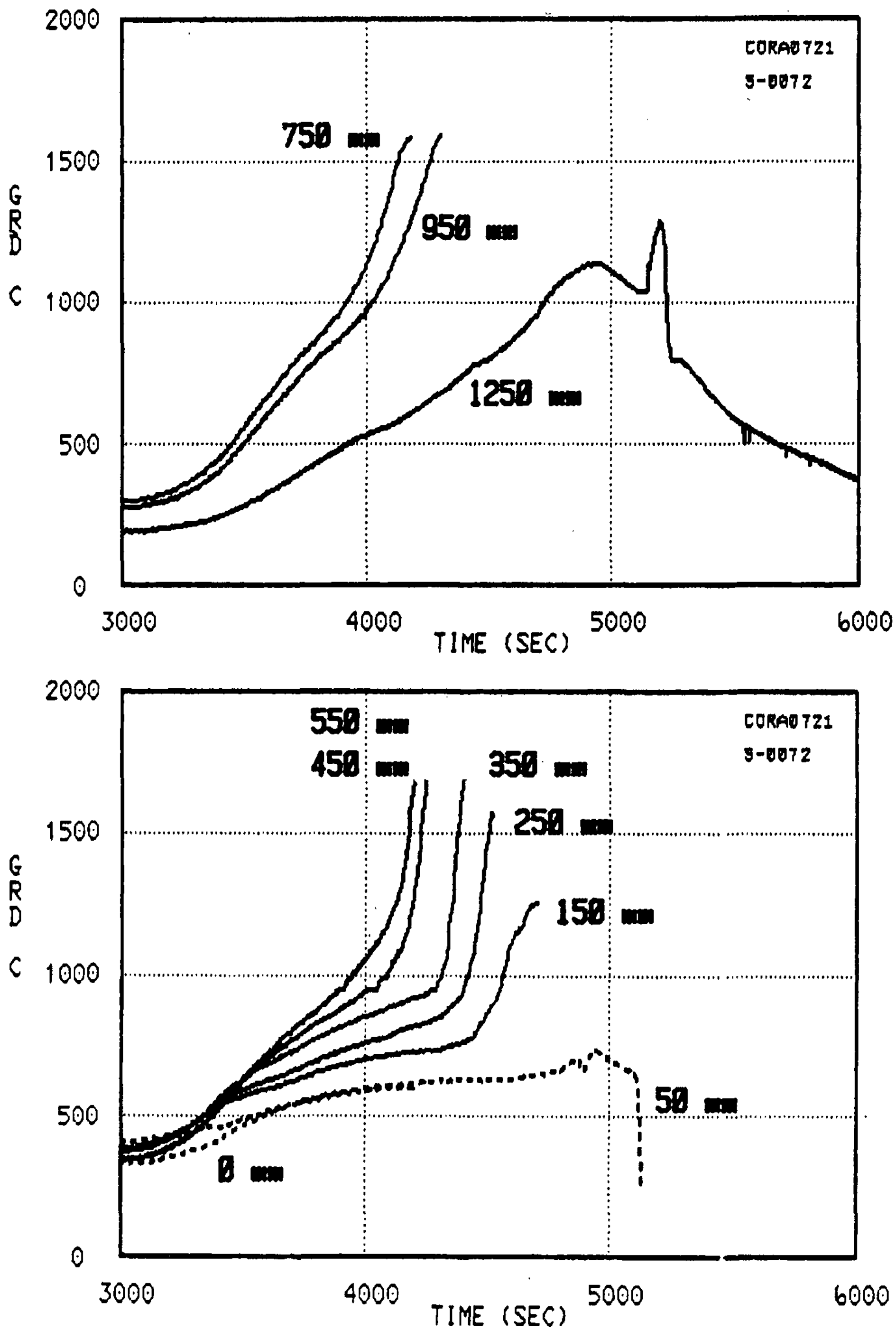


Fig.16: Temperatures of unheated rods during CORA-12

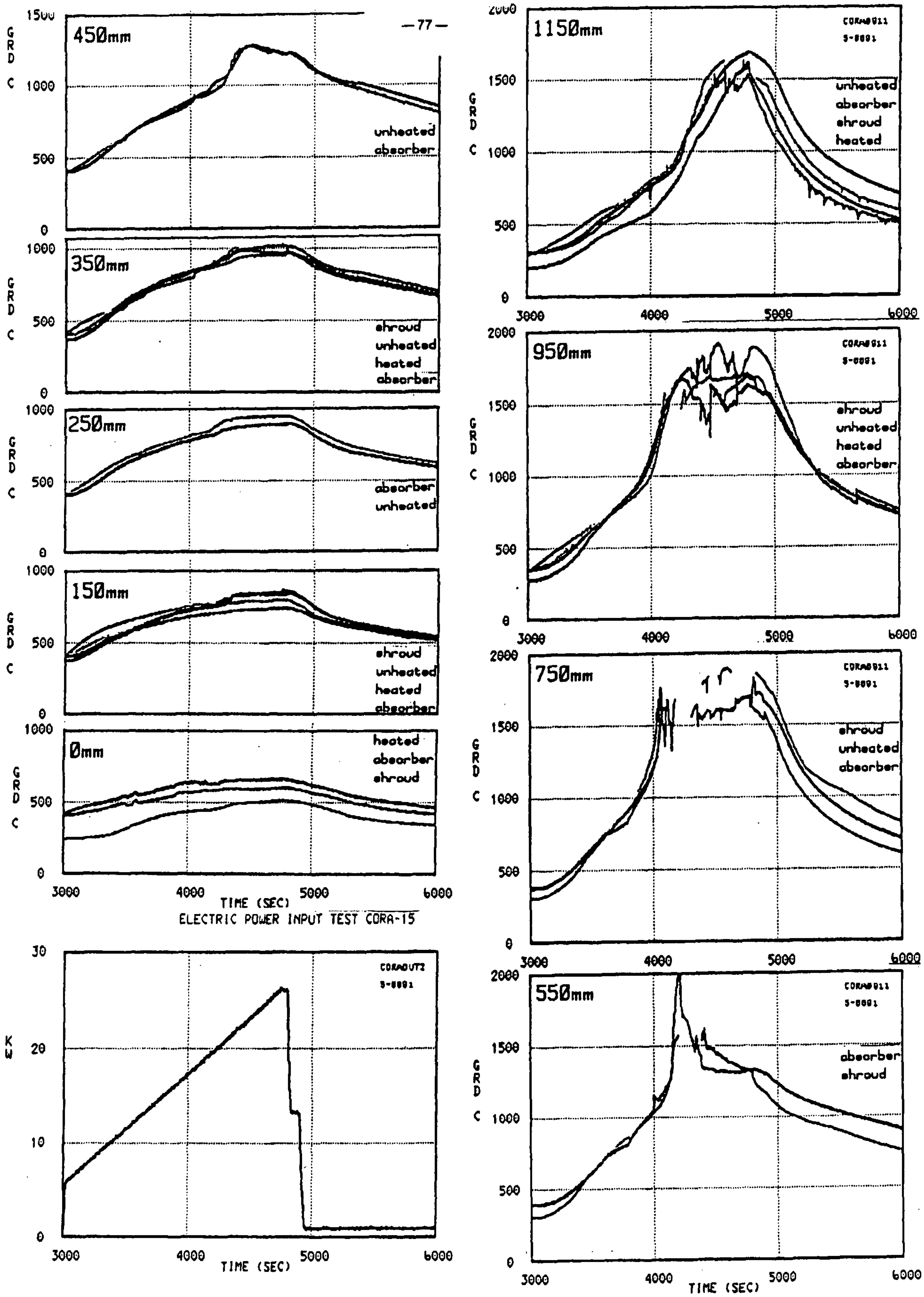


Fig.17: Temperatures at different elevations during CORA-15

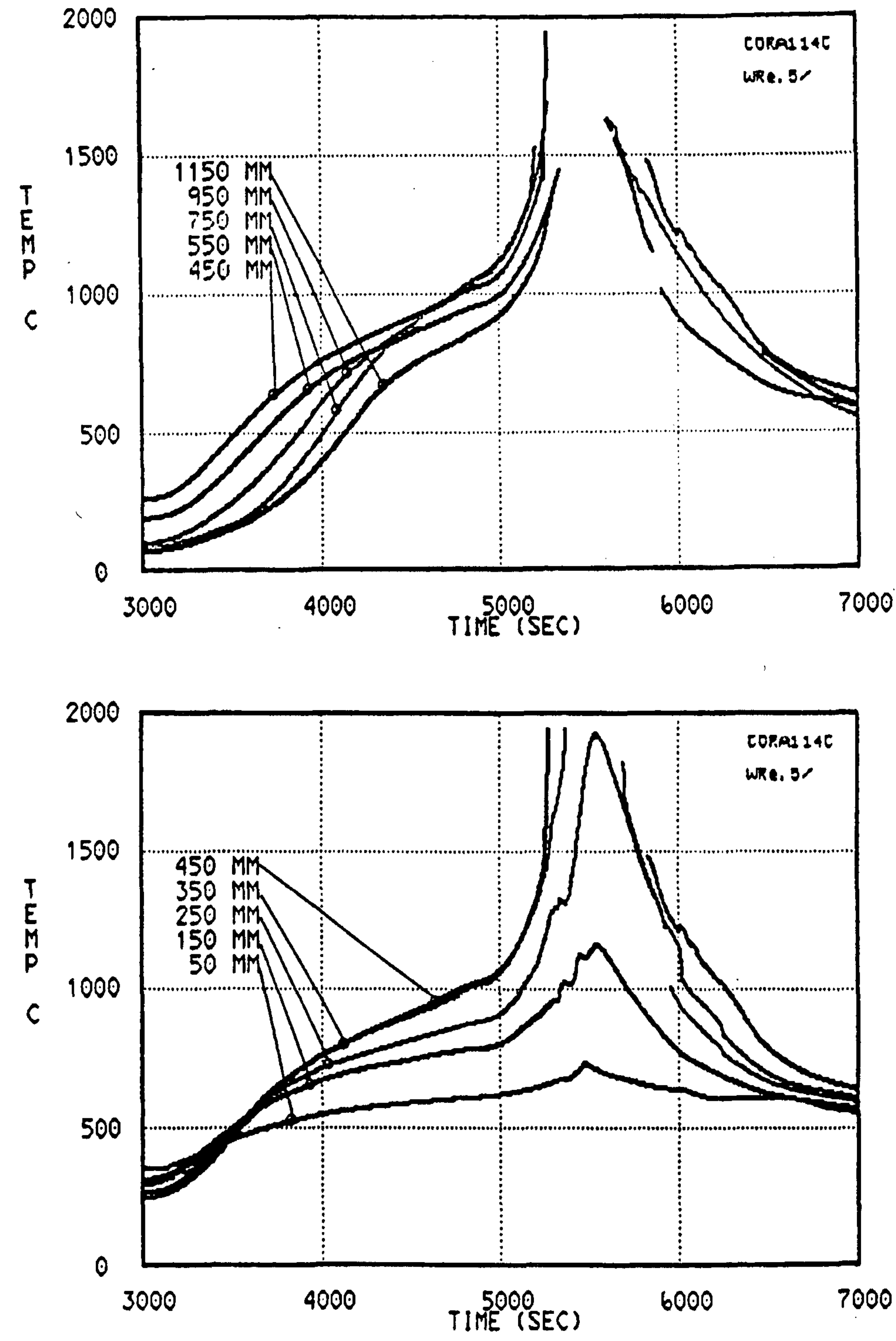
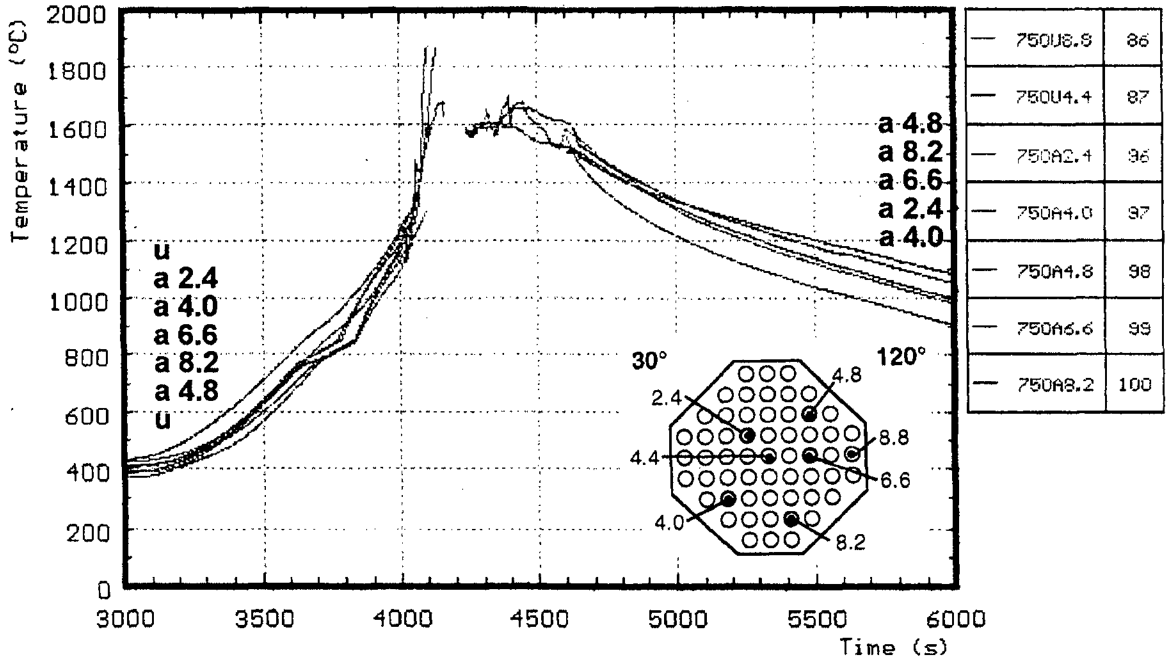
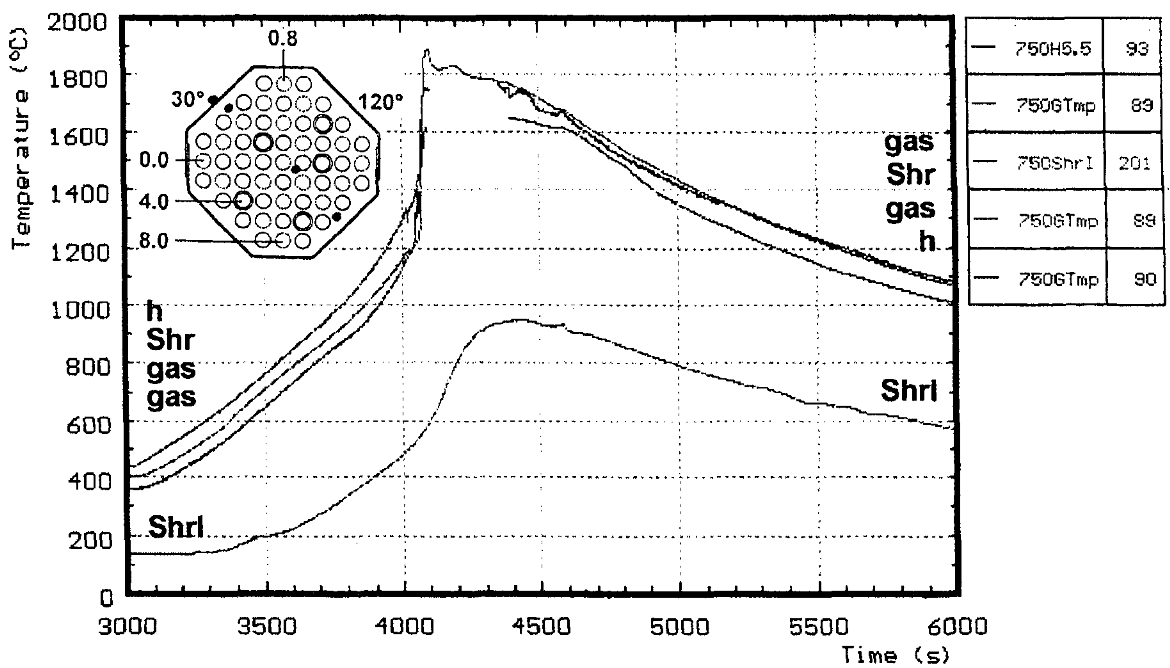


Fig.18: Temperatures of unheated rods during CORA-9





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

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

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

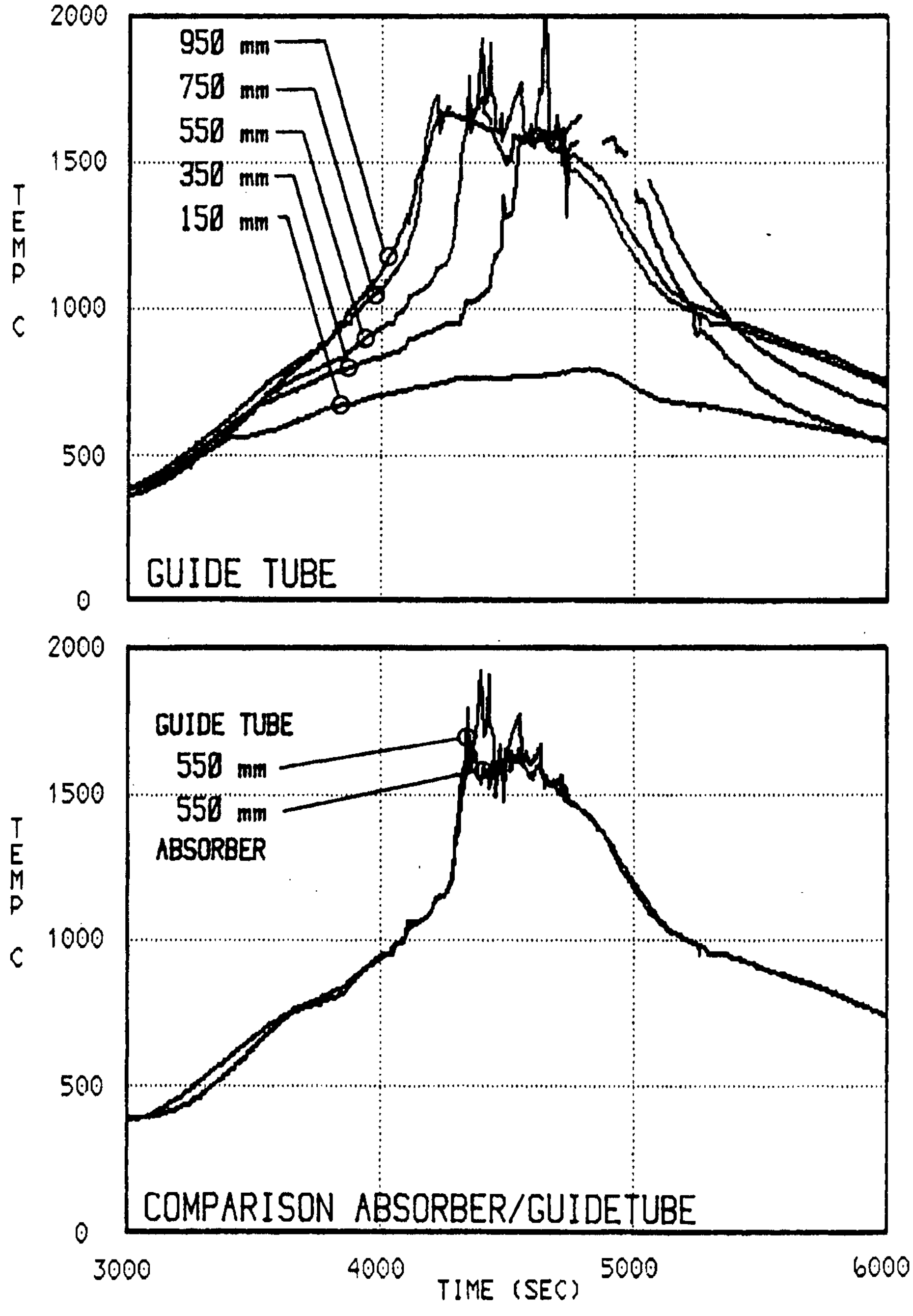


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

Appendix N Figure 37. Temperatures of the Heated Rods (CORA-13) and Figure 39. Temperatures of the Unheated Rods (CORA-13)[2]

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

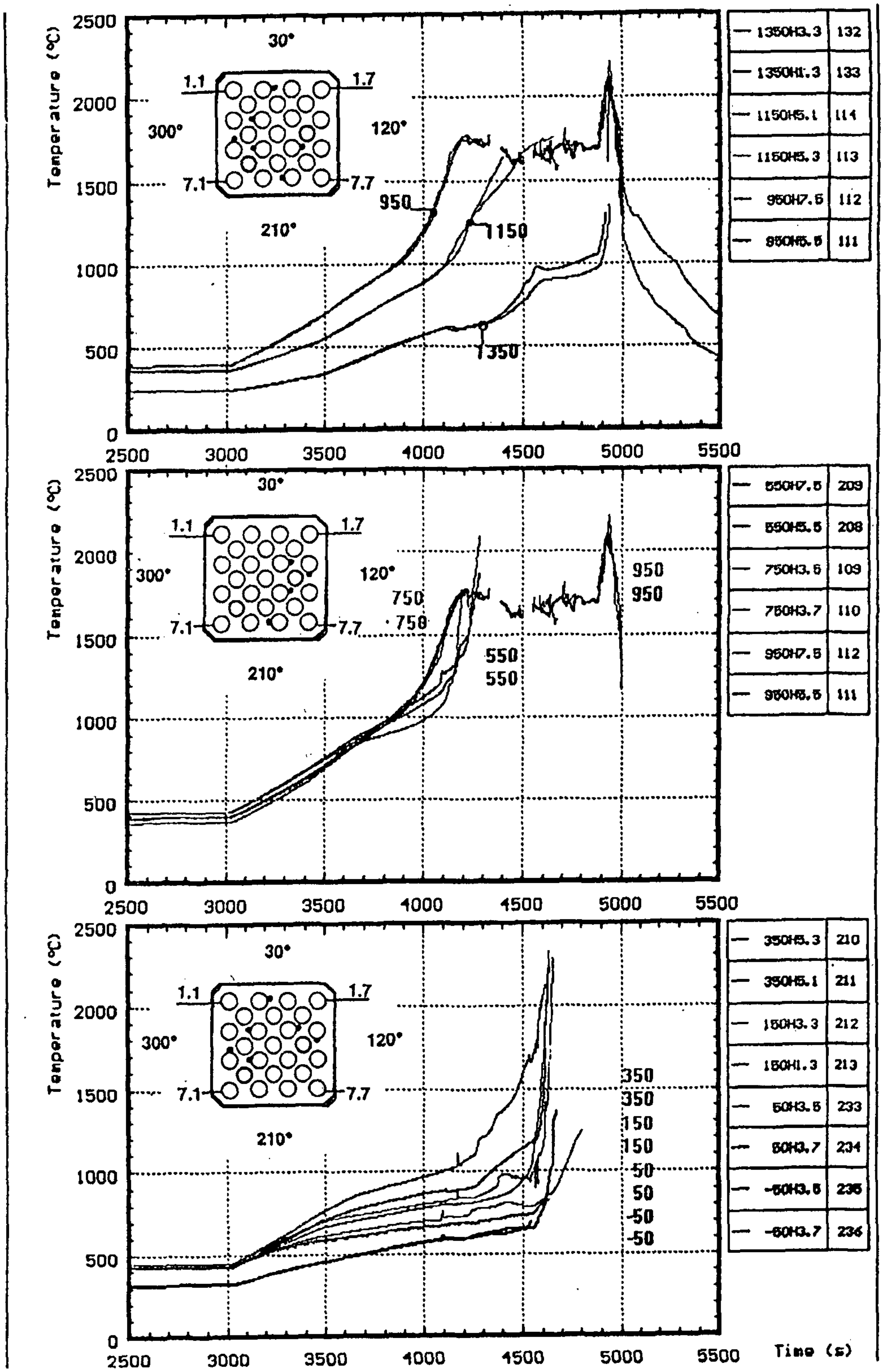


Fig. 37: Temperatures of the heated rods (CORA-13)

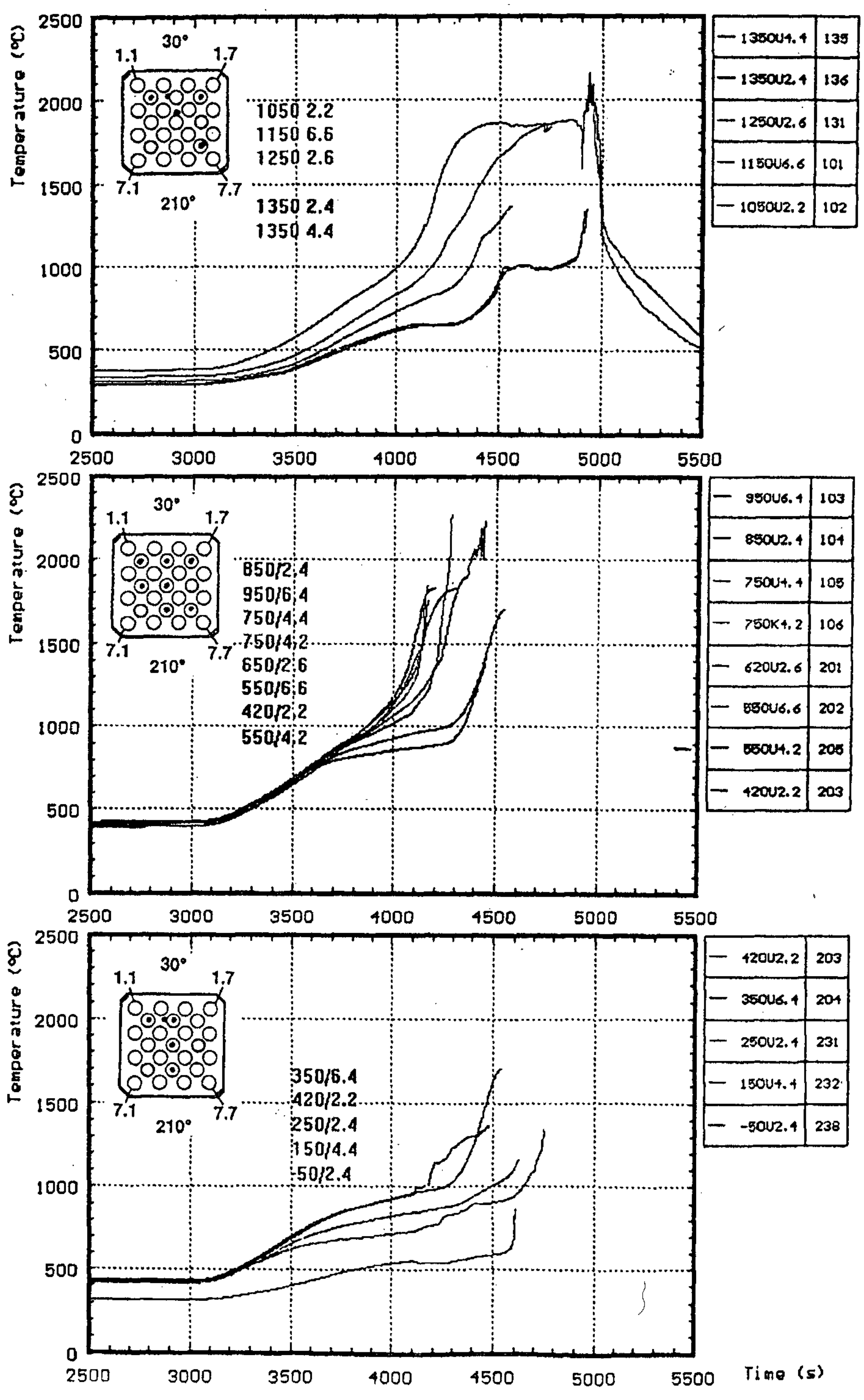


Fig. 39: Temperatures of the unheated rods (CORA-13)

Appendix O Table 10. Zircaloy Oxidation, Energy Release, and Hydrogen Production during Various CORA Tests[3]

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

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

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

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

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

Appendix P Figure A8.9. Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[4] and Figure A8.10. Analysis of Zr2K Thermal Response[5]

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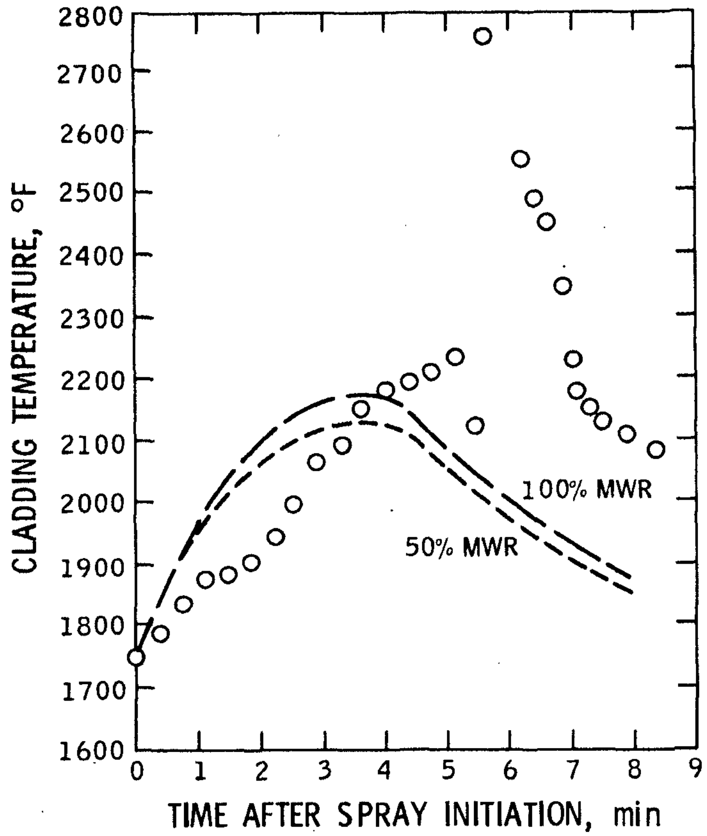
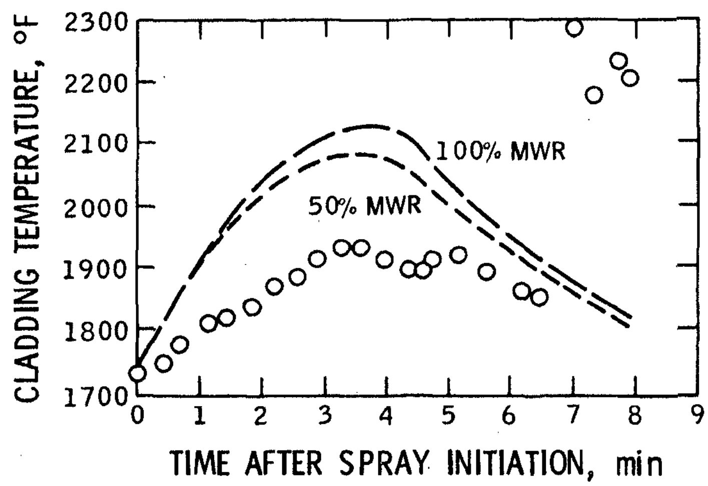
[4] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-25; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13197, June 1971, Figures A-11 and A-12, as the source of this information.

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



Bundle Zr2K Rod 24 Midplane Thermal Response Prediction

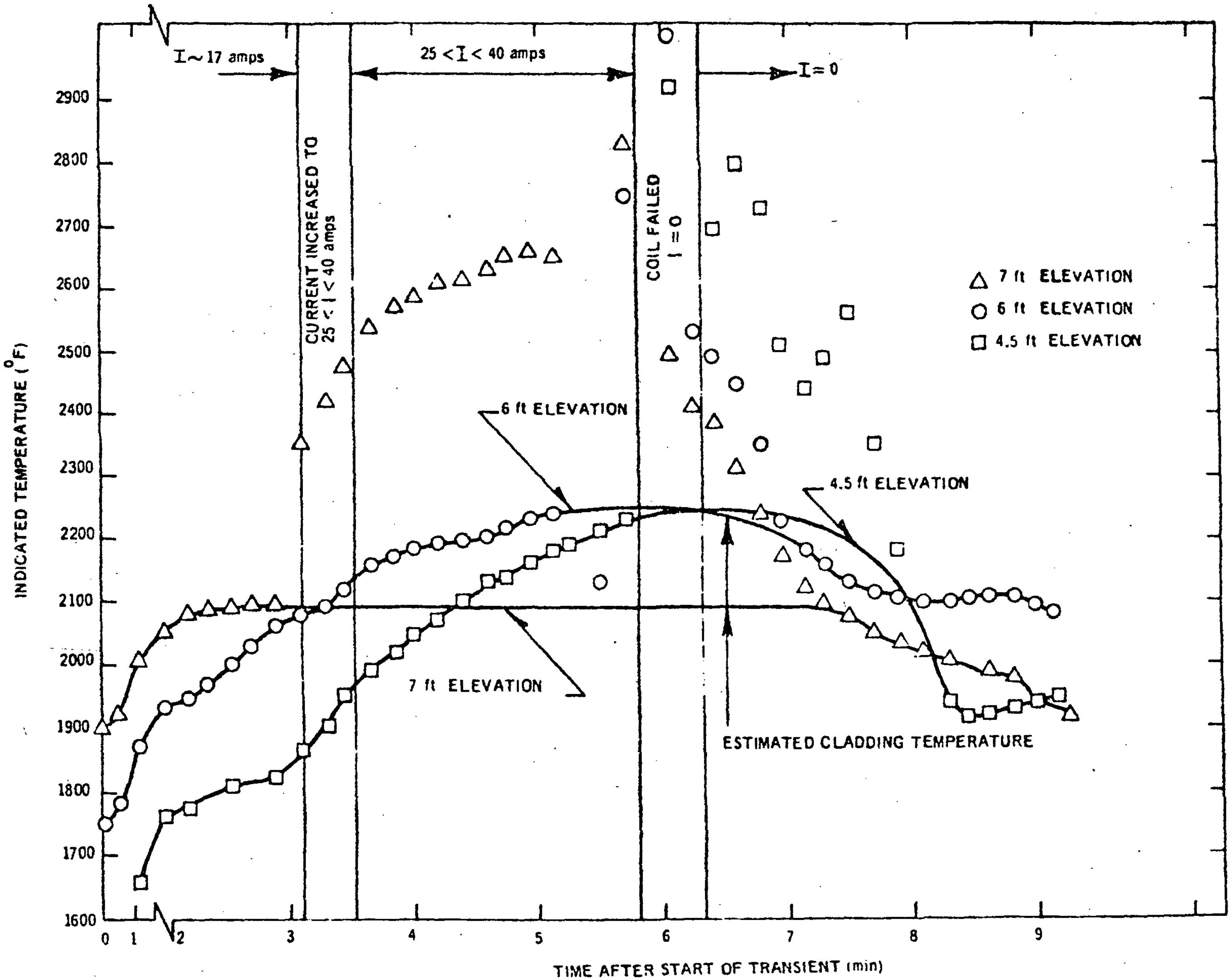


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

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

Figure A8.10

Analysis of Zr2K Thermal Response



(After Figure 12, 54, by permission.)

Rulemaking Comments

From: mel2005@columbia.edu
Sent: Thursday, January 21, 2010 10:23 AM
To: Rulemaking Comments
Subject: Attn: Rulemakings and Adjudications Staff
Attachments: Comments on Risk-Informed Changes 2010.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is a cover letter and my response, dated January 20, 2010, to the NRC's notice of solicitation of public comments on risk-informed changes to loss-of-coolant accident technical requirements, published in the Federal Register, August 10, 2009.

In a separate e-mail, as supplementary information to my response to the NRC's notice of solicitation of public comments, I have sent Appendix Q "Excerpts of Various Papers Not Located in the NRC's ADAMS Documents that are Quoted in Commentator's Responses to the NRC's Notice of Solicitation of Public Comments on Risk-Informed Changes to LOCA Technical Requirements."

The supplementary information has been sent separately due to e-mail-size limits at rulemaking.comments@nrc.gov.

Sincerely,

Mark Leyse

Received: from mail1.nrc.gov (148.184.176.41) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Thu, 21 Jan 2010
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