EDO Principal Correspondence Control

FROM: DUE: 01/12/11 EDO CONTROL: G20100729
DOC DT: 12/10/10
FINAL REPLY:

Mark Edward Leyse
New York, NY

TO:

Borchardt, EDO

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

Leeds, NRR

DESC:

2.206 - Lower Licensing Basis Peak Cladding Temperatures of Oyster Creek and Nine Mile Unit 1 (EDATS: OEDO-2010-1001)

ROUTING:

Borchardt
Weber
Virgilio
Ash
Muessle
OGC/GC
Dean, RI
Burns, OGC
Mensah, NRR
Scott, OGC
Kotzalas, OEDO

DATE: 12/13/10

ASSIGNED TO: CONTACT:

NRR Leeds

SPECIAL INSTRUCTIONS OR REMARKS:

Template: EDO-001

ERIDS: EDO-01

EDATS

Electronic Document and Action Tracking System

EDATS Number: OEDO-2010-1001 **Source:** OEDO

General Information

Assigned To: NRR **OEDO Due Date:** 1/12/2011 11:00 PM

Other Assignees: **SECY Due Date:** NONE

Subject: 2.206 - Lower Licensing Basis Peak Cladding Temperatures of Oyster Creek and Nine Mile Point Unit 1

Description:

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

ADAMS Accession Numbers - Incoming: NONE **Response/Package:** NONE

Other Information

Cross Reference Number: G20100729 **Staff Initiated:** NO

Related Task: **Recurring Item:** NO

File Routing: EDATS **Agency Lesson Learned:** NO

OEDO Monthly Report Item: NO

Process Information

Action Type: 2.206 Review **Priority:** Medium

Sensitivity: None

Signature Level: NRR **Urgency:** NO

Approval Level: No Approval Required

OEDO Concurrence: NO

OCM Concurrence: NO

OCA Concurrence: NO

Special Instructions:

Document Information

Originator Name: Mark Edward Leyse **Date of Incoming:** 12/10/2010

Originating Organization: Citizens **Document Received by OEDO Date:** 12/13/2010

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

Incoming Task Received: Letter

December 10, 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 Temperatures of Oyster Creek Nuclear Generating Station ("OCNGS") and Nine Mile Point Unit 1 ("NMP-1") in Order to Provide Necessary Margins of Safety—to Help Prevent Meltdowns—in the Event of Loss-of-Coolant Accidents ("LOCA") and to have the Licensees of OCNGS and NMP-1 Demonstrate that OCNGS and NMP-1's BWR/2 Emergency Core Cooling Systems would Effectively Quench the Fuel Cladding in the Event of LOCAs

Dear Mr. Borchardt:

The attached petition for an enforcement action is submitted pursuant to 10 C.F.R. § 2.206 by Mark Edward Leyse ("Petitioner"). 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 licensees of OCNGS and NMP-1 to lower the licensing basis peak cladding temperatures ("LBPCT") of OCNGS and NMP-1, in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of loss-of-coolant accidents ("LOCA"). Experimental data indicates that OCNGS and NMP-1's LBPCTs of 2150°F[1] and 2149°F,[2] respectively, do not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Such data indicates that OCNGS and NMP-1's LBPCTs must be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety.

Petitioner also requests that NRC order the licensees of OCNGS and NMP-1 to demonstrate that OCNGS and NMP-1's BWR/2 emergency core cooling systems ("ECCS") would effectively quench the fuel cladding in the event of LOCAs, and prevent partial or complete meltdowns. Experimental data indicates that OCNGS and NMP-1's

---

[1] Exelon, "Oyster Creek Nuclear Generating Station 10 C.F.R. § 50.46 Annual Report for 2009," June 4, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101550497, Attachment 1, p. 2.

[2] Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports," December 7, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML020170430, p. 1. "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," states that for prior 10 C.F.R. § 50.46 changes or error corrections for previous years ΔPCT was 0°F, for NMP-1; see Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," January 28, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML1003340589 , Attachment 1, p. 2,

EDO --G20100729

ECCSs would not effectively quench the fuel cladding in the event of LOCAs, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

This 10 C.F.R. § 2.206 petition is similar to a 10 C.F.R. § 2.206 petition, dated June 7, 2010 (ADAMS Accession Number: ML101610121), that Petitioner authored and submitted on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("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. However, this 10 C.F.R. § 2.206 petition requests an additional enforcement action that was not requested in the VYNPS 10 C.F.R. § 2.206 petition.

On October 27, 2010, NRC published in the Federal Register that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner authored and submitted on behalf of NEC—requesting that NRC order the licensee of VYNPS to lower the LBPCT of VYNPS—meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95.[3]

This 10 C.F.R. § 2.206 petition addresses both generic issues and issues that are specific to OCNGS and NMP-1, the only two U.S. plants with BWR/2 ECCSs. The generic issues this 10 C.F.R. § 2.206 petition addresses—regarding the metal-water reaction and the eutectic reactions between different assembly components—are also addressed in PRM-50-93 and PRM-50-95; however, this 10 C.F.R. § 2.206 petition addresses the fact that it has never been conclusively demonstrated that BWR/2 ECCSs—specific to OCNGS and NMP-1—would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

However, it should be clarified that it has also never been conclusively demonstrated that BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if maximum cladding temperatures reached between approximately 1832°F and 2200°F.

Clearly, BWR heat transfer experiments need to be conducted with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Such BWR heat transfer experiments need to be conducted in experiments modeling BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs.

(It is noteworthy that there should be a regulation stipulating minimum allowable amounts of coolant to be supplied to each fuel bundle in the BWR core, in the event of a LOCA.[4])

---

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

[4] "Resolution of Generic Safety Issues: Item A-16: Steam Effects on BWR Core Spray Distribution" states that "to ensure the health and safety of the public, [BWR] core spray systems

To uphold its congressional mandate to protect the lives, property, and environment of the people of New York and New Jersey, NRC must not allow OCNGS and NMP-1's LBPCTs to remain at elevated temperatures that would not provide necessary margins of safety, in the event of LOCAs. If implemented, the enforcement action proposed in this petition would help improve public and plant worker safety.

Petitioner would like to acknowledge that David Lochbaum of Union of Concerned Scientists helped with this petition.

Respectfully submitted,

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

---

must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores."

December 10, 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 TEMPERATURES OF OYSTER CREEK NUCLEAR GENERATING STATION ("OCNGS") AND NINE MILE POINT UNIT 1 ("NMP-1") IN ORDER TO PROVIDE NECESSARY MARGINS OF SAFETY—TO HELP PREVENT MELTDOWNS—IN THE EVENT OF LOSS-OF-COOLANT ACCIDENTS ("LOCA") AND TO HAVE THE LICENSEES OF OCNGS AND NMP-1 DEMONSTRATE THAT OCNGS AND NMP-1'S BWR/2 EMERGENCY CORE COOLING SYSTEMS WOULD EFFECTIVELY QUENCH THE FUEL CLADDING IN THE EVENT OF LOCAS

TABLE OF CONTENTS

December 10, 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
EXELON GENERATION COMPANY, LLC. (Oyster Creek Nuclear Generating Station; Facility Operating License No. DPR-16)	:	Docket No. _____
CONSTELLATION ENERGY (Nine Mile Point Unit 1; Facility Operating License No. DPR-63)		

MARK EDWARD LEYSE
Petitioner

10 C.F.R. § 2.206 REQUEST TO LOWER THE LICENSING BASIS PEAK CLADDING TEMPERATURES OF OYSTER CREEK NUCLEAR GENERATING STATION ("OCNGS") AND NINE MILE POINT UNIT 1 ("NMP-1") IN ORDER TO PROVIDE NECESSARY MARGINS OF SAFETY—TO HELP PREVENT MELTDOWNS—IN THE EVENT OF LOSS-OF-COOLANT ACCIDENTS ("LOCA") AND TO HAVE THE LICENSEES OF OCNGS AND NMP-1 DEMONSTRATE THAT OCNGS AND NMP-1'S BWR/2 EMERGENCY CORE COOLING SYSTEMS WOULD EFFECTIVELY QUENCH THE FUEL CLADDING IN THE EVENT OF LOCAS

I. REQUEST FOR ACTION

This petition for an enforcement action is submitted pursuant to 10 C.F.R. § 2.206 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."

Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") order the licensees of Oyster Creek Nuclear Generating Station ("OCNGS") and Nine Mile Point Unit 1 ("NMP-1") to lower the licensing basis peak cladding

temperatures ("LBPCT") of OCNGS and NMP-1, in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of loss-of-coolant accidents ("LOCA"). Experimental data indicates that OCNGS and NMP-1's LBPCTs of 2150°F[1] and 2149°F,[2] respectively, do not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Such data indicates that OCNGS and NMP-1's LBPCTs must be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety.

Petitioner also requests that NRC order the licensees of OCNGS and NMP-1 to demonstrate that OCNGS and NMP-1's BWR/2 emergency core cooling systems ("ECCS") would effectively quench the fuel cladding in the event of LOCAs, and prevent partial or complete meltdowns. Experimental data indicates that OCNGS and NMP-1's ECCSs would not effectively quench the fuel cladding in the event of LOCAs, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

II. STATEMENT OF PETITIONER'S INTEREST

Petitioner is submitting this 10 C.F.R. § 2.206 petition because experimental data demonstrates that OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, need to both be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Petitioner also is submitting this 10 C.F.R. § 2.206 petition because the licensees of OCNGS and NMP-1 need to demonstrate that OCNGS and NMP-1's BWR/2 ECCSs would effectively quench the fuel cladding in the event of LOCAs, and prevent partial or complete meltdowns. Experimental data indicates that OCNGS and NMP-1's ECCSs

---

[1] Exelon, "Oyster Creek Nuclear Generating Station 10 C.F.R. § 50.46 Annual Report for 2009," June 4, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101550497, Attachment 1, p. 2.

[2] Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports," December 7, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML020170430, p. 1. "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," states that for prior 10 C.F.R. § 50.46 changes or error corrections for previous years ΔPCT was 0°F, for NMP-1; see Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," January 28, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML1003340589 , Attachment 1, p. 2,

would not effectively quench the fuel cladding in the event of LOCAs, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

This 10 C.F.R. § 2.206 petition is similar to a 10 C.F.R. § 2.206 petition, dated June 7, 2010 (ADAMS Accession Number: ML101610121), that Petitioner authored and submitted on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("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. However, this 10 C.F.R. § 2.206 petition requests an additional enforcement action that was not requested in the VYNPS 10 C.F.R. § 2.206 petition.

This 10 C.F.R. § 2.206 petition addresses both generic issues and issues that are specific to OCNGS and NMP-1, the only two U.S. plants with BWR/2 ECCSs. The generic issues this 10 C.F.R. § 2.206 petition addresses—regarding the metal-water reaction and the eutectic reactions between different assembly components—are also addressed in PRM-50-93 and PRM-50-95; however, this 10 C.F.R. § 2.206 petition addresses the fact that it has never been conclusively demonstrated that BWR/2 ECCSs—specific to OCNGS and NMP-1—would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

However, it should be clarified that it has also never been conclusively demonstrated that BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if maximum cladding temperatures reached between approximately 1832°F and 2200°F.

Clearly, BWR heat transfer experiments need to be conducted with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Such BWR heat transfer experiments need to be conducted in experiments modeling BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs.

(It is noteworthy that there should be a regulation stipulating minimum allowable amounts of coolant to be supplied to each fuel bundle in the BWR core, in the event of a LOCA.[3])

On March 15, 2007, Petitioner 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[4] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[5] by Union of Concerned Scientists ("UCS"). In 2008, NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[6] And in 2009, NRC published "Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of proposed rulemaking, addressing four objectives: the fourth being the issues raised in PRM-50-84.[7]

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

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

---

[3] "Resolution of Generic Safety Issues: Item A-16: Steam Effects on BWR Core Spray Distribution" states that "to ensure the health and safety of the public, [BWR] core spray systems must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores."

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

[5] 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.

[6] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.

[7] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

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

Petitioner also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis."[8]

On November 17, 2009, Petitioner submitted PRM-50-93. PRM-50-93 requests that NRC make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments;[9] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[10, 11]

Additionally, PRM-50-93 requests that 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.[12] These same requirements also need to

---

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

[9] 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.

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

[11] 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.

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

apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[13]

PRM-50-93 was discussed briefly in ANS's *Nuclear News*'s March 2010 issue[14] and in *Inside NRC*'s July 30, 2010 issue.[15] PRM-50-93 was also 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 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.[16]

On October 27, 2010, NRC published in the Federal Register that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner authored and submitted on behalf of NEC—requesting that NRC order the licensee of VYNPS to lower the LBPCT of VYNPS—meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[17]

Petitioner would like to acknowledge that David Lochbaum of UCS helped with this petition.

III. FACTS CONSTITUTING THE BASIS FOR PETITIONER'S REQUEST

> [Consolidated National Intervenors's] direct testimony concluded that a near thermal runaway condition existed in [BWR-FLECHT] Test ZR-2. It

---

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

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

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

[15] Suzanne McElligott, *Inside NRC*, July 30, 2010.

[16] 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.

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

> 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.[18]—Henry W. Kendall and Daniel F. Ford

A. ECCS Evaluation Calculations for OCNGS and NMP-1 are Non-Conservative

OCNGS and NMP-1's LBPCTs were both determined by Appendix K to Part 50 ECCS evaluation calculations that used the Baker-Just correlation.

(Regarding the Baker-Just correlation, 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 approximately half of more than 50 LOCA calculations that the NRC performed with RELAP5/Mod3 that used the Baker-Just correlation predicted autocatalytic (runaway) oxidation to commence when cladding temperatures increased to above approximately 2600°F, 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 correlation is non-conservative for use in analyses that calculate 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 correlation is non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

It is also significant that "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory" ("In-Vessel Phenomena—CORA"), presented in 1991, explicitly states "[c]ladding oxidation [in the

[18] 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.

CORA-16 experiment] was not accurately predicted by available correlations."[19] (In 1991, the Baker-Just correlation was among the available correlations.)

Discussing "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[20] 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].[21]

So, in the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations"[22] and "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted."[23] This indicates that available Zircaloy oxidation kinetics models—including the Appendix K model—are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

---

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

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

[21] *Id.*

[22] 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.

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

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

The alleged conservatism of OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, 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.

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.[24]

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

[24] 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.

> 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.[25]

It is significant that "Compendium of ECCS Research for Realistic LOCA Analysis" states that assessing the conservatism of the 2200°F PCT limit, as a boundary that would prevent autocatalytic oxidation from occurring, can be accomplished by analyzing data from multi-rod severe accident tests, because such data, 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.[26]

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."[27] So at the point when peak cladding temperatures increased at a rate of greater than 10 K/sec. during the

---

[25] *Id.*

[26] 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.

[27] 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.

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.[28])

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].[29]

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].[30]

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

---

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

[29] 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.

[30] 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.

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.

C. Experiments that Indicate OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, would Not Provides Necessary Margins of Safety to Help Prevent Partial or Complete Meltdowns, in the Event of LOCAs

> 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.[31]—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?[32]—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.[33]—S. Hagen, *et al.*

In this section, Petitioner will discuss data from multi-rod severe fuel damage experiments and two multi-rod thermal hydraulic experiments that indicates OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, would not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of a LOCA.

---

[31] 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.

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

[33] 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.

Petitioner will discuss: 1) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures below OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively; 2) experiments 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 3) one experiment in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at a temperature of 2275°F or lower.

(Petitioner will also discuss a postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the Three Mile Island Unit 2 accident.)

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 OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, Respectively, a Postulation that Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at 1000°C in the Three Mile Island Unit 2 Accident, and a Discussion of One Multi-Rod Thermal Hydraulic Experiment

OCNGS and NMP-1's LBPCTs of 2150°F[34] and 2149°F,[35] respectively, would not provide a necessary margin of safety to help prevent a partial or complete meltdown,

[34] Exelon, "Oyster Creek Nuclear Generating Station 10 C.F.R. § 50.46 Annual Report for 2009," June 4, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML101550497, Attachment 1, p. 2.

[35] Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports," December 7, 2001, located at: www.nrc.gov, Electronic Reading

in the event of a LOCA. Experimental data indicates that OCNGS and NMP-1's LBPCTs must be decreased to temperatures lower than 1832°F in order to provide a necessary margin of safety.

a. The CORA-2 and CORA-3 Experiments

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),[36] leading the CORA-2 and CORA-3 bundles to maximum temperatures of 2000°C and 2400°C, respectively.[37]

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)],[38] 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.[39]

Room, ADAMS Documents, Accession Number: ML020170430, p. 1. "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," states that for prior 10 C.F.R. § 50.46 changes or error corrections for previous years ΔPCT was 0°F, for NMP-1; see Constellation Nuclear, "Nine Mile Point Nuclear Station Units 1 and 2 10 C.F.R. § 50.46(a)(3)(ii) Annual Reports for 2009," January 28, 2010, , located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML1003340589 , Attachment 1, p. 2,

[36] 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.

[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)," Forschungszentrum Karlsruhe, KfK 4378, September 1990, Abstract.

[38] 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.

[39] 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.

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...[40]

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."[41]

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.[42]

---

[40] 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.

[41] 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.

[42] 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.

Therefore, data from the CORA 2 and CORA 3 experiments indicates that OCNGS and NMP-1's LBPCTs must be decreased to temperatures 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.
>
> 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. 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.[43]

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

[43] 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.

> 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.[44]

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.
>
> As 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. 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.

---

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

[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.

(This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[46]

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)],[47] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C."[48] So the CORA 2 and CORA 3 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].[49]

(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).)

---

[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.

[47] 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.

[48] 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.

[49] 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.

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.[50]

So there are papers that report the autocatalytic oxidation of Zircaloy cladding by steam commences at temperatures below OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively. Therefore, in the event of LOCAs at OCNGS and NMP-1, if peak cladding temperatures reached temperatures between approximately 1832°F and 2150°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.[51]

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

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

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

---

[50] 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.

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

Second, Petitioner does not intend to use Dr. Henry's postulation that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident to support Petitioner's argument that OCNGS and NMP-1's LBPCTs should both be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Third, Petitioner is discussing what Dr. Henry postulated, because Petitioner finds it compelling that Dr. Henry postulated that an autocatalytic reaction commenced at 1000°C in the TMI-2 accident. In Dr. Henry's presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 American Nuclear Society/European Nuclear Society International Meeting, November 11, 2007,[52] Dr. Henry states that "[a]t about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal runaway state."[53]

Fourth, information presented in "TMI-2: A Textbook in Severe Accident Management," regarding the Zircaloy-steam reaction and core damage phenomena, does pertain to this 10 C.F.R. § 2.206 petition.

Fifth, it is significant that in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in this 10 C.F.R. § 2.206 petition—including the CORA experiments and LOFT LP-FP-2 experiment.

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

---

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

[53] *Id.*

"TMI-2: A Textbook in Severe Accident Management": Dr. Powers and Fauske, D.Sc., are also clearly very knowledgeable about severe accident phenomena.)

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

> Fuel Cladding Oxidation
>
> - As the boil-off of the water in the core continued, the uncovered region continued to heatup with the highest cladding/fuel temperatures being at about the 3/4-core height location.
>
> - Increasing temperatures caused the Zircaloy oxidation rate to increase which was accompanied by an increased release rate of chemical energy.
>
> - *At about 1000°C, the oxidation energy release rate equaled the decay power. From this point on, the core was in a thermal-runaway state.* During this interval the Zircaloy reaction was limited by the rate of steam generated in the covered part of the core which decreased as the water level decreased [emphasis added].[54]

So Dr. Henry postulated that runaway oxidation commenced at approximately 1000°C. And another one of the presentation slides from "TMI-2: A Textbook in Severe Accident Management," states that "[t]he chemical energy release [from the oxidation of the Zircaloy fuel cladding by steam] caused the core to overheat faster and eventually melt or liquefy the individual constituents."[55]

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

> Fuel Cladding Oxidation
>
> - The Zr in the Zircaloy cladding will oxidize in a high temperature steam environment: hydrogen and energy (heat) are released by this reaction:
>
> $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \Delta H_R$
>
> - The heat of reaction, ΔH_R, is about 6.5 MJ/kg.
>
> - At about 1000°C, the rate of chemical energy release approximately equals the decay power.

---

[54] *Id.*
[55] *Id.*

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

- Therefore, the core damage was generally caused by the cladding oxidation.[56]

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

Example: Core Heatup Rate Escalation Due to Cladding Oxidation

- Important Tests:

- Out-of-Reactor: CORA

- In-Reactor: [PBF] SFD, FLHT, LOFT LP-FP-2, and PHEBUS[57]

So in "TMI-2: A Textbook in Severe Accident Management," Dr. Henry cites some of the same experiments that are discussed in this 10 C.F.R. § 2.206 petition—including the CORA experiments and LOFT LP-FP-2 experiment. And it is compelling that Dr. Henry postulated that autocatalytic oxidation of Zircaloy cladding by steam commenced at 1000°C in the TMI-2 accident—368°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(As stated above, the alleged conservatism of OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, is predicated on the premise that the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F would provide a necessary margin of safety in the event of LOCA.)

c. National Research Universal Thermal-Hydraulic Experiment 1

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

[56] *Id.*
[57] *Id.*

TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[58]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[59] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[60] "low enough that superheated steam conditions [would] exist at the loop inlet instrument location. The superheat requirement [was] imposed so that meaningful steam temperatures [could] be measured."[61])

In TH-1 test no. 130, there was a reflood rate of 0.7 in./sec. At the start of reflood, the PCT was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[62]

In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F.[63] So because of the heat generated from the metal-water reaction, the peak cladding temperature increased by 190°F, after the reactor shutdown.

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

---

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

(It is significant that TH-1 test no. 130 was driven by an amount of fission heat that would simulate decay heat: the average fuel rod power of TH-1 test no. 130 would have been 0.37 kW/ft,[64] in the pre transient phase of the test.)

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

The data of TH-1 test no. 130 indicates, in the event of a LOCA, at either OCNGS or NMP-1, with high probability, if peak cladding temperatures reached temperatures of approximately 1850°F, the Zircaloy cladding would begin to rapidly oxidize, and that—with the combination of heat generated by the metal-water reaction and decay heat—the oxidation would become autocatalytic and cladding temperatures would start increasing at a rate of tens of degrees Fahrenheit per second. Within a period of approximately 60 seconds peak cladding temperatures would increase to 3000°F or greater; the melting point of Zircaloy is approximately 3308°F.[65]

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

d. 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;"[66] 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

[64] *Id.*, p. 10.
[65] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," p. 3-1.
[66] 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.

per second) due to the exothermal heat produced by the Zry cladding oxidation in steam environment."[67]

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 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.[68]

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.

---

[67] *Id.*, p. 1.
[68] *Id.*, p. i.

> 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.[69]

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.

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,[70] 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.[71]

And regarding "experiment-specific analytical modeling at [ORNL] for CORA-16,"[72] "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].[73]

---

[69] *Id.*, p. 5.
[70] 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.
[71] *Id.*, p. 2.
[72] *Id.*, p. 3.
[73] *Id.*

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."[74]

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...[75]

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," analyses that used 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."[76]

e. 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.[77]

---

[74] 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.
[75] *Id.*
[76] L. J. Ott, Oak Ridge National Laboratory, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.
[77] 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

(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 with solid UO_2 pellets to correspond to LWR fuel rods.[78] 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].[79]

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].[80]

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).

[78] 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.

[79] 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.

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

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 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].[81]

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*

[81] 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.

> *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].[82]

So "Behavior of AgInCd Absorber Material in Zry/UO_2 Fuel Rod Simulator Bundles Tested at High Temperatures in the CORA Facility" and "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" both state that temperature escalations due to the exothermic Zircaloy-steam reaction began at approximately 1100°C (2012°F). "Results of SFD Experiment CORA-13 (OECD International Standard Problem 31)" also states that "having reached about 1000°C [(1832°F)], the exothermal Zry/steam reaction adds an increasing contribution to the energy input, resulting in a temperature escalation."[83] 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."[84]

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].[85]

---

[82] *Id.*, p. 12.
[83] *Id.*
[84] 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.
[85] *Id.*, p. 5.

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)...[86]

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."[87]

"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."[88]

---

[86] *Id.*, p. 7.
[87] 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.
[88] *Id.*, p. 87.

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."[89] 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 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).

f. 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."[90]

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."[91] 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.[92]

---

[89] *Id.*, p. 83.
[90] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.
[91] 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.
[92] *Id.*

The LOFT LP-FP-2 experiment had an initial heatup rate of ~1 K/sec.[93] It is significant that "heatup rates [of 1 K/s or greater] are typical of severe accidents initiated from full power."[94] 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).[95]

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.[96]

---

[93] *Id.*

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

[95] 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.

[96] 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).[97, 98]

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;[99] 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).[100]

Therefore, after the onset of rapid oxidation—after a heating rate of ~1 K/sec.[101]—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

---

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

[98] 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.

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

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

[101] 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.

rate of approximately 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)]."[102]

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."[103] 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)]."[104] 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).[105]

---

[102] 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.

[103] 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.

[104] 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.

[105] 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;[106] 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."[107]

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..."[108] 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."[109] 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.

[106] 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.

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

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

[109] 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.[110]

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.[111] 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,"[112] 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.[113, 114]

---

[110] 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.

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

[112] 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.

[113] 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.

[114] 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,"[115] t]his event is believed to be the rupture of the control rod cladding.[116]

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.[117]

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.

---

[115] 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.
[116] A. B. Wahba, "Instrumentation Capabilities during the TMI-2 Accident and Improvements in Case of LP-FP-2," p. 136.
[117] *Id.*, p. 147.

2. 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].[118]

Furthermore, Figure 1[119] 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),[120] and Figure 13[121] 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.[122]

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"),[123] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[124] conducted with a

---

[118] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 195.
[119] See Appendix F Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.
[120] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 196.
[121] See Appendix F Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core.
[122] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 205.
[123] The principal technical spokesmen of Consolidated National Intervenors were Henry Kendall and Daniel Ford of Union of Concerned Scientists.
[124] 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."[125]

(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].[126]

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].[127]

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

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

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

[127] *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...[128] 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.[129]

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

---

[128] 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.

[129] 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."[130] 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.[131] 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%,[132] of the total energy input.[133]

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."[134])

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.' "[135, 136] It is significant that "in the CORA

---

[130] 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.

[131] 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.

[132] 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.

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

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

[135] 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:"[137] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[138]

Furthermore, the graphs of "Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies"[139] and "Analysis of Zr2K Thermal Response"[140] 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

---

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

[137] 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.

[138] 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.

[139] 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.

[140] 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'[141] associated with the Zr2K test."[142] 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.

(It is noteworthy that in the AEC's ECCS rulemaking hearing, the AEC agreed with GE that the thermocouple measurements of the cladding-temperature excursions taken during the Zr2K test were not valid.

Regarding the maximum cladding temperature in the Zr2K test, the AEC Commissioners concluded:

> The conditions in [the BWR FLECHT Zr-2 test] were stated to be significantly more severe than the conditions reasonably expected to prevail during a postulated BWR LOCA, even for the "hot" bundle (Exhibit 1148, p. P-15). In Zr-2, the maximum cladding temperature was approximately 2250°F, and 39 out of the 49, or 80%, of the heater rods perforated (Exhibit 1069, pp. 53-54).[143])

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,

---

[141] 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.

[142] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," pp. A8-24, A8-27.

[143] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1104-1105. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

> 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].[144]

Discussing the "well defined inter-rod correlations"[145] that occurred during "the extreme temperature excursion,"[146] "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

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

---

[144] *Id.*, p. A8-19.
[145] *Id.*
[146] *Id.*, p. A8-21.

> 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].[147]

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

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

> Based upon analysis of the material presented, it appears unquestionable that the [thermocouple] response was badly affected by short circuits and equipment malfunction. The net result is that it is not possible to certify that [metal-water] reactions were insignificant in the measured thermal transient, but the case for near "thermal runaway" proposed by the CNI is also unconvincing. It is probable that most of the dramatic [thermocouple] slope changes, as well as several of the other [thermocouple] aberrations 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].[148]

Indeed, "a more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable."[149] 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

---

[147] *Id.*, pp. A8-21, A8-23.
[148] *Id.*, p. A8-27.
[149] *Id.*

"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.' "[150, 151]

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 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].[152]

(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.")

---

[150] 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.

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

[152] 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.

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.[153] 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].[154, 155]

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 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]."[156]

---

[153] 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.

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

[155] 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.

[156] *Id.*

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 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].[157]

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]."[158]

---

[157] *Id.*, pp. 5.12, 5.14.
[158] *Id.*

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.[159]

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[160]—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-of-coolant accident."[161] 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.[162] 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.).[163]

---

[159] *Id.*, p. 5.41.
[160] 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.
[161] 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.
[162] *Id.*, p. 3.1.
[163] *Id.*, pp. 4.1-4.2.

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.[164]

"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;"[165] *i.e.*, the cladding temperature excursion began at 17:09:42. However, "FLHT-1 Test Report" also states that the cladding temperature excursion began 18 seconds latter at 17:10:00—when the cladding temperature was 1700 K.[166] 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]

---

[164] *Id.*, p. v.
[165] *Id.*, p. 4.6.
[166] *Id.*, p. 4.11

10 K/sec., signaling the onset of an autocatalytic oxidation reaction."[167] 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)."[168] According to a different account, in the LOFT LP-FP-2 experiment, the onset of rapid oxidation occurred at approximately 1500 K (2240°F).[169] 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.[170]

Furthermore, although the graphs of "Typical Cladding Temperature Behavior"[171] and "Pseudo Sensor Readings for Fuel Peak Temperature Region"[172, 173] are not large enough to clearly delineate what the temperature values were at given times during the FLHT-1 test, the graphs' cladding-temperature values are consistent with the postulation that the temperature excursion began at a temperature far lower than 1700 K, at a temperature closer to 1520 K (see Appendix 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

---

[167] 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.

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

[169] 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.

[170] 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.

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

[172] *Id.*, p. 5.3.

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

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.[174]

First, it is obvious from the above description and from Figures 4.1 and 5.4 that when cladding temperatures reached approximately 1475 K (2200°F)—and the coolant flow rate was increased—that "a stabilized condition" was not achieved. Cladding temperatures continued to rise. This is clearly stated: "The flow was increased in steps and reached a maximum of about 15 kg/hr. (34 lb/hr.). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly…"[175]

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

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

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."[176] 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)."[177]

(It is noteworthy that other reports state that the MT-6B test had a PCT of 1400 K (2060°F)[178] and 1280°C (2336°F) (1553 K).[179] So the MT-6B test may have actually demonstrated that the Zircaloy oxidation rate was easily controllable by adding more coolant at cladding temperatures of either 2060°F (1400 K) or 1280°C (2336°F) (1553 K).)

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

> Once the power is set, the test will be started through its transient operation. *The term transient is somewhat of a misnomer*; operation will consist of a series of preplanned, discrete flow-reduction steps. The size and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate.
> ...

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

> 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].[180]

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."[181]

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"[182] and "Pseudo Sensor Readings for Fuel Peak Temperature Region,"[183] 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

---

[180] W. N. Rausch, *et al.*, "Full-Length High-Temperature Severe Fuel Damage Test 1," pp. 4.3-4.5.
[181] *Id.*, p. 4.6.
[182] *Id.*, p. 4.7.
[183] *Id.*, p. 5.3.

> 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.[184]

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.[185]

It is also noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis" states that "[i]n the [FLHT-1] test, completed in March 1985, 12 ruptured zircaloy-clad rods were subjected to an autocatalytic temperature excursion. From the measurements made on the full-length rods during the test, the autocatalytic reaction was initiated in the 2500-2600°F (1371-1427°C) temperature region."[186]

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.

---

[184] *Id.*, pp. 4.6-4.7.
[185] *Id.*, p. 4.7.
[186] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

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.[187]

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].[188]

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.)"[189] "Degraded Core Quench: A Status Report" states that the temperature excursion occurred in steam-rich conditions, after an initial heatup phase in pure helium (up to 1000°C), and that the PCT was approximately 1900 K, during the first oxidation phase. The PHEBUS B9R-2 test had a second oxidation phase and temperature escalation.[190]

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

---

[187] 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.

[188] *Id.*

[189] 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.

[190] 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.

cladding temperatures were below 1477 K (2200°F)[191] (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 claddings. The so liquefied zircaloy interacted with the UO_2 fuel pellets [emphasis added].[192]

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.

---

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

[192] 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.

> 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].[193]

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.[194]

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 boride phase is clearly visible as a border around the B_4C-particles. The B_4C-particles are dissolved chemically by it.[195]

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.*[196] 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

---

[193] *Id.*, p. 10.
[194] *Id.*, p. 11.
[195] *Id.*, p. 12.
[196] W. Hering, P. Hofmann, "Material Interactions During Severe LWR Accidents; Summary of Separate-Effects Test Results," KfK 5125, 1994.

> 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].[197]

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.[198]

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.[199] 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."[200]

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."[201]

---

[197] 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.
[198] *Id.*, p. 15.
[199] *Id.*, p. 11.
[200] *Id.*, p. 15.
[201] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," Oak Ridge National Laboratory, 1997, pp. 4-5.

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,[202] 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.[203]

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"[204] 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.)

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].[205]

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.

---

[202] 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.
[203] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 8.
[204] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 196.
[205] 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.

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].[206]

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.

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),[207] 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)[208]—is further evidence that OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, would

---

[206] 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.

[207] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," pp. 4-5.

[208] 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.

not provide a necessary margin of safety to help prevent a partial or complete meltdown, in the event of a LOCA.

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

a. 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."[209]

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 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.[210]

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

[209] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 7.
[210] *Id.*, pp. 6-6, 6-7, 6-8.

higher discharge burnups, longer lived components, and provide improved plant operating margins.[211]

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 burnup[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)."

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

First, current 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 current 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 BWR 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.

---

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

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[212]...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;[213] 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[214] are not applicable.[215]

So the results of the CORA-16, CORA-17, and CORA-18 experiments provide a good indication of the damage current 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.

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

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

---

[212] R. P. Elliott, "Constitution of Binary Alloys," First Supplement, McGraw-Hill Book Company, 1965.

[213] 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.

[214] *Id.*

[215] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 8.

> metal-water reaction I am concerned with at the moment is the reaction between the Zircaloy-Inconel eutectic and steam, I am concerned to find out how the Applicant's analysis contained in the computer code, which does not consider this, how it would be different if it did.
>
> Leonard M. Trosten: I thank you for the explanation. I recognize this as being one of the principal points of concern in the critique by the Union of Concerned Scientists...[216]—IP-2 licensing hearing, November 1971

It is significant that "[t]he chemical reaction between Inconel and Zircaloy influences the meltdown of the reactor core in the vicinity of Inconel grid spacers."[217]

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

> *Grid spacers can have a significant impact on the progression of damage in a reactor core during a severe accident.* ... The impact of grid spacers on damage progression has been revealed by out-of-pile experiments in Germany[218] and Japan,[219] in-pile experiments at the PBF facility in Idaho,[220] and by examinations of the damaged Three Mile Island (TMI-2) core.[221] *The experiments in Germany and Japan have revealed the existence of chemical interactions between Inconel and Zircaloy that take place at temperatures as low as 1273 K [(1832°F)]*, more than 200 K lower than the melting temperature of Inconel. *Thus in a reactor core*

---

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

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

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

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

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

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

with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers [emphasis added].[222]

It is significant that grid spacers would effect the progression of damage in a reactor core during a LOCA if temperatures were to reach approximately 2012°F;[223] and significant that experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F: well below OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

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

> The Zircaloy/stainless steel (1.4919; corresponds to [stainless steel] Type 316 with 18 wt% Ni and 8 wt% Cr) interactions are important with respect to the contact between the absorber rod cladding and the Zircaloy guide tube and between the Inconel spacer grid and the Zircaloy fuel rod cladding. In both cases, the iron-zirconium and the nickel-zirconium phase diagrams show that due to eutectic interactions, early melt formation has to be expected, which initiates the melt progression within the fuel assembly at low temperatures. *Liquid phases form at temperatures <1000°C; however, the reaction kinetics become significant only above 1100°C.* This was seen in the CORA tests, where fuel rod bundles were heated up to complete meltdown. In all cases, the damage of the bundle was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions. *Localized liquefaction of these components started around 1200°C.*[224]
>
> The reaction kinetics between Zircaloy and stainless steel can be divided into a reaction zone growth rate in Zircaloy and one in stainless steel, as shown in Fig. 11. One can see that the Zircaloy is attacked more strongly than the stainless steel. Oxide layers on the Zircaloy cladding outside diameter delay the chemical interactions between Zircaloy and steel, but they cannot prevent them. *The influence of oxide layers becomes less important at temperatures >1100°C, since the dissolution of the protecting ZrO_2 layers occurs rather fast and the stainless steel is then in contact with metallic Zircaloy or oxygen-stabilized α-Zr(O).*[225]

---

[222] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," p. 427.
[223] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[224] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[225] P. Hofmann, M. Markiewicz,, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Stainless Steel at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 5106, 1994.

> In a first approach, the reaction behavior of Zircaloy with Inconel 718 is comparable to that with Type 316 stainless steel.[226] *At temperatures <1100°C, Inconel attacks the Zircaloy faster than stainless steel; above 1100°C, the situation is the reverse.* In both cases, the melting of a relatively large quantity of Zircaloy with limited melting of the adjacent stainless steel or Inconel takes place. *During heat-up of the stainless steel/Zircaloy and Inconel/Zircaloy reaction systems, a sudden and complete liquefaction of the specimens occurs at temperatures slightly above 1250°C. This may be the reason that melt progression in a fuel rod bundle initiates at absorber rod cladding (stainless steel)/Zircaloy guide tube contact areas and Inconel spacer grid/Zircaloy fuel rod contact locations*[227] [emphasis added].[228]

It is significant that in the CORA tests, in which fuel rod bundles were heated up to complete meltdowns, that "the damage of the [bundles] was initiated due to Zircaloy/stainless steel and Zircaloy/Inconel interactions"[229] and that "[l]ocalized liquefaction of these components started around 1200°C [(2192°F)]."[230] It was also observed in the CORA tests that "[l]iquid phases form at temperatures <1000°C [(1832°F)]" and that "the reaction kinetics become significant only above 1100°C [(2012°F)]."[231]

So clearly, in the event of a LOCA, BWR core component damage could commence at temperatures lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

F. BWR Thermal Hydraulic Experiments and Core Spray Cooling

> 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

---

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

been demonstrated nor has it been proven ineffective.[232]—J. W. McConnell

It seems that after the BWR-FLECHT program was concluded about forty years ago that there have not been any BWR heat transfer experiments conducted with parameters realistic enough to conclusively demonstrate that BWR core spray systems would be effective, in the event of a LOCA. Perhaps all of the primary BWR heat transfer experiments conducted after the BWR-FLECHT program was concluded were conducted with multi-rod Inconel 600 bundles.

So it seems that after the BWR-FLECHT program was concluded that there have not been any BWR Zircaloy multi-rod heat transfer experiments conducted with bundles that were heated up to peak cladding temperatures of 2150°F: the values of OCNGS and NMP-1's LBPCTs are 2150°F and 2149°F, respectively. Unfortunately, it seems that it has never been conclusively demonstrated that OCNGS and NMP-1's ECCSs would work effectively if their cores' peak cladding temperatures reached their respective LBPCTs. This is highly problematic, because, if a multi-rod Zircaloy bundle were heated up to maximum temperatures between 1832°F and 2150°F, it would (with high probability) incur autocatalytic oxidation. In the event of a LOCA, if autocatalytic oxidation occurred, at either OCNGS or NMP-1, it would lead to a partial or complete meltdown.

To overcome the impression left from the BWR FLECHT program, BWR heat transfer experiments need to be conducted with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Such BWR heat transfer experiments need to be conducted in experiments modeling BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs.

(It is noteworthy that there should be a regulation stipulating minimum allowable amounts of coolant to be supplied to each fuel bundle in the BWR core, in the event of a LOCA.[233])

---

[232] 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.

[233] "Resolution of Generic Safety Issues: Item A-16: Steam Effects on BWR Core Spray Distribution" states that "to ensure the health and safety of the public, [BWR] core spray systems

1. Appendix K BWR Heat Transfer Coefficients

Appendix K to Part 50, ECCS Evaluation Models, I(D)(6), *Post-Blowdown Phenomena, Heat Removal by the ECCS, Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling*, states:

> Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:
>
> a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.
>
> b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot {}^{\circ}F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.
>
> c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot {}^{\circ}F^{-1}$ shall be applied to all fuel rods.

It is significant that Appendix K convective heat transfer coefficients for BWR Zircaloy fuel rods under spray cooling are based on data from the BWR Full Length Emergency Cooling Heat Transfer ("FLECHT") tests—from tests conducted with stainless steel electrically heated fuel rod simulators.

Regarding the fact that Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are based on BWR FLECHT tests conducted with stainless steel electrically heated fuel rod simulators, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> *From the BWR FLECHT tests there is information on the heat transfer coefficients for both the convective heat flow to the water droplets and steam and for the reflood phase.* The FLECHT tests were made with an electrically heated mock-up of a 7 x 7 rod array complete with its channel box. *The convective heat transfer coefficients were determined from the*

---

must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores."

residue of a thermal balance after all of the known inputs and outputs were calculated. The factors considered were the electrical heat input, the rate of change of the heat content of the rods as calculated from their temperature history, and the calculated radiation from the rods to each other and to the channel walls. The residue from these inputs and outputs was ascribed to convective heat transfer. The convective heat transfer coefficients so determined could not be very accurate because their calculation involved taking the difference between two large numbers. The coefficients so obtained are small and are about what one would expect from the mechanisms of natural convection and radiation to steam (Exhibit 1113, p. 16-14).

The values of the calculated convective heat transfer coefficients depend to some extent upon the value used for the thermal emissivity of the stainless steel, since the convective heat transfer is obtained after subtracting the radiative heat transfer from the total. Theoretically a high value of the emissivity leads to a low calculated convective heat transfer coefficient. Values of the emissivity measured after the tests ranged from 0.6 to 0.9 (Exhibit 461, p. 81 and Exhibit 1113, p. 16-14), and to add conservatism to the calculation, the Interim Policy Statement required the use of the highest measured emissivity, 0.9, for the calculation of the convective heat transfer coefficients. However, it turned out that this resulted in a higher coefficient (less conservative) for the critical inner rods, with a higher estimated standard error (Exhibit 461, Table 2). After reviewing the derivation of the coefficients as given in Exhibit 461, we believe that those originally listed as best estimates by General Electric are the most credible and should be used. The effect of this change on the peak cladding temperature will be small, about five degrees according to Exhibit 461.

There has been a great deal of criticism of the BWR FLECHT tests, particularly by the Consolidated National Intervenors (Exhibit 1041, Chapter 5), and both General Electric and the Regulatory Staff have defended them (Closing Statements). However, for the purpose of calculating the maximum cladding temperature, only the derived heat transfer coefficients are of any great importance. The values obtained have always been known to have a high statistical error; furthermore, the values are low and reasonable, and there seems little to be gained by renewing the controversy over the manner of conducting and interpreting all features of the tests.

The high but inevitable statistical error of the coefficients for the inner rods (1.5 ± 1.0 BTU/hr·ft^2·°F) is bothersome and leads to an estimated error band of as much as ±200°F in the calculated peak temperature in some circumstances (Exhibit 1113, p. 16-36). *The test bundle SS2N was used to derive the heat transfer coefficients*; another test bundle, SS4N,

> resulted in cladding temperatures 200°F higher than those of the bundle used as a standard; one half of this discrepancy could be explained by test differences, with the other half left to be attributed to statistical variations (Exhibit 1113, p. 16-38). The problem of these large statistical errors in the convective heat transfer coefficients is compensated to some extent by the fact that the coefficients were determined at atmospheric pressure, whereas the reactor would be at some elevated pressure at which the heat transfer would be improved (Exhibit 1113, p. 16-26).
>
> The evidence for the value 25 BTU/hr·ft^2·°F of the two phase reflooding heat transfer coefficient is sketchy (Exhibit 1032, p. II 6.3-51), but it is applied for only a short time because the high reflood rate would quickly quench the core, and the exact value is of little significance [emphasis added].[234]

So "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[t]he [BWR FLECHT] test bundle SS2N was used to derive the [Appendix K] heat transfer coefficients"[235] for BWR Zircaloy fuel rods.

(In the name "SS2N," "SS" stands for "stainless steel" and "N" stands for "Nichrome.")

And also regarding Appendix K heat transfer coefficients for BWRs, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "the heat transfer coefficients utilized in the GE core spray and reflood calculation model,[236] were derived on the basis of the SS2N test series."[237, 238]

---

[234] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1125-1126. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[235] *Id.*, p. 1126.

[236] 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, p. 58.

[237] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.

[238] 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. A8-10.

In the BWR FLECHT SS2N test series, conducted from August to October 1969, three steady state tests were conducted with a peak power of 150 kW and coolant rates of 1.0-2.45 gallons/min.; 24 transient tests were conducted with peak powers of 100-250 kW, coolant rates of 2.45-5.0 gallons/min., and initial temperatures of 865-1850°F; eight combined spray and flooding tests were conducted with peak powers of 235-250 kW, coolant rates of 2.0-3.5 gallons/min. and 2.0-6.0 in./sec., and initial temperatures of 1335-1870°F.[239]

In the BWR FLECHT tests, five tests were conducted with Zircaloy electrically heated fuel rod simulators; however, Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are not based on the data from the five Zircaloy tests.

Explaining the purpose of the five BWR FLECHT Zircaloy tests "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> [I]t was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions. The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.[240]

Discussing the PWR FLECHT tests, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> [T]he Commission sees no basis for concluding that the heat transfer mechanism is different for zircaloy and stainless steel, and believes that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods.[241]

---

[239] 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. A8-5.

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

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

It is significant that the Atomic Energy Commission, also concluded that heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods in BWRs.

Regarding the problems with the heat transfer coefficients derived from the SS2N experiments with stainless steel fuel rod simulators, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> It seems probable that the difference between test and theory results from rigid adherence by GE to a time-dependant model of heat transfer coefficients which were derived from their SS2N tests and adopted as their "design model."[242] The design analysis method, based on the SS2N time history, apparently did not permit accommodation of the idiosyncrasies of the Zr2K test experience with its rod heater failures and [thermocouple] equipment malfunctions. Consequently, the predicted results might not reasonably be expected to correspond well with the reality of the Zr2K test. Whether or not design basis production of LOCA thermal histories would agree well with an actual transient also remains to be shown. Results imply that the GE thermal analysis method may be a weak predictive tool and more effort appears to be needed in model development. However, it does appear that with sufficient analysis, FLECHT results would be adequate to form a basis for demonstrating the development of conservative analytical design methods.[243]

2. Appendix K BWR Heat Transfer Coefficients for New BWR Fuel Assembly Designs

It is significant that Appendix K specifies that its BWR heat transfer coefficients are to be used for fuel rods in a 7 x 7 fuel assembly array. Since Appendix K was written, new BWR fuel assembly designs have come into use, so Appendix K BWR heat transfer coefficients have been converted so that they can also apply to new BWR fuel assembly designs.

---

[242] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.

[243] 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, pp. A8-27, A8-28.

Discussing the application of heat transfer coefficients to various BWR fuel assembly designs, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" states:

> Although the channel size has not changed significantly since the 1970's, the BWR fuel assembly designs have changed in many ways. These changes have resulted in a larger number of smaller diameter fuel rods as well as various non-boiling water channel designs. ... Spray heat transfer tests have been performed (*e.g.*, the BWR FLECHT test program) from which convective spray heat transfer coefficients have been derived. CENPD-283-P-A...provides a summary of these tests *and describes how the spray cooling heat transfer coefficients are applied to various fuel geometries*. ... The BWR FLECHT tests, which simulated a 7x7 array, showed that the convective coefficients are dependant on the location of the fuel rod relative to its proximity to the channel enclosure (corner rod, outer row rod, or interior rod). Table 6-2 lists the heat transfer coefficients that are acceptable for use in an Appendix K analysis of 7x7 fuel [emphasis added].[244]

(Table 6-2, Appendix K Spray Cooling Heat Transfer Coefficients, states that the values for heat transfer coefficients are: for corner rods—17.0 $W/m^2 \cdot K$, for side rods—19.9 $W/m^2 \cdot K$, for inner rods—8.5 $W/m^2 \cdot K$, and for channel—28.4 $W/m^2 \cdot K$.[245])

It is significant that BWR FLECHT spray heat transfer coefficients for 7x7 fuel assembly arrays have been converted so that they can be used for 8x8 fuel assembly arrays.[246] It certainly stands to reason that BWR FLECHT spray heat transfer coefficients for 7x7 fuel assembly arrays have also been converted so that they can be used for 9x9 and 10x10 fuel assembly arrays.

---

[244] John A. Blaisdell, Westinghouse, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-NP-A, November 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050390435, p. 30.

[245] John A. Blaisdell, Westinghouse, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-NP-A, November 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050390435, p. 31.

[246] John A. Blaisdell, Westinghouse, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-NP-A, November 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050390435, p. 31.

3. Criticisms of the BWR FLECHT Tests

Discussing one of Henry Kendall and Daniel Ford's, of Consolidated National Intervenors ("CNI"),[247] criticisms of the BWR-FLECHT tests, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The first complaint [regarding 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].[248]

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].[249]

(It is noteworthy that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400°K. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of

---

[247] Henry Kendall and Daniel Ford of Union of Concerned Scientists were the principal technical spokesmen of Consolidated National Intervenors, in the AEC ECCS rulemaking hearing.
[248] 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, pp. A8-2, A8-6.
[249] *Id.*, p. A8-6.

Zircaloy;"[250] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425°K. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain*" [emphasis added].[251])

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 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]."[252]

So J. W. McConnell concluded that "the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven."[253]

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

> [Another] reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating the [metal-water] reaction from the physical processes of cooling rods

---

[250] 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.
[251] *Id.*, p. 4.4.
[252] 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.
[253] *Id.*

> which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the assumption that the radiative and convective heat transfer processes for heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions.* The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.
>
> The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, *questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms* [emphasis added].[254]

(It is significant that in the ECCS rulemaking hearing, the Atomic Energy Commission ("AEC") Commissioners did not seem concerned about decoupling the zirconium-water reaction from cladding heat transfer mechanisms. The AEC Commissioners merely concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients. Regarding the AEC Commissioners' conclusion, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

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

[254] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-7.

coefficients, which depend on conditions in the coolant outside the rod [emphasis added].[255]

So the AEC Commissioners concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients, maintaining that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside fuel rods. (Petitioner discusses the fallacy of the AEC Commissioners' conclusion in the following section.))

It is significant that J. W. McConnell concluded that "from a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective"[256] in the BWR-FLECHT program.

4. The Fallacy of the AEC Commissioners' Conclusion that the Heat Generated from the Exothermic Zirconium-Water Reaction would Not Affect the Coolant Outside Fuel Rods

To discuss the fallacy of the AEC Commissioners' conclusion that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside fuel rods, Petitioner will discuss PWR FLECHT Run 9573. Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.

Run 9573 was part of the PWR FLECHT test program; however, the exothermic zirconium-water reaction that occurred in the test is pertinent to both PWR and BWR Zircaloy fuel rods in LOCA environments. It is significant that a Zircaloy assembly used in the BWR FLECHT program—the Zr2K test assembly—also incurred autocatalytic oxidation. (The BWR FLECHT Zr2K test is discussed in the following section.)

---

[255] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1123-1124. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[256] 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.

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

> [T]he Commission sees no basis for concluding that the heat transfer mechanism is different for zircaloy and stainless steel, and believes that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods. It is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[257]

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"] when compared to the equivalent stainless steel tests."[258] The NRC also stated, regarding the results of FLECHT run 9573, that the AEC Commissioners were not "concern[ed] about the zirconium-water reaction models."[259]

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

> The second reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating the [metal-water] reaction from the physical processes of cooling rods which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the*

---

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

[258] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 16-17.

[259] *Id.*, p. 17.

assumption that the radiative and convective heat transfer processes for heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions. The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.

The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms [emphasis added].[260]

And opining on the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The reasonable conclusion was reached that the effect of the difference between Zircaloy and stainless steel, if any, would be small. There is a difference, of course, in the rate of heat generation from steam oxidation, but this heat is deposited within the metal under the surface of the oxide film. *The presence of this heat source should not affect the heat transfer coefficients, which depend on conditions in the coolant outside the rod.*[261]

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

---

[260] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-7.

[261] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1123-1124. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

It is significant that within the first 18.2 seconds of FLECHT run 9573,[262] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[263] *i.e.*, more heat was transferred into the bundle midplane than was removed from that location. In petition for rulemaking 50-76 ("PRM-50-76"), Robert H. Leyse, the principal engineer in charge of directing the Zircaloy FLECHT tests and one of the authors of "PWR FLECHT Final Report," states that "[t]he negative heat transfer coefficients [occurring within the first 18.2 seconds of run 9573] were calculated as a result of a heat transfer condition during which more heat was being transferred into the heater than was being removed from the heater[; used in the FLECHT tests to simulate fuel rods]. And the reason for that condition was that the heat generated from Zircaloy-water reactions at the surface of the heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[264]

So the heat generated from the exothermic oxidation reaction of the Zircaloy cladding (and Zircaloy spacer grids) was transferred from the cladding's reacting surface inward. Indeed, the Zircaloy-cladding heater rods were very hot internally, where the thermocouples were located; yet, nonetheless, the heater rods became a heat sink.[265]

Additionally, the exothermic oxidation reaction of the Zircaloy heated a mixture of steam and hydrogen, and entrained water droplets. Westinghouse agrees with this claim; in its comments regarding PRM-50-76, Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[266]

---

[262] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[263] *Id.*, p. 3-98.
[264] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.
[265] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.
[266] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

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

> At the time of the initial [heater element] failures, midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure).[267]

Therefore, it is reasonable to conclude that a superheated mixture of steam and hydrogen, and entrained water droplets, caused heating of Zircaloy cladding in the midplane location of the fuel rod. It is also reasonable to conclude that the "negative heat transfer coefficients [that] were observed at the bundle midplane for 5...thermocouples"[268]—the occurrence of more heat being transferred into the bundle midplane than was removed from that location—within the first 18.2 seconds of FLECHT run 9573, were caused by an exothermic zirconium-water reaction. Additionally, it is reasonable to conclude that "the impression left from [FLECHT] run 9573" cannot be separated from concerns about zirconium-water reaction models.

Furthermore, because, as Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam,"[269] the AEC Commissioners' conclusion that "the presence of...heat [generated from the exothermic zirconium-water reaction] should not affect...heat transfer coefficients, which depend on conditions in the coolant outside the rod"[270] is erroneous. Clearly, the exothermic zirconium-water reaction affects the coolant outside the cladding by heating a mixture of steam and hydrogen, and entrained water droplets; therefore, the zirconium-water reaction cannot legitimately be separated from cladding heat transfer mechanisms.

---

[267] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[268] *Id.*, p. 3-98.
[269] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[270] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

5. More Criticisms of the BWR FLECHT Tests

Regarding the BWR-FLECHT Program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"—section V.G.2., CNI Guidance for Commission on Information Needs for LOCA Analysis, FLECHT Program, BWR-FLECHT—states:

> The BWR-FLECHT Program was carried out by the General Electric Company (GE) under a subcontract of the Idaho Nuclear Corporation, itself a contractor to the test program sponsor, the AEC. Roger Griebe of [Aerojet] coordinated the program, as project engineer, for [Aerojet], GE and the AEC. The UCS devoted a very substantial effort to an independent analysis of the BWR-FLECHT Program and its weaknesses. To CNI's knowledge this has been the only independent review of the program which has been carried out and made available in the public literature. CNI believes that the program failures and weaknesses that are identified in CNI testimony (Exhibit 1041) are overwhelmingly supported by the testimony of the knowledgeable engineers in the AEC contract laboratories who were associated and familiar with the elements of the program. CNI testimony sets out the case in substantial detail. In brief, the program was characterized by narrow scope, limited range of parameters investigated (many inappropriate to the tasks at hand), the use of incorrect materials, crude and incompetent instrumentation and operating techniques (with consequent major equipment malfunctions), and, as a culminating weakness, expansive and overgenerous interpretations. These latter, in CNI's view, misrepresented badly technical information available from the test results. In particular, in a test series of over 150 tests only one, ZR-2, simulated fuel rod swelling and rupture and the associated channel blockage which would be expected to occur under LOCA circumstances. It was a unique test, a circumstance which should not have occurred, and was highly defective. The importance of test ZR-2 was reflected in the hearing record in the extensive time taken by participants to discuss and to try to illuminate the nature and sources of the test weaknesses and to determine reliability what the test had to say.
>
> J.O. Zane of [Aerojet] did not believe ZR-2 was a demonstration of the ability of BWR ECCS to operate in a LOCA (Tr. 6415-6423).
>
> C.G. Lawson of ORNL said test ZR-2 was borderline and more tests were required employing pressurized fuel rods as in ZR-2 (Tr. 5719-5725).
>
> P.L. Rittenhouse of ORNL stated that it was unreasonable and arbitrary to conclude that test ZR-2 shows flow blockage would not inhibit the spray cooling system (Tr. 4757).

Roger Griebe, the engineer at [Aerojet] with perhaps the most familiarity with the BWR-FLECHT Program, said that General Electric did not have the enthusiasm he felt necessary to conduct the tests (Tr. 6935-6945) and that he could not personally defend the General Electric conclusions (Tr. 7006). He said that GE "overstated" points, became carried away with impressions not verified by the technical data, and that the General Electric conclusion that protection was provided by the ECCS against all break sizes was not completely supported in the FLECHT data (Tr. 7100 et. seq.). In cross-examination he stated that he felt the GE reporting of the data was "tremendously slanted" (Tr. 7117). [Aerojet]-GE-AEC internal memorandum released by CNI bearing on the conduct of the BWR-FLECHT tests tells an even more dismal story of the conduct and interpretation of these tests than is contained either in CNI testimony or in the oral transcript. Based on the careful reading of the memorandum in the light of CNI's analysis of the tests and of the cross-examination of [Aerojet] and GE witnesses, CNI has concluded that in effect GE tried to approach elements of the test program, and attempted to interpret the results, in ways wholly inconsistent with the technical content of the test program.

These [Aerojet] memos, incorporated in Exhibit 1153, note:

"This was not [a] satisfactory demonstration test—the same need exists today—in fact, the need is greater because margin appears to be less than originally expected."

"[GE's] role in this program can only be described as a conflict of interest... Because the GE systems are marginally effective...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."

"...the close coupling between GE-FLECHT Project Group [the testing group] and GE licensing group has precluded pursuing a completely objective experimental program in an expedient manner."

An internal investigation of the failure of some of the GE design test apparatus to function properly, concluded:

"...the 'why' of the situation has come down to the simple fact that we believed GE was doing the job for which they were paid... the GE effort in heater development has been demonstrated to be seriously inadequate."

CNI's conclusion has been that it has proven inappropriate and damaging for the AEC to have established a policy of letting industry do the testing

to check out the industry's own claims regarding safety system performance. The inherent conflict of interest has led to a testing program of narrow scope and poor quality.

One should note the letter of June 30, 1970 (Exhibit 1029) from William B. Cottrell, Director of the ORNL Nuclear Safety Program. In this letter to A.J. Pressesky:

"The Commission's position in its support of nuclear safety research is *seriously compromised* by relegating significant portions of the nuclear safety research and development program to the same industry it would license [emphasis added]."

Later in the letter, Cottrell cites examples known to him wherein a reactor vendor when given the responsibility for undertaking safety research on the reactor he was selling failed to get to the heart of the safety problems in question: In regard to fuel rod swelling in LOCA circumstances, the reactor vendors, on the basis of their own in-house R&D, concluded that the diametrical swelling of the fuel rods during the LOCA would be less than 30%. This they later increased to 60%. ORNL experiments subsequently demonstrated that swelling greater than 100% was possible under realistic conditions over significant portions of the core. Additionally, vendors maintained that embrittlement would not occur in the LOCA and hence its consequences need not be considered in evaluating the accident. Cottrell noted that ORNL experiments have been much more pessimistic in this regard.

With regard to [the] BWR-FLECHT Program, CNI concluded that in effect *GE tried to sabotage elements of the program* and attempted to interpret the results in ways utterly inconsistent with the test program's technical content. It is CNI's conclusion that the judgments set forth in its testimony are now even more powerfully supported by the hearing record [emphasis not added]. 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.[271]

(It is noteworthy that despite the testimony of a number of safety experts that ZR-2 did not demonstrate the ability of BWR ECCS to work effectively, the AEC Commissioners concluded:

[T]he data of the Zr-2 BWR FLECHT experiment were cited as *evidence for the effectiveness of spray cooling*, although no temperature

[271] *Id.*, pp. 5.37-5.41.

measurements were made at the positions of maximum bulging. We believe that additional assessments need to be made of these effects.

In addition to the primary heat transfer effects of taking into consideration the swelling and rupture of the cladding, there would be important secondary effects arising from the steam oxidation of the cladding by the steam. Higher temperatures would lead to increased oxidation, which would contribute to a further increase in temperature, and the opening in the cladding would allow oxidation on the inside, again increasing the calculated temperature [emphasis added].[272])

6. Criticisms of GE's BWR Core Spray Tests

Regarding problems with BWR core spray cooling, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing"—section V.I., CNI Guidance for Commission on Information Needs for LOCA Analysis, BWR Core Spray—states:

> CNI identified the weaknesses in simulations of BWR core spray cooling of a full-length bundle as implemented in GE's uniquely defective test ZR-2 (Exhibit 1041, p. 5.39). CNI raised the possible existence of core spray diversion mechanisms that could in a BWR LOCA result in spray starvation of the central and hotter fuel bundles of a core. Similar concerns have been raised by Aerojet (Exhibit 1032, p. 124). GE diligence in resolving these concerns leaves a substantial amount to be desired. Cross-examination established (Tr. 13,925 et. seq.) that GE had done *no experiments* to determine spray droplet size distribution *either* at spray nozzles or at bundle tops (Tr. 13,953-13,956). They have done *no experiments* to determine the degree of superheat of the ejected steam. However, they "assume" the steam is saturated. They have done *no experiments* to determine steam temperature at bundle exit *or* to determine steam velocity at the bundle top. Moreover, the GE analytical model *does not furnish* a prediction of the velocity *nor compute entrainment* of spray drops by upward streaming steam. Since GE believes superheating "will not occur in the reactor" they carried out no calculations to see if superheating results in velocity increases. GE, however, stated that in some of the ZR FLECHT tests a thermocouple was placed in the exit flue (steam exhaust line) some distance from the bundle exit. Temperatures at that point would surely be lower than at the bundle top. The thermocouple

[272] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1106. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

showed temperatures [of up] to 250°F. This information effectively invalidates the GE statement that no experimental information contradicted the saturation assumption especially in view of their not setting forth experimental results confirming the assumption. A hint that the exit steam velocities may be very great is given by Roger Griebe's observations that the steam plume and apparent steam exit velocity from test ZR-3 and 4 were unexpectedly and uniquely large (Staff reference 16.20 in the Regulatory Staff Supplementary Testimony). These observations raise some unresolved questions about the applicability of the GE core spray tests which are discussed next.

GE carried out spray tests using upward *air*flow, *with no heating*, to attempt to simulate spray operation (Tr. 13,919-13,925), but in view of the remarks above, CNI believes the test results to be inapplicable. Lawson of ORNL has criticized the tests because of the difference between steam and air on droplet entrainment (Tr. 5790-5795).

CNI shares the Aerojet view that spray may not work (Exhibit 1032). CNI believes that GE has not made adequate attempts to establish a contrary view, a situation which may reflect the fact that the contrary may not be supportable. The GE attitude toward test conduct and the interpretation of test data is well established with respect to the BWR-FLECHT tests by the [Aerojet]-GE-AEC internal memoranda placed in the record by CNI. These are discussed in [the FLECHT Section], above. It is shown that GE made a poor accommodation to the conflict of interest inherent in their assumption of responsibility for carrying out the tests. GE's diligence in core spray effectiveness tests is no better than in FLECHT, and their conclusions no better supported. As pointed out in [Section VII, "The Regulatory Failure,"] there is no assurance available from the FLECHT program that a BWR bundle can be cooled successfully at the spray rates employed in the tests and with the identified weaknesses in spray injection simulation (Exhibit 1041, p. 5.39) that reduced the conservatism of the test. Spray diversion in a BWR LOCA would reduce even further the controllability of the accident and it is without question a possibility which has not been eliminated. It requires prompt resolution.[273]

Regarding core spray distribution, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

Another example relates to the distribution of core spray assumed by GE. GE performed tests involving air up-flow with non-heated simulated bundles as the basis for its core spray distribution assumptions. In June of 1972, the Regulatory Staff ask[ed] GE to describe the basis upon which it could conclude that these air up-flow tests are applicable to the reactor situation. These tests were performed many years ago [before 1973] by

[273] *Id.*, pp. 5.43-5.45.

> GE and they have been the basis upon which GE boiling water reactor emergency core cooling systems have been evaluated for several years, and they are the basis upon which the model approved by the Regulatory Staff in June 1971 determines now much emergency cooling water is delivered to the core. In asking this question, the Regulatory Staff raise[d] the most fundamental doubt about the kind of review that it made of the GE LOCA analysis during all [of] these years in which it has been allowing GE reactors to operate.[274]

7. More Recent BWR Thermal Hydraulic Experiments have been Conducted with Inconel 600 Fuel Rod Simulators

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.[275]

Petitioner would add that such a new BWR-FLECHT program would have to be conducted 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[276]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

Unfortunately, it seems that none of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Zircaloy fuel assemblies.

Perhaps all of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Inconel 600 fuel rod simulators. For example, the Two-Loop Test Apparatus ("TLTA") facility had electrically heated Inconel 600 fuel

---

[274] *Id.*, pp. 7.5-7.6.
[275] *Id.*, p. 5.41.
[276] 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.

rod simulators,[277] the Rig of Safety Assessment ("ROSA") III facility had electrically heated Inconel 600 fuel rod simulators,[278] and the Full Integral Simulation Test ("FIST") facility had electrically heated Inconel 600 fuel rod simulators.[279]

Additionally, the BWR FIX-II test facility had electrically heated Inconel 600 fuel rod simulators[280] and the NUPEC BWR Full-Size Fine-Mesh Bundle Test ("BFBT") facility had electrically heated Inconel 600 fuel rod simulators.[281]

Petitioner has not been able to locate information identifying the cladding material that was used in the fuel rod simulators in the 30° Steam Sector Test Facility ("SSTF"); in the SSTF, it is doubtful that Zircaloy was used as the fuel rod simulator cladding material. The SSTF experiments used steam injection to simulate core heat[282] the maximum temperature of the steam was 800 F.[283]

(It is noteworthy that many of the papers reporting on BWR heat-transfer experiments do not mention what type of cladding material was used in the fuel rod simulators in the experiments they describe. For example, Petitioner has not located any papers that state what type of cladding material is used in the fuel rod simulators in the Purdue University Multidimensional Integral Test Assembly ("PUMA") facility. Most

---

[277] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, August 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072480029, p. 5-27.

[278] Y. Koizumi, M. Iriko, T. Yonomoto, K. Tasaka, "Experimental Analysis of the Power Curve Sensitivity Test Series at ROSA-III," Nuclear Engineering and Design, 86, 1985, pp. 268, 270.

[279] General Electric, "BWR Full Integral Simulation Test (FIST) Program Facility Description Report" NUREG/CR-2576, EPRI NP-2314, GEAP-22054, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML073461126, pp. 2-32, 2-37; and Siemens, "EXEM BWR-2000 ECCS Evaluation Model," EMF-2361 (NP), October 2000, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML003772936, p. 5-2.

[280] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, August 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072480029, pp. 5-119, 5-129.

[281] B. Neykov, F. Aydogan, L. Hochreiter, K. Ivanov, H. Utsuno, K. Fumio, E. Sartori, "NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT) Benchmark," Volume I: Specifications, NEA/NSC/DOC(2005)5, June 2005, pp. 15, 34.

[282] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.5-11.

[283] NRC, (Appendix A) "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053620415, Appendix A, p. A-208.

likely, the PUMA facility—currently investigating BWR-related problems—uses Inconel 600 fuel rod simulators: the Rod Bundle Heat Transfer ("RBHT") facility at Penn State University—currently investigating PWR-related problems—uses Inconel 600 fuel rod simulators.[284] Also, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, which describes many experimental facilities, does not mention what type of cladding material was used in the fuel rod simulators at the experimental facilities it describes.)

It is significant that Inconel 600 does not oxidize nearly as much as Zircaloy in the design-basis accident temperature range.

Discussing Inconel 600's resistance to oxidation, "INCONEL alloy 600," states:

> INCONEL alloy 600 is widely used in the furnace and heat-treating fields for retorts, boxes, muffles, wire belts, roller hearths, and similar parts which require resistance to oxidation and to furnace atmospheres. ... The alloy's resistance to oxidation and scaling at 1800°F (980°C) is shown in Figure 11.[285]

Figure 11 of "INCONEL alloy 600," depicts a graph of the results of cyclic oxidation tests at 1800°F (980°C), in which there were alternating intervals of 15 minutes of heating and 5 minutes of cooling in air: Inconel 600 oxidized less than stainless steel (type 304), stainless steel (type 309), and Inconel 800HT. Inconel 600 oxidized very little over a period of 1000 hours of cyclic exposure time.

Additionally, in an Advisory Committee on Reactor Safeguards, subcommittee meeting on thermal hydraulic phenomena, on July 7, 2008, a participant, Mr. Kelly, discussing LOCA phenomena, stated that Inconel has "almost no oxidation."[286]

Henry Kendall and Daniel Ford's criticisms of the BWR FLECHT tests conducted with stainless steel fuel rod simulators would also apply to BWR thermal hydraulic experiments conducted since the early 1970s with Inconel 600 fuel rod simulators. To conclusively demonstrate that BWR ECCSs would be effective, in the

---

[284] Donald R. Todd, Cesare Frepoli, Lawrence E. Hochreiter, "Development of a COBRA-TF Model for the Penn State University Rod Bundle Heat Transfer Program," 7th International Conference on Nuclear Engineering, Tokyo, Japan, April 19-23, 1999, ICONE-7827, p. 3.

[285] Special Metals Corporation, "INCONEL alloy 600," www.specialmetals.com, SMC-027, 2008, p. 11.

[286] Mr. Kelly, NRC, Advisory Committee on Reactor Safeguards, Transcript of Subcommittee Meeting on Thermal Hydraulic Phenomena, July 7, 2008, p. 168.

event of a LOCA, it would be necessary to conduct BWR heat transfer experiments with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Experiments with Inconel 600 fuel rod simulators are inadequate.

Furthermore, interpretations of the results of experiments conducted with Inconel 600 fuel rod simulators would most likely lead the interpreters to false conclusions. For example, a multi-rod Inconel 600 bundle heated up to peak cladding temperatures of 2150°F—the value of OCNGS's LBPCT—would not incur autocatalytic oxidation; however, a multi-rod Zircaloy bundle heated up to peak cladding temperatures between 1832°F and 2150°F would (with high probability) incur autocatalytic oxidation.

IV. CONCLUSION

Petitioner requests that NRC order the licensees of OCNGS and NMP-1 to lower the LBPCTs of OCNGS and NMP-1 in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Experimental data demonstrates that OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively, do not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Such data demonstrates that OCNGS and NMP-1's LBPCTs need to both be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety.

Petitioner also requests that NRC order the licensees of OCNGS and NMP-1 demonstrate that OCNGS and NMP-1's BWR/2 ECCSs would effectively quench the fuel cladding in the event of LOCAs, and prevent partial or complete meltdowns. Experimental data indicates that OCNGS and NMP-1's ECCSs would not effectively quench the fuel cladding in the event of LOCAs, if the fuel cladding reached OCNGS and NMP-1's LBPCTs of 2150°F and 2149°F, respectively.

To uphold its congressional mandate to protect the lives, property, and environment of the people of New York and New Jersey, NRC must not allow OCNGS and NMP-1's LBPCTs to remain at elevated temperatures that would not provide necessary margins of safety, in the event of LOCAs. 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

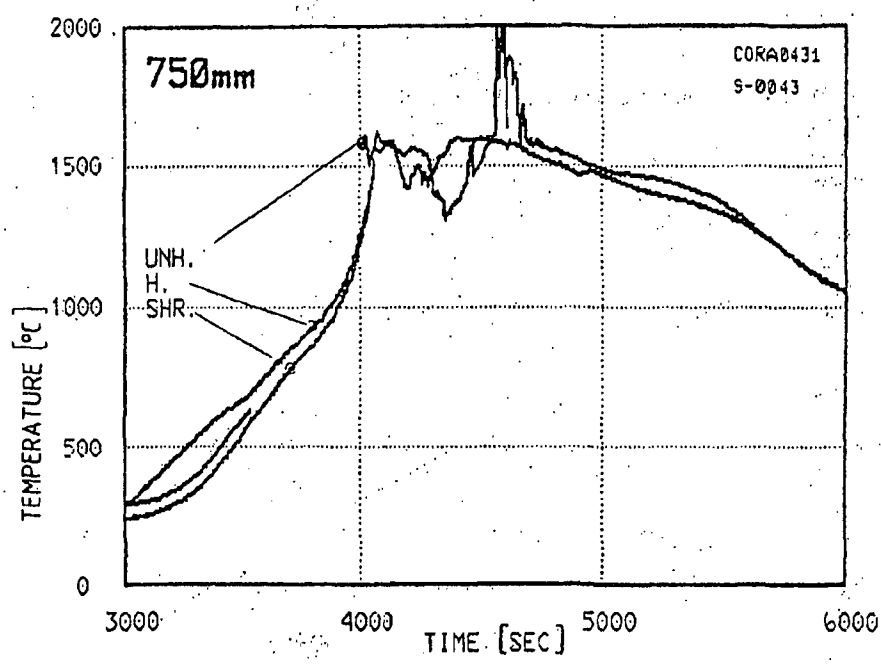
Respectfully submitted,

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

Dated: December 10, 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.



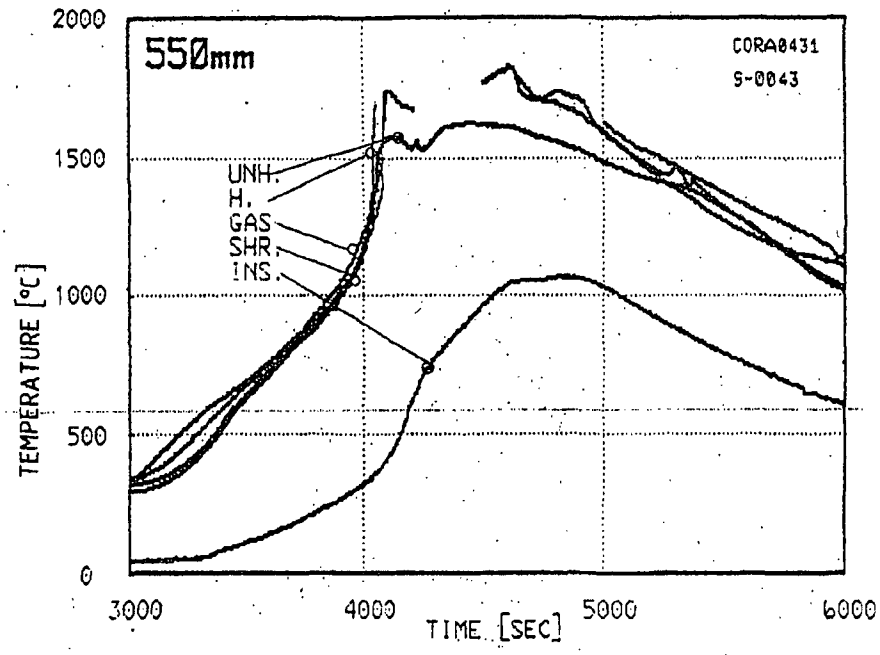
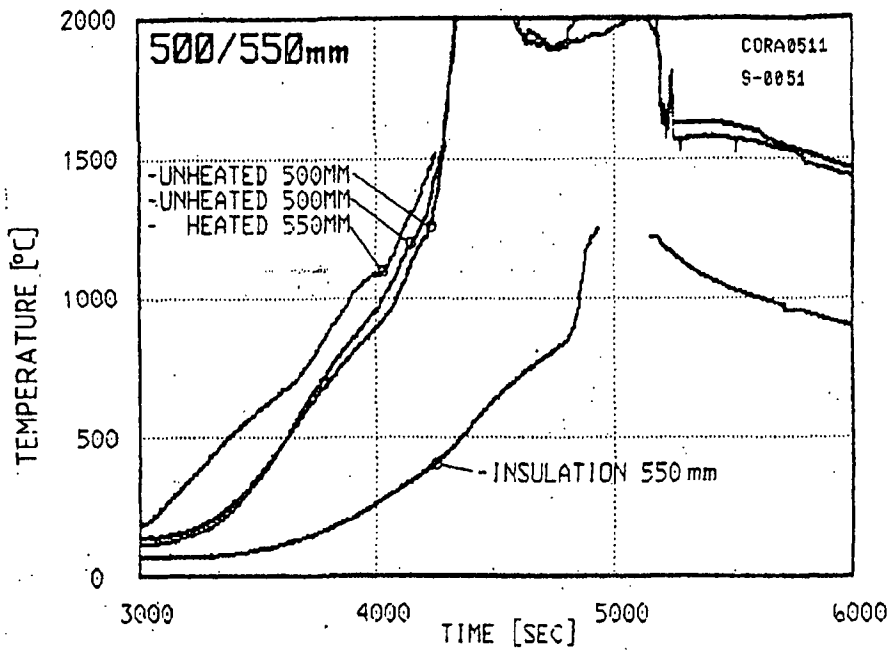


Fig. 12. Temperatures during test CORA-2 at 500 mm and 750 mm elevation. Temperatures of heated (H) and unheated rod (UNH), atmosphere (gas), shroud (SHR), and outer surface of shroud insulation (INS)



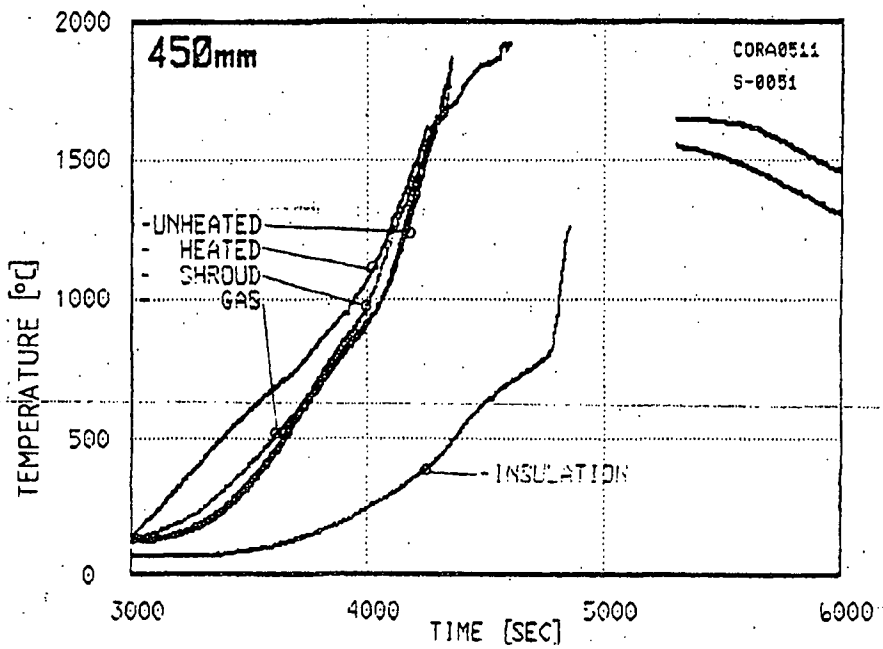


Fig. 13. Temperatures measured during test CORA-3 at 450 mm and 550 mm elevation

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.

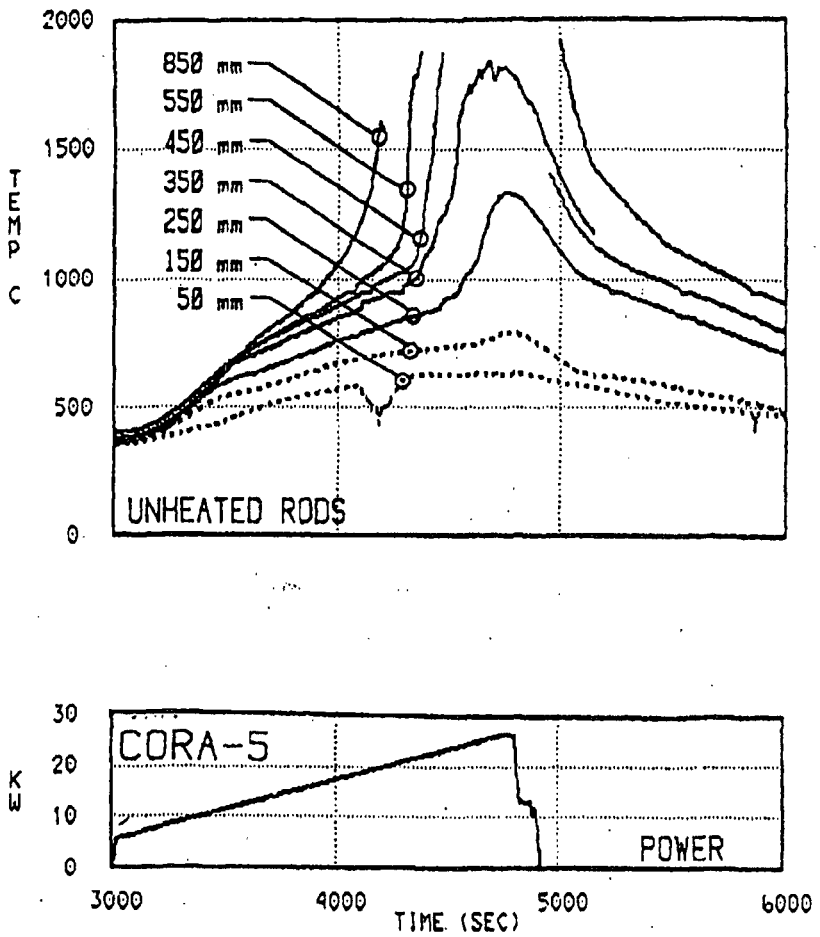


Fig.15: Temperatures of unheated rods and power history of CORA-5

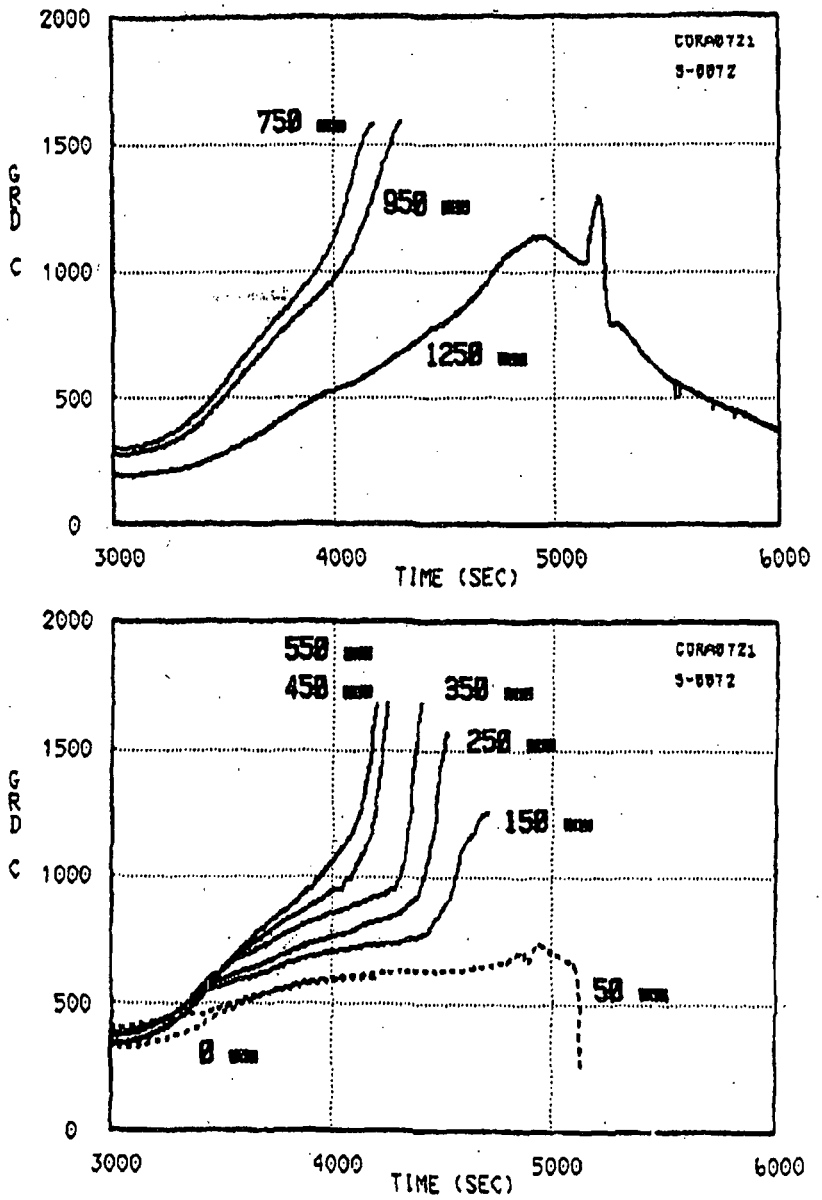


Fig.16: Temperatures of unheated rods during CORA-12

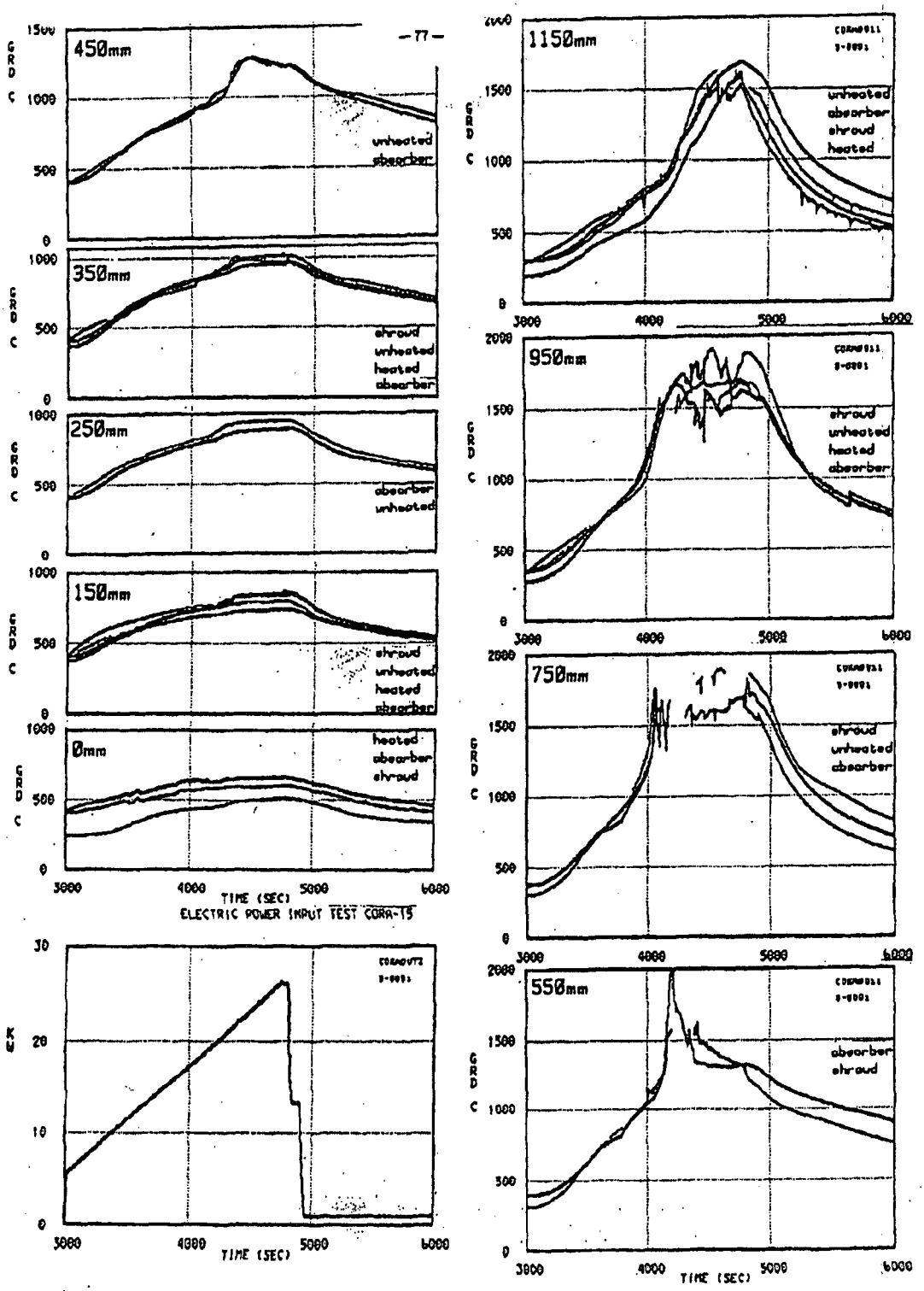
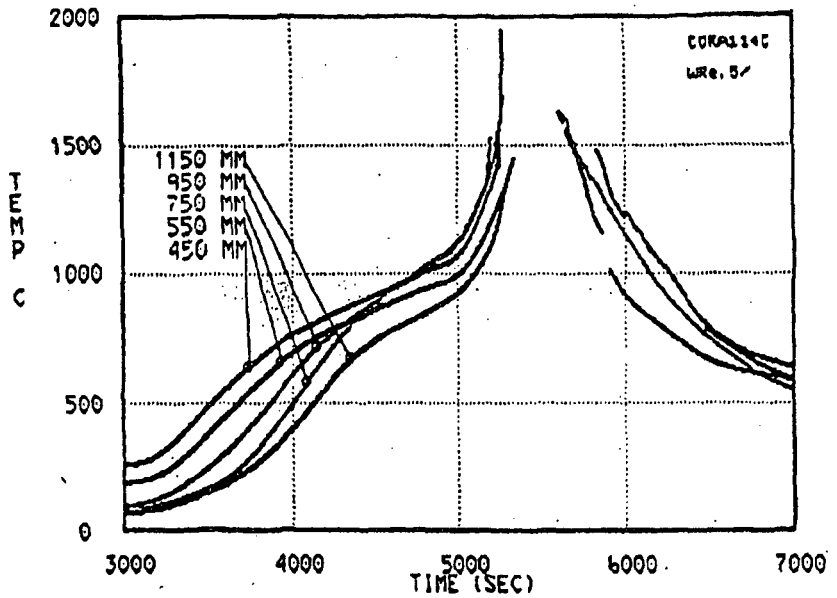


Fig.17: Temperatures at different elevations during CORA-15



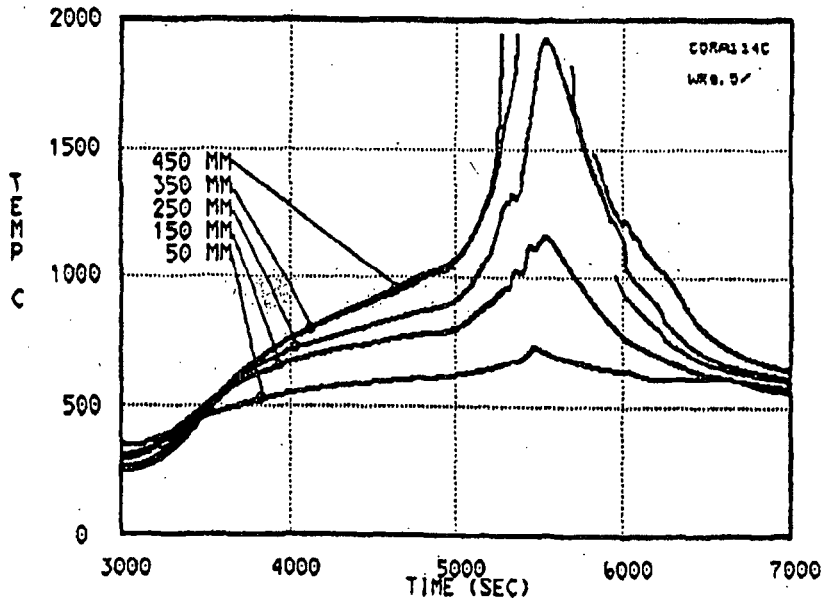
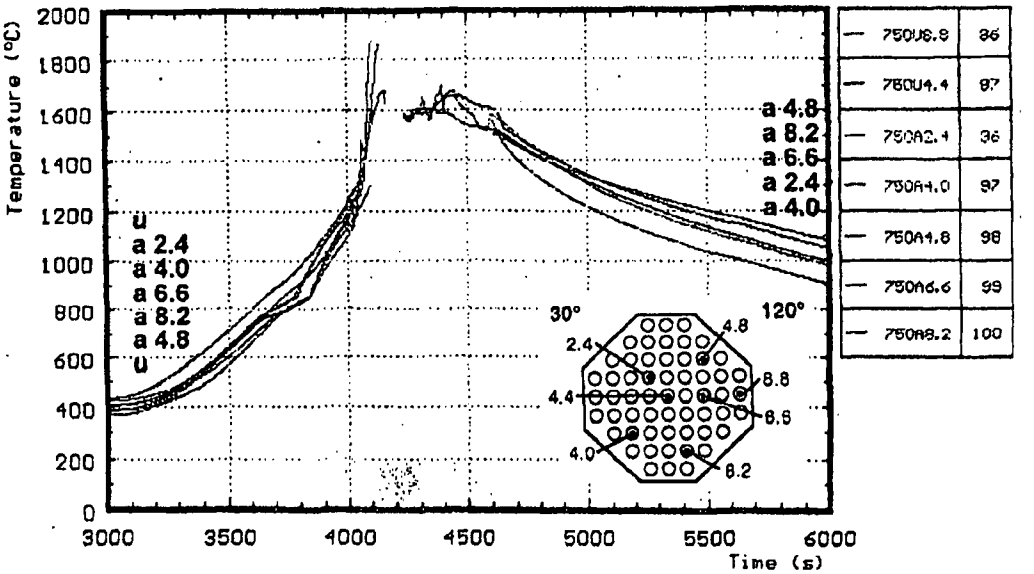
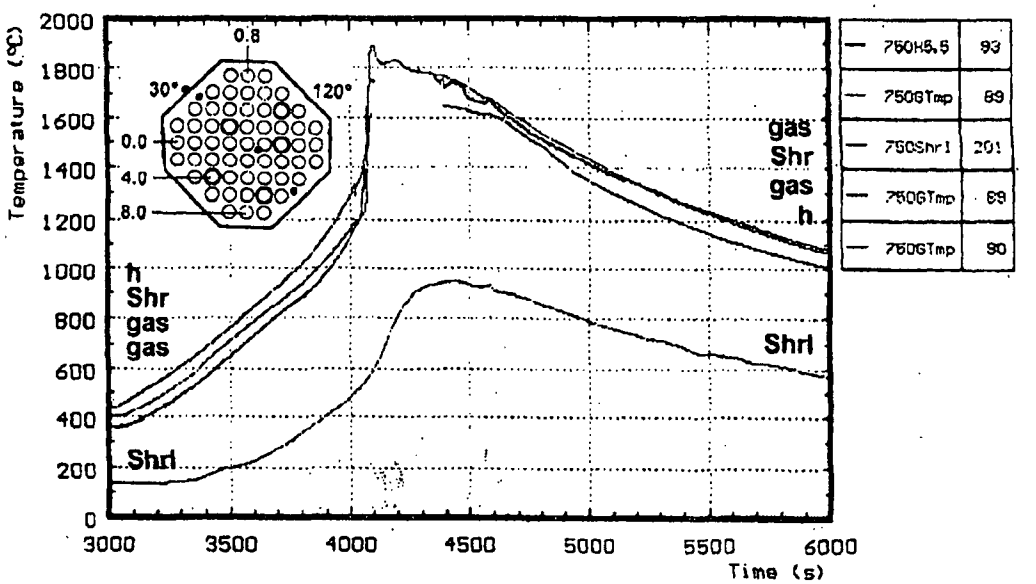


Fig.18: Temperatures of unheated rods during CORA-9





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)

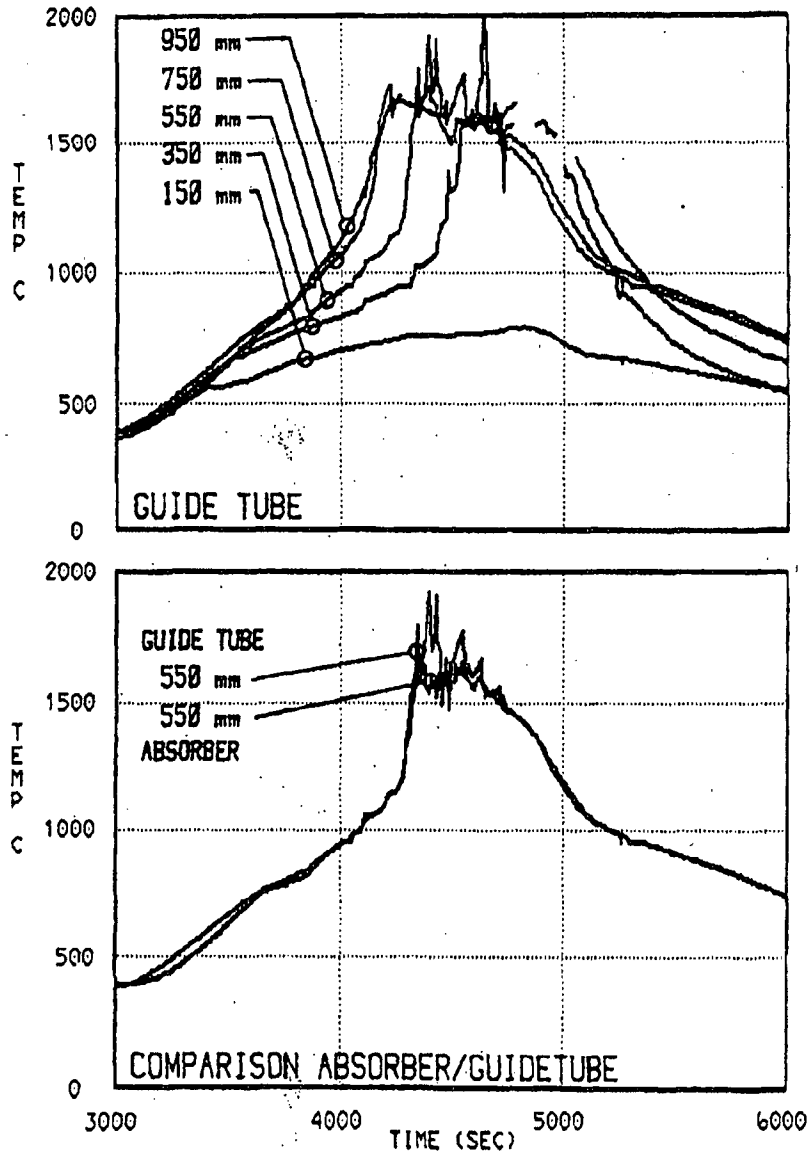


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

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.

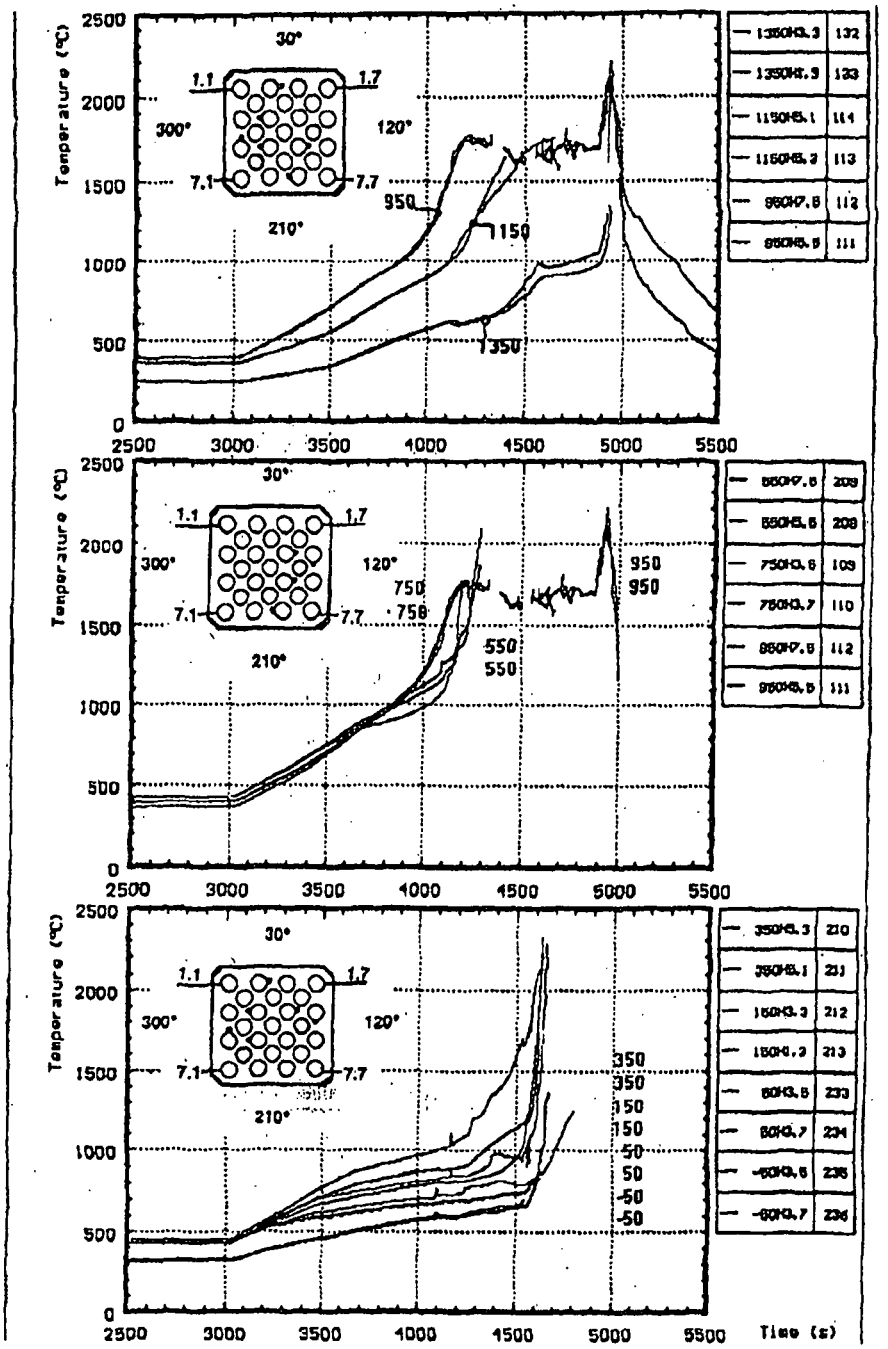


Fig. 37: **Temperatures of the heated rods (CORA-13)**

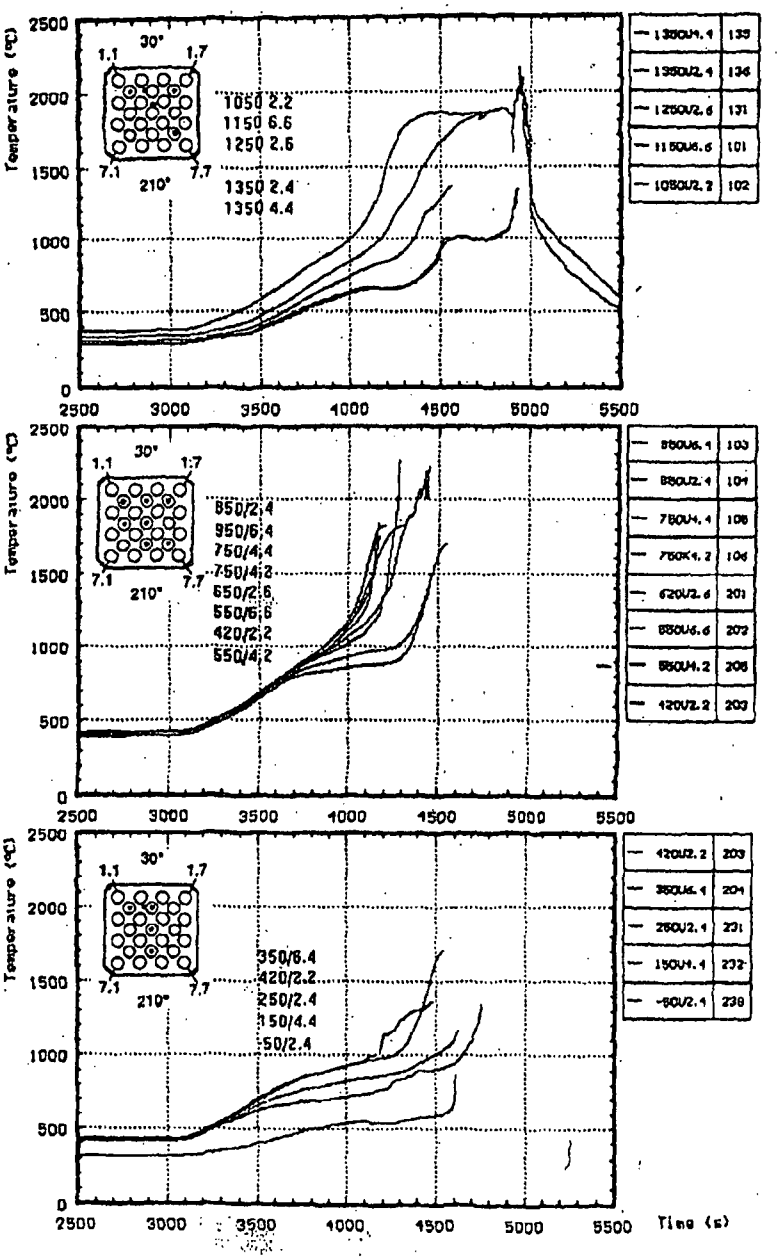


Fig. 39: Temperatures of the unheated rods (CORA-13)

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.

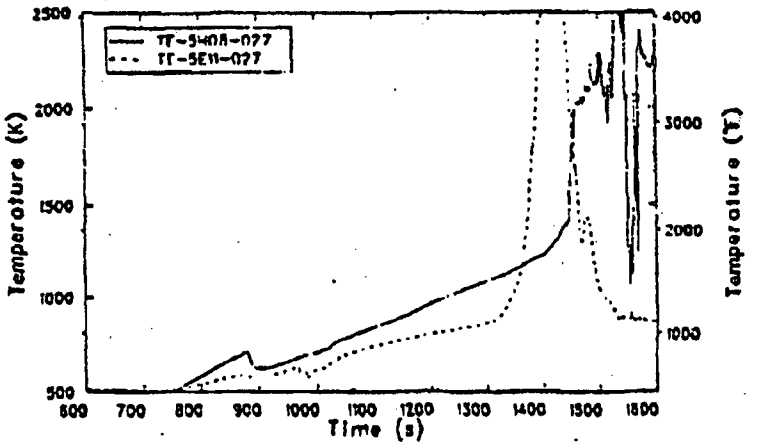


Figure 3.7 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5.

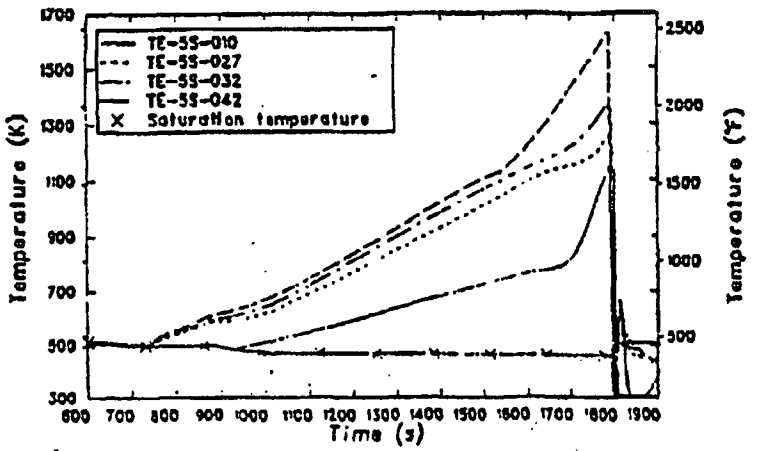


Figure 3.8 Comparison of four external wall temperatures at the 1.07-, 0.81-, 0.69-, and 0.25-m (42-, 32-, 27-, and 10-in.) elevations on the south side of the flow shroud.

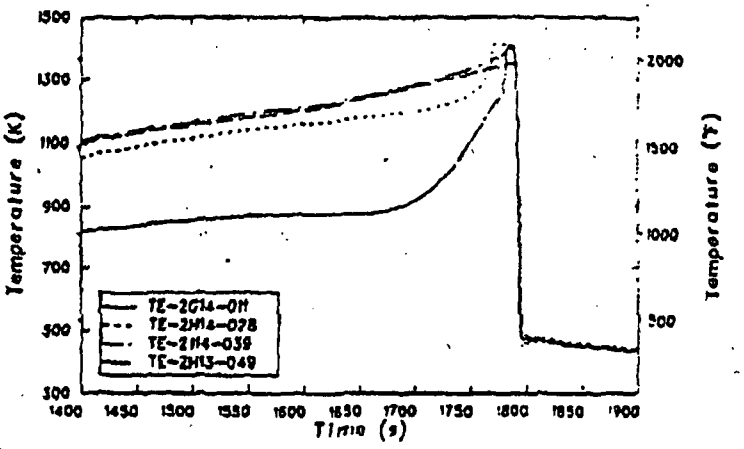


Figure 3.9 Comparison of cladding temperatures at the 1.24-, 0.99-, 0.71-, and 0.28-m (49-, 39-, 28-, and 11-in.) elevations in Fuel Assembly 2.

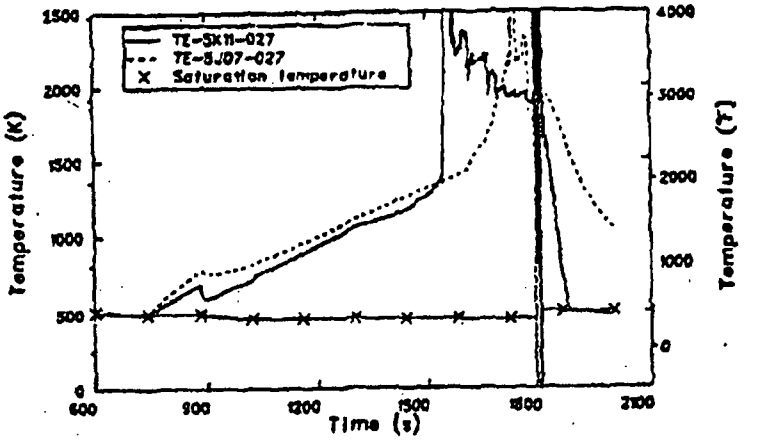


Figure 3.10 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5 with saturation temperature.

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.

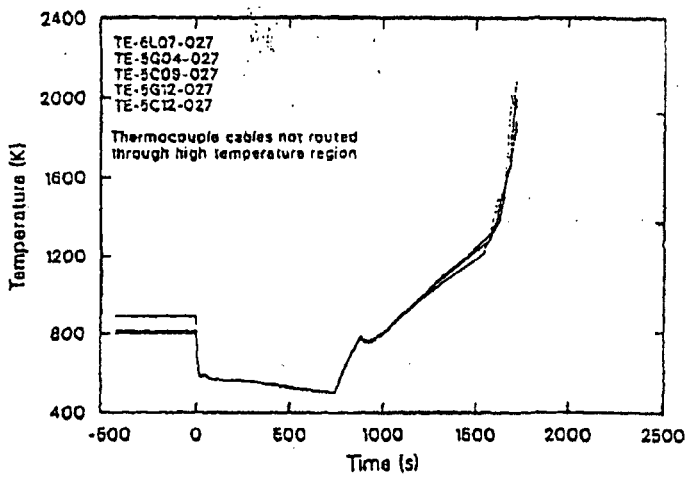


Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

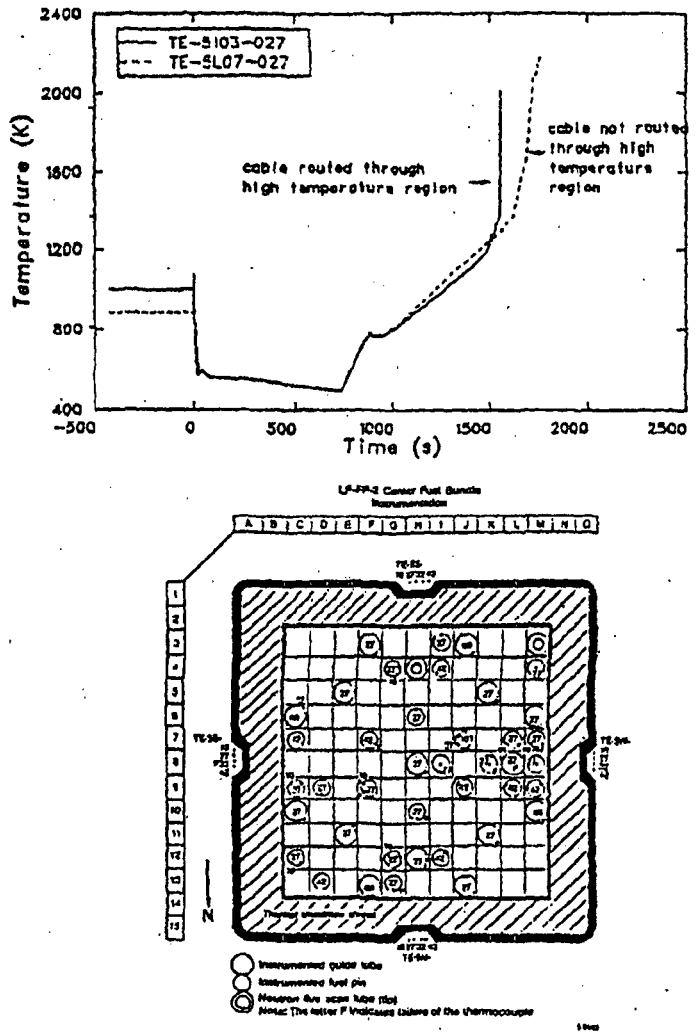


Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

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.

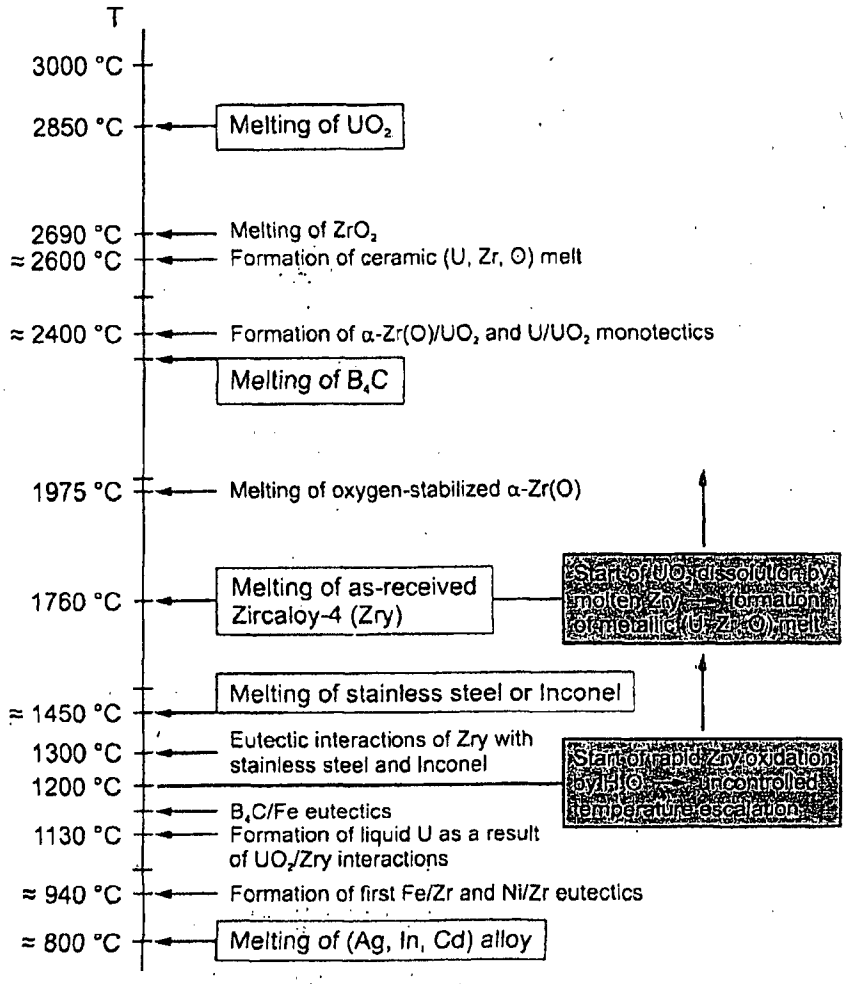


Fig. 1. LWR severe accident-relevant melting and chemical interaction temperatures which result in the formation of liquid phases.

- 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

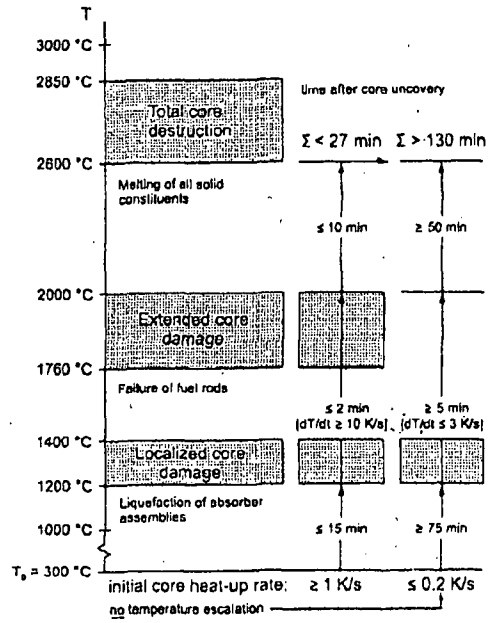


Fig. 13. Dependence of the temperature regimes on liquid phase formation on the initial heat-up rate of the core. Small heat-up rates drastically reduce the amount of molten Zircaloy (1800–2000°C) and give more time for possible accident management measures.

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].

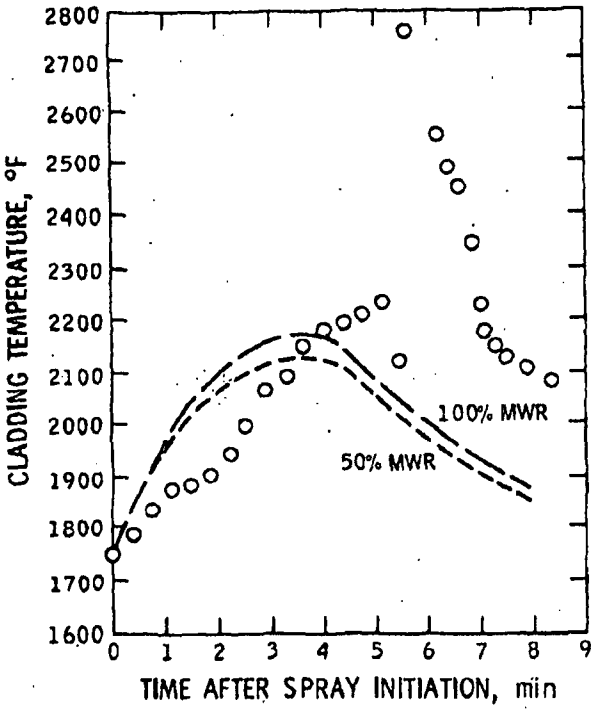
The CORA-10 test simulated the behavior of a rod bundle with additional cooling at its lower end (TMI-2 conditions) [34]. Fig. 14 depicts the axial bundle temperature profile at different times and the material relocation. One can recognize the influence of the higher heat losses at the lower end (30 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].

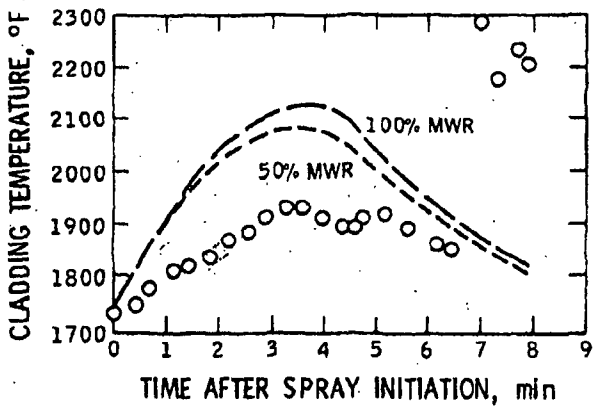
Appendix G Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[7] and Figure A8.10 Analysis of Zr2K Thermal Response[8]

[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

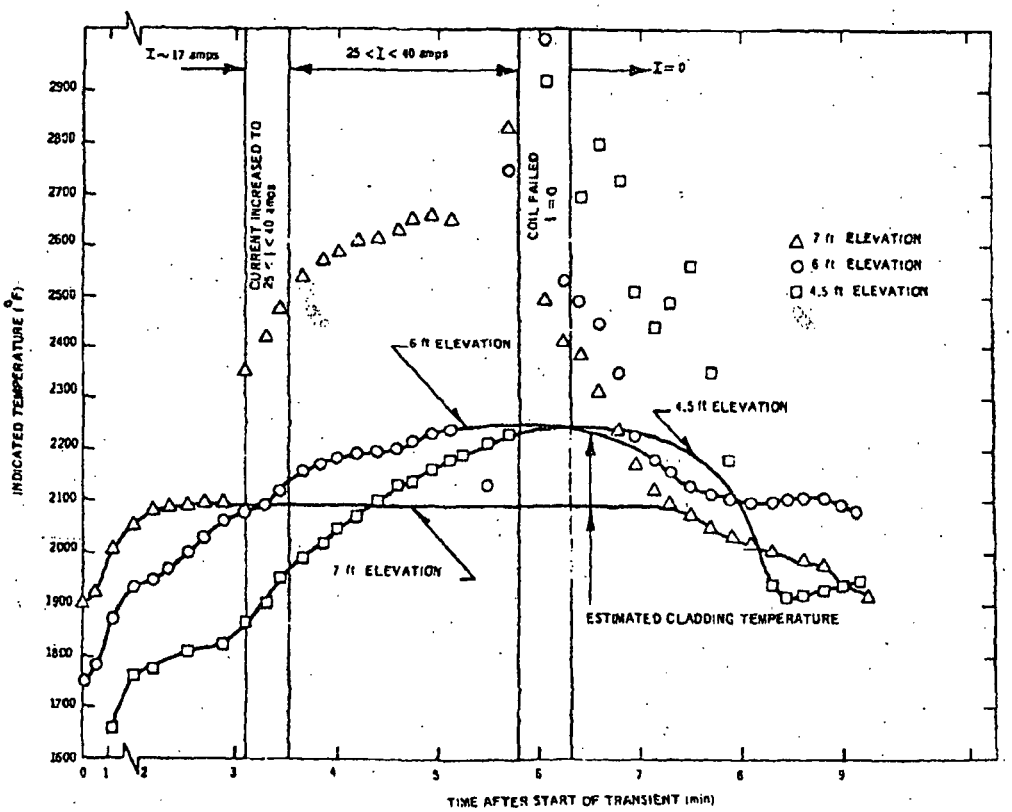


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

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

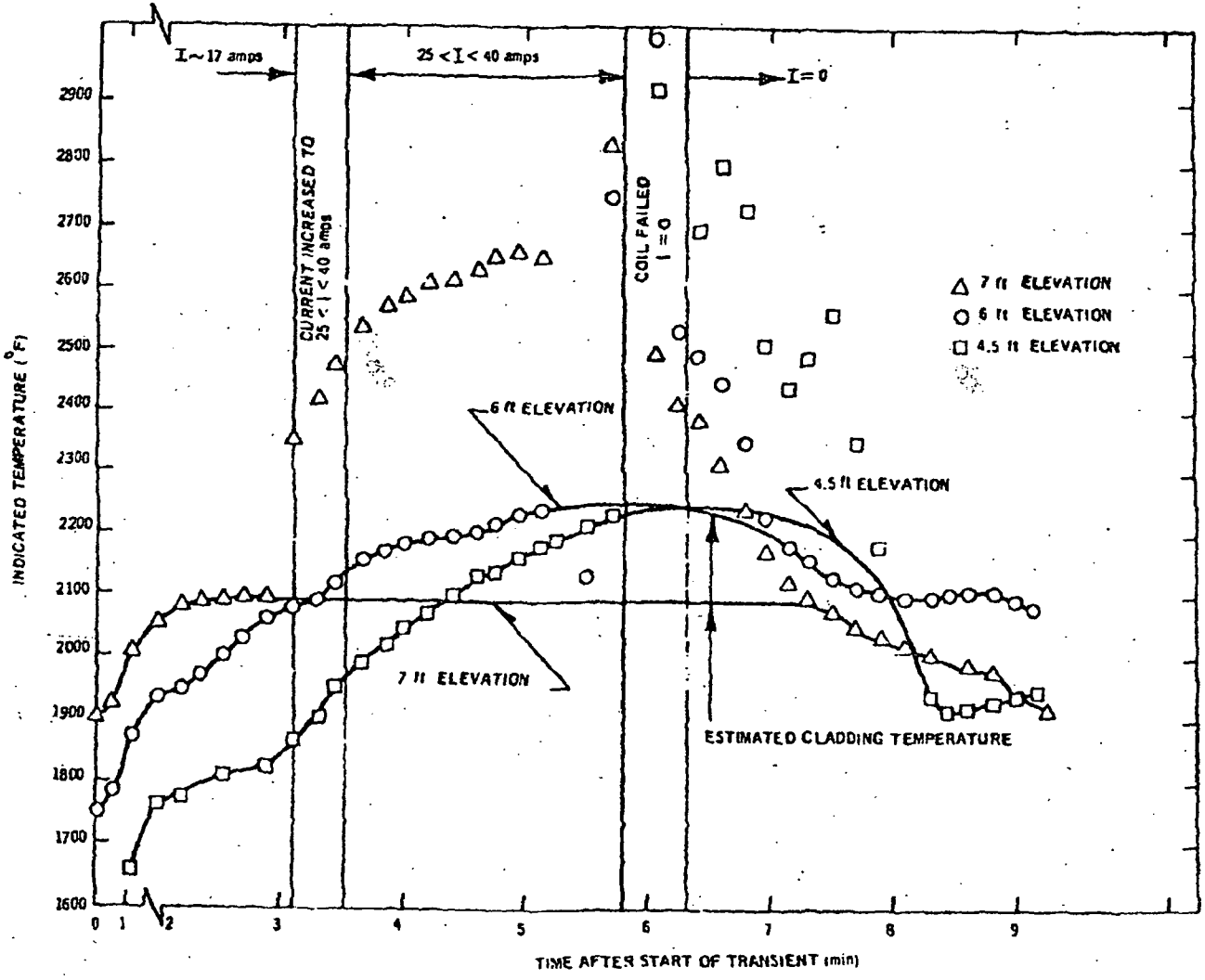
Analysis of Zr2K Thermal Response



(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.

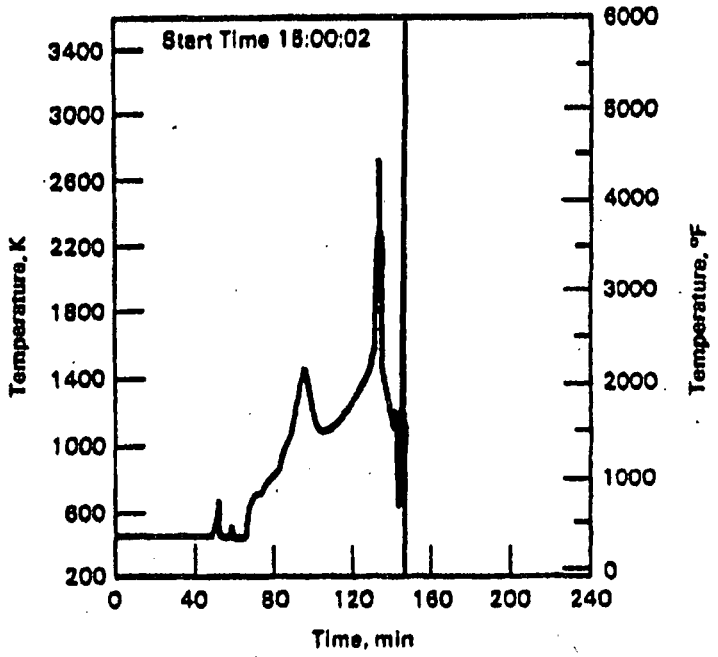


FIGURE 4.1. Typical Cladding Temperature Behavior

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.

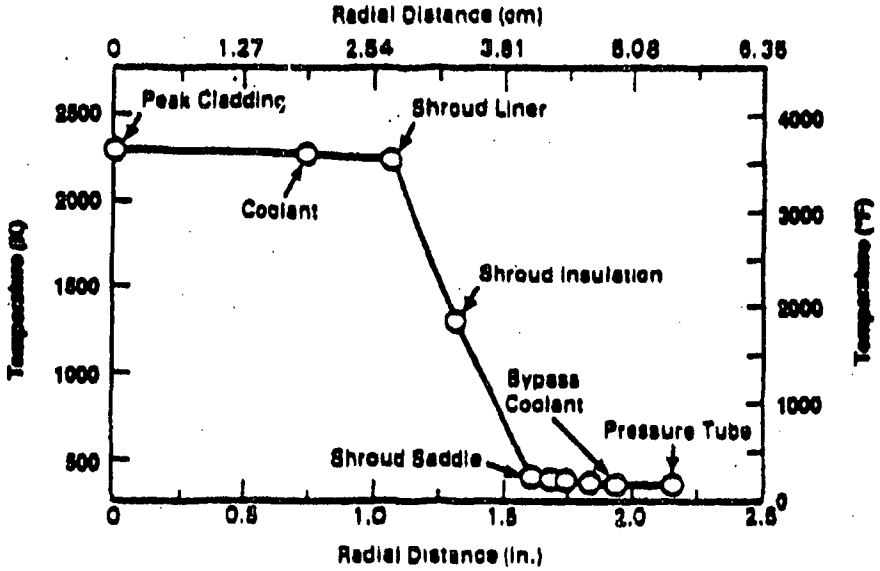


FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

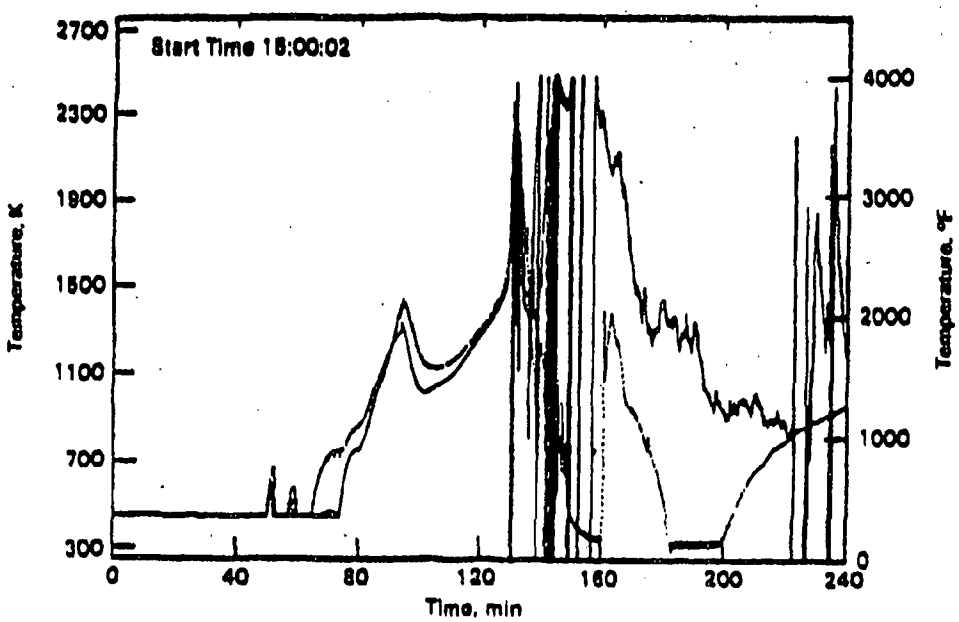


FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

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.

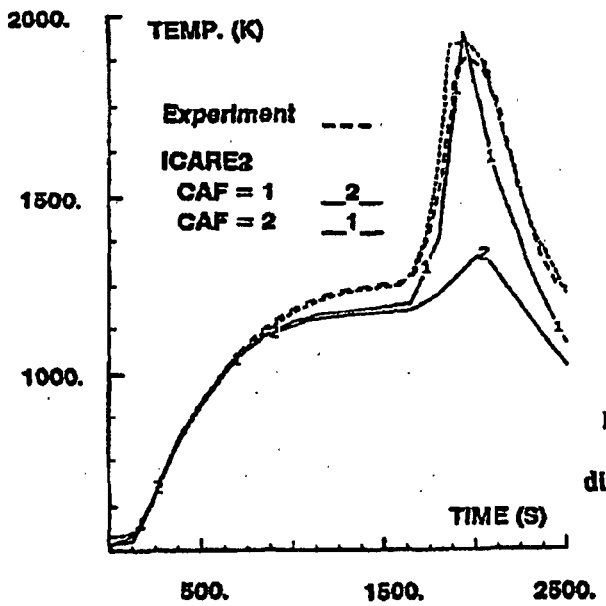


Fig.1 :
Sensitivity calculation on the B9R test. Temperature escalation at the hot level (0.6 m) with different Contact Area Factors (CAF)

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