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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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

Attention: Rulemakings and Adjudications Staff

Subject: Response to the U.S. Nuclear Regulatory Commission's ("NRC") notice of solicitation of public comments on PRM-50-93; NRC-2009-0554

Dear Ms. Vietti-Cook:

Enclosed is Mark Edward Leyse's, Petitioner's, second response to the NRC's notice of solicitation of public comments on PRM-50-93, published in the Federal Register, January 25, 2010.

Respectfully submitted,

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Annette L. Vietti-Cook
Secretary
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Washington, DC 20555-0001

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

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

I. Statement of Commentator's ("Petitioner") Interest

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

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] These same requirements also need to

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

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

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

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

apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[5]

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

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

PRM-50-84 was summarized briefly in the American Nuclear Society's *Nuclear News*'s June 2007 issue[6] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[7] by the Union of Concerned Scientists.

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

In these comments on PRM-50-93, Petitioner provides supplementary information to section III.C.2. of PRM-50-93.

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

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

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

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

> There doesn't seem to be any magic temperature at which you get some autocatalytic reaction that runs away. It's simply a matter of heat balances: how much heat from the chemical process and how much can you pull away.[8]—Dr. Ralph Meyer

> ...I have seen some calculations...dealing with heat transfer of single rods versus bundles which says, well, on heat transfer effects, I just don't learn anything from single rod tests. So I really have to go to bundles, and even multi-bundles to understand the heat transfer. The question we're struggling with now is a modified question. Is there more we need to do to understand what goes on in the reactor accident?[9]—Dr. Dana A. Powers

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

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

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

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[8] Dr. Ralph Meyer, NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Transcript, April 4, 2001. In the transcript the second sentence was transcribed as a question; however, the second sentence was clearly not phrased as a question.

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

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

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

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

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

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

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

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

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

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

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[13] *Id.*

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

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

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

And regarding "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[17] "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> The predicted and observed cladding thermal response are in excellent agreement *until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) oxidation to be underpredicted.*
>
> ... Dr. Haste pointed out that he is chairing a committee (for the OECD) which is preparing a report on the state of the art with respect to Zircaloy oxidation kinetics. He will forward material addressing the low-temperature Zircaloy oxidation problems encountered in the CORA-16 analyses to ORNL [emphasis added].[18]

So in the ORNL SCDAP/RELAP5 calculations performed for the CORA-16 experiment, a rod bundle experiment, "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) oxidation to be underpredicted."[19] This indicates that in the early '90s there were deficiencies in SCDAP/RELAP5 calculations of Zircaloy oxidation kinetics in the 900-1200°C temperature range. And such deficiencies in ECCS evaluation calculations for LOCAs continue to this day (2010).

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

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

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

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

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

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

(It is noteworthy that according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F), at the 0.69 m. elevation.)

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

In PRM-50-93, regarding RELAP5/Mod3 calculations using the Baker-Just and Cathcart-Pawel equations, Petitioner quoted, "Acceptance Criteria and Metal-Water Reaction Correlations," Attachment 2 of "Research Information Letter 0202, Revision of 10 C.F.R. 50.46 and Appendix K;" the document states:

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

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

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

> cladding temperature could be raised to 2300°F without affecting this sensitivity and without reducing the margin that the Commission would have perceived in 1973.
>
> To explore this sensitivity further, we performed more than 50 LOCA calculations with RELAP5/Mod3. In about half of the cases, the Baker-Just equation was used for the metal-water heat generation rate, and in the other half, the Cathcart-Pawel equation was used. Reactor power just prior to the LOCA was varied parametrically to simulate incremental variations in decay heat. The highest peak cladding temperature observed with the Baker-Just equation was about 2600°F; when the temperature went above this value, it continued to the melting point without turning around at some peak value. This indicated that runaway temperatures could not be prevented above about 2600°F for the parameters used in these calculations. The highest peak cladding temperature without runaway observed in corresponding calculations with the Cathcart-Pawel equation was about 2700°F. Each series of calculations done with the two metal-water models always showed peak cladding temperatures without runaway to be at least 100°F higher with Cathcart-Pawel, which is consistent with the temperature difference in the rate equations. Thus in these calculations, the margin between 2300°F and the calculational instability using Cathcart-Pawel was always equal to or greater than the margin between 2200°F and the calculational instability using Baker-Just.[23]

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

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

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With

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

> the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec.[24]

A maximum heating rate of 15°K/sec. indicates that an autocatalytic oxidation reaction commenced. "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10°K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[25] So at the point when peak cladding temperatures increased at a rate of greater than 10°K/sec. during the CORA experiments, autocatalytic oxidation reactions commenced—at cladding temperatures between 2012°F and 2192°F.

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

(It is noteworthy that Schanz is one of the coauthors of "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," which states that "[w]ith the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec."[27])

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

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

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

[27] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, p. 83.

2. Inductive Heating and Furnace Experiments

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

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

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

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

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

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

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

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[28] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, 2003, p. 2.

[29] V. F. Urbanic and T. R. Heidrick, "High-Temperature Oxidation of Zircaloy-2 and Zircaloy-4 in Steam," Journal of Nuclear Materials 75, 1978, p. 252.

[30] Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, located at: www.nrc.gov,

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

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

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

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

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

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

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

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

[33] *Id.*

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

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

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

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

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

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

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

[37] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, April 1992, located at:

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

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

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

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

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

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

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

[40] The experimental data that 50.46(b)(1) and 50.46(b)(2) are primarily based on is reported on in Hobson, D. O. and Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972 and Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, 26 March, 1973.

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

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

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

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

And regarding the significant exothermic heat of oxidation of Zircaloy that was not well recognized in the Hobson/Rittenhouse furnace experiments, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report" states:

> It is important to realize that in the early experiments of oxidation of Zircaloys at high temperatures,[47] specimen temperatures were not

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[43] Hobson, D. O., Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972.
[44] *Id.*
[45] Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," ANS Topical Meeting on Water Reactor Safety, 1973, Salt Lake City, pp. 274-288.
[46] Nuclear Energy Agency, OECD, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," NEA No. 6846, 2009, p. 103.
[47] Hesson, J. C., *et al.*, "Laboratory Simulations of Cladding-Steam Reactions Following Loss-of-Coolant Accidents in Water-Cooled Power Reactors," Argonne National Laboratory, ANL-7609, January 1970; Hobson, D. O., Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by

> measured directly; *e.g.*, by using spot-welded thermocouples. Likewise, specimen temperatures in the experiment of Baker-Just[48] were determined indirectly. *Before [the] mid-1970s, it appears that the effect of the large exothermic heat of oxidation of [Zircaloy] was not well recognized by the investigators.* In Hobson's experiments,[49] the temperature of [the] Zircaloy tube being oxidized was assumed to be the same as the temperature of the uniform central zone of the high-temperature furnace. This assumption would be reasonable for low temperatures; e.g., <800°C. *However, at higher temperatures—e.g., >1100°C—high rate of self-heat generation from oxidation causes actual specimen temperature significantly higher than that of the furnace temperature. In this respect, actual oxidation temperature of a Zircaloy tube reported in Hobson's experiment is believed to be significantly higher; e.g., 1200°C vs. 1260°C* [emphasis added].[50]

It is significant that, according to "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," in the Hobson/Rittenhouse furnace experiments, the temperature of a Zircaloy tube would have been approximately 1260°C when the furnace temperature was 1200°C. So in the Hobson/Rittenhouse furnace experiments, the radiative heat losses of the Zircaloy tube specimens to the furnace environment—that apparently at 1200°C was approximately 60°C lower than the specimen temperature—would have affected the specimens' oxidation kinetics in the experiments.

(It is noteworthy that "[b]efore [the] mid-1970s, it appears that the effect of the large exothermic heat of oxidation of [Zircaloy] was not well recognized by the investigators,"[51] because the Baker-Just equation—required by Appendix K to Part 50 I(A)(5)—which calculates the rate of energy release from the metal-water reaction, dates back to 1962.)

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

[48] Baker, L., Just, L. C., "Studies of Metal-Water Reactions at High Temperatures. III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," Argonne National Laboratory, ANL-6548, May 1962.

[49] Hobson, D. O., Rittenhouse, P. L., "Embrittlement of Zircaloy Clad Fuel Rods by Steam During LOCA Transients," Oak Ridge National Laboratory, ORNL-4758, January 1972 and Hobson, D. O., "Ductile-Brittle Behavior of Zircaloy Fuel Cladding," ANS Topical Meeting on Water Reactor Safety, 1973, Salt Lake City, pp. 274-288.

[50] Nuclear Energy Agency, OECD, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," NEA No. 6846, 2009, p. 38.

[51] *Id.*

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

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

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

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

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

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

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

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[52] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079, p. 79.

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

> oxygen, nitrogen, or hydrogen; nor small variations in alloy composition significantly influence the isothermal rate of oxidation of Zircaloy-4. Obviously, however, both steam temperature and flow rate are important parameters in heat transfer calculations, and *any failure to remove the heat of the Zircaloy-steam reaction from the fuel cladding can result in an increase in the temperature of the cladding* [emphasis added].[54]

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

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

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

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

The NRC states that "temperature control is absolutely necessary in reaction kinetics experiments such as [those conducted with the MaxiZWOK and MiniZWOK]."[59] But clearly, it would not be possible to investigate the oxidation kinetics of Zircaloy fuel-cladding bundles under isothermal conditions at temperatures between 1000°C and 1200°C. If such an attempt were made, it would not be possible to meet the experimental

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[54] *Id.*, pp. 118-119.
[55] Robert H. Leyse, PRM-50-76, May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 5.
[56] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, p. 14.
[57] *Id.*, p. 15.
[58] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, p. 14.
[59] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 14.

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

In the MaxiZWOK experiment, at 1000°C (1832°F), the Zircaloy specimen was able to return to its steady-state value and the specimen self-heating was able to dissipate; however, in a Zircaloy bundle experiment between 1000°C and 1200°C there would be a temperature excursion.

It is significant that regarding the uncontrollable Zircaloy-steam reaction that would occur in the event of a LOCA, "Current Knowledge on Core Degradation Phenomena, a Review" sates:

> Oxidation of Zircaloy cladding materials by steam becomes a significant heat source which increases with temperature; *if the heat removal capability is lost*, it determines a feedback between temperature increase and cladding oxidation [emphasis added].[60]

Furthermore, Figure 1[61] of the same paper depicts that the "start of rapid [Zircaloy] oxidation by H_2O [causes an] uncontrolled temperature escalation," at 1200°C (2192°F),[62] and Figure 13[63] of the same paper depicts that if the initial heat up rate is 1 K/sec. or greater, a cladding temperature excursion would commence at 1200°C (2192°F), in which the rate of increase would be 10 K/sec. or greater.[64]

It is significant that "if the heat removal capability is lost [from the oxidation of Zircaloy cladding materials by steam], it determines a feedback between temperature increase and cladding oxidation;"[65] and that "any failure to remove the heat of the Zircaloy-steam reaction from the fuel cladding can result in an increase in the temperature of the cladding."[66]

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

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

It is significant that in the Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting, on September 29, 2003, Dr. Dana A. Powers stated:

> ...I have seen some calculations...dealing with heat transfer of single rods versus bundles which says, well, on heat transfer effects, I just don't learn anything from single rod tests. So I really have to go to bundles, and even multi-bundles to understand the heat transfer.[68]

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

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

3. Multi-Rod Bundle (Assembly) Experiments

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

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

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

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

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

> One of the bases for selecting 2200°F (1204°C) as the PCT [limit] was that it provided a safe margin, or conservatism, away from an area of zircaloy oxidation behavior known as the autocatalytic regime. *The autocatalytic condition occurs when the heat released by the exothermic zircaloy-steam reaction (6.45 megajoules per kg zircaloy reacted) is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant.* This reaction heat then further raises the zircaloy temperature, which in turn increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.
>
> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. This type of comparison implicitly includes...complex heat transfer mechanisms...and the effects of fuel rod ballooning and rupture on coolability... [E]ven though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, *when design basis heat transfer and decay heat are considered,* some margin above 2200°F exists [emphasis added].[71]

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[69] From a letter, dated February 9, 1972, from Oak Ridge National Laboratory Director Alvin Weinberg to AEC Chairman James Schlesinger; in Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, pp. 4.28-4.29.

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

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

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

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

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

Discussing how the oxidation rate of Zircaloy increases with increasing temperature in the conditions of excellent thermal insulation that multi-rod bundle tests provide, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> As already observed in previous tests [(CORA Tests B and C)],[75] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing

[72] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, pp. 30, 33.
[73] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, p. 8-2.
[74] Robert H. Leyse, "PRM-50-76," p. 1.
[75] S. Hagen et al., "Interactions between Aluminium Oxide Pellets and Zircaloy Tubes in Steam Atmosphere at Temperatures above 1200°C (Posttest Results from the CORA Tests B and C)," KfK-4313, 1988.

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

(It is noteworthy that Schanz is one of the coauthors of "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3).")

As stated above, it would not be possible to investigate the oxidation kinetics of Zircaloy fuel-cladding bundles under isothermal conditions at temperatures between 1000°C and 1200°C. If such an attempt were made, it would not be possible to meet the experimental protocol of isothermal conditions, because the energy from the exothermal Zircaloy-steam oxidation would cause a temperature excursion.

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

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

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

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

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

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

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

observations of the CORA-2 and CORA-3 experiments, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states:

> As already observed in previous tests [(CORA Tests B and C)],[79] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles. ...
>
> This explains the observation that the temperature escalation starts at the hottest position in the bundle, at an elevation above the middle. *From there, slowly moving fronts of bright light, which illuminated the bundle, were seen, indicating the spreading of the temperature escalation upward and downward.* It is reasonable to assume, that the violent oxidation essentially consumed the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, should have occurred [emphasis added].[80]

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

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

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

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

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

Regarding how the metal-water reaction propagates away from the initiation point, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states:

> The first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation. ... A cladding thermocouple at the same elevation...reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase. Note that, due to the limited number of measured cladding temperature locations, the precise location of the initiation of [the] metal-water reaction on any given fuel rod or guide tube is not likely to coincide with the location of a thermocouple. Thus, the temperature rises are probably associated with precursory heating as the metal-water reaction propagates away from the initiation point. Care must be taken in determining the temperature at which the metal-water reaction initiates, since the precursory heating can occur at a much lower temperature. It can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400 K (2060°F).[81]

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

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

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

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

[81] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," International Agreement Report, NUREG/IA-0049, pp. 30, 33.

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

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

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

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

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

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

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

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

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

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

that "[w]ith the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec,"[85] Schanz claims that "the Baker-Just Correlation will retain its importance for comparison and licensing purposes."[86]

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

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

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

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

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

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

[85] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, 1991, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230460, p. 83.
[86] G. Schanz, "Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes," FZKA 6827, p. 8.
[87] *Id.*

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

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

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

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

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

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

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[88] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, p. 6.

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

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

[91] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, pp. 117, 118.

So despite having coauthored "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," which states that "[w]ith the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15°K/sec,"[92] Schanz claims that the Cathcart-Pawel equation is of high reliability for calculating oxidation kinetics for temperatures below 2200°F.[93]

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

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

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

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

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[92] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, p. 83.

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

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

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

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

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

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

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

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

> The [LOFT LP-FP-2] experiment...provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[99]

And according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2

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

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

[99] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 3. 23.

experiment commenced at approximately 1400 K (2060°F). Also, according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" the peak measured cladding temperature reached 2100°K (3320°F) within approximately 75 seconds.

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

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

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

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

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

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

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

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

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

b. The ASTEC Code

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

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

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

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

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

> DIVA module able to calculate core degradation, corium relocation and behavior in the lower head up to vessel failure. Most DIVA models are issued from the ICARE2 IRSN mechanistic code for core degradation (Chatelard, et al., 2006), except some fast-running models that were specifically developed for ASTEC (2D core gas thermal-hydraulics and corium behavior in the lower plenum).
>
> Many partners of the SARNET network of excellence (in the 6th Framework Programme of the European Commission) were involved in ASTEC V1 code validation against experiments. This paper summarizes the main results of the validation performed on the CESAR and DIVA modules of the successive code versions up to ASTEC V1.3rev2 delivered in December 2007. Table 1 presents the selected experiments that include several International Standard Problems (ISP) of OECD/CSNI (Committee on the Safety of Nuclear Installations).[103]

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

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

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

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

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

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

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

(It is noteworthy that according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F), at the 0.69 m. elevation.)

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

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

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

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

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

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

> code to stimulate thermal-hydraulic conditions and core degradation phenomena. The ASTEC results were then compared with the results of the ICARE/CATHARE code.
>
> ASTEC simulates reasonably well the transient phase of the experiment before the reflood phase, that is, reactor system thermal-hydraulics, core uncovery and heatup, hydrogen generation and fission-product release. The total hydrogen release is in good agreement with test measurements. Instead the code needs some improvement in order to investigate the reflood phase *because temperature excursions and consequent heavy degradation of the fuel rods, hydrogen release and primary pressure increase are not reproduced by ASTEC because of the inadequate modeling.*
>
> In general, the ICARE/CATHARE results confirm the validity of the ASTEC results [emphasis added].[108]

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

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

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

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

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[108] ENEA, Nuclear Fusion and Fission, and Related Technologies Department, "2006 Progress Report," pp. 109-110.

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

[110] G. Bandini, *et al.*, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," p. 155.

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

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

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

(It is noteworthy that according to "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" the temperature excursion in the LOFT LP-FP-2 experiment commenced at approximately 1400 K (2060°F), at the 0.69 m. elevation.)

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

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

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

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

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

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

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

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

III. Conclusion

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

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

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

(Such experimental data also indicates the Baker-Just and Cathcart-Pawel equations are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA, which, in turn, indicates these equations are both non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.)

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

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

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

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

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

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

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

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

Experiment conductors and reviewers and regulators have not acknowledged that it would not be possible to investigate the oxidation kinetics of Zircaloy fuel-cladding bundles under isothermal conditions at temperatures between 1000°C and 1200°C. If such an attempt were made, it would not be possible to meet the experimental protocol of isothermal conditions, because the energy from the exothermal Zircaloy-steam oxidation would cause a temperature excursion.

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

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

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

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

[114] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079, p. 14.

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

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

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

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

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

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

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

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

[118] *Id.*

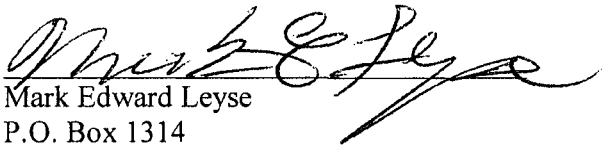
[119] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting,"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

[120] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," NUREG/CP-0119, Vol. 2, p. 83.

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

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

Respectfully submitted,



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Dated: April 12, 2010

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

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

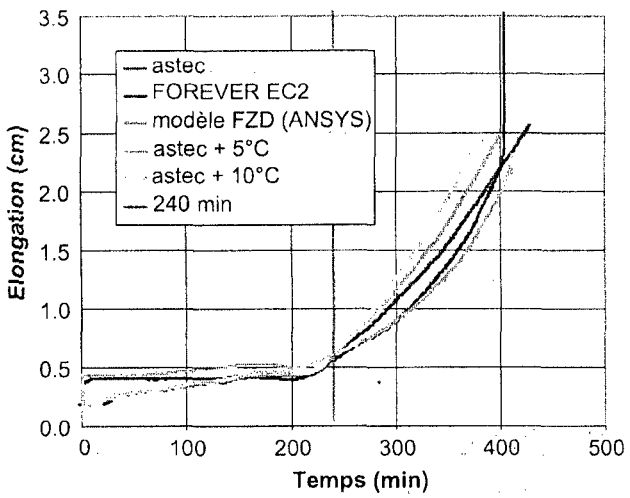

Fig. 16. FOREVER EC2 lower head displacement.

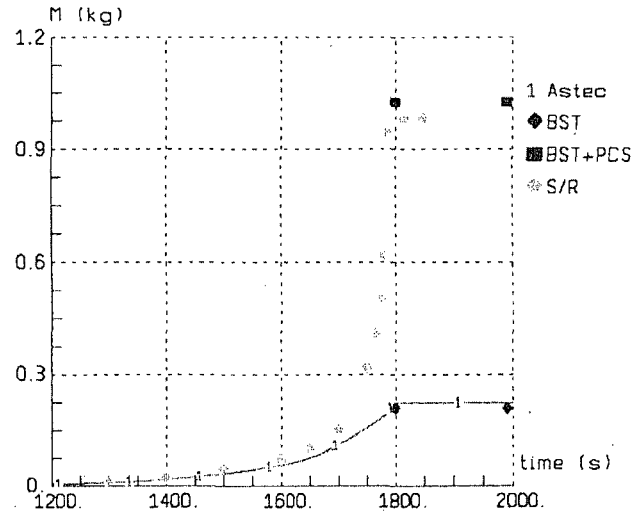

Fig. 18. LOFT-LP-FP-2 total mass of hydrogen produced.

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

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

4.1. LOFT-LP-FP-2 experiment

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

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

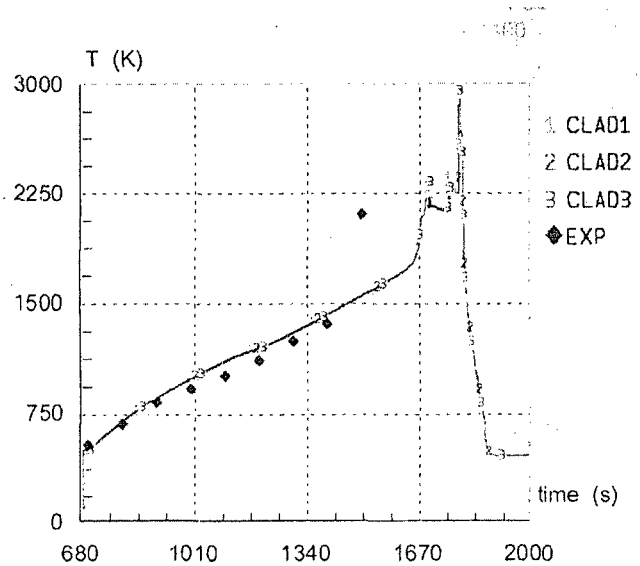

Fig. 17. LOFT-LP-FP-2 CFM clad temperature at elevation 1.067 m.

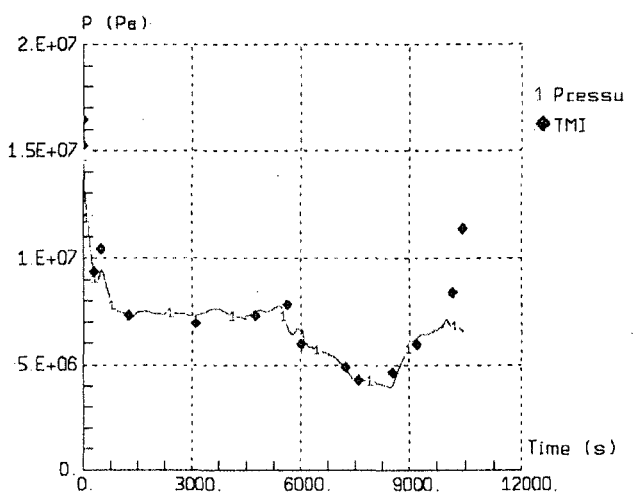

Fig. 19. TMI-2 primary system pressure.

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

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

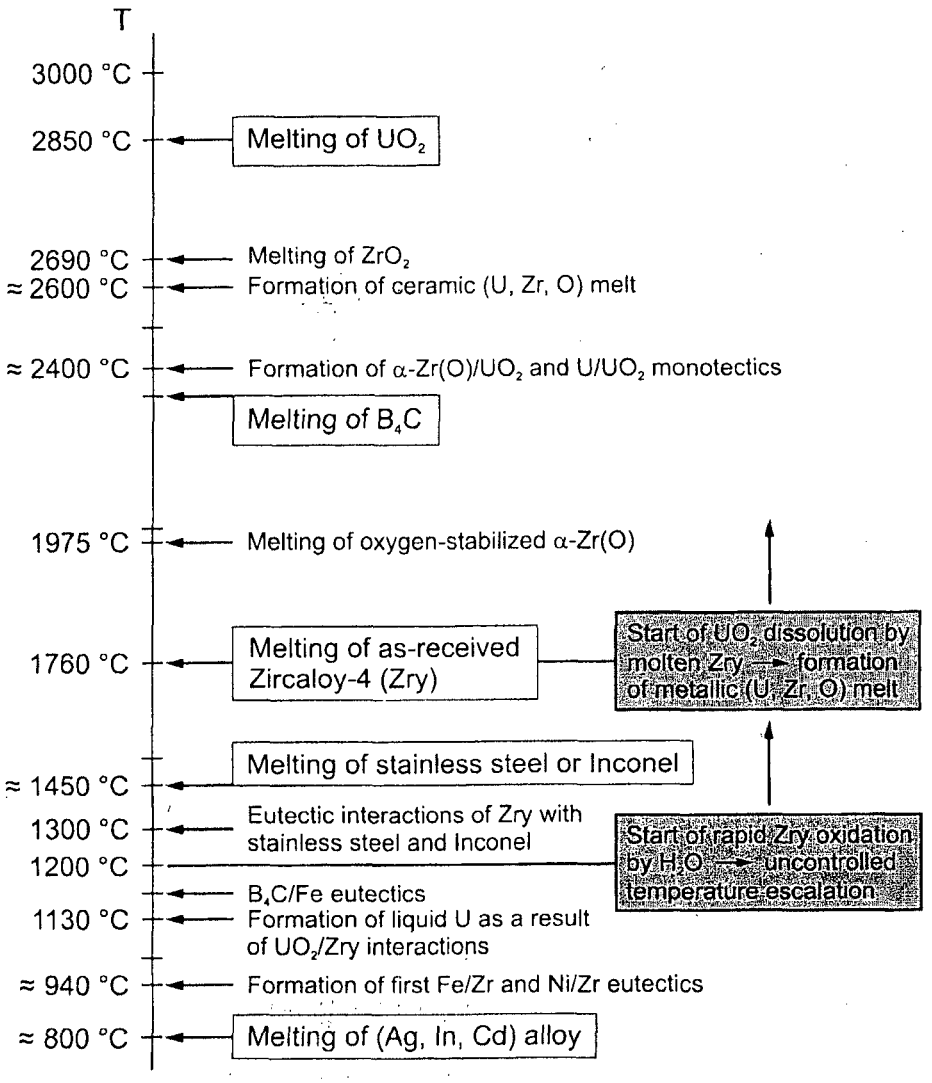

Fig. 1. LWR severe accident-relevant melting and chemical interaction temperatures which result in the formation of liquid phases.

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

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

Many of these physical and chemical processes have been identified in separate-effects tests, out-of-pile and in-pile integral severe fuel damage (SFD) experiments, and Three Mile Island Unit 2 (TMI-2) core material examinations [5–10,33]. All of these interactions are of concern in a severe accident, because relocation and/or solidification of the resulting fragments or melts may result in local cooling channel blockages of different sizes and may cause further heatup of these core regions

steam starvation. At high heat-up rates >5 K/s, the ZrO_2 layer will probably be too thin to hold the metallic melt in place and relocation will occur after mechanical and/or chemical breach of the ZrO_2 shell (Fig. 13).

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

5.3. Material distribution in integral experiments

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

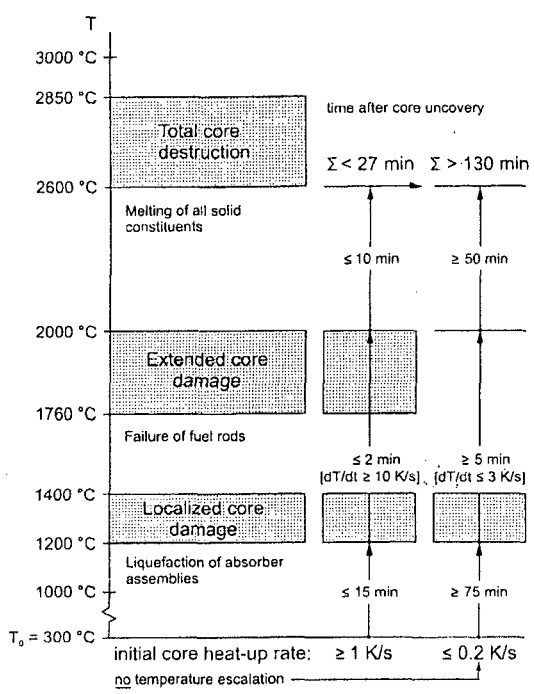

Fig. 13. Dependence of the temperature regimes on liquid phase formation on the initial heat-up rate of the core. Small heat-up rates drastically reduce the amount of molten Zircaloy (1800–2000°C) and give more time for possible accident management measures.

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

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

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

Rulemaking Comments

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

Dear Ms. Vietti-Cook:

Attached to this e-mail is a cover letter and my response, dated April 12, 2010, to the NRC's notice of solicitation of public comments on PRM-50-93, NRC-2009-0554, published in the Federal Register, January 25, 2010.

Sincerely,

Mark Leyse

Received: from mail2.nrc.gov (148.184.176.43) by OWMS01.nrc.gov
(148.184.100.43) with Microsoft SMTP Server id 8.1.393.1; Sun, 11 Apr 2010
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