EDO Principal Correspondence Control

FROM: DUE: 07/08/10 EDO CONTROL: G20100368

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FINAL REPLY:

Mark Edward Leyse
New England Coalition

TO:

Borchardt, EDO

FOR SIGNATURE OF : ** GRN ** CRC NO:

Leeds, NRR

DESC:

2.206 - Lower the Licensing BAsis Peak Caldding Temperature of Vermont Yankee in Order to Provide Necessary Margin of Safety in Event of Loss-of-Coolant Accident (EDATS: OEDO-2010-0476)

ROUTING:

Borchardt
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Scott, OGC
Burns, OGC
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DATE: 06/09/10

ASSIGNED TO: CONTACT:

NRR Leeds

SPECIAL INSTRUCTIONS OR REMARKS:

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General Information

Assigned To: NRR **OEDO Due Date:** 7/8/2010 11:00 PM

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Subject: 2.206 - Lower the Licensing Basis peak Cladding Temperature of Vermont Yankee in Order to Provide Necessary Margin of Safety In Event of Loss-of-Coolant Accident

Description:

CC Routing: RegionI; OGC; Tanya.Mensah@nrc.gov; Catherine.Scott@nrc.gov

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Other Information

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OEDO Concurrence: NO

OCM Concurrence: NO

OCA Concurrence: NO

Special Instructions:

Document Information

Originator Name: Mark Edward Leyse **Date of Incoming:** 6/7/2010

Originating Organization: New England Coalition **Document Received by OEDO Date:** 6/8/2010

Addressee: R. W. Borchardt, EDO **Date Response Requested by Originator:** NONE

Incoming Task Received: Letter

June 7, 2010

R. William Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

Subject: 10 C.F.R. § 2.206 Request to Lower the Licensing Basis Peak Cladding Temperature of Vermont Yankee Nuclear Power Station (Docket-50-271) in Order to Provide a Necessary Margin of Safety—to Help Prevent a Meltdown—in the Event of a Loss-of-Coolant Accident

Dear Mr. Borchardt:

The enclosed 10 C.F.R. § 2.206 petition is submitted on behalf of New England Coalition of Brattleboro, Vermont by Mark Edward Leyse.

10 C.F.R. § 2.206(a) states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper."

New England Coalition requests that the United States Nuclear Regulatory Commission ("NRC") order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a loss-of-coolant accident ("LOCA"). Experimental data indicates that VYNPS's LBPCT of 1960°F[1] does not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA. Such data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety.

To uphold its congressional mandate to protect the lives, property, and environment of the people of Vermont and locations within proximity of VYNPS, the NRC must not allow VYNPS's LBPCT to remain at an elevated temperature that would not provide a necessary margin of safety, in the event of LOCA. If implemented, the enforcement action proposed in this petition would help improve public and plant worker safety.

New England Coalition respectfully submits that—although revisions to the 10 C.F.R. § 50.46(b)(1) peak cladding temperature limit criterion have been proposed in a rulemaking petition—this petition is separately and appropriately brought under 10 C.F.R. § 2.206, because the concerns brought forward are plant specific, brought by a local, affected party, and have immediate bearing on safety margins at VYNPS, currently operating at its maximum permissible extended power uprate level. Furthermore, the concerns raised

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

EDO --G20100368

in the enclosed 10 C.F.R. § 2.206 petition are of an immediate nature that require prompt NRC review and action, which are available to the petitioners only through the 10 C.F.R. § 2.206 process.

New England Coalition looks forward to providing any additional information or clarification as may be required by your office or by a petition review board.

Respectfully submitted,

Mark Edward Leyse
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June 7, 2010

R. William Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

10 C.F.R. § 2.206 REQUEST TO LOWER THE LICENSING BASIS PEAK CLADDING TEMPERATURE OF VERMONT YANKEE NUCLEAR POWER STATION IN ORDER TO PROVIDE A NECESSARY MARGIN OF SAFETY—TO HELP PREVENT A MELTDOWN—IN THE EVENT OF A LOSS-OF-COOLANT ACCIDENT

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June 7, 2010

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of:	:	TO: R. WILLIAM BORCHARDT Executive Director for Operations U.S. Nuclear Regulatory Commission Washington D.C. 20555-0001
ENTERGY NUCLEAR OPERATIONS, INC. (Vermont Yankee Nuclear Power Station; Facility Operating License No. DPR-28)	: : :	
---	:	Docket No. ______

NEW ENGLAND COALITION,
Petitioner

10 C.F.R. § 2.206 REQUEST TO LOWER THE LICENSING BASIS PEAK CLADDING TEMPERATURE OF VERMONT YANKEE NUCLEAR POWER STATION IN ORDER TO PROVIDE A NECESSARY MARGIN OF SAFETY—TO HELP PREVENT A MELTDOWN—IN THE EVENT OF A LOSS-OF-COOLANT ACCIDENT

I. REQUEST FOR ACTION

This petition for an enforcement action is submitted pursuant to 10 C.F.R. § 2.206 by New England Coalition. 10 C.F.R. § 2.206(a) states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper."

Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a loss-of-coolant accident ("LOCA"). Experimental data indicates that

VYNPS's LBPCT of 1960°F[1] does not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA. Such data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety.

II. STATEMENT OF PETITIONER'S INTEREST

New England Coalition ("NEC") is a membership-supported 501(c)(3) non-profit educational organization, based in Brattleboro, Vermont, which serves the New England region of the United States. NEC was initially named New England Coalition on Nuclear Pollution; it was founded in February of 1971 by several groups of citizens and scientists from Vermont and western Massachusetts.

From the time of its founding, NEC has been an intervenor in numerous NRC licensing proceedings. NEC's legal efforts have included interventions before the NRC to challenge VYNPS's plans to increase—in 1977 and 1987—its on-site storage capacity for spent fuel. NEC has intervened before the Vermont Public Service Board in numerous VYNPS proceedings. NEC also intervened before the Vermont Environmental Court on VYNPS's thermal discharge.

Petitioner is submitting this 10 C.F.R. § 2.206 petition because VYNPS's LBPCT of 1960°F[2] must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

Petitioner respectfully submits that—although revisions to the 10 C.F.R. § 50.46(b)(1) peak cladding temperature limit criterion have been proposed in a rulemaking petition—this petition is separately and appropriately brought under 10 C.F.R. § 2.206, because the concerns brought forward are plant specific, brought by a local, affected party, and have immediate bearing on safety margins at VYNPS, currently operating at its maximum permissible extended power uprate level. Furthermore, the concerns raised in the enclosed 10 C.F.R. § 2.206 petition are of an immediate nature that require <u>prompt</u>

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

NRC review and action, which are available to Petitioner only through the 10 C.F.R. § 2.206 process.

This 10 C.F.R. § 2.206 petition is submitted on behalf of NEC by Mark Edward Leyse.

On March 15, 2007, Mark Edward Leyse submitted a petition for rulemaking, PRM-50-84 (ADAMS Accession No. ML070871368). PRM-50-84 was summarized briefly in American Nuclear Society's ("ANS") *Nuclear News*'s June 2007 issue[3] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[4] by Union of Concerned Scientists ("UCS"). In 2008, the NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.

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

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

On November 17, 2009, Mark Edward Leyse submitted a second petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the NRC make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly)

[3] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[4] 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. 2.

severe fuel damage experiments;[5] and 2) to stipulate minimum allowable core reflood rates, in the event of a LOCA.[6, 7]

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.[8] These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[9]

PRM-50-93 was discussed briefly in ANS's *Nuclear News*'s March 2010 issue[10] and commented on by UCS.

Regarding PRM-50-93, UCS states:

> In our opinion, [PRM-50-93] addresses a genuine safety problem. We believe the NRC should embark on a rulemaking process based on this petition. We are confident that this process would culminate in revised

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

[6] 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.

[7] 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.

[8] 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.

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

[10] American Nuclear Society, *Nuclear News*, March 2010, p. 36.

> regulations—perhaps not precisely the ones proposed [in PRM-50-93] but ones that would adequately resolve the issues...meticulously identified [in PRM-50-93]—that would better ensure safety in event of a loss of coolant accident.[11]

Mark Edward Leyse also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis," which was presented at ANS's 2009 Winter Meeting, November 15-19, 2009, Washington, D.C.

III. FACTS CONSTITUTING THE BASIS FOR PETITIONER'S REQUEST

> There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the [General Electric] determination to prove out their ECC systems. ... Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ...the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven. From a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective.[12]—J. W. McConnell

> [Consolidated National Intervenors's] direct testimony concluded that a near thermal runaway condition existed in [BWR-FLECHT] Test ZR-2. It is of compelling importance that Roger Griebe, the [Aerojet] project engineer for BWR-FLECHT, stated a similar interpretation of this test, which they submitted to GE, and Griebe testified, there is *no* convincing proof available from ZR-2 test data to demonstrate that this near-thermal runaway definitely did not exist.[13]—Henry. W. Kendall and Daniel F. Ford

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[11] David Lochbaum, Union of Concerned Scientists, "Comments Submitted by the Union of Concerned Scientists on the Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-93)," April 27, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101180175, p. 1.

[12] J. W. McConnell, Aerojet internal memoranda; see Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, Union of Concerned Scientists, 1974, p. 5.11.

[13] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 5.11.

A. General Electric's ECCS Evaluation Calculations that Helped Qualify the 20% Power Uprate for VYNPS are Non-Conservative

Regarding the licensing basis peak cladding temperature ("LBPCT") for VYNPS at power levels of 1593 MWt and 1912 MWt, General Electric's "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate" states:

> The LBPCT was determined based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties. The CPPU GE14 LBPCT is 1960°F at CPPU RTP and rated core flow. This is 50°F greater than the LBPCT at the pre-CPPU conditions. The CPPU GE13 LBPCT is 1940°F at CPPU RTP and rated core flow. This is 40°F greater than the LBPCT at the pre-CPPU conditions (see Table 4-3). The LBPCT for GE14 and GE13 fuel are bounding for GE9 fuel. Although the PCT changes due to CPPU are greater than the typically seen 20°F, these changes are small compared to the margin to the 2200°F licensing limit that the bounding LBPCTs of 1960°F and 1940°F provide. In addition, the effect on the LBPCT adder is negligible considering the margin to the 2200°F licensing limit. The ECCS-LOCA results for VYNPS are in conformance with the error reporting requirements of 10 CFR 50.46 through notification number 2003-003.[14]

(Table 4-3 states that before the power uprate—at 104.5% of the 2003 licensed thermal power—the LBPCT was 1910°F and 1900°F for GE14 and GE13 fuel, respectively. Table 4-3 also states that after the power uprate—at 120% of the 2003 licensed thermal power—the LBPCT would be 1960°F and 1940°F for GE14 and GE13 fuel, respectively. Additionally, Table 4-3 states that for 104.5% and 120% of the 2003 licensed thermal power the calculated total oxidation of the cladding would be lower than 3% (at any local point), respectively, and that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water and steam would be lower than 0.1%, respectively. Furthermore, Table 4-3 states that for 104.5% and 120% of the 2003 licensed thermal power there would be a coolable core geometry and core long term cooling, respectively.[15])

[14] General Electric, "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate," September 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML032580103, p. 4-12.
[15] *Id.*, p. 4-19.

So the ECCS evaluation calculations that helped qualify VYNPS's constant pressure power uprate, calculated VYNPS's LBPCT at 1960°F and 1940°F for GE14 and GE13 fuel, respectively.

It is significant that "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate" states that "[t]he LBPCT was determined based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties," because the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction for the "the calculated Appendix K PCT" would have been calculated with the Baker-Just equation.

(Regarding the Baker-Just equation, 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*, states:

> The rate of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation.)

It is significant that the Baker-Just equation calculated autocatalytic (runaway) oxidation to occur when cladding temperatures increased above approximately 2600°F—in approximately half of more than 50 LOCA calculations that the NRC performed with RELAP5/Mod3[16]—because data from severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that autocatalytic oxidation of Zircaloy cladding can commence at far lower temperatures: even more than 500 degrees Fahrenheit lower than 2600°F. Therefore, the Baker-Just equation is 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 equation is non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

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

It is also significant that regarding "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[17] a BWR severe fuel damage experiment, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> The predicted and observed cladding thermal response are in excellent agreement *until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted.*
>
> … Dr. Haste pointed out that he is chairing a committee (for the OECD) which is preparing a report on the state of the art with respect to Zircaloy oxidation kinetics. He will forward material addressing the low-temperature Zircaloy oxidation problems encountered in the CORA-16 analyses to ORNL [emphasis added].[18]

Additionally, it is significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory" ("In-Vessel Phenomena—CORA"), states that for the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations."[19]

Regarding the CORA-16 and CORA-17 experiments, "In-Vessel Phenomena—CORA" states:

> Applications of ORNL models specific to the KfK CORA-16 and CORA-17 experiments are discussed and significant findings from the experimental analyses such as the following are presented:
>
> 1) applicability of available Zircaloy oxidation kinetics correlations,
>
> 2) influence of cladding strain on Zircaloy oxidation…[20]

The Baker-Just correlation was among the "available Zircaloy oxidation kinetics correlations"—in 1991—when "In-Vessel Phenomena—CORA" was presented. So according to "In-Vessel Phenomena—CORA," the Baker-Just correlation did not accurately predict the cladding oxidation of the CORA-16 experiment. Furthermore, in

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[17] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," October 16, 1990, p. 3.
[18] *Id.*
[19] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.
[20] *Id.*

the CORA-16 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) [(1652-2192°F)] oxidation to be underpredicted."[21] This also indicates that the Baker-Just equation is non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.

Therefore, General Electric's ECCS evaluation calculations—which used the Baker-Just equation for calculating the metal-water reaction rates that would occur in the event of a LOCA—that helped qualify the constant pressure power uprate for VYNPS are non-conservative.

B. The 10 C.F.R. § 50.46(b)(1) Peak Cladding Temperature limit of 2200°F is Non-Conservative

It is significant that General Electric's "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate" states:

> The CPPU GE14 LBPCT is 1960°F at CPPU RTP and rated core flow. This is 50°F greater than the LBPCT at the pre-CPPU conditions. ... Although the PCT changes due to CPPU are greater than the typically seen 20°F, these changes are small compared to the margin to the 2200°F licensing limit that the bounding [LBPCT] of 1960°F...provide[s]. In addition, the effect on the LBPCT adder is negligible considering the margin to the 2200°F licensing limit.[22]

So the alleged conservatism of VYNPS's LBPCT of 1960°F is predicated on the premise that the 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit of 2200°F would provide a necessary margin of safety in the event of LOCA. Unfortunately, the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F would not provide a necessary margin of safety in the event of LOCA.

[21] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.

[22] General Electric, "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate," p. 4-12.

It is commonly asserted that the autocatalytic oxidation of Zircaloy would commence at cladding temperatures far greater than 2200°F, in the event of a LOCA. Discussing the 2200°F PCT limit and autocatalytic (runaway) Zircaloy oxidation, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> One of the bases for selecting 2200°F (1204°C) as the PCT [limit] was that it provided a safe margin, or conservatism, away from an area of zircaloy oxidation behavior known as the autocatalytic regime. The autocatalytic condition occurs when the heat released by the exothermic zircaloy-steam reaction (6.45 megajoules per kg zircaloy reacted) is greater than the heat that can be transferred away from the zircaloy by conduction to the fuel pellets or convection/radiation to the coolant. This reaction heat then further raises the zircaloy temperature, which in turn increases the diffusivity of oxygen into the metal, resulting in an increased reaction rate, which again increases the temperature, and so on.[23]

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

> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. This type of comparison implicitly includes...complex heat transfer mechanisms...and the effects of fuel rod ballooning and rupture on coolability... Analysis of experiments performed in the Power Burst Facility, in the Annular Core Research Reactor, and in the NEILS-CORA (facilities in West Germany) program have shown that temperatures above 2200°F are required before the zircaloy-steam reaction becomes sufficiently rapid to produce an autocatalytic temperature excursion. Another group of relevant experimental data were produced from the MT-6B and FLHT-LOCA and Coolant Boilaway and Damage Progression tests conducted in the NRU Reactor in Canada. ...even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists.[24]

It is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" states that assessing the conservatism of the 2200°F PCT limit, as a boundary

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

that would prevent autocatalytic oxidation from occurring, can be accomplished by analyzing data from multi-rod severe accident tests, because such data, in fact, indicates that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

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

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15 K/sec.[25]

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."[26] So at the point when peak cladding temperatures increased at a rate of greater than 10 K/sec. during the CORA experiments, autocatalytic oxidation reactions commenced at cladding temperatures between 2012°F and 2192°F.

(It is noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis," published in 1988, does not mention that some reports state that autocatalytic oxidation commenced in the LOFT LP-FP-2 experiment—conducted in 1985—at cladding temperatures of approximately 2060°F.[27])

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[25] 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.

[26] 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.

[27] 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:

Furthermore, recent papers still assert that the autocatalytic oxidation of Zircaloy would commence at cladding temperatures far greater than 2200°F, in the event of a LOCA. For example, "The History of LOCA Embrittlement Criteria," presented in October 2000, 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].[28]

And, for example, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," published in 2009, states:

> Last but not least important, is the large exothermic heat generated during oxidation of the cladding. At high enough temperatures, the rate of steam-cladding oxidation is so high that the heat can no longer be adequately dissipated by cooling, eventually leading to runaway oxidation. If runaway or autocatalytic oxidation is not arrested, cladding metal and [the] reactor core could melt. *Although this temperature is well above any temperature expected in a design basis loss-of-coolant accident*, such events occurred in the...Three Mile Island [accident] [emphasis added].[29]

So, clearly, many people who are concerned with nuclear safety issues still have not acknowledged that in multi-rod bundle experiments, like the LOFT LP-FP-2 experiment and CORA experiments, the onset of runaway oxidation commenced at cladding temperatures lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

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

[28] 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.

[29] Nuclear Energy Agency, OECD, "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions: State-of-the-Art Report," NEA No. 6846, 2009, p. 26.

C. Experiments that Indicate VYNPS's LBPCT of 1960°F for GE14 Fuel would Not Provide a Necessary Margin of Safety to Help Prevent a Partial or Complete Meltdown, in the Event of 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.[30]—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?[31]—Dr. Dana A. Powers

> As already observed in previous tests, the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above [1832°F]. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles.[32]—S. Hagen, *et al.*

In this section, Petitioner will discuss data from multi-rod severe fuel damage experiments and one multi-rod thermal hydraulic experiment that indicates VYNPS's LBPCT of 1960°F for GE14 fuel would not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

Petitioner will discuss: 1) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures below VYNPS's LBPCT of 1960°F; 2) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures of 2060°F or lower; 3) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures of

[30] 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.

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

[32] 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.

approximately 2192°F (approximately at the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F); and 4) one experiment in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at a temperature of 2275°F or lower.

It is noteworthy that some of the multi-rod severe fuel damage experiments discussed in this section simulated pressurized water reactor ("PWR") fuel assemblies. There would definitely be differences in how the different ECCSs and core components of boiling water reactors ("BWR") and PWRs (*e.g.*, the BWR boron carbide (B_4C) absorber versus the PWR Ag-In-Cd absorber) would affect the progression of a LOCA. However, the temperatures at which the autocatalytic oxidation of Zircaloy cladding by steam would commence during a LOCA at a BWR and PWR would be similar, as the results of multi-rod severe fuel damage experiments that simulated BWR and PWR fuel assemblies indicate.

1. Multi-Rod Severe Fuel Damage Experiments in which the Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at Temperatures below VYNPS's LBPCT of 1960°F

VYNPS's 10 C.F.R. § 50.46(a)(3)(ii) annual report for 2009 states that VYNPS's LBPCT is 1960°F for GE14 fuel.[33]

VYNPS's LBPCT of 1960°F for GE14 fuel would not provide a necessary margin of safety to help prevent a partial or complete meltdown, in the event of a LOCA. Experimental data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety.

It is significant that the CORA-2 and CORA-3 experiments, initiated with a temperature ramp rate of 1 K/sec, had temperature excursions, due to the exothermal Zircaloy-steam reaction, that commenced at approximately 1000°C (1832°F),[34] leading

[33] Entergy, "VYNPS 10 C.F.R. § 50.46(a)(3)(ii) Annual Report for 2009," p. 2.
[34] See Appendix A Fig. 12. Temperatures during Test CORA-2 at [550] mm and 750 mm Elevation and Fig. 13. Temperatures Measured during Test CORA-3 at 450 mm and 550 mm Elevation.

the CORA-2 and CORA-3 bundles to maximum temperatures of 2000°C and 2400°C, respectively.[35]

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

> As already observed in previous tests [(CORA Tests B and C)],[36] 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.[37]

So the CORA 2 and CORA 3 experiments demonstrated that temperature escalations due to the rapid oxidation of Zircaloy can commence at temperatures as low as 1000°C (1832°F).

Regarding cladding temperature escalations that occur because of the exothermic metal-water reaction, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures" states:

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15 K/sec.[, after an initial heatup rate of about 1 K /sec.] The maximum temperatures attained are about 2000°C...[38]

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

[36] 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.

[37] 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.

[38] 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.

It is significant that "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures" states that in the CORA Experiments, at cladding temperatures between 1100°C and 1200°C (2012°F to 2192°F), the cladding began to rapidly oxidize and cladding temperatures started increasing at a maximum rate of 15°C/sec. (27°F/sec.), because "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[39]

So when the CORA 2 and CORA 3 experiments had cladding temperature escalations because of the exothermic metal-water reaction, which commenced at approximately 1000°C (1832°F), local cladding temperatures would have increased at a maximum rate of 15°C/sec. (27°F/sec.). And within a period of approximately 60 seconds peak cladding temperatures would have increased to above 3000°F; the melting point of Zircaloy is approximately 3308°F.[40]

Therefore, data from the CORA 2 and CORA 3 experiments indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

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

> In the CORA experiments test bundles of usually 16 electrically heated fuel rod simulators and nine unheated rods are subjected to temperature transients of a slow heatup rate in a steam environment. Thus an accident sequence is simulated, which may develop from a small-break loss-of-coolant accident of an LWR.
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> CORA-2 and CORA-3 were the first "Severe Fuel Damage" experiments of the program with UO_2 pellet material. The transient tests were performed on August 6, 1987, and on December 3, 1987, respectively.

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

[40] 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.

> Both test bundles did not contain absorber rods. Therefore, CORA-2 and CORA-3 can serve as reference experiments for the future tests, in which the influence of absorber rods will be considered. An aim of CORA-2, as a first test of its kind, was also to gain experience in the test conduct and posttest handling of UO_2 specimens. CORA-3 was performed as a high-temperature test. With this test the limits of the electric power supply unit could be defined
>
> The transient phases of CORA-2 and CORA-3 were initiated with a temperature ramp rate of 1 K/sec. The temperature escalation due to the exothermal [Zircaloy]-steam reaction started at about 1000°C, leading the bundles to maximum temperatures of 2000°C and 2400°C for tests CORA-2 and CORA-3, respectively.[41]

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

> The high-temperature shield is located within the pressure tube. Through a number of holes in the shield, the test bundle is being inspected during the test by several video and still cameras. The holes are also used for temperature measurements by two-color pyrometers complementing the thermocouple readings at elevated temperatures.[42]

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

> The tests CORA-2 and CORA-3 have been successfully conducted, accompanied by measurements and visual observations and evaluated by micro-structural and compositional analyses. On the basis of this information and the expertise from separate-effects investigations the following interpretation of the sequence of mechanisms during the degradation of the bundles is given.

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

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

As already observed in previous tests [(CORA Tests B and C)],[43] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C. This temperature escalation is due to the additional energy input from the exothermal [Zircaloy]-steam oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles. An effectively moderated escalation would be observed for smaller initial heatup rates, because the growth of protective scale during steam exposure counteracts by decreasing the oxidation rate of the material.

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

A first melting process starts already at about 1250°C at the central grid spacer of Inconel, due to diffusive interaction in contact with Zry cladding material, by which the melting temperatures of the interaction partners (ca. 1760°C for Zry, ca. 1450°C for Inconel) are dramatically lowered towards the eutectic temperature, where a range of molten mixtures solidifies. (This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[44]

It is significant that "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)" states "[a]s already observed in previous tests [(CORA Tests B and C)],[45] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C."[46] So the CORA 2 and CORA 3

[43] 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.
[44] 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.
[45] 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.
[46] 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.

experiments were not the only CORA experiments to have temperature excursions that commenced at 1000°C, because of the autocatalytic oxidation of Zircaloy cladding by steam.

It is also significant that one passage from "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" states:

> The temperature rise shows the same general features already found in earlier tests. With the increase of the electrical power input, first the temperature rises proportional to the power. *Having reached about 1000°C, the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation* [emphasis added].[47]

(Elsewhere "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" states that temperature escalations due to the exothermic Zircaloy-steam reaction began at approximately 1100°C (2012°F).)

Additionally, it is significant that "Degraded Core Quench: Summary of Progress 1996-1999" states that the autocatalytic oxidation of Zircaloy cladding by steam commences at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or greater.[48]

So there are papers that report the autocatalytic oxidation of Zircaloy cladding by steam commences at temperatures below VYNPS's LBPCT of 1960°F. Therefore, in the event of a LOCA at VYNPS, if peak cladding temperatures reached temperatures between approximately 1832°F and 1960°F—there is experimental data that indicates—the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a maximum rate of 27°F/sec. Within a period of approximately 60 seconds peak cladding temperatures would increase to above 3000°F; the melting point of Zircaloy is approximately 3308°F.[49]

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[47] 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, p. 12.

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

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

2. Multi-Rod Severe Fuel Damage Experiments in which the Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at Temperatures of 2060°F or Lower

a. The Autocatalytic Zircaloy-Steam Reaction in the BWR CORA Experiments: CORA-16, CORA-17, and CORA-18

It is significant that "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states that in the CORA-16, CORA-17, and CORA-18 "[t]he temperature escalation due to the exothermal zircaloy(Zry)-steam reaction started at about 1100°C [(2012°F)], leading the bundles to maximum temperatures of approximately 2000°C;"[50] and states that "[t]he transient of a SFD-type accident is initiated by a slow temperature rise in the order of 0.5 [to] 1.0 K/sec., followed by a rapid temperature escalation (several tens of degrees Kelvin per second) due to the exothermal heat produced by the Zry cladding oxidation in steam environment."[51]

Regarding the BWR CORA experiments the abstract of "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The CORA experiments carried out in an out-of-pile facility at the Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany, are part of the "Severe Fuel Damage" (SFD) program.
>
> The experimental program was to provide information on the failure mechanisms of Light Water Reactor (LWR) fuel elements in a temperature range from 1200°C to 2000°C and in a few cases up to 2400°C.
>
> In the CORA experiments two different bundle configurations were tested: PWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor) bundles. The BWR-type bundles consisted of 18 fuel rod simulators (heated and unheated rods), an absorber blade of steel containing eleven absorber rods filled with boron carbide powder. The larger bundle CORA-18 contained the same number of absorber rods but was made up of 48 fuel rod simulators. All BWR bundles were surrounded by a zircaloy shroud and the absorber blades by a channel box wall on each

[50] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, p. i.

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

side, also made of zircaloy. The test bundles were subjected to temperature transients of a slow heatup rate in a steam environment. Thus, an accident sequence was simulated, which may develop from a small-break loss-of-coolant accident of a LWR.

The transient phases of the tests were initiated with a temperature ramp rate of 1 K/sec. The temperature escalation due to the exothermal zircaloy(Zry)-steam reaction started at about 1100°C, leading the bundles to maximum temperatures of approximately 2000°C.[52]

Regarding the percentage of additional energy from the exothermic zirconium-steam reaction during the escalation phase of the CORA tests, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> In the escalation phase; *i.e.*, starting from about 1100°C, the slow temperature rise was followed by a rapid increase caused by the energy from the exothermal zirconium-steam reaction which becomes significant at the temperature mentioned and in addition—the electric power input. The contribution of the exothermal heat to the total energy; *i.e.*, electrical and chemical power, is generally between 30 and 50%. For CORA-16, CORA-17, and CORA-18 the chemical reaction contributes to 48, 44, and 33 %, respectively.[53]

So the percentage of oxidation energy from the exothermic zirconium-steam reaction was between 33 and 48% of the total energy input during the escalation phase of the CORA-16, CORA-17, and CORA-18 experiments. And the cladding temperature escalation (tens of degrees Fahrenheit per second) from the exothermal Zircaloy-steam reaction commenced at approximately 2012°F, in the CORA-16, CORA-17, and CORA-18 experiments.

[52] *Id.*, p. i.
[53] *Id.*, p. 5.

Regarding the temperature excursion in the CORA-18 experiment (and two PWR CORA experiments), the document, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," which is partly a report on the 1990 CORA Workshop at KfK GmbH, Karlsruhe, FRG, October 1-4, 1990,[54] 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.[55]

And regarding "experiment-specific analytical modeling at [ORNL] for CORA-16,"[56] "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division" states:

> The predicted and observed cladding thermal response are in excellent agreement *until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted.*
>
> ... Dr. Haste pointed out that he is chairing a committee (for the OECD) which is preparing a report on the state of the art with respect to Zircaloy oxidation kinetics. He will forward material addressing the low-temperature Zircaloy oxidation problems encountered in the CORA-16 analyses to ORNL [emphasis added].[57]

It is significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory" ("In-Vessel Phenomena—CORA"), states that for the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations."[58]

Regarding the CORA-16 and CORA-17 experiments, "In-Vessel Phenomena—CORA" states:

> Applications of ORNL models specific to the KfK CORA-16 and CORA-17 experiments are discussed and significant findings from the experimental analyses such as the following are presented:

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[54] 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.
[55] *Id.*, p. 2.
[56] *Id.*, p. 3.
[57] *Id.*
[58] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory," Presented at Cooperative Severe Accident Research Program, Semiannual Review Meeting, Bethesda, Maryland, May 6-10, 1991.

1) applicability of available Zircaloy oxidation kinetics correlations,

2) influence of cladding strain on Zircaloy oxidation...[59]

The Baker-Just and Cathcart-Pawel correlations were among the "available Zircaloy oxidation kinetics correlations"—in 1991—when "In-Vessel Phenomena—CORA" was presented. So according to "In-Vessel Phenomena—CORA," the Baker-Just and Cathcart-Pawel correlations did not accurately predict the cladding oxidation of the CORA-16 experiment. Furthermore, in the CORA-16 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) [(1652-2192°F)] oxidation to be underpredicted."[60]

b. The Autocatalytic Zircaloy-Steam Reaction in the PWR CORA Experiments

At least two papers on the PWR CORA experiments state that in some of the CORA experiments there were cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding that commenced at approximately 2012°F.[61]

(The PWR CORA experiments were conducted to study severe accident sequences, with electrically heated bundles of 2-meter long fuel rod simulators, held in place by three spacer grids (two Zircaloy, one Inconel), and surrounded by a shroud. The electric heating was done with tungsten heating elements, installed in the center of annular UO_2 pellets, which, in turn, were sheathed by PWR Zircaloy-4 cladding. The total available heating power was 96kW, which had the capability of being distributed among three bundles of the fuel rod simulators. There were also unheated rods, filled

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[59] *Id.*

[60] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.

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

with solid UO_2 pellets to correspond to LWR fuel rods.[62] In the CORA experiments the initial heatup rate of the fuel rod simulators was approximately 1 K /sec., in the presence of steam.)

First, regarding cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding, the abstract of "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states:

> The transient phases of the tests were initiated with a temperature ramp rate of 1 K/sec. *The temperature escalation due to the exothermal zircaloy (Zry)-steam reaction started at about 1100°C*, leading the bundles to maximum temperatures of approximately 2000°C [emphasis added].[63]

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

> The transient of a SFD-type accident is initiated by a slow temperature rise in the order of 0.5 [to] 1.0 K/sec., followed by a *rapid temperature escalation (several tens of degrees Kelvin per second)* due to the exothermal heat produced by the cladding oxidation in steam environment [emphasis added].[64]

Second, regarding cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding, the abstract of "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" states:

> In the CORA experiments two different bundle configurations are tested: PWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor) bundles. The PWR-type assemblies usually consist of 25 rods with 16 electrically heated fuel rod simulators and nine unheated rods (full-pellet and absorber rods). Bundle CORA-13, a PWR-type assembly, contained two Ag/In/Cd-steel absorber rods. The test bundle was subjected to temperature transients of a slow heatup rate in a steam environment; *i.e.*, the transient phase of the test was initiated with a temperature ramp rate of

[62] 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. 77.

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

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

> 1 K/sec. *The temperature escalation due to the exothermal zircaloy(Zry}-steam reaction started at about 1100°C at an elevation of 850 mm (1000 sec. after [the] onset of the transient), leading to a temperature plateau of 1850°C* and after initiation of quenching to maximum temperatures of approximately 2000°C to 2300°C. CORA-13 was terminated by quenching with water from the bottom with a flooding rate of 1 cm/sec.
>
> Rod destruction started with the failure of the absorber rod cladding at about 1200°C; *i.e.*, about 250 K below the melting regime of steel. Penetration of the steel cladding was presumably caused by a eutectic interaction between steel and the zircaloy guide tube. As a consequence, the absorber-steel-zircaloy melt relocated radially outward and axially downward. Besides this melt relocation the test bundle experienced severe oxidation and partial melting of the cladding, fuel dissolution by Zry/UO_2 interaction, complete Inconel grid spacer destruction, and relocation of melts and fragments to lower elevations in the bundle. An extended flow blockage has formed at the axial midplane.
>
> Quenching of the hot test bundle by water resulted, besides additional fragmentation of fuel rods and shroud, in an additional temperature increase in the upper bundle region. Coinciding with the temperature response an additional hydrogen buildup was detected. During the flooding phase 48% of the total hydrogen [was] generated [emphasis added].[65]

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

> The temperature rise shows the same general features already found in earlier tests. With the increase of the electrical power input, first the temperature rises proportional to the power. *Having reached about 1000°C, the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation.* The escalation starts at [the] 950 mm and 750 mm elevation. For the outer fuel rod simulator [number] 3.7 the escalation is delayed at 750 mm by about 150 sec. A possible reason for this delay could be the heat losses due to the window at 790 mm adjacent to this rod. The escalation at the 550 mm elevation follows 200 sec. later. The escalation at 1150 mm develops before that at the 350 mm elevation [emphasis added].[66]

So "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" and "Results of SFD

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[65] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," KfK 5054, Abstract, p. v.
[66] *Id.*, p. 12.

Experiment CORA-13 (OECD International Standard Problem 31)" both state that temperature escalations due to the exothermic Zircaloy-steam reaction began at approximately 1100°C (2012°F). "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" also states that "having reached about 1000°C [(1832°F)], the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation."[67] Additionally, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" states that the "rapid temperature escalation[s were] several tens of degrees Kelvin per second...due to the exothermal heat produced by the cladding oxidation in [a] steam environment."[68]

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

> In the escalation phase; *i.e.*, starting from about 1100°C the slow temperature rise is followed by a rapid increase caused by the increased electric power input *and the additional energy from the exothermal zirconium-steam reaction. The contribution of this exothermal heat to the total energy input is generally between 30 and 40%* [emphasis added].[69]

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

> Based on the accumulated H_2 productions of tests CORA-15, CORA-9, and CORA-7 the oxidation energy is determined. Its percentage amounts to 30 - 45% of the total energy input (electric supply plus exothermal energy)...[70]

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[67] *Id.*

[68] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, p. 1.

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

[70] *Id.*, p. 7.

So the percentage of oxidation energy from the exothermic zirconium-steam reaction was generally between 30 and 40%, and in some cases was as high as 45%, of the total energy input during the escalation phase of the CORA tests.

A third paper on the PWR CORA experiments states that in the CORA experiments there where cladding temperature excursions due to the autocatalytic oxidation reaction of Zircaloy cladding that commenced at temperatures between approximately 2012°F and 2192°F.

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

> The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation. With the good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15 K/sec.[, after an initial heatup rate of about 1 K /sec.] The maximum temperatures attained are about 2000°C; the oxide layers formed and the consumption of the available steam set limits on the temperature escalation due to rate-controlled diffusion processes. The temperature escalation starts in the hotter upper half of the bundle and the oxidation front subsequently migrates from there both upwards and downwards."[71]

"CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures" also states that temperature escalations "continued even after shut-off of the electric power, as long as steam was available."[72]

It is also significant that the CORA experiments demonstrated that "[t]he critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; *i.e.*, on bundle insulation."[73] So with good fuel assembly insulation—like what the core of a nuclear power plant has—cladding temperature escalation, due to the exothermic Zircaloy-steam reaction, commences when cladding temperatures reach between approximately 1100°C and 1200°C (2012°F and 2192°F), and cladding temperatures start

[71] 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.
[72] *Id.*, p. 87.
[73] *Id.*, p. 83.

increasing at a maximum rate of 15°C/sec. (27°F/sec.). There is also experimental data that indicates such temperature escalations can commence when the cladding reaches temperatures as low as approximately 1000°C (1832°F).

c. The Autocatalytic Zircaloy-Steam Reaction in the LOFT LP-FP-2 Experiment

It is significant that "[t]he first recorded and qualified rapid temperature rise [in the LOFT LP-FP-2 experiment] associated with the rapid reaction between Zircaloy and water occurred at about...1400 K (2060°F) on a guide tube at the 0.69-m (27-in.) elevation."[74]

The LOFT LP-FP-2 experiment was conducted in the Loss-of-Fluid Test ("LOFT") facility at Idaho National Engineering Laboratory, on July 9, 1985. The LOFT facility was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[75] The LOFT LP-FP-2 experiment—the second and final fission product test conducted at the LOFT facility—had an 11 by 11 test assembly, comprised of 100 pre-pressurized Zircaloy 1.67 meter fuel rods; it was the central assembly, isolated from the remainder of the core—a total of nine assemblies—by an insulated shroud. The LOFT LP-FP-2 experiment combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[76]

[74] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.
[75] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 13.
[76] *Id.*

The LOFT LP-FP-2 experiment had an initial heatup rate of ~1 K/sec.[77] It is significant that "heatup rates [of 1 K/s or greater] are typical of severe accidents initiated from full power."[78] And regarding the significance of the initial heatup rate in the LOFT LP-FP-2 experiment, "Review of Experimental Results on LWR Core Melt Progression" states:

> The higher initial heating rate [of 1 K/sec.] in the LOFT [LP-]FP-2 experiment is related to the higher fraction of decay heat available following rapid blowdown of the coolant inventory in the reactor vessel. This higher heating rate leads to smaller oxide thickness on the cladding for a particular temperature and, therefore, more rapid oxidation. The increase in heating rate at the higher temperatures is the result of rapid oxidation of zircaloy and the strongly exothermic nature of the reaction (6.45 kJ/g Zr oxidized).[79]

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

> Data from [the LOFT LP-FP-2] experiment provide a wealth of information on severe accident phenomenology. The results provide important data on early phase in-vessel behavior relevant to core melt progression, hydrogen generation, fission product behavior, the composition of melts that might participate in core-concrete interactions, and the effects of reflood on a severely damaged core. The experiment also provides unique data among severe fuel damage tests in that actual fission-product decay heating of the core was used.
>
> The experiment was particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident.[80]

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[77] *Id.*

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

[79] R. R. Hobbins, D. A. Petti, D. J. Osetek, and D. L. Hagrman, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "Review of Experimental Results on LWR Core Melt Progression," in NRC "Proceedings of the Eighteenth Water Reactor Safety Information Meeting," NUREG/CP-0114, Vol. 2, 1990, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042250131, p. 7.

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

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

> The first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K on a guide tube at the 0.69-m (27-in.) elevation. This temperature is shown in Figure 3.7. A cladding thermocouple at the same elevation (see Figure 3.7) reacted earlier, but was judged to have failed after 1310 [seconds], prior to the rapid temperature increase. Note that, due to the limited number of measured cladding temperature locations, the precise location of the initiation of [the] metal-water reaction on any given fuel rod or guide tube is not likely to coincide with the location of a thermocouple. Thus, the temperature rises are probably associated with precursory heating as the metal-water reaction propagates away from the initiation point. Care must be taken in determining the temperature at which the metal-water reaction initiates, since the precursory heating can occur at a much lower temperature. It can be concluded from examination of the recorded temperatures that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400 K (2060°F).[81, 82]

Additionally, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" also states that the hottest measured cladding temperature reached 2100 K (3320°F) by 1504 ± 1 seconds;[83] and states that it was difficult to determine the PCT reached during the entire experiment—because of thermocouple failure—but that the PCT exceeded 2400 K (3860°F).[84]

Therefore, after the onset of rapid oxidation—after a heating rate of ~1 K/sec.[85]—peak cladding temperatures increased from approximately 1400 K (2060°F) to 2100 K (3320°F) within a range of approximately 75 seconds; in other words, after the onset of rapid oxidation, cladding temperatures increased at an average rate of approximately

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

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

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

[84] *Id.*, p. 33.

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

10 K/sec. (18°F/sec.). In general agreement with this postulation, "Review of Experimental Results on LWR Core Melt Progression" states that "[i]n the LOFT [LP-]FP-2 experiment, which was driven by decay heat, the heating rate started out at about 1 K/sec. and increased to about 10-20 K/sec. above 1500 K [(2240°F)]."[86]

It is significant that "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[87] So at the point when peak cladding temperatures increased at a rate of greater than 10 K/sec. during the LOFT LP-FP-2 experiment, an autocatalytic oxidation reaction commenced; and that occurred when the temperature of a Zircaloy fuel rod or guide tube reached approximately 1400 K (2060°F), or when cladding temperatures reached approximately 1500 K (2240°F).

In a different account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Degraded Core Quench: A Status Report" states that "[t]he initial heating rate in the central assembly was ~1 K/sec. with an onset to rapid oxidation at a temperature near 1500 K [(2240°F)]."[88] In a similar account, as already mentioned, "Review of Experimental Results on LWR Core Melt Progression" states that the initial heatup rate was 1 K/sec., and that the heatup rate increased to approximately 10-20 K/sec. at a cladding temperature greater than 1500 K (2240°F).[89]

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

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

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

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

And offering yet another account of the cladding-temperature excursion during the LOFT LP-FP-2 experiment, "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2" states that in the LOFT LP-FP-2 experiment that the metal-water reaction was initiated at 1450.0 ± 30 sec. after the beginning of the experiment and that at 1500 ± 1 sec, after the beginning of the experiment, the maximum cladding temperatures reached 2100 K;[90] elsewhere the same paper states that the "[m]etal-water reaction began at about 1450 seconds and [that the] hottest measured cladding temperature reached 2100 K [(3320°F)] by 1504 seconds."[91]

As quoted above, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" states that "[t]he first recorded and qualified rapid-temperature rise associated with the rapid reaction between Zircaloy and water occurred at about 1430 [seconds] and 1400 K..."[92] So it is reasonable to conclude that at some point when peak cladding temperatures were approximately 1400 K (2240°F) or 1500 K (2240°F), cladding temperatures began increasing at a rate of greater than 10 K/sec., signaling the onset of an autocatalytic oxidation reaction.

Regarding the expertise of the test design of the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> The last experiment of the OECD LOFT Project LP-FP-2, conducted on [July] 9, 1985, was a severe core damage experiment. It simulated a LOCA caused by a pipe break in the Low Pressure Injection System (LPIS) of a four-loop PWR as described in "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2."[93] The central fuel assembly of the LOFT core was specially designed and fabricated for this experiment and included more than 60 thermocouples for temperature measurements. ...

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.

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

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

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

[93] M. L. Carboneau, V. T. Berta, and S. M. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989.

> Experience available in EG&G Idaho from TMI-2 analyses and from the PBF severe fuel damage scoping test conducted in October 1982 were utilized in the design, conduction and analyses of this experiment. LP-FP-2 costs [were] $25 million out of [the] $100 million [spent] for the whole OECD LOFT project.[94]

And regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> From the analyses of core temperature measurements in [the LOFT] LP-FP-2 [experiment], the rapid increase in temperature shown in fig 14.[95] was a result of the oxidation of zircaloy which became rapid at temperatures in excess of 1400 K. Further examination of such high temperatures measured by thermocouples gave rise to the detection of a cable shunting effect which is defined in "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2,"[96] as the formation of a new thermocouple junction on the thermocouple cable due to exposure of the cable to high temperature. Experiments were designed and conducted by EG&G Idaho to examine the cable shunting effect. The results of these experiments indicate that the cladding temperature data in LP-FP-2 contain deviations from true temperature due to cable shunting after 1644 K is reached. This temperature is within the range when rapid metal-water reaction occurs. An example of such temperature deviation due to cable shunting is shown in fig. 15.[97, 98]

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94 A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," GRS-Garching, Proceedings of the OECD (NEA) CSNI Specialist Meeting on Instrumentation to Manage Severe Accidents, Held at Cologne, F.R.G. March 16-17, 1992, p. 133.

95 See Appendix E Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation.

96 M. L. Carboneau, V. T. Berta, and S. M. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989.

97 See Appendix E Fig. 15 Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM.

98 A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 135.

Additionally, regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> More phenomena were detected from the analyses of the recorded behavior of the 60 thermocouples in the CFM together with other thermocouples and measuring systems in the LOFT nuclear reactor.
>
> After the first indication of [the] metal-water reaction at 1430 [seconds] several instruments indicated a common event at 1500 [seconds]. These instruments included gross gamma monitor, momentum flux meter in the downcomer, upper tie plate and guide tube thermocouples. [According to "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2,"[99] t]his event is believed to be the rupture of the control rod cladding.[100]

And regarding the durability of pressure sensors, thermocouples, and radiation monitors in the LOFT-LP-FP-2 experiment and TMI-2 accident, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2" states:

> Both in TMI-2 and [LOFT] LP-FP-2 only [a] few types of sensors were able to withstand the consequences of severe accidents and were able to deliver information for post-accident analysis. These were pressure sensors, thermocouples, and radiation monitors. Advanced instrumentation technology have proven to be able to utilize these three types of sensors in redundant and diverse instrumentation of Light Water Reactors (LWR) to manage severe accidents.[101]

It is significant that "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" state that the temperature excursion in the LOFT LP-FP-2 experiment, as a result of the autocatalytic oxidation reaction of Zircaloy cladding, commenced at approximately 1400 K (2060°F)—well below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

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[99] M. L. Carboneau, V. T. Berta, and S. M. Modro, "Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2," OECD LOFT-T-3806, OECD, June 1989.
[100] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 136.
[101] *Id.*, p. 147.

3. Multi-Rod Severe Fuel Damage Experiments and One Multi-Rod Thermal Hydraulic Experiment in which the Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at Temperatures of Approximately 2192°F (Approximately at the 10 C.F.R. § 50.46(b)(1) PCT Limit of 2200°F) and One Experiment in which Autocatalytic Oxidation Commenced at a Temperature of 2275°F or Lower

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].[102]

Furthermore, Figure 1[103] 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),[104] and Figure 13[105] 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.[106]

a. The Autocatalytic Zircaloy-Steam Reaction in the BWR FLECHT Zr2K Test

It is significant that during the AEC's ECCS rulemaking hearing, conducted in the early '70s, that Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[107] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[108] conducted with a

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[102] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 195.
[103] See Appendix F Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.
[104] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 196.
[105] See Appendix F Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core.
[106] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 205.
[107] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists.
[108] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-17; this paper cites Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated

Zircaloy assembly. Among other things, "CNI claimed that the [Zr2K] test showed that near 'thermal runaway' conditions resulted from [metal-water] reactions, in spite of the 'failed' heater rods. They compared test results for SS2N [(conducted with a stainless steel assembly)] with Zr2K, showing satisfactory correlation during approximately the first five minutes of the test with substantial deviations (Zr2K temperatures greater than SS2N) during the subsequent periods of substantial heater failures."[109]

(The BWR FLECHT Zr2K test was a thermal hydraulic experiment; however, in some respects it resembled a severe fuel damage experiment. In the BWR FLECHT Zr2K test the Zircaloy assembly incurred autocatalytic oxidation.)

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

> The first complaint [of the BWR-FLECHT tests] was that although all BWR fuel rods are manufactured of a zirconium...alloy, Zircaloy, only 5 of the 143 FLECHT tests utilized [Zircaloy] rods. The remaining 138 tests were conducted with stainless steel...rods. *Since...[Zircaloy] reacts exothermically with water at elevated temperatures, contributing additional energy to that of the decaying fission products, the application of water to the core has the potential of increasing the heat input to the fuel rods rather than cooling them, as desired.* The small number of [Zircaloy] tests in comparison with the total test program was seriously faulted by the CNI [emphasis added].[110]

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

> The [stainless steel] rods were apparently chosen primarily for their durability. They could be used repeatedly in testing (for 30 or 40 individual tests) without substantial changes in response over the series. ...
>
> On the other hand, *as a result of metal-water reactions, [Zircaloy] rods could be used only once* and then had to be subjected to a destructive post-mortem examination after the test [emphasis added].[111]

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

[109] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-18.

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

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

General Electric ("GE") argued that the exothermic metal-water reactions were insignificant in the thermal response of the Zircaloy heater rods. Regarding this issue, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> Attempts by GE to show that [metal-water] reactions were insignificant in the thermal response of the rods were not overly convincing since they did not evaluate actual dynamic heat rate inputs but depended instead upon arbitrarily time averaged heat inputs over arbitrary time intervals...[112] Gross estimates were made of the total energy contributed to the thermal transient through the [metal-water] reaction of 1/4 B/inch of cladding length (based upon the maximum observed depth of ZrO_2 penetration for the Zr2K experiment of 1.8 mils). This was compared with a design total delivered decay power to the center of the maximum peaked rod over the 24 minute spray cooling transient of 29.7 B/inch (14.5 B/inch over the first 10 minutes). Thus, GE inferred the total [metal-water] reaction to be 5-10 percent of the decay energy depending upon which of the two time periods was used in the estimation. They acknowledge that the rate of [metal-water reaction] energy addition is more significant than the comparisons with [the] total energy shown above, but state that rate information cannot be obtained from the Zr2K data. Irrespective of the validity of this observation, it seems that comparisons with rod input energy increments taken over 10 to 24 minute intervals are too insensitive to be adequate indications of the significance of the [metal-water reaction] energy contribution. No feeling of confidence is gained that [metal-water] reactions were unimportant as a result of this GE analysis. However, the case for [metal-water reaction] induced thermal runaway in the Zr2K test is equally weak.[113]

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

[112] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, Appendix A.

[113] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-18, A8-19.

rate of 15 K/sec."[114] Furthermore, during the escalation phase of BWR CORA experiments (CORA-16, CORA-17, and CORA-18), the percentage of oxidation energy from the exothermic Zircaloy-water reaction was 48, 44, and 33 %, respectively, of the total energy input.[115] And during the escalation phase of the PWR CORA experiments, the percentage of oxidation energy from the exothermic Zircaloy-water reaction was generally between 30 and 40%, and in some cases was as high as 45%,[116] of the total energy input.[117]

So during the Zr2K test it is highly probable that—like the CORA experiments—the energy from the exothermic Zircaloy-water reaction was between 30 and 48% of the total energy input, not between 5 and 10% as GE estimated. (It is noteworthy that GE "acknowledge[d] that the rate of [metal-water reaction] energy addition [was] more significant than the[ir] comparisons with [the] total energy...but state[d] that rate information [could not] be obtained from the Zr2K data."[118])

Second, when taking into account data from the CORA experiments and other severe fuel damage experiments, it is highly probable that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI...implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[119, 120] It is significant that "in the CORA

[114] P. Hofmann, S. Hagen, G. Schanz, G. Schumacher, L. Sepold, Idaho National Engineering Laboratory, EG&G Idaho, Inc., "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," in NRC "Proceedings of the Nineteenth Water Reactor Safety Information Meeting," p. 83.
[115] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, p. 5.
[116] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material in Zry/ UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," FZKA 7448, 2008, p. 7.
[117] *Id.*, p. 5.
[118] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-19.
[119] Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, p. 5.63.

test facility, [cladding] temperature escalation start[ed] between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a maximum heating rate of 15 K/sec:"[121] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[122]

Furthermore, the graphs of "Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies"[123] and "Analysis of Zr2K Thermal Response"[124] depict thermocouple measurements taken during the Zr2K test that resemble thermocouple measurements taken during severe fuel damage experiments: the graphs depict temperature excursions that began when cladding temperatures reached between approximately 2100 and 2200°F. The graphs depict cladding-temperature values at separate points in approximately 20-second intervals; in some cases the temperature increases by several hundred degrees Fahrenheit within approximately 20 seconds, indicating the onset of temperature excursions, at rates greater than 10 K/sec (see Appendix G Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies and Figure A8.10 Analysis of Zr2K Thermal Response).

It is significant that GE concluded that the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test were not valid. GE stated

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

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

[122] 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.

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

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

"that the 'erratic thermocouple outputs do not represent actual cladding temperatures, but are the result of equipment malfunctions'[125] associated with the Zr2K test."[126] However, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies it is highly probable that GE's claim that the thermocouple measurements did not represent actual cladding temperatures is erroneous; after all, the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test resemble thermocouple measurements of cladding-temperature excursions taken during severe fuel damage experiments.

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

> One of the more difficult aspects of evaluation of Zr2K test results is associated with the fundamental data for the tests, the recorded thermocouple...responses. *GE has been very liberal with their accreditation of observed [thermocouple] responses as erratic.* However, several proffered examples of erratic response seem to show well defined inter-rod correlations. Under such circumstances, "unexplained" might be a better description for the observed [thermocouple] behavior than "erratic" [emphasis added].[127]

Discussing the "well defined inter-rod correlations"[128] that occurred during "the extreme temperature excursion,"[129] "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

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

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

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

The relationships described above seem to indicate a systematic correlation between the electrical anomalies of the "failed" rods and temperature extremes for the bundle [emphasis added].[130]

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

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

Based upon analysis of the material presented, it appears unquestionable that the [thermocouple] response was badly affected by short circuits and equipment malfunction. The net result is that it is not possible to certify that [metal-water] reactions were insignificant in the measured thermal transient, but the case for near "thermal runaway" proposed by the CNI is

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

> also unconvincing. It is probable that most of the dramatic [thermocouple] slope changes, as well as several of the other [thermocouple] aberrations associated with the test, were short-circuit induced rather than [metal-water] reactions. *However, more results seem to be systematically correlatable between rods [than] the GE test analysis is willing to concede. This leads to uncertainty over the proper interpretation of [the] results. A more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable* [emphasis added].[131]

Indeed, "a more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable."[132] However, when taking into account data from the CORA experiments and other severe fuel damage experiments conducted with Zircaloy assemblies more than a decade after the Zr2K test, it is clear that GE's claim that the metal-water reactions were insignificant during the Zr2K test is erroneous and that CNI's claim the Zr2K test nearly incurred a "thermal runaway" oxidation reaction, an autocatalytic oxidation reaction, is correct. In fact, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "CNI...implied that the test was on the verge of 'thermal runaway' and was saved only as a 'consequence of the extensive heater failures that occurred.' "[133, 134]

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

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

reached between approximately 2100 and 2200°F—incurred an autocatalytic oxidation reaction.

Furthermore, it is significant that in the AEC's ECCS rulemaking hearing, Dr. Roger Griebe, the Aerojet project engineer for BWR-FLECHT, testified that "there is *no* convincing proof available from [Zr2K] test data to demonstrate that [a] near-thermal runaway [condition] definitely did not exist [in the Zr2K test] [emphasis not added].[135]

(In "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," the BWR-FLECHT Zr2K test is termed "Test ZR-2;" therefore, in the passages below the BWR-FLECHT Zr2K test will be termed "Test ZR-2.")

Regarding Dr. Roger Griebe's testimony, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> CNI's direct testimony concluded that a near thermal runaway condition existed in Test ZR-2.[136] It is of compelling importance that Roger Griebe, the [Aerojet] project engineer for BWR-FLECHT, stated a similar interpretation of this test, which they submitted to [General Electric ("GE")], and Griebe testified, there is *no* convincing proof available from ZR-2 test data to demonstrate that this near-thermal runaway definitely did not exist [emphasis not added].[137, 138]

And regarding Aerojet internal memoranda that provide commentary on the BWR-FLECHT program consistent with that presented by CNI, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] internal memoranda provide commentary on the BWR-FLECHT program quite consistent with that presented by CNI. Thus, for example, J. W. McConnell (who will be co-author, with Dr. Griebe, of the as-yet-unpublished BWR-FLECHT final report from [Aerojet]) wrote:
>
> "There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the GE determination to

[135] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, Union of Concerned Scientists, 1974, p. 5.11.

[136] Daniel F. Ford and Henry. W. Kendall, Union of Concerned Scientists, "An Evaluation of Nuclear Reactor Safety," Volume I, Direct Testimony prepared in behalf of the Consolidated National Intervenors, USAEC Docket RM-50-l, 23 March 1972, p. 5.63.

[137] Official Transcript of the AEC's Emergency Core Cooling Systems Rulemaking Hearing, pp. 7138-7139.

[138] Daniel F. Ford and Henry. W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 5.11.

> prove out their ECC systems. Their role in this program can only be described as a conflict of interest as is the Westinghouse portion of PWR-FLECHT. Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ... A combination of poor data acquisition and transmission, faulty test approaches (probably caused by crude test facilities) and the marginal nature of these tests has produced a large amount of questionable data. It appears probable that the results of these tests can be interpreted. *But the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven.* From a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective [emphasis added]."[139]

Additionally, regarding Dr. Griebe's review of the data presented by GE regarding the maximum cladding history of ZR-2, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> It is important to note that GE's interpretation of Test ZR-2 is based on a bundle maximum cladding temperature curve that CNI contended in its direct testimony constituted false reporting of the test data. The basis that GE asserts for the correctness of its reported maximum temperature curve are the thermocouple data available from Sanborn strip recorders that were used by GE. It is important to note that the GE report published on Test ZR-2 (Exhibit 133) does not present any reporting of the strip data. Moreover, the Board turned down CNI's request for discovery that the data be made available. Finally, Dr. Roger Griebe, who had the Sanborn tapes available, was addressed an interrogatory by CNI concerning what the test data established to be the true maximum cladding temperature curve for Test ZR-2. Dr. Griebe's answer, which presented detailed documentation from the Sanborn strip data, completely confirmed CNI's position that the maximum cladding temperature curve used in GE analysis of ZR-2 is false and that the much more severe temperature history from Exhibit 125 is, in fact, the correct data for Test ZR-2, as CNI had asserted.
>
> Dr. Griebe's review of the data presented by GE regarding the maximum cladding history of ZR-2 provides quite precise technical support for his testimony earlier that GE "tremendously slanted" BWR-FLECHT data "towards the lower temperatures and towards the interpretation GE obviously presented in their report" (Tr. 7127). ...
>
> CNI's interpretation of both the correct maximum cladding temperature curve and their more reasonable assessment of the test was concurred in by Dr. Griebe. Yet the Regulatory Staff provides *no commentary*

[139] *Id.*

> *whatsoever on either the issue of the correct temperature curve for ZR-2 or the issue of the existence of a near thermal runaway condition* [emphasis added].[140]

Indeed, it is unfortunate that the AEC Regulatory Staff did not provide commentary "on either the issue of the correct temperature curve for ZR-2 or the issue of the existence of a near thermal runaway condition [in the ZR-2 test]."[141]

Regarding the prospect of planning and conducting a new BWR-FLECHT program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> No recovery from the defects in the BWR-FLECHT Program are possible without a new program of greater scope being planned and carried out, like a new PWR-FLECHT Program, carried out in a way essentially free of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[142]

Petitioner, would add that such a new BWR-FLECHT program would have to be conduced with Zircaloy fuel assemblies. It would also be necessary that the PCTs of such tests exceeded those of the PWR Thermal-Hydraulic Experiment 1 ("TH-1") tests, conducted at Chalk River in the early '80s, where the test planners—"for safety purposes"—did not want the maximum PCTs of the TH-1 tests to exceed 1900°F[143]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

b. The Autocatalytic Zircaloy-Steam Reaction in the NRU Reactor Full-Length High-Temperature 1 Test

The first full-length high-temperature severe fuel damage ("FLHT-1") test was conducted at the National Research Universal ("NRU") reactor at Chalk River, Ontario, Canada, by Pacific Northwest Laboratory ("PNL"), "to evaluate degraded core behavior and the progression of light water reactor ("LWR") fuel damage resulting from [a] loss-

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[140] *Id.*, pp. 5.12, 5.14.
[141] *Id.*
[142] *Id.*, p. 5.41.
[143] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024, p. 3-3.

of-coolant accident."[144] The FLHT-1 test was part of the PNL Coolant Boilaway and Damage Progression program. The FLHT-1 test used an assembly comprised of 12 fuel rods that were 3.7-meters in length.[145] During the test the nominal fuel rod linear power was 0.524 kW/m (0.160 kW/ft.) and the nominal bundle power was 23 kW (22 Btu/sec.).[146]

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

> This report presents a summary of the FLHT-1 test operations. The test was performed on March 2, 1985. In the report, the actual test operations and data are compared to the planned operations and predicted test behavior. ... The test plan called for a gradual temperature increase to approximately 2150 K (3400°F). However, during the test, the fuel cladding began to rapidly oxidize, causing local bundle temperatures to rapidly increase from about 1700 K (2600°F) to 2275 K (3635°F), at which time the test was terminated. Much of the Zircaloy cladding in the central region (axially) of the 3.7-m-long (12-ft) fuel bundle was heavily oxidized, and some Zircaloy cladding melted.[147]

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

"FLHT-1 Test Report" states that "[t]he reactor power was decreased at approximately 17:11:07, 85 seconds after the start of the [cladding temperature] excursion;"[148] *i.e.*, the cladding temperature excursion began at 17:09:42. However, "FLHT-1 Test Report" also states that the cladding temperature excursion began 18

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[144] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, p. v.
[145] *Id.*, p. 3.1.
[146] *Id.*, pp. 4.1-4.2.
[147] *Id.*, p. v.
[148] *Id.*, p. 4.6.

seconds latter at 17:10:00—when the cladding temperature was 1700 K.[149] The difference of 18 seconds is highly significant, because it means that the cladding temperatures were much lower than 1700 K when the temperature excursion actually began.

"Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods and the Relevancy to LWR Severe Accident Melt Progression Safety Issues" states that during the FLHT-1, -2, -4, and -5 tests that "[t]he heatup phase of the tests culminated near 1700 K in a rapid [cladding] temperature escalation, [greater than] 10 K/sec., signaling the onset of an autocatalytic oxidation reaction."[150] So if peak cladding temperatures increased at a rate of greater than 10 K/sec. during the FLHT-1 test, it is highly probable that 18 seconds before 17:10:00—when the peak cladding temperature was 1700 K (2600°F)—the peak cladding temperature was approximately 1520 K (~2275°F) or lower.

This is reasonable to postulate; after all, another severe fuel damage experiment—LOFT LP-FP-2—demonstrated "that the oxidation of Zircaloy by steam becomes rapid at temperatures in excess of 1400 K (2060°F)."[151] According to a different account, in the LOFT LP-FP-2 experiment, the onset of rapid oxidation occurred at approximately 1500 K (2240°F).[152] Additionally, "Degraded Core Quench: Summary of Progress 1996-1999," states that autocatalytic (runaway) oxidation of Zircaloy cladding by steam occurs at temperatures of 1050°C to 1100°C (1922°F to 2012°F) or higher.[153]

Furthermore, although the graphs of "Typical Cladding Temperature Behavior"[154] and "Pseudo Sensor Readings for Fuel Peak Temperature Region"[155, 156] are not large

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[149] *Id.*, p. 4.11

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

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

[152] 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.

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

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

enough to clearly delineate what the temperature values were at given times during the FLHT-1 test, the graphs' cladding-temperature values are consistent with the postulation that the temperature excursion began at a temperature far lower than 1700 K, at a temperature closer to 1520 K (see Appendix H Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region). The slopes of the lines of the cladding-temperature value plots in the graphs become nearly vertical, when the cladding-temperature values reach approximately 1520 K, indicating the onset of the temperature excursion, at a rate of 10 K/sec. or greater.

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

> Typical cladding temperature behavior at one position in the assembly during the test is shown in Figure 4.1. At about 60 to 70 min. along the abscissa, a temperature increase [commenced] when the [bundle coolant] flow rate was about 9 kg/hr. (20 lb/hr.). The [cladding] temperature increased until about 95 min. and [reached] 1450 K (2150°F), at which time the bundle coolant [flow] rate was increased to 18 kg/hr. (40 lb/hr.) to stabilize the temperature. However, the [cladding] temperature rapidly dropped to about 1060 K (1450°F). The bundle coolant flow rate was then decreased through a series of steps to a minimum of 9 kg/hr. (20 lb/hr.). This action stopped the temperature decrease and started another temperature rise. When the temperature reached about 1475 K (2200°F), the bundle coolant flow [rate] was again increased to stop the temperature ramp. This led to a stabilized condition. The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly until the test director requested that the reactor power be reduced to zero power.[157]

First, it is obvious from the above description and from Figures 4.1 and 5.4 that when cladding temperatures reached approximately 1475 K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was not achieved. Cladding temperatures continued to rise. This is clearly stated: "The flow was increased in steps

[155] *Id.*, p. 5.3.
[156] Pseudo sensor readings are the averages of the readings of two or more thermocouples.
[157] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," p. 4.6.

and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly..."[158]

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

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

In "Compendium of ECCS Research for Realistic LOCA Analysis," discussing an earlier NRU reactor test, the NRC states that "[t]he MT-6B test...showed that at cladding temperatures of 2200°F (1204°C) the zircaloy oxidation rate was easily controllable by adding more coolant."[159] Furthermore, the test conductors would have thought "the zircaloy oxidation rate was easily controllable" at cladding temperatures far above 2200°F (1477 K): "[t]he [FLHT-1] test plan called for a gradual [cladding] temperature increase [up] to approximately 2150 K (3400°F)."[160]

(It is noteworthy that other reports state that the MT-6B test had a PCT of 1400 K (2060°F)[161] and 1280°C (2336°F) (1553 K).[162] So the MT-6B test may have actually demonstrated that the Zircaloy oxidation rate was easily controllable by adding more

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

coolant at cladding temperatures of either 2060°F (1400 K) or 1280°C (2336°F) (1553 K).)

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

> Once the power is set, the test will be started through its transient operation. *The term transient is somewhat of a misnomer*; operation will consist of a series of preplanned, discrete flow-reduction steps. The size and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate. ...
>
> The bundle [coolant] flow rate will then be decreased in a series of precalculated flow steps... The duration of the time between steps is dictated by the time needed to reach near steady state and also by *the requirement that the Zircaloy-steam reaction be limited.* About 14 steps, each of about 1/2 hr. duration, are expected. *The last flow reduction step will be calculated to give a peak cladding temperature of about 2150 K (3400°F).* ...
>
> The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150 K (3400°F) [emphasis added].[163]

Indeed, the test conductors must have been taken by surprise when they could not control the zircaloy oxidation rate by increasing the coolant flow rate. They realized that there was no way to terminate the cladding-temperature increase—after peak cladding temperatures reached approximately 1475 K (2200°F)—short of reducing the reactor power to zero power, as they did "85 seconds after the start of the [cladding temperature] excursion."[164]

It is important to remember that the events described above occurred within a period of approximately 85 seconds: peak cladding temperatures increased from approximately 1520 K (~2275°F) or lower to approximately 2275 K (3635°F), within approximately 85 seconds. Additionally, as discussed above, in the graphs of "Typical Cladding Temperature Behavior"[165] and "Pseudo Sensor Readings for Fuel Peak

[163] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," pp. 4.3-4.5.
[164] *Id.*, p. 4.6.
[165] *Id.*, p. 4.7.

Temperature Region,"[166] the slopes of the lines of the cladding-temperature value plots of the FLHT-1 test become nearly vertical, after the cladding-temperature values reach approximately 1520 K, indicating that only a short time period passed before temperatures reached approximately 2275 K (3635°F).

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

> The reactor power was decreased at approximately 17:11:07, 85 sec. after the start of the excursion (approximately 131 minutes in Figure 4.1). The reactor reached 10% of the initial power approximately 35 sec. later and reached low neutron level in another 30 sec.
>
> There were two Indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.[167]

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

> A review of the thermocouple data led to the conclusion that the temperatures were not rising after the reactor shutdown. Typical cladding, coolant, and liner temperatures immediately after the reactor shutdown are shown in Figures 4.2, 4.3, and 4.4, starting at 17:12:00. The temperatures shown are somewhat erratic and show noise (probably associated with some thermocouple damage), but the general trend is downward, indicating an effective shutdown.
>
> Additional Indications of an effective test shutdown are shown by the saddle temperature, MMPD [(molten material penetration detector)] response, and bypass coolant power (radial heat loss) after the reactor power shutdown. Typical data from these sources are shown in Figures 4.5 through 4.7. All three of these indicators show steadily decreasing temperatures.[168]

It is also noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis" states that "[i]n the [FLHT-1] test, completed in March 1985, 12 ruptured zircaloy-clad rods were subjected to an autocatalytic temperature excursion. From the

[166] *Id.*, p. 5.3.
[167] *Id.*, pp. 4.6-4.7.
[168] *Id.*, p. 4.7.

measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500-2600°F (1371-1427°C) temperature region."[169]

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

c. The Autocatalytic Zircaloy-Steam Reaction in the PHEBUS B9R Test

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[170]

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

> During the B9R-2 test, an *unexpected strong escalation of the Zr-water reaction occurred* at mid-bundle elevation during the steam injection. Considerable heatup rates of 20 to 30 K/sec. were measured in this zone with steam starved conditions at upper levels. Post Irradiation Examinations (PIE) show cladding failures and considerable deformations (about 70%) [emphasis added].[171]

And offering a different account of the elevation at which the temperature excursion occurred during the PHEBUS B9R-2 test, "Degraded Core Quench: A Status Report" states that the B9R-2 test had "an unexpected high oxidation escalation in the upper bundle zone (20 to 30 K/sec.)"[172] "Degraded Core Quench: A Status Report" states that the temperature excursion occurred in steam-rich conditions, after an initial

[169] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.
[170] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042230126, p. 311.
[171] *Id.*
[172] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," p. 14.

heatup phase in pure helium (up to 1000°C), and that the PCT was approximately 1900 K, during the first oxidation phase. The PHEBUS B9R-2 test had a second oxidation phase and temperature escalation.[173]

Neither paper states what peak cladding temperatures were at the outset of the autocatalytic oxidation reaction; however, a graph of the cladding-temperature values at the 0.6 meter "hot-level" indicates that the autocatalytic oxidation reaction began when cladding temperatures were below 1477 K (2200°F)[174] (see Appendix I Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF)).

D. The Damage BWR Fuel Assembly Components Incurred at "Low Temperatures" in the BWR CORA Experiments: CORA-16, CORA-17, and CORA-18

1. The Liquefaction of Fuel Assembly Components at "Low Temperatures" in the BWR CORA Experiments: CORA-16, CORA-17, and CORA-18

Regarding the damage process that started in the upper bundles of the BWR CORA experiments at relatively low temperatures, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The conduct of tests CORA-16, CORA-17, and CORA-18 resulted in a behavior typical for BWR-type CORA experiments: *The flame front; i.e., the temperature escalation developed first above the axial centerline and then moved to the upper and lower part of the bundle. The damage process started in the upper bundle region with melting of the absorber blade by interaction of boron carbide and steel at about 1200°C.* The resulting melt attacked the zircaloy channel box walls by the steel-zirconium interaction. After destruction of the walls the melt was able to penetrate the coolant channels starting the interaction with the rod

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[173] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, Department of Safety Research, Research Center of Cadarache France, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," p. 311.
[174] *Id.*, p. 312.

> claddings. The so liquefied zircaloy interacted with the UO_2 fuel pellets [emphasis added].[175]

And regarding the liquefaction of bundle components that began at approximately 1200°C in the CORA-16 experiment, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> When the BWR bundle CORA-16 was heated to a maximum temperature of 2000°C, *liquid reaction products have formed as early as from 1200°C* on, due to the chemical interactions of the bundle components, some of them occurring even well below the melting point of the components. Liquefaction of the bundle components, beginning at 1200°C, could be visualized by means of the ten video-cameras installed, simultaneously to the temperature measurements, and characterized with a view to temperature [emphasis added].[176]

And regarding the B_4C-stainless steel reaction that began at approximately 1000°C in the CORA-16 experiment, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The B_4C absorber material enters into a reaction with its steel cladding, beginning at approximately 1000°C, and liquefies the cladding very quickly above 1200°C.[177]

And also regarding the B_4C-stainless steel reaction in the CORA-16 experiment, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The various axial transverse micro-sections of the CORA-16 bundle to which different temperatures can be attributed reflect the material behavior as a function of the temperature. The CORA 16-08 transverse micro-section…, prepared from a section outside the heated bundle zone, clearly shows the onset of the chemical interactions of B_4C and stainless steel (type AISI 316) at temperatures ranging from 1100 to 1200°C. B_4C reacts with stainless steel eutectically while forming liquid phases. The

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[175] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," Forschungszentrum Karlsruhe, FZKA 7447, 2008, p. 6.
[176] *Id.*, p. 10.
[177] *Id.*, p. 11.

> boride phase is clearly visible as a border around the B_4C-particles. The B_4C-particles are dissolved chemically by it.[178]

Additionally, regarding the B_4C-stainless steel reaction in the CORA-16 experiment, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The determination of critical temperatures beyond which the reaction products are liquid and thus easily amenable to relocation, is of particular importance. *B_4C, melting point approximately 2350°C, can be liquefied from approximately 1250°C on due to chemical interactions with the Fe, Cr, and Ni steel components.*[179] This process was observed with video-cameras during the heating phase of the BWR bundle CORA 16. The subsequent relocation of the B_4C-containing melt produces relatively large axial sections of bundles containing no more B_4C absorber material. Under realistic accident conditions flooding of the overheated, partly destroyed reactor core with boron-free water might give rise to criticality problems [emphasis added].[180]

And summarizing the results of the CORA-16, CORA-17, and CORA-18 experiments, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility" states:

> The destructive post-test examinations of the bundle showed strong chemical interactions over the whole bundle length.
>
> The presence of B_4C absorber material causes the formation of a "low temperature" melt at around 1250°C that attacks the zircaloy channel box and the zircaloy fuel rod cladding. The liquefaction is due to an interaction between B_4C and steel (of the absorber rod cladding and the absorber blade). ... The liquefied B_4C/[stainless steel] absorber blade relocates completely from the upper half of the CORA test bundle; *i.e.*, the absorber material is missing in the upper regions of fuel elements whereas it is concentrated at the bottom. This fact may cause recriticality problems with the injection of unborated emergency cooling water into a dried-out reactor core.[181]

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[178] *Id.*, p. 12.
[179] W. Hering, P. Hofmann, "Material Interactions During Severe LWR Accidents; Summary of Separate-Effects Test Results," KfK 5125, 1994.
[180] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," FZKA 7447, p. 13.
[181] *Id.*, p. 15.

So in the CORA-16 experiment, the B_4C-stainless steel reaction began at approximately 1000°C and the stainless steel cladding of the B_4C absorber material liquefied very quickly above 1200°C.[182] And in the CORA-16, CORA-17, and CORA-18 experiments "[t]he presence of B_4C absorber material cause[d] the formation of a 'low temperature' melt at around 1250°C that attack[ed] the zircaloy channel box and the zircaloy fuel rod cladding."[183]

Regarding the B_4C-stainless steel reaction, "Advanced BWR Core Component Designs and the Implications for SFD Analysis" states that the "strong chemical attack of the stainless steel by B_4C at ~1200°C with complete liquefaction by 1250°C...contrasts with the expected failure of the BWR control blade by melting at 1375°-1425°C."[184]

And regarding the B_4C/stainless steel control blade (control rod) liquefaction, "Advanced BWR Core Component Designs and the Implications for SFD Analysis" states:

> Given the constituents of the control blade (*i.e.*, B, C, Fe, Ni, Cr, and minor impurities) and referring to standard references,[185] several binary combinations (B/Fe and B/Ni) show low melting eutectics (from 1000° to 1150°C), and this is the reason that the control blade liquefies ~200°C lower than the melting range of stainless steel.[186]

Additionally, "Current Knowledge on Core Degradation Phenomena, a Review," Fig. 1. "LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases"[187] depicts: 1) that Fe/Zr and Ni/Zr eutectics commence at 940°C (1724°F) and 2) that B_4C/Fe eutectics commence at temperatures between 1130°C (2066°F) and 1200°C (2192°F). (See Appendix F Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.)

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[182] *Id.*, p. 11.
[183] *Id.*, p. 15.
[184] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," Oak Ridge National Laboratory, 1997, pp. 4-5.
[185] M. Hansen, "Constitution of Binary Alloys," McGraw-Hill Book Company, 1958 and R. P. Elliott, "Constitution of Binary Alloys," First Supplement, McGraw-Hill Book Company, 1965.
[186] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 8.
[187] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 196.

And comparing the BWR CORA-17 experiment with the PWR CORA-12 and CORA-13 experiments (which used typical PWR bundles and Ag-In-Cd absorber), "Degraded Core Quench: A Status Report" states:

> The earlier starting and stronger reaction in the [CORA-17] BWR test can be interpreted as being due to the additional influence of the boron carbide [(B_4C)] absorber. *This material has an exothermic reaction rate three times larger than that of Zircaloy* and produces [four] to [eight] times more hydrogen [emphasis added].[188]

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

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

> Immediately after quenching BWR test bundle CORA-17 experiences a modest increase for 20 sec. and changed then in a steep increase resulting in the highest temperature and hydrogen peaks of the three tests [(CORA-12, CORA-13, CORA-17)]. CORA-17 also showed a temperature increase in the lower part of the bundle... We interpret this earlier starting and stronger reaction [as being] due to the influence of the boron carbide, the absorber material of the BWR test.
>
> *B_4C has an exothermic reaction energy [four] to [five] times larger than Zry* and produces about [six] times more hydrogen. Probably the hot remained columns of B_4C (seen in the non-quench test CORA-16) react, early in the quench process with the increased upcoming steam. *The bundle temperature, raised by this reaction increases the reaction rate of the remained metallic Zry* (exponential dependence) [emphasis added].[189]

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

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188 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. 16.

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

Clearly, the fact that there would be complete liquefaction of the stainless steel of the BWR control blade at approximately 1250°C (2282°F), instead of at temperatures between 1375 and 1425°C (2507 and 2597°F),[190] is a significant nuclear power safety issue. And, clearly, data from the CORA-16 experiment—*i.e.*, the B_4C-stainless steel reaction beginning at approximately 1000°C (1832°F) and the stainless steel cladding of the B_4C absorber material liquefying very quickly above 1200°C (2192°F)[191]—is further evidence that VYNPS's LBPCT of 1960°F for GE14 fuel would not provide a necessary margin of safety to help prevent a partial or complete meltdown, in the event of a LOCA.

2. The Damage GE14 Fuel Assemblies and Current BWR Core Component Designs would, with High Probability, Incur in a LOCA

a. GE14 Fuel Assemblies and Current BWR Core Component Designs

It is significant that the CORA-16, CORA-17, and CORA-18 experiments were conducted with assemblies "modeled on the BWR core component designs circa 1985; that is, the 8x8 fuel assembly with two water rods (fuel rod and water rods having diameters of 12.27 and 15.0 mm, respectively) and a cruciform control blade constructed of B_4C-filled tubelets."[192]

VYNPS's GE14 fuel is a 10x10 fuel assembly of 78 full-length Zircaloy-2 fuel rods, 14 part length rods, and two large central water rods.[193]

And regarding the control rods (control blades, absorbers) that are currently used in BWRs, "ABWR General Description: Core and Fuel Design" states:

> [C]ruciform shaped control rods are configured for insertion between every four fuel assemblies, comprising a module or "cell." ... Typically, the cruciform control rods contain stainless steel tubes in each wing of the cruciform filled with boron carbide (B_4C) powder compacted to approximately 75% of theoretical density. The tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes

[190] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," pp. 4-5.
[191] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber Tested under Severe Fuel Damage Conditions in the CORA Facility," FZKA 7447, p. 11.
[192] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 7.
[193] General Electric, "ABWR General Description: Core and Fuel Design," Chapter 6, pp. 6-2, 6-4.

> into individual longitudinal compartments. ... The tubes are held in cruciform array by a stainless steel sheath extending the full length of the tubes. ... In addition to boron carbide, hafnium absorber may be placed in the highest burnup locations of select control rods, the full length outside edge of each wing and, optionally, the tip of each wing. Hafnium is a heavy metal with excellent neutron absorbing characteristics and does not swell at high burnups.[194]

Regarding fuel designs and core components developed after the BWR CORA experiments were conducted, "Advanced BWR Core Component Designs and the Implications for SFD Analysis" states:

> Generally [nuclear power plant] operating trends have been towards longer operating cycle lengths (18-24 months) and higher discharge burnups (approaching 50,000 MWd/MTU for BWRs). These trends have brought pressure on the fuel fabricators to develop fuel designs that offer higher discharge burnups, longer lived components, and provide improved plant operating margins.[195]

And "Advanced BWR Core Component Designs and the Implications for SFD Analysis" also provides a partial list of fuel design and core component improvements made after the BWR CORA experiments were conducted and explains their benefits; among the fuel design and core component improvements listed are: 1) "smaller (diametrically) fuel rods (*i.e.*, 9x9 and 10x10 fuel rod arrays) [that allow] higher bumup[s] with a lower linear heat generation rate [thus providing] lower pellet and cladding operating temperatures and [less] cladding corrosion;" 2) "larger water rods (or more water rods, or water crosses) [that increase] hot excess and cold shutdown (ridging) [and provide] reactivity differences [that improve] neutron efficiency [and] moderation;" 3) "using high purity stainless steel tubing in the control blade [to increase] rod life [and decrease] B_4C/stainless steel swelling/cracking problems;" 4) "using hafnium at the control blade wing edges and at the top of the control blade [to reduce] swelling at high burnups (as compared to B_4C) [and to provide] longer rod life;" and 5) "solid control blade construction (*i.e.*, no outside blade sheath)."

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194 *Id.*, pp. 6-6, 6-7, 6-8.

195 L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 7.

b. The Damage GE14 Fuel Assemblies and Current BWR Core Component Designs would, with High Probability, Incur in a LOCA

First, GE14 fuel assemblies are Zircaloy fuel assemblies, so they would, in the event of a LOCA, with high probability, incur autocatalytic oxidation, if they reached temperatures between approximately 1832°F and 2192°F; in such a case, local cladding temperatures of the GE14 fuel assemblies would escalate at tens of degrees Fahrenheit per second. In the CORA 2 and CORA 3 experiments, the Zircaloy fuel assemblies incurred autocatalytic oxidation when cladding temperatures reached 1832°F, and in the CORA-16, CORA-17, and CORA-18 experiments, the 8x8 Zircaloy fuel assemblies incurred autocatalytic oxidation when cladding temperatures reached 2012°F.

Second, current control rods would, with high probability, liquefy at temperatures between 1200°-1250°C, like the control rods did in the BWR CORA experiments.

Regarding how control blade components with hafnium content would, with high probability, liquefy if they reached temperatures between approximately 1200°C and 1250°C, "Advanced BWR Core Component Designs and the Implications for SFD Analysis" states:

> Elliott[196]...indicates that [hafnium] may form low melting eutectics with Fe and Ni, although these systems are less definitive than the boron systems. Thus, if Elliott is correct, then the new BWR control blade (with hafnium) may behave the same as the control blade as currently modeled;[197] however, there is the possibility that the hafnium may not interact with the stainless steel sheath of the control blade. For this postulate, the inner portion of the blade (where the B_4C-filled tubelets are positioned) will probably liquefy at 1200°-1250°C and relocate (interacting with the control blade and Zircaloy channel wall at lower elevations); but the blade wing tips (containing the hafnium) might remain intact in the core until the stainless steel or the hafnium melts. For this case, the recriticality issue is again raised, since neutron-absorbing material (hafnium) might remain in the core after the B_4C portion of the control blade has exited the core; also, for this case, even the advanced control blade models[198] are not applicable.[199]

[196] R. P. Elliott, "Constitution of Binary Alloys," First Supplement, McGraw-Hill Book Company, 1965.
[197] F. P. Griffin, "BWR Control Blade/Channel Box Model for SCDAP/RELAP5: Damage Progression Theory and User Guide," letter report (ORNL/NRC/LTR-96/20) to Dr. Yi-Shung Chen, Accident Evaluation Branch, Division of Systems Research, RES, USNRC, July 12, 1996.
[198] *Id.*

So the results of the CORA-16, CORA-17, and CORA-18 experiments provide a good indication of the damage GE14 fuel assemblies and current BWR core components would incur in the event of a LOCA, if the cladding reached temperatures between approximately 1832°F and 2192°F.

IV. CONCLUSION

Petitioner requests that the NRC order the licensee of VYNPS to lower the LBPCT of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA. Experimental data indicates that VYNPS's LBPCT of 1960°F[200] does not provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA. Such data indicates that VYNPS's LBPCT must be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety.

To uphold its congressional mandate to protect the lives, property, and environment of the people of Vermont and locations within proximity of VYNPS, the NRC must not allow VYNPS's LBPCT to remain at an elevated temperature that would not provide a necessary margin of safety, in the event of LOCA. If implemented, the enforcement action proposed in this petition would help improve public and plant worker safety.

To: R. William Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

[199] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 8.
[200] Entergy, "VYNPS 10 C.F.R. § 50.46(a)(3)(ii) Annual Report for 2009," p. 2.

Respectfully submitted,

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New York, NY 10025
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Dated: June 7, 2010

Appendix A Fig. 12. Temperatures during Test CORA-2 at [550] mm and 750 mm Elevation and Fig. 13. Temperatures Measured during Test CORA-3 at 450 mm and 550 mm Elevation[1]

[1] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Interactions in Zircaloy/UO_2 Fuel Rod Bundles with Inconel Spacers at Temperatures above 1200°C (Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," KfK 4378, pp. 79, 80.

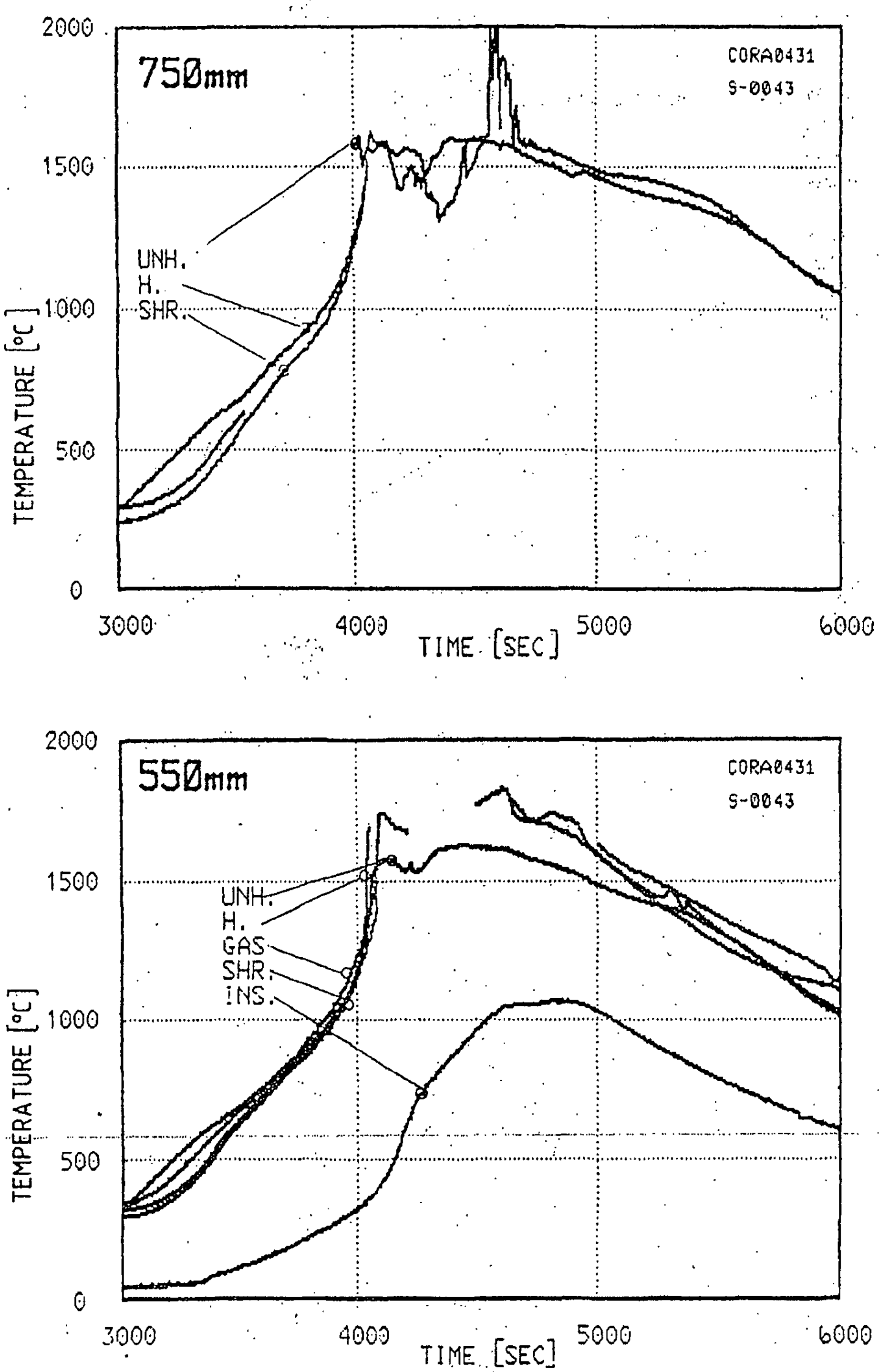


Fig. 12. Temperatures during test CORA-2 at 500 mm and 750 mm elevation. Temperatures of heated (H) and unheated rod (UNH), atmosphere (gas), shroud (SHR), and outer surface of shroud insulation (INS)

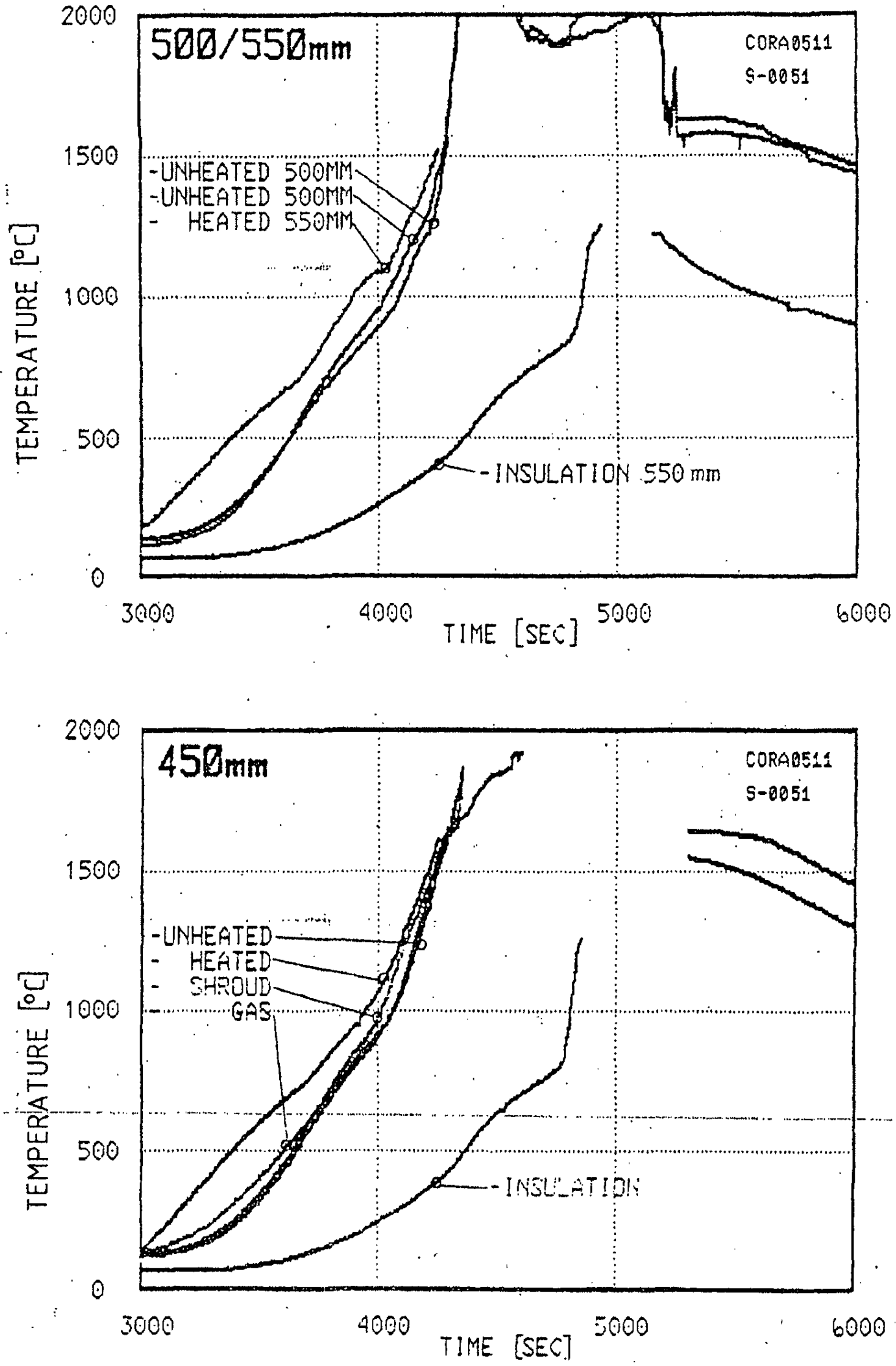


Fig. 13. Temperatures measured during test CORA-3 at 450 mm and 550 mm elevation

Appendix B Figure 15. Temperatures of Unheated Rods and Power History of CORA-5, Figure 16. Temperatures of Unheated Rods during CORA-12, Figure 17. Temperatures at Different Elevations during CORA-15, Figure 18. Temperatures of Unheated Rods during CORA-9, Figure 19 CORA-7; Temperatures at Elevations Given (750 mm), and Figure 20 Temperatures of Guide Tube and Absorber Rod during Test CORA-5[2]

[2] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, Institut für Materialforschung Programm Nukleare Sicherheitsforschung, Forschungszentrum Karlsruhe GmbH, Karlsruhe, "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility," 2008, pp. 75-80.

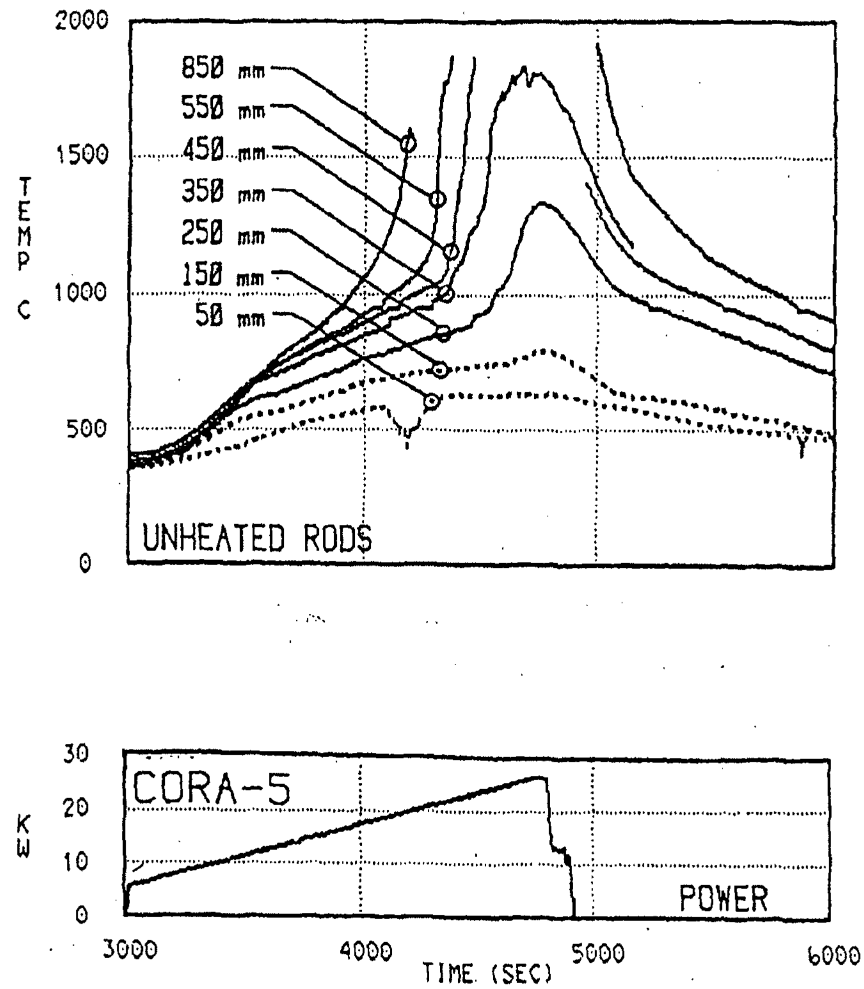


Fig.15: **Temperatures of unheated rods and power history of CORA-5**

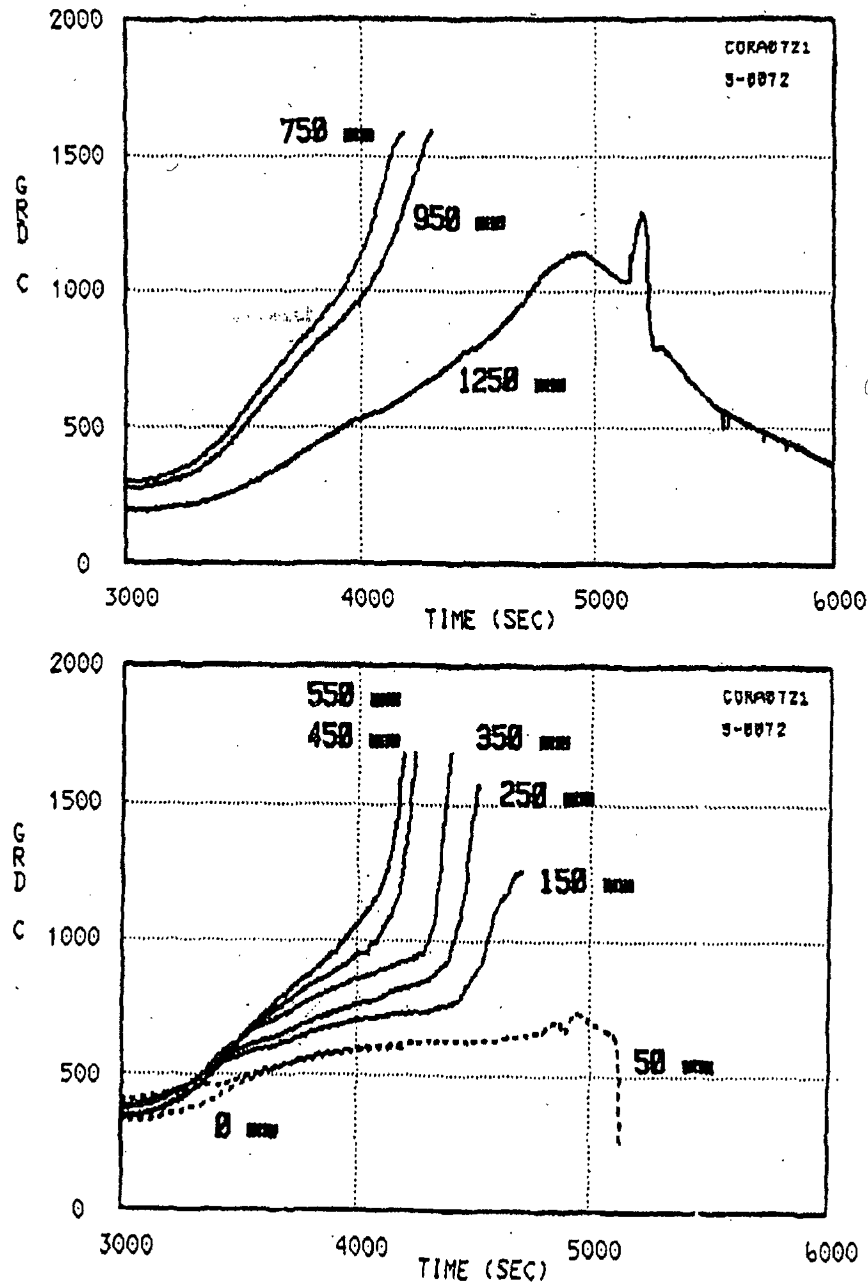


Fig.16: Temperatures of unheated rods during CORA-12

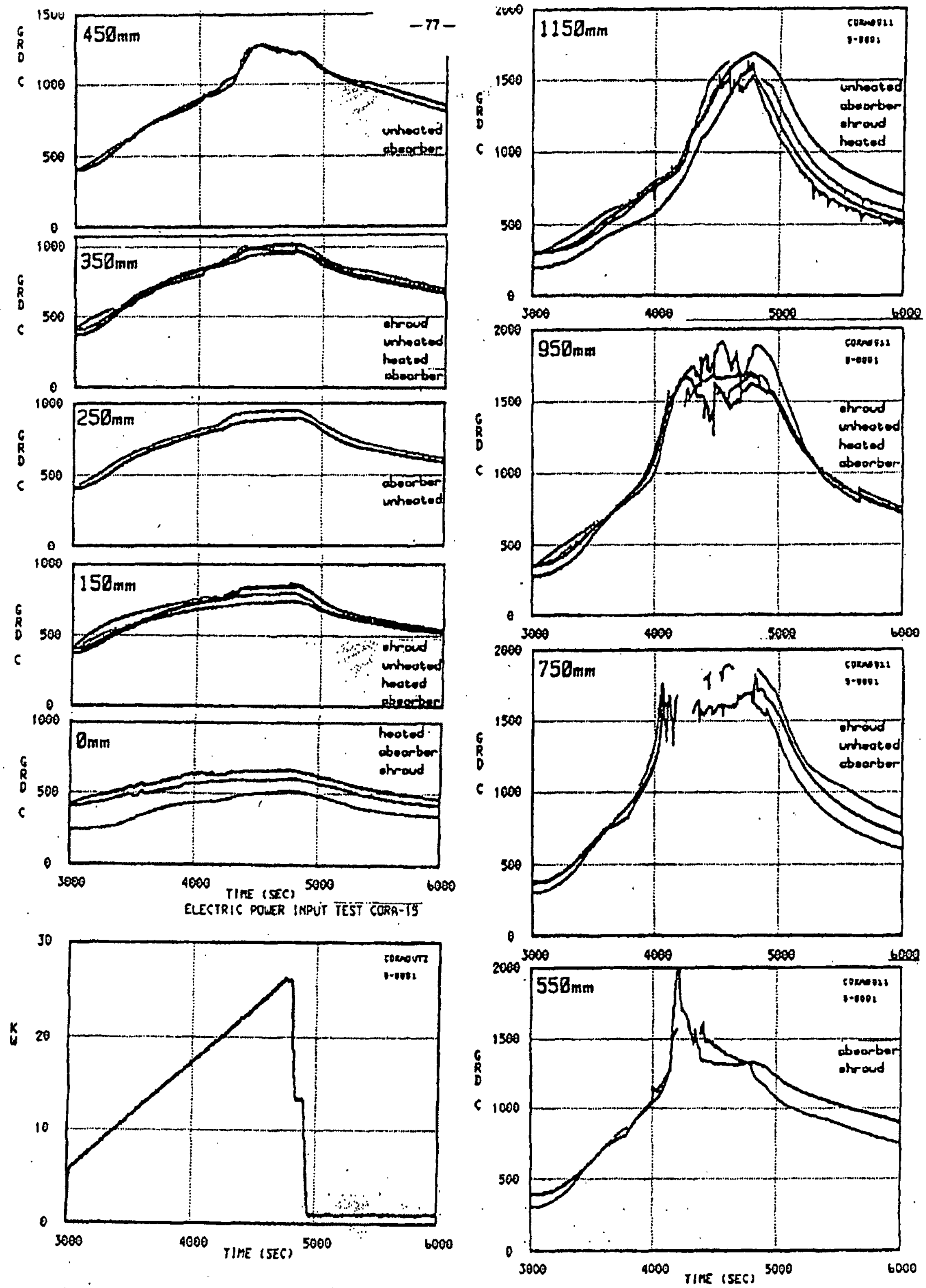


Fig.17: Temperatures at different elevations during CORA-15

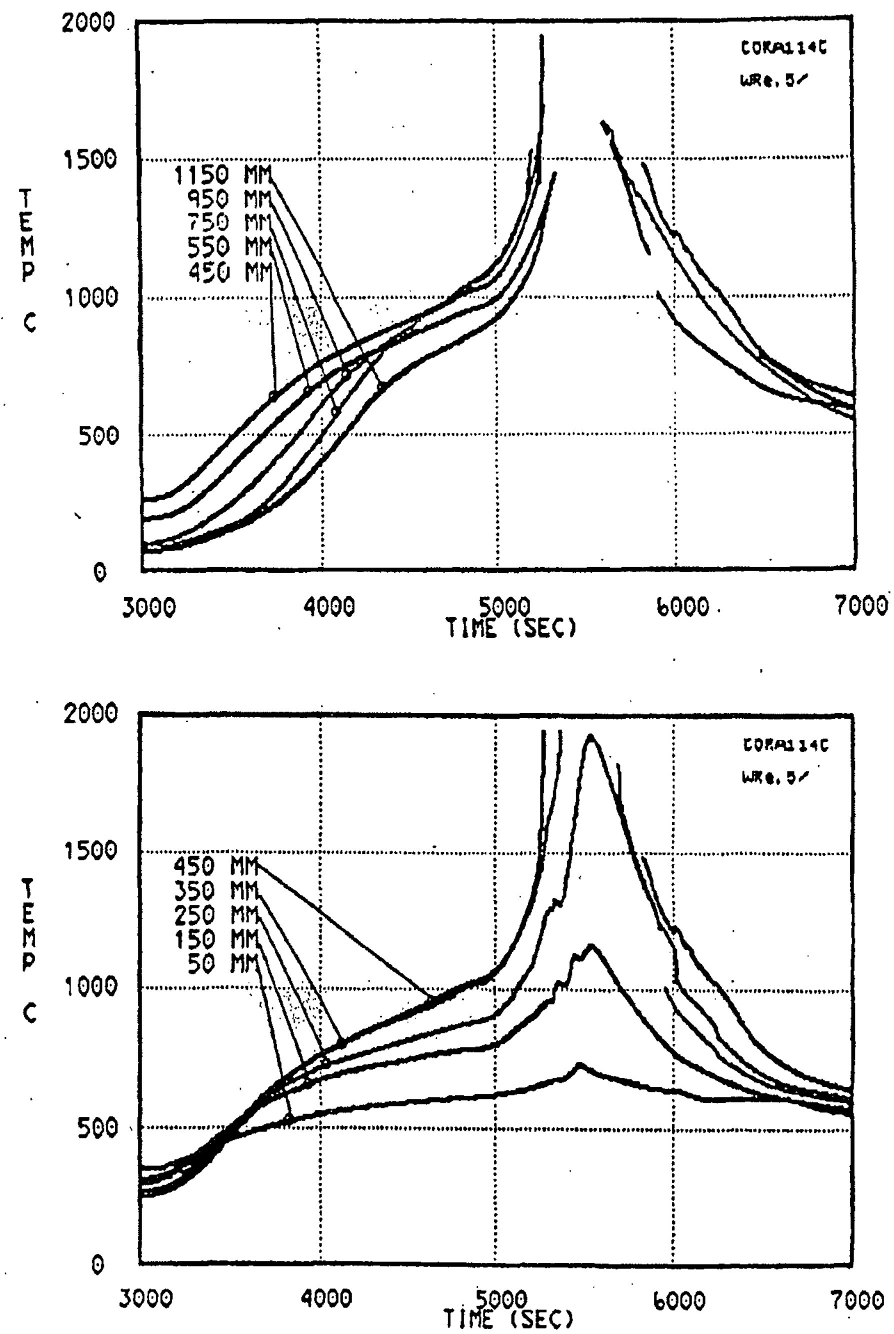
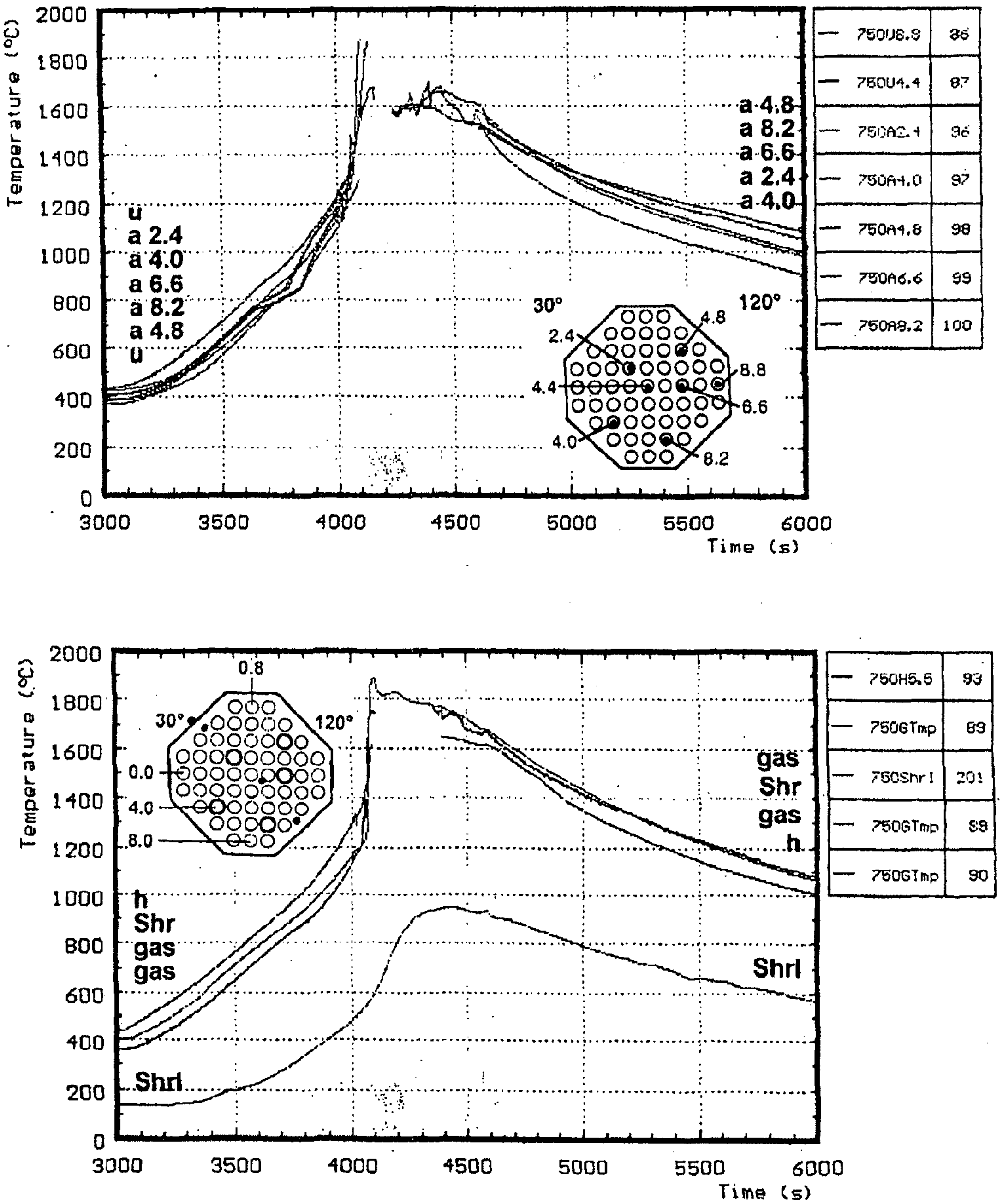


Fig.18: Temperatures of unheated rods during CORA-9



h : heated rods
u : unheated rods
a : in absorber

shr : outer side of shroud
shrl : on shroud insulation
gas : gas temperature

Fig. 19: CORA-7; Temperatures at elevations given (750 mm)

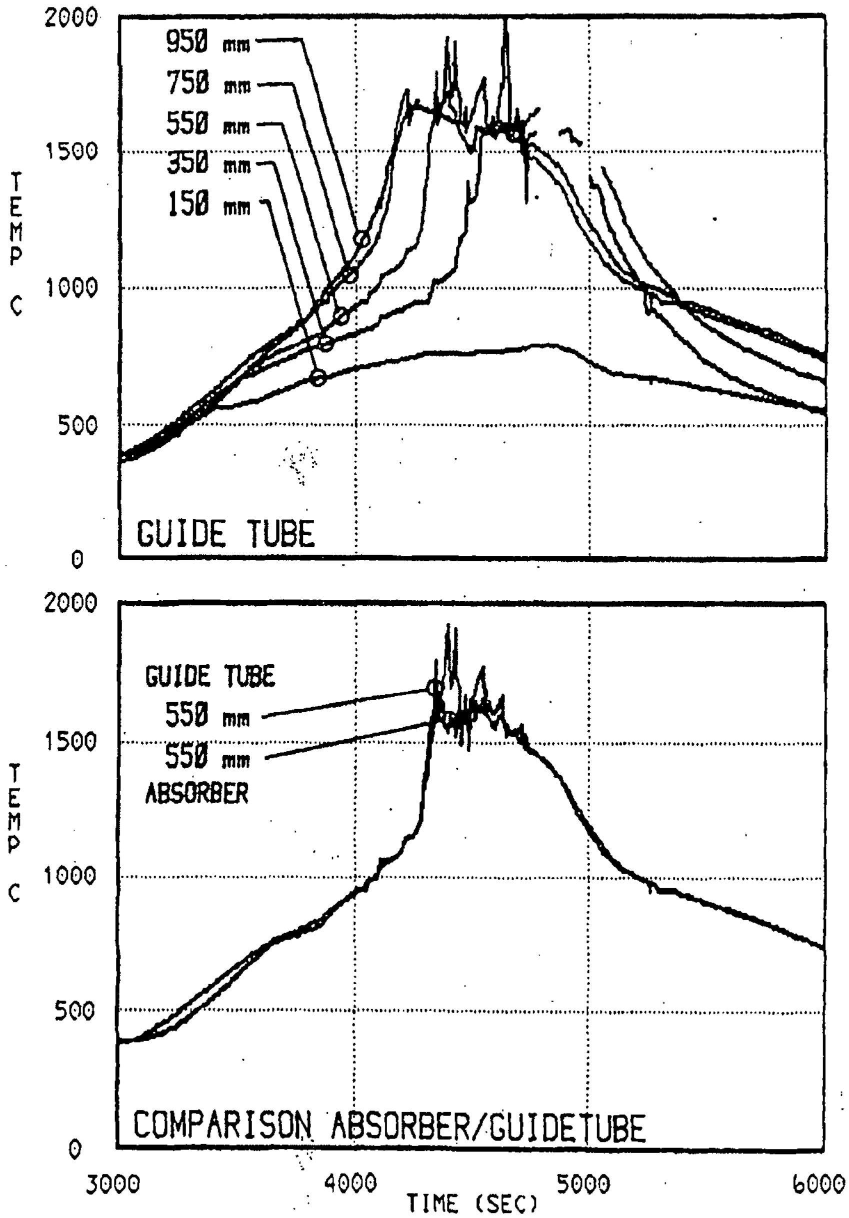


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

Appendix C Figure 37. Temperatures of the Heated Rods (CORA-13) and Figure 39. Temperatures of the Unheated Rods (CORA-13) [3]

[3] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, Kernforschungszentrum Karlsruhe, "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)," 1993, pp. 76, 78.

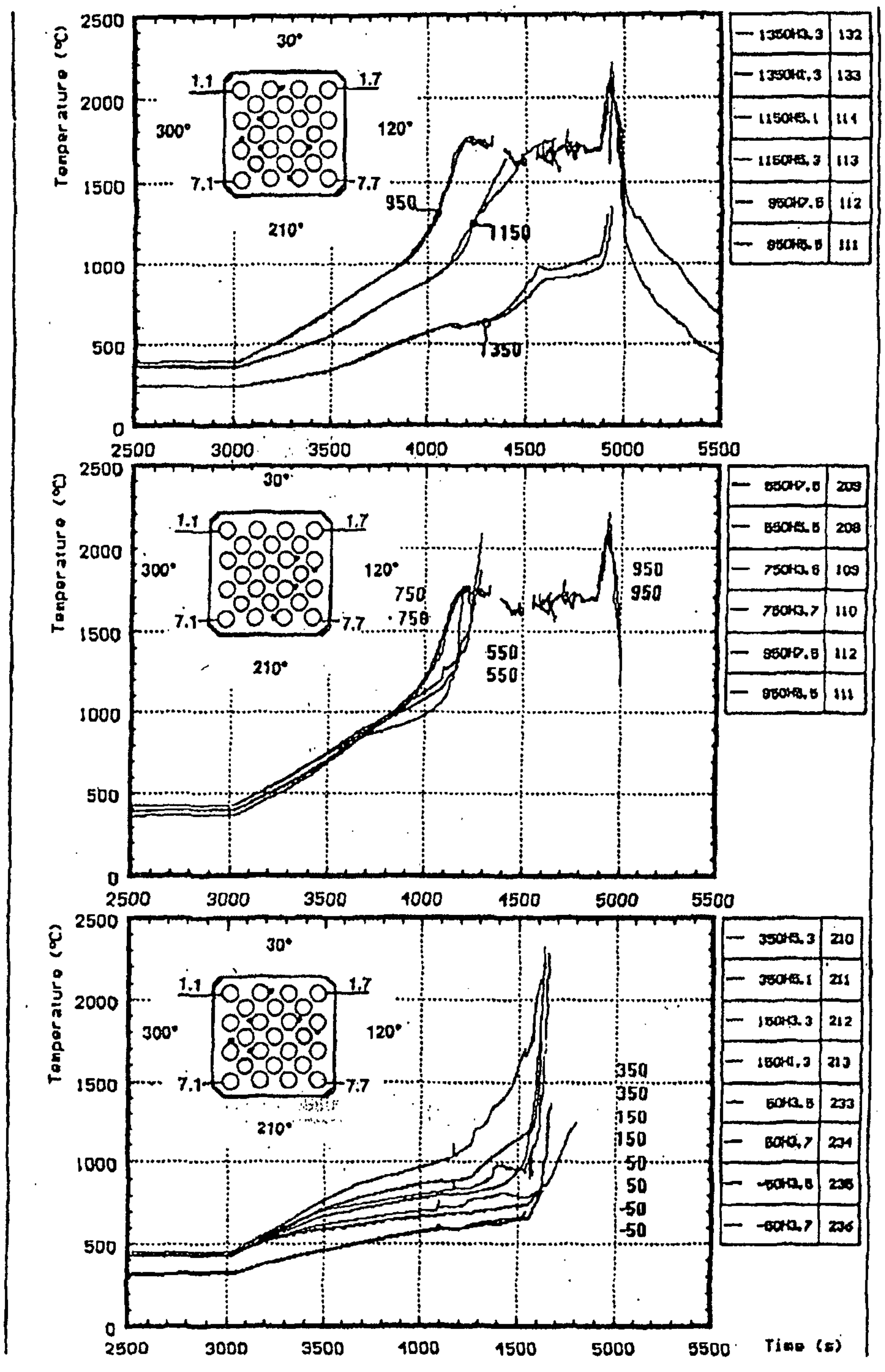


Fig. 37: Temperatures of the heated rods (CORA-13)

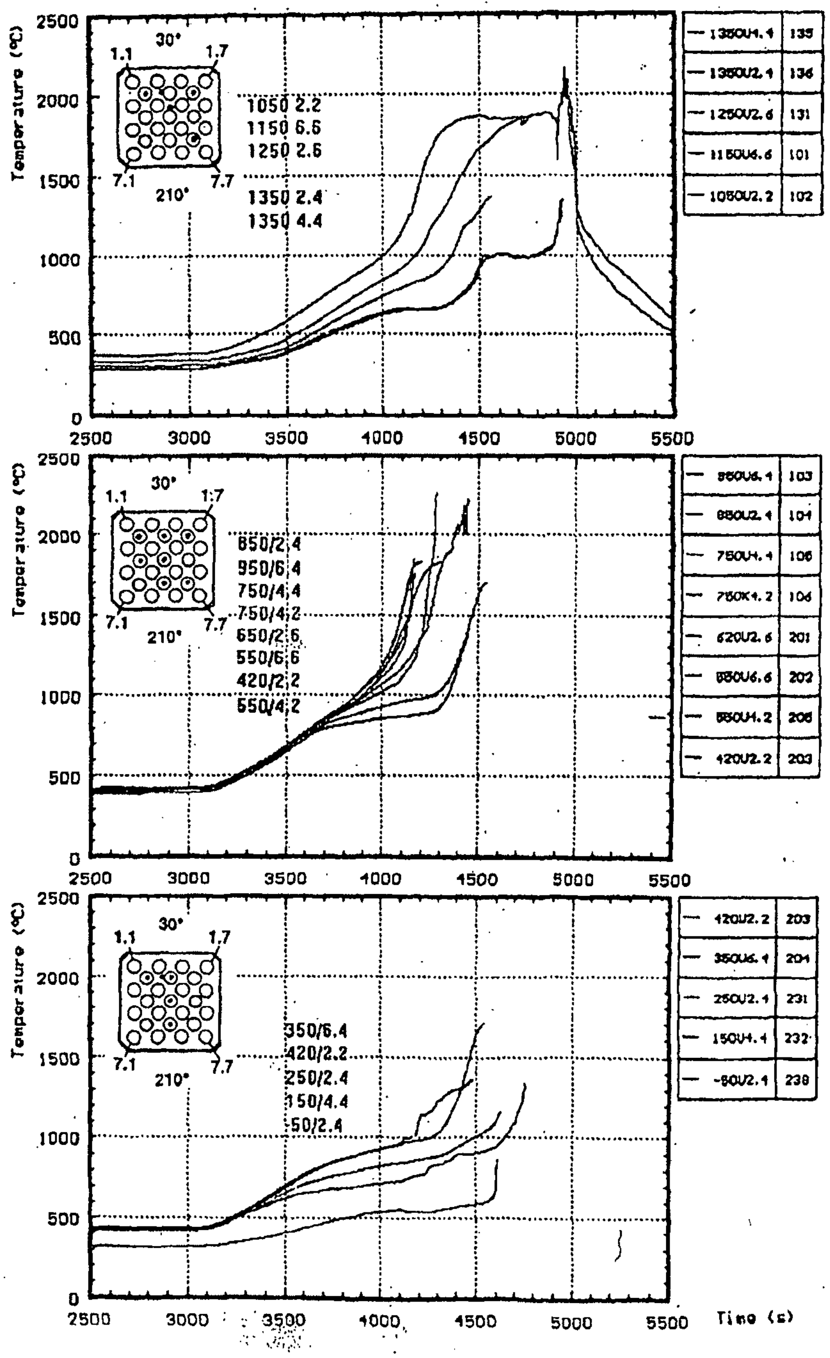


Fig. 39: **Temperatures of the unheated rods (CORA-13)**

Appendix D Figure 3.7. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 and Figure 3.10. Comparison of Two Cladding Temperatures at the 0.69-m (27-in.) Elevation in Fuel Assembly 5 with Saturation Temperature (Graphs of Cladding Temperature Values During the LOFT LP-FP-2 Experiment)[4]

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

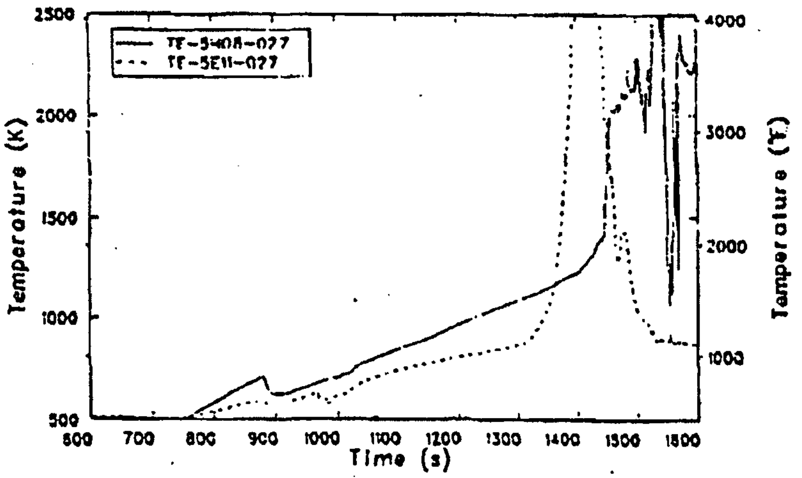


Figure 3.7 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5.

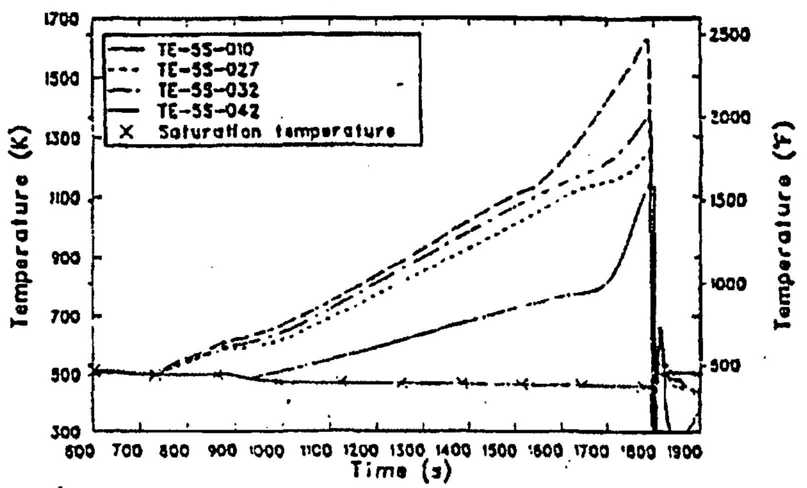


Figure 3.8 Comparison of four external wall temperatures at the 1.07-, 0.81-, 0.69-, and 0.25-m (42-, 32-, 27-, and 10-in.) elevations on the south side of the flow shroud.

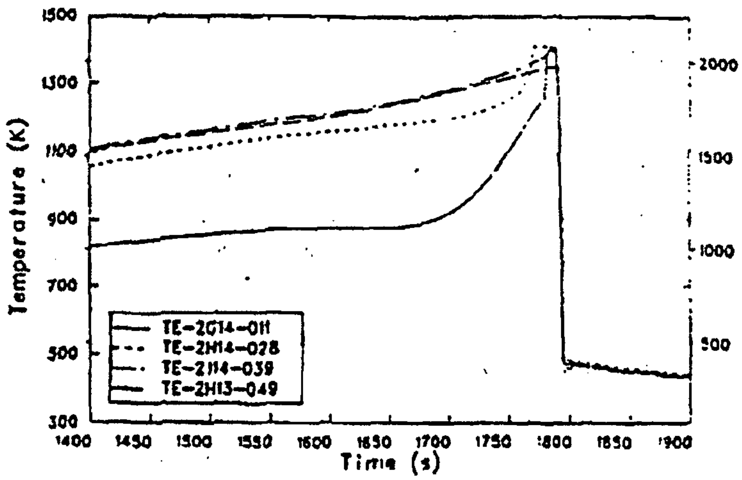


Figure 3.9 Comparison of cladding temperatures at the 1.24-, 0.99-, 0.71-, and 0.28-m (49-, 39-, 28-, and 11-in.) elevations in Fuel Assembly 2.

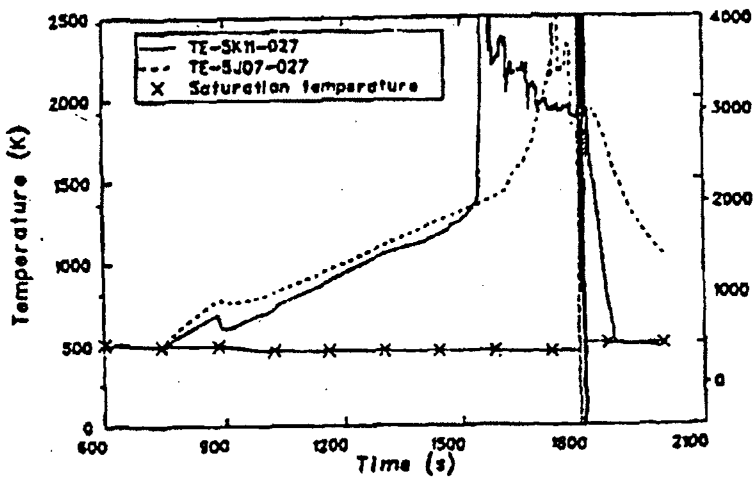


Figure 3.10 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5 with saturation temperature.

Appendix E Fig. 14. CFM Fuel Cladding Temperature at the 0.686 m. (27 in.) Elevation and Fig. 15 Comparison of Temperature Data with and without Cable Shunting Effects at the 0.686 m. (27 in.) Elevation in the CFM[5]

[5] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," GRS-Garching, Proceedings of the OECD (NEA) CSNI Specialist Meeting on Instrumentation to Manage Severe Accidents, Held at Cologne, F.R.G. March 16-17, 1992, pp. 143, 144.

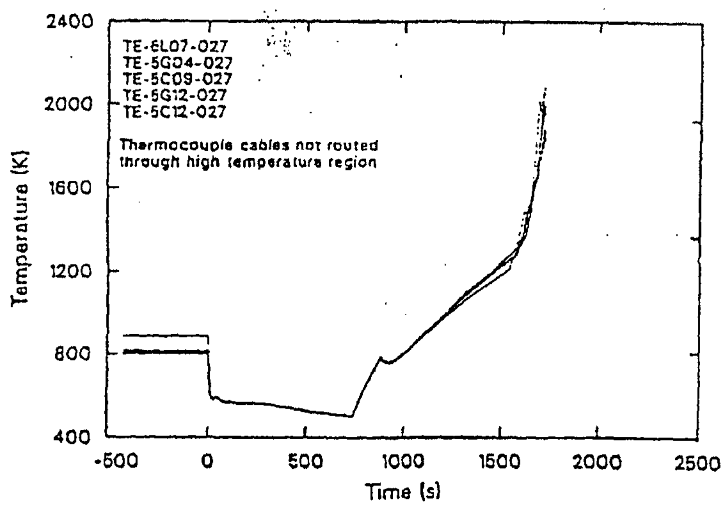


Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

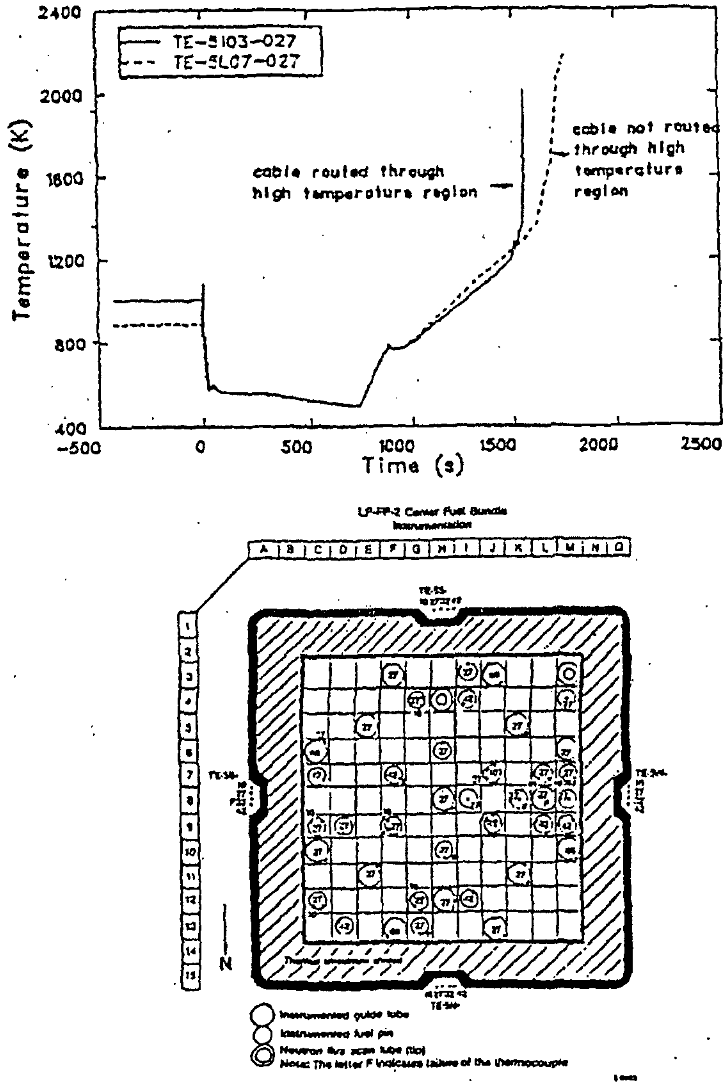


Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

Appendix F 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[6]

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

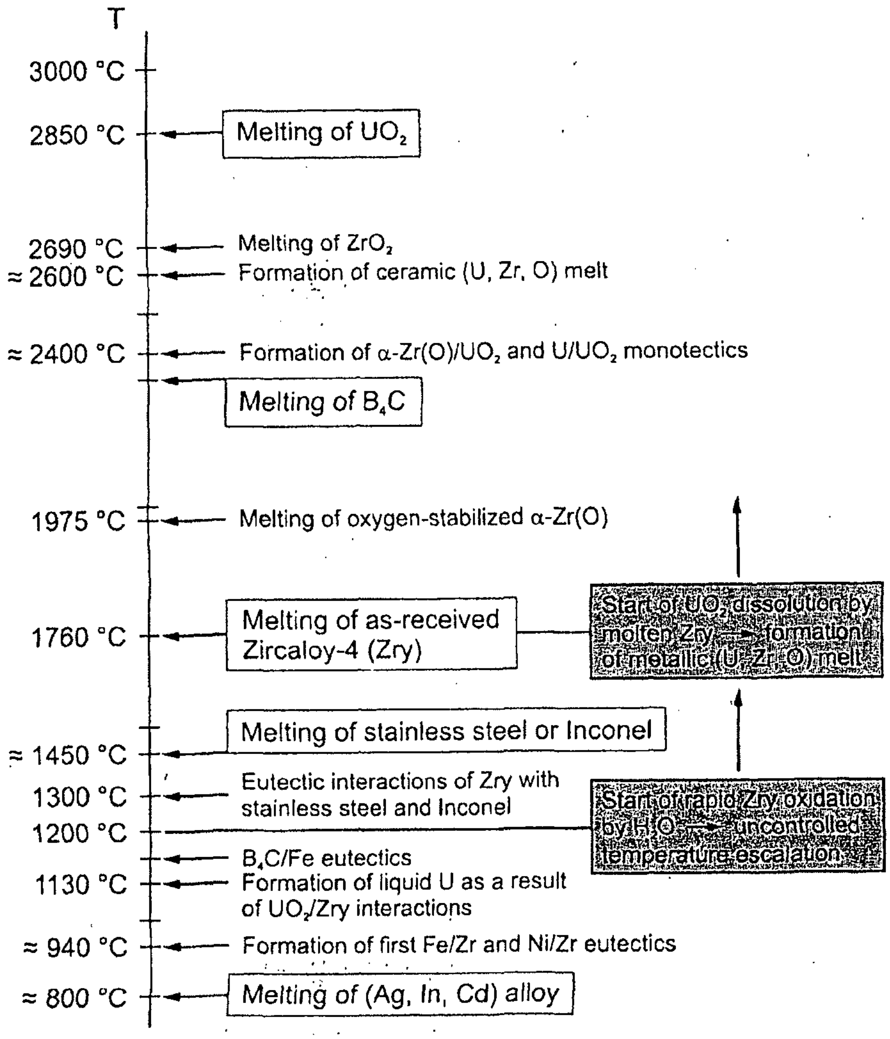


Fig. 1. LWR severe accident-relevant melting and chemical interaction temperatures which result in the formation of liquid phases.

- 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

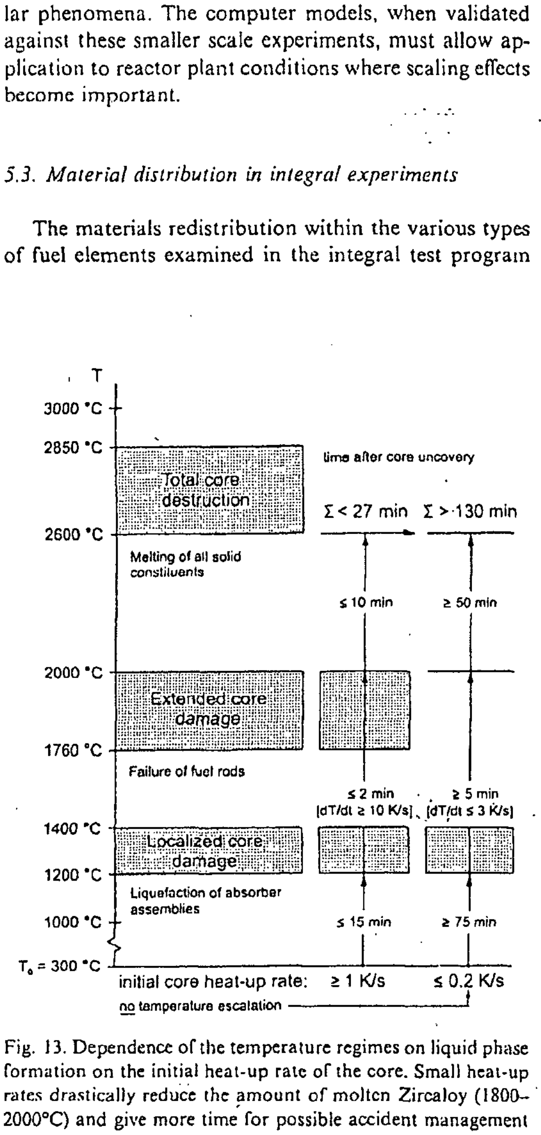


Fig. 13. Dependence of the temperature regimes on liquid phase formation on the initial heat-up rate of the core. Small heat-up rates drastically reduce the amount of molten Zircaloy (1800–2000°C) and give more time for possible accident management measures.

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].

Appendix G Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[7] and Figure A8.10 Analysis of Zr2K Thermal Response[8]

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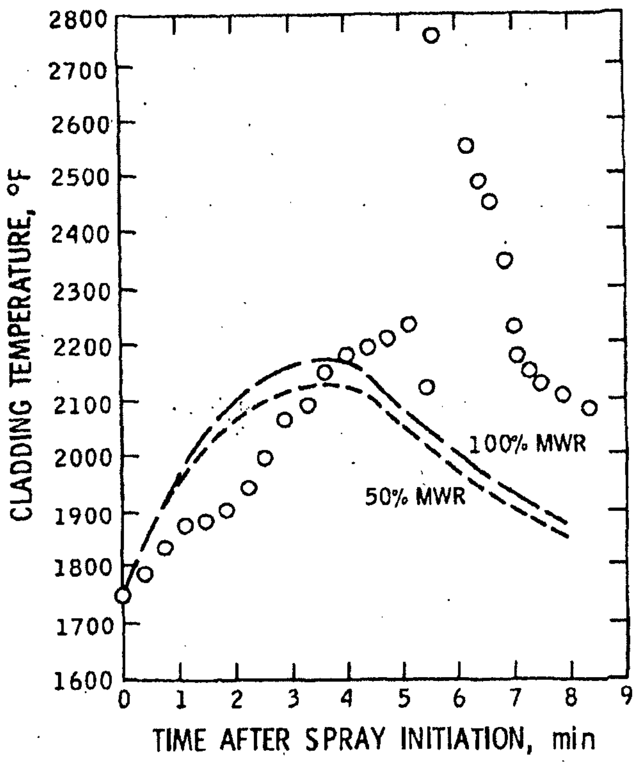
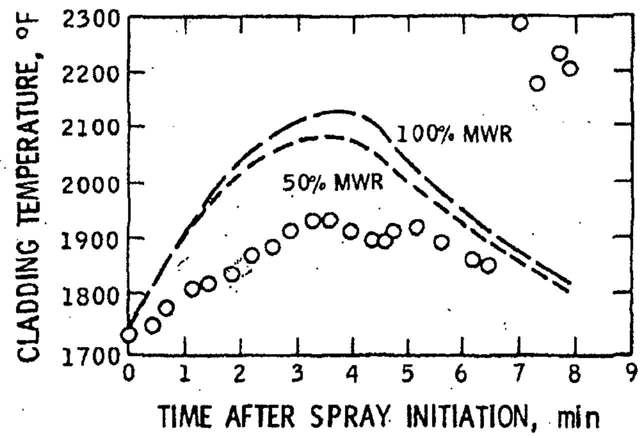
[7] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-25; this paper cites J. D. Duncan and J. E. Leonard, "Emergency Cooling in Boiling Water Reactors Under Simulated Loss-of-Coolant Conditions," (BWR-FLECHT Final Report), General Electric Co., San Jose, CA, GEAP-13197, June 1971, Figures A-11 and A-12, as the source of this information.

[8] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-26; this paper cites J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, Figure 12, as the source of this information.



Bundle Zr2K Rod 24 Midplane Thermal Response Prediction

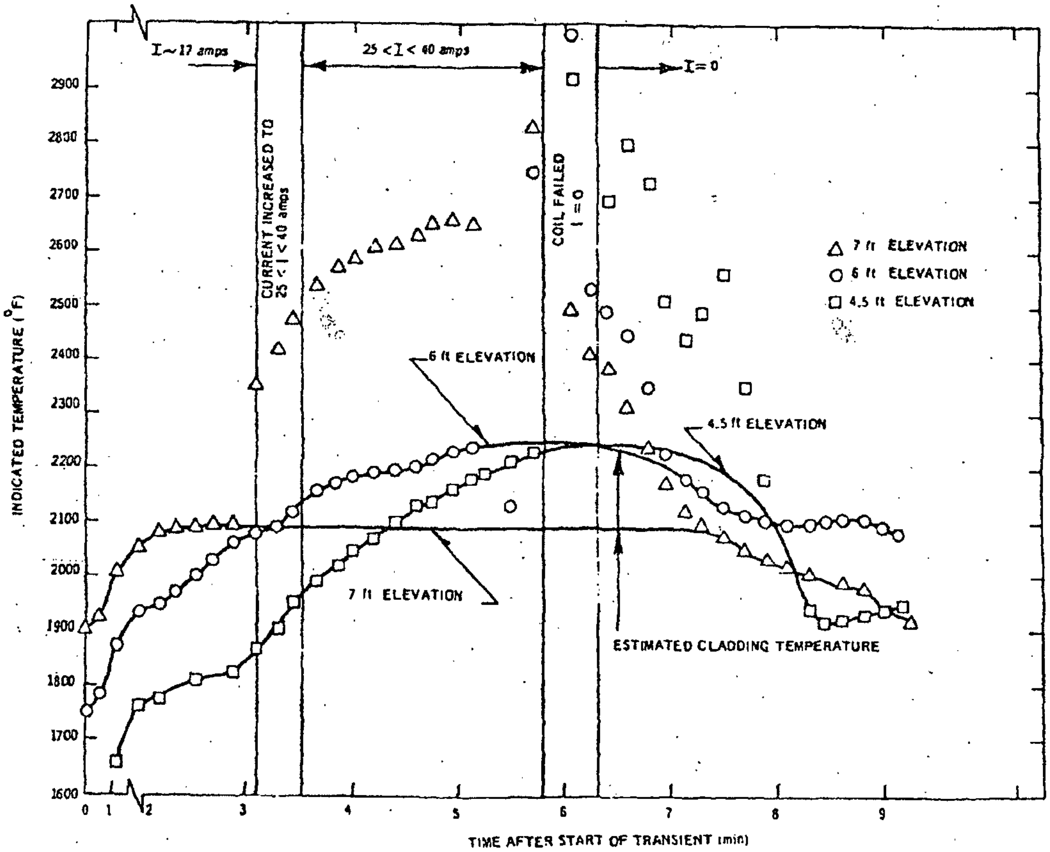


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies (After Figures A-11 and A-12 from 52 by permission.)

Figure A8.10

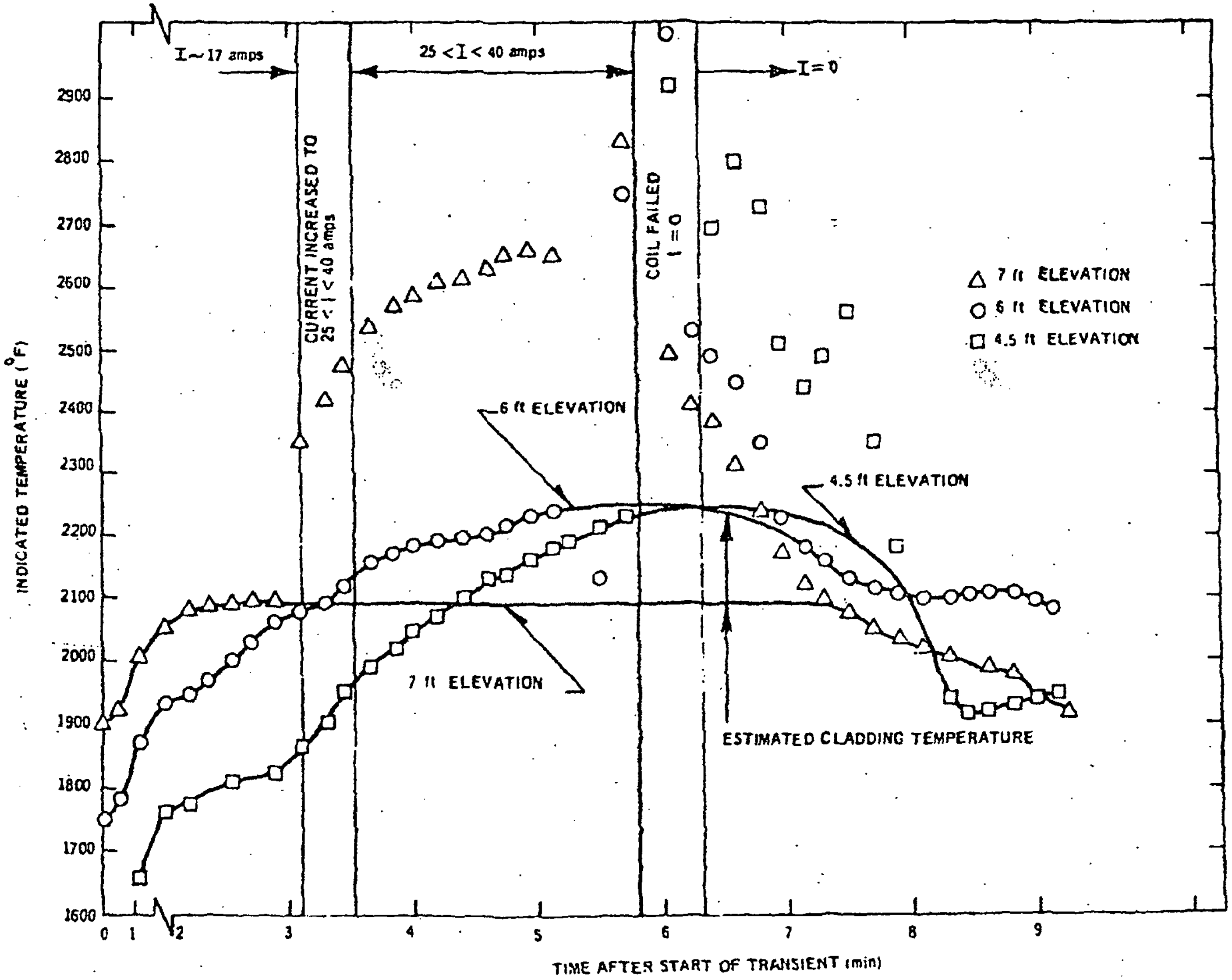
Analysis of Zr2K Thermal Response



(After Figure 12, 54, by permission.)

Figure A8.10

Analysis of Zr2K Thermal Response



(After Figure 12, 54, by permission.)

Appendix H Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region[9] (Graphs of Cladding Temperature Values During the FLHT-1 Test)[10]

[9] Pseudo sensor readings are the averages of the readings of two or more thermocouples.
[10] W. N. Rausch, G. M. Hesson, J. P. Pilger, L. L. King, R. L. Goodman, F. E. Panisko, Pacific Northwest Laboratory, "Full-Length High-Temperature Severe Fuel Damage Test 1," August 1993, pp. 4.7, 5.3.

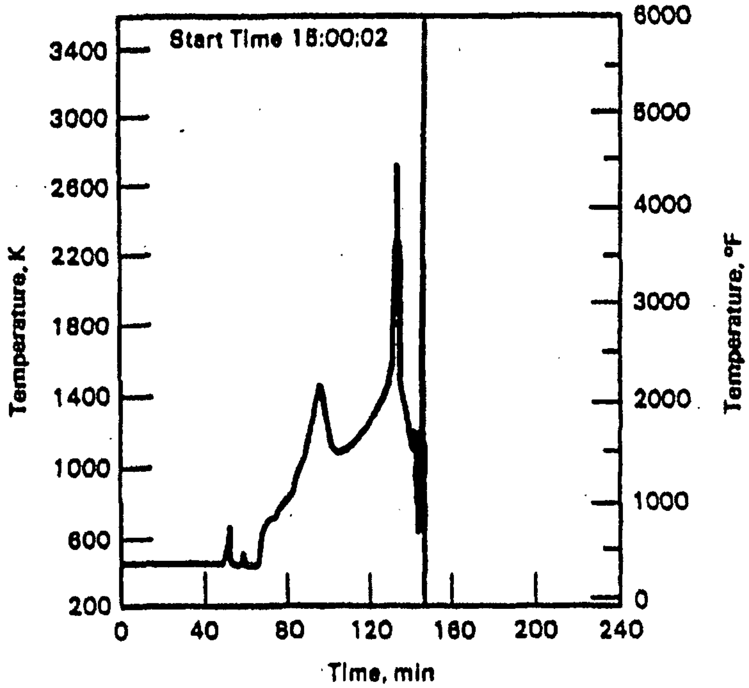


FIGURE 4.1. Typical Cladding Temperature Behavior

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

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

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

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

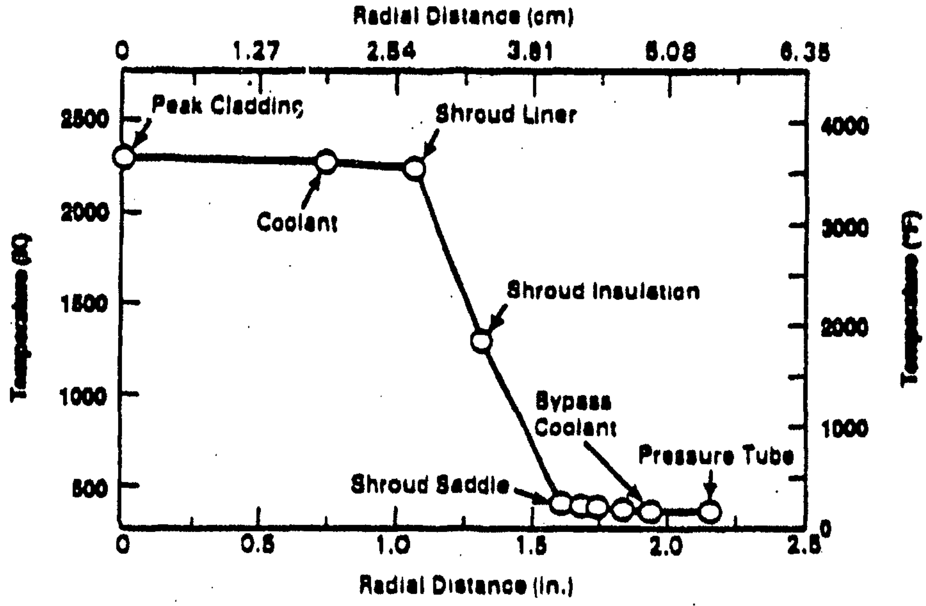


FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

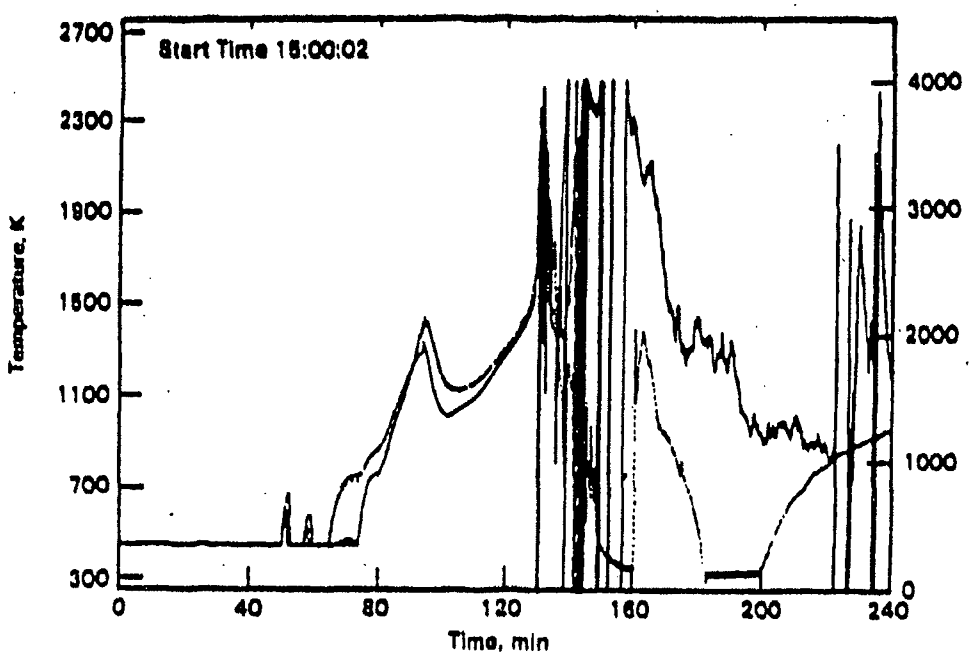


FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

Appendix I Figure 1. Sensitivity Calculation on the B9R Test: Temperature Escalation at the Hot Level (0.6 m) with Different Contact Area Factors (CAF) [11]

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

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

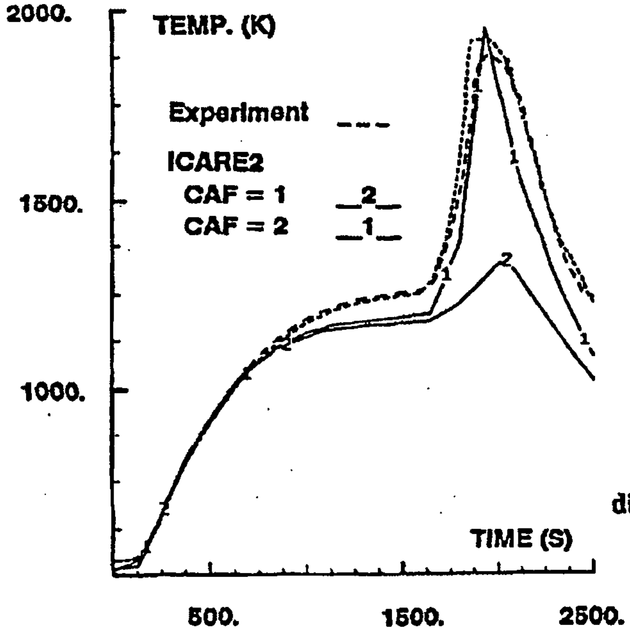


Fig.1 :
Sensitivity calculation on the B9R test. Temperature escalation at the hot level (0.6 m) with different Contact Area Factors (CAF)

3.2.2 PHEBUS C3 + test

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

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

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