EDO Principal Correspondence Control

FROM: DUE: 04/28/11

EDO CONTROL: G20110218
DOC DT: 03/28/11
FINAL REPLY:

Deborah Brancato
Riverkeeper, Inc.

TO:

Borchardt, EDO

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

Leeds, NRR

DESC:

2.206 - Lower Licensing Basis Peak Cladding Temperatures of Indian Point Units 2 and 3 (EDATS: OEDO-2011-0223)

DATE: 03/29/11

ASSIGNED TO: NRR

CONTACT: Leeds

ROUTING:

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

SPECIAL INSTRUCTIONS OR REMARKS:

EDATS

Electronic Document and Action Tracking System

EDATS Number: OEDO-2011-0223 **Source:** OEDO

General Information

Assigned To: NRR **OEDO Due Date:** 4/28/2011 11:00 PM

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

Subject: 2.206 - Lower Licensing Basis Peak Cladding Temperatures of Indian Point Units 2 and 3

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: G20110218 **Staff Initiated:** NO

Related Task: **Recurring Item:** NO

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

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

Sensitivity: None

Signature Level: NRR **Urgency:** NO

Approval Level: No Approval Required

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OCM Concurrence: NO

OCA Concurrence: NO

Special Instructions:

Document Information

Originator Name: Deborah Brancato **Date of Incoming:** 3/28/2011

Originating Organization: Riverkeeper, Inc. **Document Received by OEDO Date:** 3/29/2011

Addressee: R. W. Borchardt, EDO **Date Response Requested by Originator:** 4/28/2011

Incoming Task Received: 2.206

March 28, 2011

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 INDIAN POINT UNITS 2 AND 3 ("IP-2 AND -3") IN ORDER TO PROVIDE NECESSARY MARGINS OF SAFETY—TO HELP PREVENT MELTDOWNS—IN THE EVENT OF LOSS-OF-COOLANT ACCIDENTS ("LOCAS") AND TO HAVE THE LICENSEE OF IP-2 AND -3 DEMONSTRATE THAT IP-2 AND -3'S EMERGENCY CORE COOLING SYSTEMS WOULD EFFECTIVELY QUENCH THE FUEL CLADDING IN THE EVENT OF LOCAS

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March 28, 2011

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of:	:	TO: R. WILLIAM BORCHARDT
	:	Executive Director for Operations
ENTERGY CORPORATION	:	U.S. Nuclear Regulatory Commission
(Indian Point Nuclear Generating	:	Washington D.C. 20555-0001
Units No. 2 and 3; Facility Operating	:	
Licenses DPR-26 and DPR-64)	:	Docket No. ______

RIVERKEEPER,
Petitioner

10 C.F.R. § 2.206 REQUEST TO LOWER THE LICENSING BASIS PEAK CLADDING TEMPERATURES OF INDIAN POINT UNITS 2 AND 3 ("IP-2 AND -3") IN ORDER TO PROVIDE NECESSARY MARGINS OF SAFETY—TO HELP PREVENT MELTDOWNS—IN THE EVENT OF LOSS-OF-COOLANT ACCIDENTS ("LOCAS") AND TO HAVE THE LICENSEE OF IP-2 AND -3 DEMONSTRATE THAT IP-2 AND -3'S 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 Riverkeeper. 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."

Riverkeeper (hereinafter "Petitioner") requests that United States Nuclear Regulatory Commission ("NRC") order the licensee of Indian Point Units 2 and 3 ("IP-2 and -3") to lower the licensing basis peak cladding temperatures ("LBPCT") of IP-2 and -3 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

demonstrates that IP-2 and -3's LBPCTs of 1937°F[1] and 1961°F,[2] respectively, do not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Such data demonstrates that IP-2 and -3's LBPCTs need to both be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety.

Second, Petitioner requests that NRC order the licensee of IP-2 and -3 to determine how far below 1832°F the LBPCT values of IP-2 and -3 needs to be lowered to in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Third, Petitioner requests that NRC order the licensee of IP-2 and -3 to lower both of IP-2 and -3's LBPCTs to 1600°F—in the interim—before conservative LBPCT values for IP-2 and -3 are determined.

Fourth, Petitioner requests that NRC order the licensee of IP-2 and -3 to demonstrate that IP-2 and -3 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 IP-2 and -3's ECCSs may not effectively quench the fuel cladding in the event of LOCAs, if fuel cladding temperatures approached or reached IP-2 and -3 LBPCTs of 1937°F and 1961°F, respectively.

II. STATEMENT OF PETITIONER'S INTEREST

Petitioner is a member-supported, not-for-profit organization dedicated to protecting the Hudson River and its tributaries.[3] Since its inception in 1966, Petitioner has used litigation, science, advocacy, and public education to raise and address concerns relating to the Indian Point nuclear power plant, located on the eastern bank of the Hudson River in Buchanan, NY. Petitioner is headquartered in Ossining, New York,

[1] IP-2's current LBPCT is primarily based on ECCS evaluation calculations with Westinghouse's ASTRUM methodology, bounded in the 1996 WCOBRA/TRAC code. NRC authorized the use of the ASTRUM methodology for IP-2's ECCS evaluation calculations in 2006. Since 2006 there have been minor changes to IP-2's LBPCT value; see section III.C.5.a. of this document.

[2] IP-3's current LBPCT is primarily based on the ECCS evaluation calculations that helped qualify IP-3's 2005 stretch power uprate. Since 2005 there have been minor changes to IP-3's LBPCT value; see section III.C.5.b. of this document.

[3] *See generally,* Riverkeeper.org, Our Story, http://www.riverkeeper.org/ourstory_index.php (last visited March 24, 2011).

approximately 10 miles from the Indian Point facility, and has numerous members that reside within at least 50 miles of the plant.[4]

For almost a decade, Petitioner has taken an active role in bringing to light critical and problematic safety issues which have notoriously plagued Indian Point, and called for improvements to the facility to ensure the safe operation of the plant. For example, in 2001, Petitioner filed an enforcement petition with the NRC pursuant to 10 C.F.R. § 2.206 seeking enhanced safety and security measures in light of the September 11th terrorist attacks;[5] in 2004, Petitioner commissioned an expert analysis of the potential consequences of a severe accident/terrorist event occurring at Indian Point, which described the catastrophic radioactive releases that can occur as the result of such incidents;[6] and in 2007, Petitioner filed a petition to intervene in the Indian Point license renewal proceeding, raising several technical safety concerns relating to the lack of adequate analysis concerning severe accident mitigation alternatives, fatigue of metal components, and flow-accelerated corrosion of piping at the plant.[7]

Based on Petitioner's vested interest in the safe operation of Indian Point, Petitioner is personally affected and aggrieved by the continued operation of the plant without implementation of the specific measures identified in this request.

Petitioner is submitting this 10 C.F.R. § 2.206 petition because experimental data demonstrates that IP-2 and -3 LBPCTs of 1937°F and 1961°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.

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[4] *See* Riverkeeper.org, Contact Us, http://www.riverkeeper.org/contact/ (last visited March 24, 2011).

[5] In the Matter of Entergy Corporation (Indian Point Nuclear Power Station, Units No. 2 and 3; Facility Operating Licenses DPR-26 and DPR-64, Section 2.206 Request for Emergency Shutdown Of Indian Point Units 2 and 3 (Nov. 8, 2001), *available at*, ADAMS Accession No. ML013480179.

[6] Edwin S. Lyman, PhD, *Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack at the Indian Point Nuclear Plant* (Union of Concerned Scientists, September 2004), *available at*, http://www.riverkeeper.org/wp-content/uploads/2011/03/Chernobyl-on-the-Hudson_indianpointhealthstudy.pdf (hereinafter "Lyman, *Chernobyl on the Hudson?* "). A copy of this report is annexed hereto as Appendix K.

[7] Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant, Docket Nos. 50-247-LR, 5-286-LR (November 30, 2007), *available at*, ADAMS Accession No ML073410093.

Second, Petitioner is submitting this 10 C.F.R. § 2.206 petition because it needs to be determined how far below 1832°F the LBPCT values of IP-2 and -3 need to be lowered to in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Third, Petitioner is submitting this 10 C.F.R. § 2.206 petition because IP-2 and -3's LBPCTs need to both be lowered to 1600°F—in the interim—before conservative LBPCT values for IP-2 and -3 are determined.

Fourth, Petitioner is submitting this 10 C.F.R. § 2.206 petition because experimental data indicates that IP-2 and -3's ECCSs may not effectively quench the fuel cladding in the event of LOCAs, if fuel cladding temperatures approached or reached IP-2 and -3 LBPCTs of 1937°F and 1961°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 Mark Leyse wrote and submitted on behalf of New England Coalition, 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.

This 10 C.F.R. § 2.206 petition is also similar to a 10 C.F.R. § 2.206 petition, dated December 10, 2010 (ADAMS Accession Number: ML103481100), that Mark Leyse submitted, requesting that NRC order the licensees of Oyster Creek Nuclear Generating Station ("OCNGS") and Nine Mile Point Unit 1 ("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.

The OCNGS and NMP-1 10 C.F.R. § 2.206 petition also requested that NRC order the licensees of OCNGS and NMP-1 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.

This 10 C.F.R. § 2.206 petition requests enforcement actions—two additional enforcement actions—that were not requested in the VYNPS 10 C.F.R. § 2.206 petition and in the OCNGS and NMP-1 10 C.F.R. § 2.206 petition.

Furthermore, this 10 C.F.R. § 2.206 petition raises safety issues that were not raised in the VYNPS 10 C.F.R. § 2.206 petition or in the OCNGS and NMP-1 10 C.F.R. § 2.206 petition: safety issues regarding pressurized water reactor ("PWR") core components; *e.g.*, PWR (Ag,In,Cd) alloy absorber rods. (The VYNPS 10 C.F.R. § 2.206 petition as well as the OCNGS and NMP-1 10 C.F.R. § 2.206 petition raised safety issues regarding ("BWR") core components; *e.g.*, BWR B_4C absorber blades.) This 10 C.F.R. § 2.206 petition also raises safety issues regarding the fact that the reflood heat transfer coefficients used in ECCS evaluation calculations for IP-2 and -3 are *not* based on data from thermal hydraulic experiments conducted with zirconium alloy multi-rod bundles.

Revisions to NRC's 10 C.F.R. § 50.46(b)(1) peak cladding temperature ("PCT") limit criterion have been requested by Mark Leyse in a petition for rulemaking, PRM-50-93, dated November 17, 2009 (ADAMS Accession No. ML093290250); however, this petition for an enforcement action is separately and appropriately submitted pursuant to 10 C.F.R. § 2.206. The safety issues raised in this 10 C.F.R. § 2.206 petition affect IP-2 and -3 and need prompt resolution to protect the lives, property, and environment of the people of New York. Furthermore, this 10 C.F.R. § 2.206 petition is submitted by a local, affected party and the safety issues raised in this petition are of an immediate nature and require prompt NRC review and action—available to Petitioner only through the 10 C.F.R. § 2.206 process.

A. Plant Specific Issues

This 10 C.F.R. § 2.206 petition addresses generic issues (regarding metal-water reaction rates, eutectic reactions between different assembly components, and reflood heat transfer coefficients); however, this 10 C.F.R. § 2.206 petition should also be considered plant specific because of the location of IP-2 and -3.

This 10 C.F.R. § 2.206 petition is plant specific, because New York City is located less than 25 miles south of IP-2 and -3 and more than 17 million people live within a 50-mile radius of IP-2 and -3.[8]

This 10 C.F.R. § 2.206 petition is also plant specific, because IP-2 and -3 were built within one or two miles of the Ramapo seismic zone: a "system [that] is not so

[8] *See* Lyman, *Chernobyl on the Hudson?*, *supra* Note 6, at 23.

much a single fracture as a braid of smaller ones, where quakes emanate from a set of still ill-defined faults."[9] In fact, the area around Indian Point is susceptible to an earthquake of 7.0 in magnitude on the Richter scale.[10] The owner of Indian Point, Entergy Nuclear Operations, Inc. (hereinafter "Entergy"), indicates that Units 2 and 3 were allegedly build to only withstand a 6.0 magnitude earthquake.[11] Even if this alleged, as yet unsubstantiated estimate were true, a 7.0 magnitude earthquake is approximately 30 times more powerful than a 6.0. Thus, IP-2 and -3 are not capable of withstanding earthquakes that could reasonably occur in the area. Indeed, an NRC report dated August 2010 (in conjunction with supplemental data regarding power plants not reviewed in the report) reveals that IP-3 has the highest risk of seismic related core damage than any other nuclear power plant in the country.[12] Based on the foregoing, an earthquake occurring in proximity of IP-2 and -3 could cause one or two LOCAs. It would be reasonable to claim that the probability of LOCAs at IP-2 and -3 is higher than it is for most other nuclear power plants licensed by NRC.

Given IP-2 and -3's higher probability for LOCAs and the fact that New York City is located less than 25 miles south of IP-2 and -3 and more than 17 million people live within a 50-mile radius of IP-2 and -3, the safety issues raised in this 10 C.F.R. § 2.206 petition, need prompt resolution. Both of these concerns are discussed in a study

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[9] Lynn R. Sykes, John G. Armbruster, Won-Young Kim, & Leonardo Seeber, Observations and Tectonic Setting of Historic and Instrumentally Located Earthquakes in the Greater New York City–Philadelphia Area, Bulletin of the Seismological Society of America, Vol. 98, No. 4, pp. 1696–1719, August 2008 (hereinafter "Sykes, *Earthquakes in New York*"); The Earth Institute, Columbia University, "Earthquakes May Endanger New York More than Thought, Says Study: Indian Point Nuclear Power Plant Seen as Particular Risk," Press Release Posted on The Earth Institute website, August 21, 2008, *available at*, http://www.earth.columbia.edu/articles/view/2235 (last visited March 24, 2011) (hereinafter "Columbia Earth Institute Earthquake Study Press Release").

[10] Sykes, *Earthquakes in New York*; Columbia Earth Institute Earthquake Study Press Release.

[11] *See, e.g.*, CBS New York, *Japan Crisis Raises Concerns About Indian Point Plant*, March 15, 2011, *available at*, http://newyork.cbslocal.com/2011/03/15/japan-crisis-raises-concerns-about-indian-point-power-plant/ (last visited March 24, 2011) ("Indian Point spokesman Jerry Nappi said . . . the plant is built to withstand approximately a 6.0 magnitude earthquake.").

[12] *See* Generic Issue 199 (GI-199), Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants Safety/Risk Assessment, August 2010, at Appendix D (Seismic Sore-Damage Frequencies), *available at*, ADAMS Accession Nos. ML100270639, ML100270756; Bill Dedman, *What are the odds? US nuke plants ranked by quake risk*, MSNBC.com, March 17, 2011, *available at*, http://www.msnbc.msn.com/id/42103936/ns/world_news-asia-pacific/ (last visited March 24, 2011).

conducted by Columbia University's Earth Institute, as memorialized in "Observations and Tectonic Setting of Historic and Instrumentally Located Earthquakes in the Greater New York City–Philadelphia Area." The study states: "Indian Point is situated at the intersection of the two most striking linear features marking the seismicity and also in the midst of a large population that is at risk in case of an accident… This is clearly one of the least favorable sites in our study area from an earthquake hazard and risk perspective."[13]

B. Information Regarding Author of Petition

Mark Leyse wrote this 10 C.F.R. § 2.206 petition on behalf of Riverkeeper. (Deborah Brancato of Riverkeeper helped write section II.A. regarding plant specific issues.)

On March 15, 2007, Mark Leyse submitted a petition for rulemaking, PRM-50-84 (ADAMS Accession No. ML070871368). PRM-50-84 was summarized briefly in American Nuclear Society's ("ANS") *Nuclear News*'s June 2007 issue[14] and commented on and deemed "a well-documented justification for…recommended changes to the [NRC's] regulations"[15] by Union of Concerned Scientists ("UCS"). In 2008, NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[16] 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.[17]

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

[13] Sykes, *Earthquakes in New York*, *supra* Note 9, at 1717.
[14] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[15] 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.
[16] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.
[17] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

10 C.F.R. § 50.46(b) ECCS acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.

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

Mark Leyse also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis."[18]

On November 17, 2009, Mark Leyse 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;[19] and 2) to stipulate minimum allowable core reflood rates, in the event of a LOCA.[20, 21]

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*

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[18] 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.

[19] 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.

[20] It can be extrapolated from experimental data from Thermal-Hydraulic Experiment 1, conducted in the National Research Universal reactor at Chalk River, Ontario, Canada, 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.

[21] 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.

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.[22] These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[23]

PRM-50-93 was discussed briefly in ANS's *Nuclear News*'s March 2010 issue[24] and in *Inside NRC*'s July 30, 2010 issue.[25] 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.[26]

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, Mark Leyse authored and submitted on behalf of New England Coalition—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

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

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

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

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

[26] 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.

C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[27]

Presently, Mark Leyse has almost completed a new 10 C.F.R. § 2.802 petition for rulemaking requesting revisions to Appendix K to Part 50—ECCS Evaluation Models I(D)(5) and I(D)(6).[28, 29]

C. Information Regarding Petition's Reviewer

This 10 C.F.R. § 2.206 petition was reviewed by Robert H. Leyse. Robert H. Leyse is a nuclear engineer and chemical engineer with decades of experience that is pertinent to this petition.

Robert H. Leyse

Education:

B.S. in Chemical Engineering, University of Wisconsin at Madison (1950).

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[27] 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.

[28] The new 10 C.F.R. § 2.802 petition will request that NRC revise Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized Water Reactors*, to require that reflood heat transfer coefficients shall be based on data from thermal hydraulic experiments conducted with full-length zirconium alloy multi-rod bundles (comprised of either fuel rods sheathing UO_2 fuel or realistic fuel rod simulators). In such experiments the zirconium alloy multi-rod bundles must be heated up to peak cladding temperatures of at least 2200°F, with a realistic range of different reflood rates. These same requirements will also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.

[29] The new 10 C.F.R. § 2.802 petition will request that NRC revise 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*, to require that convective heat transfer coefficients for boiling water reactor fuel rods under spray cooling shall be based on data from thermal hydraulic experiments conducted with full-length zirconium alloy multi-rod bundles (comprised of either fuel rods sheathing UO_2 fuel or realistic fuel rod simulators). In such experiments the zirconium alloy multi-rod bundles should be heated up to peak cladding temperatures of at least 2200°F, with a realistic range of different amounts of coolant supplied to each multi-rod bundle, with various combinations of different spray and reflood rates. These same requirements will also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.

Employment:

General Electric at Hanford Works (August 1950 to November 1952), Argonne National Laboratory (November 1952 to August 1954), Dupont Savannah River Plant (August 1954 to September 1955), U.S. Navy, San Francisco Naval Shipyard (September 1955 to November 1958), Argonne National Laboratory (November 1958 to January 1960), General Electric at Vallecitos (January 1960 to August 1967), Westinghouse Electric at Pittsburgh (August 1967 to April 1974), Gilbert Commonwealth Associates, Jackson Michigan (April 1974 to February 1976), Scandpower Inc., Halden Norway and Rockville Maryland (February 1976 to April 1979), Consulting with Westinghouse Electric on location during the Three Mile Island Unit 2 accident (April 1979), Electric Power Research Institute, Nuclear Safety Analysis Center and Exploratory Research (May 1979 to August 1997), Independent Consultant (August 1997 to Present).

Three of Robert H. Leyse's 10 C.F.R. § 2.802 Petitions for Rulemaking:

1) PRM-50-73, September 4, 2001: Addressing the fact that 10 C.F.R. § 50.46 and Appendix K to Part 50 do not address the impact of crud on coolability during a fast moving large break LOCA.

(The information presented in Robert H. Leyse's PRM-50-73 was essential for Mark Leyse's research and writing of PRM-50-84; NRC is currently considering issues raised in PRM-50-84 in its rulemaking process.)

2) PRM-50-76, May 8, 2002: Addressing the fact that the Baker-Just and Cathcart-Pawel correlations were not developed to consider how heat transfer would affect zirconium-water reaction kinetics in the event of a LOCA.

(The information presented in Robert H. Leyse's PRM-50-76 was essential for Mark Leyse's research and writing of PRM-50-93; NRC is currently reviewing PRM-50-93.)

(In the course of producing his public comments on Mark Leyse's PRM-50-93, Robert H. Leyse became aware that NRC staff had never studied the basic references of ANL-6548,[30] the report regarding the Baker-Just correlation. (The Baker-Just correlation is required by Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Metal-Water Reaction Rate*.) In NRC's technical review of PRM-50-76, NRC staff did not review the basic references of ANL-6548. Robert H. Leyse's actions in prompting

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

NRC to acquire the basic references[31] of ANL-6548 are well documented: see the letter from T. J. McGinty to Robert H. Leyse, dated April 16, 2010 (ADAMS Accession Number: ML100950085). Robert H. Leyse submitted Comment 13 on PRM-50-93 (ADAMS Accession Number: ML101020563), emphasizing that PRM-50-93 is based on sound science and that NRC staff had not had access to the reports (discussing experiments that the Baker-Just correlation is primarily based on) cited in ANL-6548, until March 2010.

Based on his analysis of the key reports referenced in ANL-6548, that NRC staff had never studied, Robert H. Leyse stated to the ACRS Subcommittee on Plant License Renewal, September 8, 2010 (ADAMS Accession Number: ML102530135), that "[i]t is absurd to license the emergency cooling of tons of zirconium alloy, having thousands of square feet of interfacial surface area, based on the limited investigations that yielded the Baker-Just equation.")

3) PRM-50-78, September 9, 2002: Addressing the need for regulations regarding the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants.

Robert H. Leyse's Work Experience that is Especially Pertinent to this Petition:

1) At Argonne National Laboratory, Robert H. Leyse initiated studies of the thermal impact of scale (crud) on the zirconium alloy cladding of the EBWR.

2) At General Electric at Vallecitos, Robert H. Leyse conducted experiments with single-rod Zircaloy cladding specimens and single-rod stainless steel cladding specimens in an induction-heated furnace, in a steam environment, with temperature measured by optical pyrometry.

3) At Westinghouse at Pittsburgh, Robert H. Leyse was the principal engineer in charge of directing the FLECHT multi-rod bundle Zircaloy experiments and the FLECHT multi-rod bundle stainless experiments.

(The data reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," is important for ECCS evaluation calculations, required for all holders of operating licenses for nuclear power plants. Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized*

[31] One of which is the report by Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957.

Water Reactors, states that "[f]or reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"].")

4) Westinghouse Electric had an active role during the early stages of the Three Mile Island Unit 2 accident. H. N. Andrews who led the Westinghouse activities hired Robert H. Leyse as a consultant because he was aware of Leyse's work on the PWR FLECHT experiments, as a Westinghouse principal engineer in the late 1960s. Reporting to Andrews at the TMI-2 site, Leyse drew on his experience from FLECHT run 9573 and showed Andrews photographs of a severely damaged portion of the bundle from FLECHT run 9573.[32] Leyse outlined a range of likely conditions of the core and pointed out the need for continued cooling via forced circulation, emphasizing that heat removal problems would become substantially less challenging with time. Leyse's projection of the condition of the core turned out to be reasonably accurate as a substantial amount of FLECHT run 9573-type debris was found when the core was examined several years latter.

Partial List of Robert H. Leyse's Reports and Publications:

1) R. H. Leyse, Argonne National Laboratory, "Scale on Fuel Elements in EBWR": a report incorporated into "EBWR Operational History," ANL-6229.

2) C. R. Breden, I. Charak, R. H. Leyse, Argonne National Laboratory, "Water Chemistry and Fuel Element Scale in EBWR," ANL-6136, November 1960.

3) V. C. Hall Jr., R. H. Leyse, "Radiation Levels in EBWR," Nucleonics, Vol. 19, No. 3, March 1, 1961.

4) R. H. Leyse, General Electric, Vallecitos Atomic Laboratory, "Fuel Element Abrasion in the General Electric Test Reactor," APED-4454, November 26, 1963.

5) R. H. Leyse, General Electric, Vallecitos Atomic Laboratory, "Zircaloy-2 and Type-304 Stainless Steel at 2000°F in Water-Steam for Brief Times," APED-4413, April 27, 1964.

6) J. Cermak, R. Farman, A. Kitzes, R. Leyse, H. Skreppen, "PWR FLECHT Final Test Plan," WCAP-7288, January 1969, located at:

[32] See Appendix J for photographs of the bundle from FLECHT Run 9573.

www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070750190.

7) J. O. Cermak, A. S. Kitzes, F. F. Cadek, R. H. Leyse, D. P. Dominicis, “PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,” WCAP-7435, January 1970, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780075.

8) F. F. Cadek, D. P. Dominicis, R. H. Leyse, “PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report,” WCAP-7544, September 1970, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070750189.

9) F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, “PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,” WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083.

10) F. F. Cadek, D. P. Dominicis, H. C. Yeh, R. H. Leyse, “PWR FLECHT Final Report Supplement,” WCAP-7931, October 1972, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780086.

11) R. H. Leyse, R. D. Smith: “Gamma Thermometer Developments for Light Water Reactors,” IEEE Transactions on Nuclear Science, Vol.N5.26, No. 1, February 1979, pp. 934–943.

12) Robert H. Leyse, “Microscale Heat Transfer to Subcooled Water,” Annals of the New York Academy of Sciences, Microgravity Transport Processes in Fluid, Thermal, Biological Sciences II, 974, 2001, pp. 260-273, located at:
http://www3.interscience.wiley.com/journal/118947467/abstract

13) Robert H. Leyse, “Unmet Challenges for SCDAP/RELAP5-3D: Analysis of Severe Accidents for Light Water Nuclear Reactors with Heavily Fouled Cores,” Presentation at 2003 RELAP5 International Users Seminar, West Yellowstone, Montana, located at:
www.inl.gov/relap5/rius/yellowstone/leyse.pdf

14) Robert H. Leyse and Phani K. Meduri, Gopinath R. Warrier, and Vijay K. Dhir of UCLA, “Microscale Phase Change Heat Transfer at High Heat Flux,” Proceedings of the 5th International Conference on Boiling Heat Transfer, Montego Bay, Jamaica, 2003, located at: boiling.seas.ucla.edu/Publications/Conf_LMWD2003

15) Robert H. Leyse, “Nuclear Power Blog,” located at: http://nuclearpowerblog.blogspot.com

III. FACTS CONSTITUTING THE BASIS FOR PETITIONER’S REQUEST

Petitioner requests that NRC order the licensee of IP-2 and -3 to lower the LBPCTs of IP-2 and -3 in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Experimental data—from multi-rod severe fuel damage experiments conducted in the aftermath of the Three Mile Island Unit 2 accident—demonstrates that IP-2 and -3’s LBPCTs of 1937°F and 1961°F, respectively, do not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. For example, it is reported that in some multi-rod severe fuel damage experiments the fuel rods incurred runaway oxidation when they locally reached temperatures as low as approximately 1832°F.

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

(It is noteworthy that the Baker-Just correlation—used in Appendix K to Part 50 ECCS evaluation calculations—is primarily based on data from Lemmon and Bostrom’s experiments,[33] conducted with inductively heated Zircaloy-2 specimens. In Lemmon’s experiments, “Lemmon measured the rates of reaction between Zircaloy-2 and steam in the temperature range 1000-1700°C by inductively heating specimens in steam at 50 psia and measuring the rate of hydrogen evolution.”[34] (Bostrom’s experiments were

[33] G. Schanz, “Recommendations and Supporting Information on the Choice of Zirconium Oxidation Models in Severe Accident Codes,” FZKA 6827, 2003, p. 2.
[34] V. F. Urbanic and T. R. Heidrick, “High-Temperature Oxidation of Zircaloy-2 and Zircaloy-4 in Steam,” Journal of Nuclear Materials 75, 1978, p. 252.

conducted in a temperature range above that of design basis accidents: 1300-1860°C.[35]) Lemmon's specimen was a Zircaloy-2 cylinder that was 2 inches long and 0.5 inches in diameter.[36])

The 1971 IP-2 licensing hearing transcript states:

> James S. Moore: No. There are no large-scale tests for the core. You are talking about a very complex chain of events. You are ending up with a zirc-water reaction. And you have to start with the loss of coolant and go through the blowdown, the reflood, the heat-up, the time and temperature, and then the zirc-water reaction.
>
> Daniel Ford: Right. Now in terms of simply your experimental philosophy do you see the necessity, since there are as you note so many complicated factors behind any independent phenomenon, do you see the necessity, the experimental necessity for the kind of integral test that I am talking about or do you think that you can just test individual small components of the problem, you know, assuming all the input from other phenomena?
>
> James S. Moore: I believe it is my opinion that we can properly bound the calculation without a total completely integrated test.
> …
>
> Daniel Ford: I am talking about the water reactor safety program which has a variety of experiments on different… Using a variety of different equipment to simulate loss-of-coolant accident[s]. And I am talking about some of the large-scale experiments that are planned to take place [in] 1975 or so in which we will actually have a live reactor and have it subjected to loss-of-coolant transients and see what happens. I am talking about whether or not that is necessary in Mr. Moore's opinion, whether that would make a substantive contribution to the confirmation of these results on metal-water reactions inasmuch as they depend on all the other phenomena of the transient. I am asking him whether that is necessary or whether you can simply take Baker-Just's correlation, which is derived from experimental data that is completely outside of the context of nuclear systems? I am asking him whether we should have these kinds of integral experiments or whether we can just take empirical correlations and just use them with no hesitation…
>
> James S. Moore: I count at least four or five questions in it. Do I think it necessary, do I think it would contribute?

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[35] *Id*.

[36] Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100570218, p. C-4.

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

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

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

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

Daniel Ford: Yes.

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

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

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

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

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

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

James S. Moore: No specific experiment, complete integrated experiment.[37]

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[37] 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:

Unfortunately, to this day, nearly 40 years after the original IP-2 licensing hearing, the metal-water reaction rates predicted to occur in the event of LOCAs at IP-2 and -3 still have *not* been confirmed by data from large-scale integral experiments. In fact, now in 2010, there is a preponderance of metal-water-reaction-rate data from multi-rod severe fuel damage experiments like the LOFT LP-FP-2 experiment; nevertheless, IP-2 and -3's ECCS evaluation calculations still use metal-water reaction rate correlations that were derived from the data of single-rod experiments.

It is significant that for the LOFT LP-FP-2 experiment, "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production."[38]

The LOFT LP-FP-2 experiment, conducted in 1985, is considered "particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[39] In the LOFT LP-FP-2 experiment, "[t]he first recorded and qualified rapid temperature rise associated with the rapid reaction between Zircaloy and water occurred at about…[2060°F]"[40]—approximately 140°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

A. Entergy's Best-Estimate ECCS Evaluation Calculations for IP-2 and -3 are Non-Conservative

When Entergy performed ECCS evaluation calculations for the safety evaluations that helped qualify the power uprates of IP-2 and -3 (authorized by NRC in 2004 and

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www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, pp. 2550-2553.

[38] Report by Nuclear Energy Agency Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

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

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

2005, respectively),[41] the calculations were done with Westinghouse's WCOBRA/TRAC code.[42]

The WCOBRA/TRAC code is a best estimate code: best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC's Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."[43]

Regarding metal-water reaction rates, Regulatory Guide 1.157, 3.2.5, Metal-Water Reaction Rate, states:

> The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be calculated in a best-estimate manner. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis. For rods calculated to rupture their cladding during the loss-of-coolant accident, the oxidation of the inside of the cladding should be calculated in a best-estimate manner.[44]

And regarding the best-estimate model evaluation procedure for metal-water reaction rates, Regulatory Guide 1.157, 3.2.5.1, Model Evaluation Procedure for Metal-Water Reaction Rate, states:

> Correlations to be used to calculate metal-water reaction rates at less than or equal to 1900°F should:
>
> a. Be checked against a set of relevant data, and
>
> b. Recognize the effects of steam pressure, pre-oxidation of the cladding, deformation during oxidation, and internal oxidation from both steam and UO_2 fuel.
>
> The data of ["Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies"[45]] are considered acceptable for calculating the rates of

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[41] The power uprates for IP-2 and -3 were 3.26 % and 4.85%, respectively.
[42] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, pp. 6-7. Such calculations were also done for IP-3.
[43] NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989 (hereinafter "Regulatory Guide 1.157").
[44] *Id*., p. 6.
[45] J. V. Cathcart et al., "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977.

> energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F.[46]

(It is noteworthy that the Cathcart-Pawel correlation is based on data from "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies." Cathcart and Pawel's experiments were conducted in two different furnaces with Zircaloy-4 PWR tube specimens. In the MaxiZWOK furnace, the specimen was 18 inches long (only a small segment of that tube—in close proximity to the thermocouple stations—served as the specimen); in the MiniZWOK furnace, the specimen was about 1.2 inches long.[47])

It is significant that approximately half of more than 50 LOCA calculations that the NRC performed with RELAP5/Mod3 that used the Cathcart-Pawel correlation predicted autocatalytic (runaway) oxidation to commence when cladding temperatures increased to above approximately 2700°F,[48] 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 600 degrees Fahrenheit lower than 2700°F. Therefore, the Cathcart-Pawel 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 Cathcart-Pawel 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," presented in 1991, explicitly states "[c]ladding oxidation [in the CORA-16 experiment] was not accurately

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[46] NRC, Regulatory Guide 1.157, p. 6.

[47] J. V. Cathcart, R. E. Pawel, *et al*., "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079, pp. 12, 15.

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

predicted by available correlations."[49] (In 1991, the Cathcart-Pawel correlation was among the available correlations.)

Discussing "experiment-specific analytical modeling at [Oak Ridge National Laboratory ("ORNL")] for CORA-16,"[50] 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].[51]

So, in the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations"[52] 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."[53] This indicates that available Zircaloy oxidation kinetics models—including best-estimate models—are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

Furthermore, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," published in 2001, states that

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[49] 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 (hereinafter "In-Vessel Phenomena—CORA").
[50] 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 (hereinafter "Report of Foreign Travel of L. J. Ott").
[51] *Id*.
[52] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA."
[53] L. J. Ott, "Report of Foreign Travel of L. J. Ott," p. 3.

"[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[54]

In more detail, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I states:

> Reflooding and quenching of the uncovered core is the most important accident management measure to terminate a severe accident transient. If the core is overheated, this measure can lead to increased oxidation of the Zircaloy cladding which in turn can trigger a temperature escalation. Relatively short flooding and quenching times can thereby lead to high hydrogen source rates which must be taken into account in risk analysis and in the design of hydrogen mitigation systems.
>
> Until recently, the experimental database on quenching phenomena was rather scarce. The available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the few available tests (CORA, LOFT LP-FP-2).[55]

This also indicates that available Zircaloy oxidation kinetics models—including best-estimate models—are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

(It is noteworthy that " "GAMA Perspective Statement on In-Vessel Hydrogen Sources," [was] prepared by B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), on the basis of information collected from GAMA [Working Group on the Analysis and Management of Accidents] members and the previous Principal Working Group on Coolant System Behaviour (PWG2). It was endorsed by GAMA in April 2001 and approved for publication by CSNI in June 2001."[56])

(It is also noteworthy that "[GAMA] is mainly composed of technical specialists in the areas of coolant system thermal-hydraulics, in-vessel protection, containment protection, and fission product retention. Its general functions include the exchange of information on national and international activities in these areas, the exchange of detailed technical information, and the discussion of progress achieved in respect of

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[54] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I, B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 9 (hereinafter: "In-Vessel and Ex-Vessel Hydrogen Sources," Part I).

[55] *Id.*

[56] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, p. 5.

specific technical issues. Severe accident management is one of the important tasks of the group."[57])

Therefore, the ECCS evaluation calculations that helped qualify the power uprates for IP-2 and -3—which used best-estimate models for calculating the metal-water reaction rates that would occur in the event of a LOCA—are non-conservative.

1. Best-Estimate ECCS Evaluation Calculations

Regulatory Guide 1.157 states that "the terms 'best-estimate' and 'realistic' have the same meaning."[58] And regarding best-estimate calculations, Regulatory Guide 1.157 states:

> A best-estimate calculation uses modeling that attempts to realistically describe the physical processes occurring in a nuclear reactor. There is no unique approach to the extremely complex modeling of these processes. The NRC has developed and assessed several best-estimate advanced thermal-hydraulic transient codes. These include TRAC-PWR, TRAC-BWR, RELAP-5, COBRA, and the FRAP series of codes… These codes reasonably predict the major phenomena observed over a broad range of thermal-hydraulic and fuel tests. …
>
> A best-estimate model should provide a realistic calculation of the important parameters associated with a particular phenomenon to the degree practical with the currently available data and knowledge of the phenomenon. *The model should be compared with applicable experimental data* and should predict the mean of the data, rather than providing a bound to the data. …
>
> A best-estimate code contains all the models necessary to predict the important phenomena that might occur during a loss-of-coolant accident. *Best-estimate code calculations should be compared with applicable experimental data* (*e.g.*, separate-effects tests and integral simulations of loss-of-coolant accidents) to determine the overall uncertainty and biases of the calculation. In addition to providing input to the uncertainty evaluation, *integral simulation data comparisons should be used to ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted* [emphasis added].[59]

[57] *Id.*, p. 3.
[58] NRC, Regulatory Guide 1.157, p. 1, footnote 1.
[59] *Id.*, p. 3.

So a best-estimate ECCS evaluation calculation is supposed to "be compared with applicable experimental data"[60] and "ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted,"[61] however, data from multi-rod severe fuel damage experiments indicates that, with high probability, autocatalytic oxidation would commence at cladding temperatures far lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

For example, in the event of a LOCA at either IP-2 or IP-3, experimental data indicates that if peak cladding temperatures increased to approximately 1832°F or greater, with high probability, the Zircaloy cladding would begin to rapidly oxidize, and cladding temperatures would start increasing at a rate of tens of degrees Fahrenheit per second.[62] Within a period of approximately 60 seconds, peak cladding temperatures would increase to approximately 3000°F[63] or greater; the melting point of Zircaloy is approximately 3308°F.[64]

Regarding best-estimate ECCS evaluation calculations and safety issues, Regulatory Guide 1.157 states:

> It was also found that some plants were being restricted in operating flexibility by limits resulting from conservative Appendix K requirements. Based on the research performed, it was determined that these restrictions could be relaxed through the use of more realistic calculations *without adversely affecting safety*. …
>
> *Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations* [emphasis added].[65]

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[60] *Id*.
[61] *Id*.
[62] 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.
[63] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p 23.
[64] 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.
[65] NRC, Regulatory Guide 1.157, p. 2.

Indeed, safety would be best served if decisions concerning the limits within which nuclear reactors are permitted to operate were actually based on realistic calculations. For example, realistic ECCS evaluation calculations of the metal-water reaction rates would be based on data from multi-rod severe fuel damage experiments. Furthermore, data from such experiments indicates that IP-2 and -3's LBPCTs of 1937°F and 1961°F, respectively, are both non-conservative.

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

The alleged conservatism of IP-2 and -3 LBPCTs of 1937°F and 1961°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. 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.[66]

[66] 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.

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

> Assessment of the conservatism in the PCT limit can be accomplished by comparison to multi-rod (bundle) data for the autocatalytic temperature. This type of comparison implicitly includes…complex heat transfer mechanisms…and the effects of fuel rod ballooning and rupture on coolability… Analysis of experiments performed in the Power Burst Facility, in the Annular Core Research Reactor, and in the NEILS-CORA (facilities in West Germany) program have shown that temperatures above 2200°F are required before the zircaloy-steam reaction becomes sufficiently rapid to produce an autocatalytic temperature excursion. Another group of relevant experimental data were produced from the MT-6B and FLHT-LOCA and Coolant Boilaway and Damage Progression tests conducted in the NRU Reactor in Canada. …even though some severe accident research shows lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists.[67]

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. [68]

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[67] *Id.*

[68] 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.

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

(It is noteworthy that "Compendium of ECCS Research for Realistic LOCA Analysis," published in 1988, does not mention that some reports state that autocatalytic oxidation commenced in the LOFT LP-FP-2 experiment—conducted in 1985—at cladding temperatures of approximately 2060°F.[70])

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].[71]

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[69] 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 (hereinafter "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods").

[70] 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.

[71] 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.

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].[72]

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

1. Data from Thermal Hydraulic Experiments Demonstrates that the Zircaloy-Steam Reaction is Not Negligible below 1900°F

In Atomic Energy Commission ("AEC") responses to questions submitted by Anthony Z. Roisman, pertaining to the IP-2 licensing hearing, AEC stated:

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[73]

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

[72] 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.

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

It is significant that data from multi-rod bundle Thermal-Hydraulic Experiment 1 ("TH-1") test no. 130 demonstrates that the Zircaloy-steam reaction is *not* negligible below 1900°F.

In TH-1 test no. 130—conducted with a full-length Zircaloy multi-rod fuel assembly—there was a reflood rate of 0.74 in./sec.[74] At the start of reflood, the peak cladding temperature ("PCT") was 998°F, and in the test the overall PCT was 2040°F—an increase of 1042°F.[75]

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,[76] in the pre-transient phase of the test.

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.[77] 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 runaway oxidation if the reactor had not shutdown when the PCT was approximately 1850°F. In such a case, the overall PCT would have far exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

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[74] 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, Abstract, p. v (hereinafter "Prototypic Thermal-Hydraulic Experiment"). The Abstract states that the lowest reflood rate in the TH-1 tests was 1.88 cm/ sec (0.74 in./sec); the Summary states that the lowest reflood rate in the TH-1 tests was 0.74 in./sec; page 13 states that the reflood rate of TH-1 test no. 130 was 0.7 in./sec: so the value of "0.7 in./sec," given on page 13, was rounded off from 0.74 in./sec.
[75] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," p. 13.
[76] *Id.*, p. 10.
[77] *Id.*

So data from multi-rod bundle thermal hydraulic experiments demonstrates that the Zircaloy-steam reaction is not negligible below 1900°F.

C. Recent LBPCTs of IP-2 and -3

1. IP-2's LBPCT Calculated in 2001 with Westinghouse's 1996 WCOBRA/TRAC Computer Code

In 2001, IP-2 had a LBPCT of 2188°F in a computer simulated large break ("LB") LOCA—only 12°F lower than the requirements of 10 C.F.R. § 50.46(b)(1).[78]

2. IP-2's LBPCTs Calculated with WCOBRA/TRAC for IP-2's 2004 Stretch Power Uprate

The ECCS evaluation calculations that helped qualify IP-2's 2004 stretch power uprate, calculated IP-2's LBPCTs at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies for a postulated LB LOCA.[79]

3. IP-3's LBPCT Calculated with WCOBRA/TRAC for IP-3's 2005 Stretch Power Uprate

The ECCS evaluation calculations that helped qualify IP-3's 2005 stretch power uprate, calculated IP-3's LBPCT at 1944°F for a postulated LB LOCA.[80]

4. IP-2's LBPCTs Calculated in 2005 with Westinghouse's ASTRUM Methodology, Bounded in the 1996 WCOBRA/TRAC Code

In 2006, IP-2 was issued an amendment to its operating license for its LB LOCA analysis methodology: IP-2's ECCS evaluation calculations were converted to the

[78] Consolidated Edison Company of New York, Inc., "Indian Point Unit 2 – 30 Day and Annual 10 CFR 50.46 Report," April 10, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011150434, p. 1.
[79] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," October 27, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042960007, Enclosure 2, p. 18.
[80] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," March 24, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050600380, Enclosure 2, p. 16.

"Realistic Large Break LOCA Evaluation Methodology using the Automated Statistical Treatment of Uncertainty Method ("ASTRUM")."[81] With the ASTRUM methodology, bounded in the 1996 WCOBRA/TRAC code,[82] IP-2's LBPCT was calculated at 1962°F for ZIRLO cladding in 422 Vantage assemblies and at 1814°F for ZIRLO cladding in 15x15 assemblies.[83]

5. The Current LBPCTs of IP-2 and -3

a. IP-2's Current LBPCT of 1937°F

In 2006, IP-2 was issued an amendment to its operating license for its LB LOCA analysis methodology: IP-2's ECCS evaluation calculations were converted to the ASTRUM methodology, bounded in the 1996 WCOBRA/TRAC code. IP-2's LBPCT was calculated at 1962°F with the ASTRUM methodology, bounded in the 1996 WCOBRA/TRAC code.

In 2007, for IP-2, Entergy's 2006 annual 10 C.F.R. 50.46 report, reported "a 1°F large break LOCA PCT penalty associated with changes to the containment sump strainer."[84]

In 2008, for IP-2, Entergy's 2007 annual 10 C.F.R. 50.46 report, reported no adjustments for IP-2's LBPCT.[85]

[81] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Large Break Loss-of-Coolant Accident (LBLOCA) Analysis Methodology," July 24, 2006, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML061710291.
[82] Cesare Frepoli, Katsuhiro Ohkawa, Robert M. Kemper, "Realistic Large Break LOCA Analysis of AP1000 with ASTRUM," The 6th International Conference on Nuclear Thermal Hydraulics, Operations and Safety, Nara, Japan, October 4-8, 2004, pp. 7-8.
[83] See NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Large Break Loss-of-Coolant Accident (LBLOCA) Analysis Methodology," Enclosure 2, p. 3; see also Entergy, Attachment 1 to NL-05-058, "Reanalysis of Large Break Loss of Coolant Accident Using ASTRUM," April 22, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML051230311, pp. 1-2.
[84] Entergy, "Annual 10 C.F.R. 50.46 Report for Year 2006: Emergency Core Cooling System Evaluation Changes," December 4, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML073460417, p. 1.
[85] Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2007: Emergency Core Cooling System Evaluation Changes," October 14, 2008, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML083030163, p. 1.

In 2009, for IP-2, Entergy's 2008 annual 10 C.F.R. 50.46 report, reported "the elimination of the transition core penalty (26°F) [for a LB LOCA] beginning with cycle 19 which commenced with startup from the Spring 2008 refueling outage."[86]

In 2010, for IP-2, Entergy's 2009 annual 10 C.F.R. 50.46 report, reported no adjustments for IP-2's LBPCT.[87]

So IP-2's current LBPCT is 1937°F: 1962°F +1°F -26°F.[88]

b. IP-3's Current LBPCT of 1961°F

The ECCS evaluation calculations that helped qualify IP-3's 2005 stretch power uprate, calculated IP-3's LBPCT at 1944°F for a postulated LB LOCA.

In 2007, for IP-3, Entergy's 2006 annual 10 C.F.R. 50.46 report, reported no adjustments for IP-3's LBPCT.[89]

In 2008, for IP-3, Entergy's 2007 annual 10 C.F.R. 50.46 report, stated that "Entergy is reporting two adjustments for the IP-3 large break LOCA [analysis of record]. The first adjustment (+15°F) is the result of a correction to the Evaluation Model for HOTSPOT fuel relocation. The second adjustment (+2°F) accounts for the effects of the containment sump strainer modification installed during the Spring 2007 refueling outage and is not a change to the Evaluation Model."[90]

In 2009, for IP-3, Entergy's 2008 annual 10 C.F.R. 50.46 report, reported no adjustments for IP-3's LBPCT.[91]

In 2010, for IP-3, Entergy's 2009 annual 10 C.F.R. 50.46 report, stated that "the transition core penalty [for a LB LOCA] is now eliminated beginning with cycle 16

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[86] Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2008: Emergency Core Cooling System Evaluation Changes," July 21, 2009, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML092110189, p. 1.

[87] Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2009: Emergency Core Cooling System Evaluation Changes," August 4, 2010, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML102250210, p. 1.

[88] Entergy's recent annual 10 C.F.R. 50.46 reports do not provide the value of IP-2's current LBPCT.

[89] Entergy, "Annual 10 C.F.R. 50.46 Report for Year 2006: Emergency Core Cooling System Evaluation Changes," December 4, 2007, p. 1.

[90] Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2007: Emergency Core Cooling System Evaluation Changes," October 14, 2008, p. 1.

[91] Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2008: Emergency Core Cooling System Evaluation Changes," July 21, 2009, p. 1.

which commenced with startup from the Spring 2009 refueling outage."[92] Entergy's 2009 annual 10 C.F.R. 50.46 report does not provide a value for the transition core penalty that was eliminated for IP-3.

So IP-3's current LBPCT is 1961°F: 1944°F +15°F +2°F.[93]

D. Experiments that Indicate IP-2 and -3's LBPCTs of 1937°F and 1961°F, Respectively, would Not Provide 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.[94]—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?[95]—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.[96]—S. Hagen, *et al*.

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92 Entergy, "Annual 10 C.F.R. 50.46 Report for Calendar Year 2009: Emergency Core Cooling System Evaluation Changes," August 4, 2010, p. 1.

93 Entergy's recent annual 10 C.F.R. 50.46 reports do not provide the value of IP-3's current LBPCT.

94 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.

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

96 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 (hereinafter: "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3").

In this section, Petitioner will discuss data from multi-rod severe fuel damage experiments and two multi-rod thermal hydraulic experiments that indicates IP-2 and -3's LBPCTs of 1937°F and 1961°F, respectively, would not provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Petitioner will discuss: 1) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures below IP-2 and -3's LBPCTs of 1937°F and 1961°F, respectively; 2) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures of 2060°F or lower; 3) experiments in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at temperatures of approximately 2192°F (approximately at the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F); and 4) one experiment in which the autocatalytic oxidation of Zircaloy cladding by steam commenced at a temperature of 2275°F or lower.

(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 boiling water reactor ("BWR") fuel assemblies. There would definitely be differences in how the different ECCSs and core components of pressurized water reactors ("PWR") and BWRs (*e.g.*, the PWR Ag-In-Cd absorber versus the BWR boron carbide (B_4C) 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 PWR and BWR would be similar, as the results of multi-rod severe fuel damage experiments that simulated PWR and BWR fuel assemblies indicate.

1. Multi-Rod Severe Fuel Damage Experiments in which the Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at Temperatures below IP-2 and -3's LBPCTs of 1937°F and 1961°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

a. The CORA Experiments in which the Autocatalytic Oxidation of Zircaloy Cladding by Steam Commenced at Temperatures below IP-2 and -3's LBPCTs of 1937°F and 1961°F, Respectively

IP-2 and -3's LBPCTs of 1937°F and 1961°F, respectively, would not provide a necessary margin of safety to help prevent a partial or complete meltdown, in the event of a LOCA. Experimental data demonstrates that IP-2 and -3's LBPCTs must both be decreased to a temperature lower than 1832°F in order to provide a necessary margin of safety.

It is significant that the CORA-2 and CORA-3 experiments, initiated with a temperature ramp rate of 1 K/sec, had rapid temperature increases, due to the exothermal Zircaloy-steam reaction, that commenced at approximately 1000°C (1832°F),[97] leading the CORA-2 and CORA-3 bundles to maximum temperatures of 2000°C and 2400°C, respectively.[98]

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)],[99] 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

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[97] 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.

[98] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3," KfK 4378, Abstract.

[99] 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.

> oxidation, the strong increase of the reaction rate with increasing temperature, together with the excellent thermal insulation of the bundles.[100]

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…[101]

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."[102]

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

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[100] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3)," KfK 4378, p. 41.

[101] 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.

[102] 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.

seconds peak cladding temperatures would have increased to above 3000°F; the melting point of Zircaloy is approximately 3308°F.[103]

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

Providing additional information on the CORA-2 and CORA-3 experiments, the abstract of "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.[104]

[103] 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.

[104] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3" KfK 4378, Abstract.

And discussing video and still cameras that recorded the CORA-2 and CORA-3 experiments, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3" states:

> The high-temperature shield is located within the pressure tube. Through a number of holes in the shield, the test bundle is being inspected during the test by several video and still cameras. The holes are also used for temperature measurements by two-color pyrometers complementing the thermocouple readings at elevated temperatures.[105]

And discussing the interpretation of the CORA-2 and CORA-3 experiments results, "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)],[106] 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.

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

[106] 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.

> A first melting process starts already at about 1250°C at the central grid spacer of Inconel, due to diffusive interaction in contact with Zry cladding material, by which the melting temperatures of the interaction partners (ca. 1760°C for Zry, ca. 1450°C for Inconel) are dramatically lowered towards the eutectic temperature, where a range of molten mixtures solidifies. (This behavior is similar to that of the binary eutectic systems Zr-Ni and Zr-Fe with eutectic temperatures of roughly 950°C).[107]

It is significant that "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)],[108] the temperature traces recorded during the tests CORA-2 and -3 indicate an increase in the heatup rate above 1000°C."[109] So the CORA 2 and CORA 3 experiments were not the only CORA experiments to have rapid temperature increases 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].[110]

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

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[107] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3," KfK 4378, p. 41.
[108] 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.
[109] S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3," KfK 4378, p. 41.
[110] 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 (hereinafter: "Results of SFD Experiment CORA-13").

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.[111]

So there is experimental data that demonstrates, as well as papers that report, the autocatalytic oxidation of Zircaloy cladding by steam commences at temperatures below IP-2 and -3's LBPCTs of 1937°F and 1961°F, respectively. Therefore, in the event of a LOCA at either IP-2 or IP-3, with high probability, if peak cladding temperatures reached temperatures greater than approximately 1832°F, the Zircaloy cladding would begin to rapidly oxidize, 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.[112]

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.)

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[111] 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.
[112] 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 IP-2 and -3'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,[113] 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."[114]

Fourth, information presented in "TMI-2," 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," 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," one of the presentation slides states that Dr. Dana Powers sent Dr. Henry the slides Dr. Powers had used in lectures on the TMI-2 accident and that Hans Fauske, D.Sc., reviewed all of the slides presented in "TMI-2": 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," states:

Fuel Cladding Oxidation

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[113] Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 ANS/ENS International Meeting, November 11, 2007 (hereinafter: "TMI-2"), 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.

[114] *Id*.

- 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].[115]

So Dr. Henry postulated that runaway oxidation commenced at approximately 1000°C. And another one of the presentation slides from "TMI-2," 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."[116]

It is significant that one of the presentation slides from "TMI-2," 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.

- 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.[117]

[115] *Id*.
[116] *Id*.
[117] *Id*.

It is also significant that another one of the presentation slides from "TMI-2," 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[118]

So in "TMI-2," 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 IP-2 and -3 LBPCTs of 1937°F and 1961°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 TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents."[119]

(In the pre transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[120] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40

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[118] *Id*.
[119] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," NUREG/CR-1882.
[120] *Id*., p. 10.

psia):[121] "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."[122])

In TH-1 test no. 130, there was a reflood rate of 0.74 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.[123]

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.[124] 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.

(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,[125] 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 IP-2 or IP-3, with high probability, if peak cladding temperatures reached temperatures of approximately 1850°F, the Zircaloy cladding would begin to rapidly oxidize, and that—

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[121] 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.
[122] *Id.*
[123] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment, p. 13.
[124] *Id.*
[125] *Id.*, p. 10.

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.[126]

(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.)

(It is noteworthy that the reactor tripped (shutdown) in TH-1 test No. 130, because, according to the test plan, if certain parameters were exceeded for the test section, in the TH-1 tests, the reactor would automatically shutdown. It is probable that the reactor shutdown in TH-1 test No. 130, because the test guidelines for (at least) one of the five parameters (low steam flow, fuel cladding high temperature, reflood low flow, accumulators low inventory, transient termination time) was exceeded.[127])

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

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

At least two papers on the PWR CORA experiments state that in some of the CORA experiments there were rapid cladding temperature increases due to the autocatalytic oxidation reaction of Zircaloy cladding that commenced at approximately 2012°F.[128]

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

[127] 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, p. 4-16.

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

(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.[129] 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 rapid cladding temperature increases 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].[130]

And regarding the same phenomenon, "Behavior of AgInCd Absorber Material" 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].[131]

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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).

[129] 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.

[130] 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 (hereinafter: "Behavior of AgInCd Absorber Material").

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

Second, regarding rapid cladding temperature increases due to the autocatalytic oxidation reaction of Zircaloy cladding, the abstract of "Results of SFD Experiment CORA-13" 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].[132]

And regarding the same phenomenon, "Results of SFD Experiment CORA-13" 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*

[132] S. Hagen, P. Hofmann, V. Noack, G. Schanz, G. Schumacher, L. Sepold, "Results of SFD Experiment CORA-13," 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].[133]

So "Behavior of AgInCd Absorber Material" and "Results of SFD Experiment CORA-13" 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" 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."[134] Additionally, "Behavior of AgInCd Absorber Material" 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."[135]

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" 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].[136]

And elsewhere, regarding the same phenomenon, "Behavior of AgInCd Absorber Material" 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)…[137]

[133] *Id.*, p. 12.
[134] *Id.*
[135] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material," FZKA 7448, p. 1.
[136] *Id.*, p. 5.
[137] *Id.*, p. 7.

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

A third paper on the CORA experiments, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," states that in the CORA experiments there where rapid cladding temperature increases due to the autocatalytic oxidation reaction of Zircaloy cladding that commenced at temperatures between approximately 2012°F and 2192°F.

In more detail, the third paper 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."[138]

"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."[139]

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."[140] 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

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[138] 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.
[139] *Id.*, p. 87.
[140] *Id.*, p. 83.

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

b. 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;"[141] and states that "[t]he transient of a SFD-type accident is initiated by a slow temperature rise in the order of 0.5 [to] 1.0 K/sec., followed by a rapid temperature escalation (several tens of degrees Kelvin per second) due to the exothermal heat produced by the Zry cladding oxidation in steam environment."[142]

Regarding the BWR CORA experiments the abstract of "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber" 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

[141] 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 (hereinafter: "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber").
[142] *Id*., p. 1.

> 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.[143]

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" states:

> In the escalation phase; *i.e.*, starting from about 1100°C, the slow temperature rise was followed by a rapid increase caused by the energy from the exothermal zirconium-steam reaction which becomes significant at the temperature mentioned and in addition—the electric power input. The contribution of the exothermal heat to the total energy; *i.e.*, electrical and chemical power, is generally between 30 and 50%. For CORA-16, CORA-17, and CORA-18 the chemical reaction contributes to 48, 44, and 33 %, respectively.[144]

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.

[143] *Id.*, p. i.
[144] *Id.*, p. 5.

Regarding the rapid temperature increase in the CORA-18 experiment (and two PWR CORA experiments), the document, "Report of Foreign Travel of L. J. Ott," which is partly a report on the 1990 CORA Workshop at KfK GmbH, Karlsruhe, FRG, October 1-4, 1990,[145] 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.[146]

And regarding "experiment-specific analytical modeling at [ORNL] for CORA-16,"[147] "Report of Foreign Travel of L. J. Ott" 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].[148]

It is significant that "In-Vessel Phenomena—CORA," states that for the CORA-16 experiment, "[c]ladding oxidation was not accurately predicted by available correlations."[149]

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…[150]

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[145] L. J. Ott, "Report of Foreign Travel of L. J. Ott," Cover Page.
[146] *Id*., p. 2.
[147] *Id*., p. 3.
[148] *Id*.
[149] L. J. Ott, W. I, van Rij, "In-Vessel Phenomena—CORA."
[150] *Id*.

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."[151]

(It is noteworthy that "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, published in 2001, states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[152])

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

It is significant that "[t]he first recorded and qualified rapid temperature rise [in the LOFT LP-FP-2 experiment] associated with the rapid reaction between Zircaloy and water occurred at about…1400 K (2060°F) on a guide tube at the 0.69-m (27-in.) elevation."[153]

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."[154] 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

[151] L. J. Ott, "Report of Foreign Travel of L. J. Ott," p. 3.
[152] Report by Nuclear Energy Agency Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.
[153] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 30.
[154] 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.

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.[155]

The LOFT LP-FP-2 experiment had an initial heatup rate of ~1 K/sec.[156] It is significant that "heatup rates [of 1 K/s or greater] are typical of severe accidents initiated from full power."[157] 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).[158]

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

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

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[155] *Id.*
[156] *Id.*
[157] 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.
[158] 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.

> 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.[159]

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).[160, 161]

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;[162] 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).[163]

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[159] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," p. 3.23.
[160] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," pp. 30, 33.
[161] 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.
[162] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 23.
[163] *Id.*, p. 33.

Therefore, after the onset of rapid oxidation—after a heating rate of ~1 K/sec.[164]—peak cladding temperatures increased from approximately 1400 K (2060°F) to 2100 K (3320°F) within a range of approximately 75 seconds; in other words, after the onset of rapid oxidation, cladding temperatures increased at an average rate of approximately 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)]."[165]

It is significant that "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods" states that "a rapid [cladding] temperature escalation, [greater than] 10 K/sec., signal[s] the onset of an autocatalytic oxidation reaction."[166] 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 rapid cladding temperature increase 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)]."[167] In a similar account, as already mentioned, "Review of Experimental Results on LWR Core Melt Progression" states that the initial

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[164] 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.
[165] 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.
[166] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[167] 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.

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).[168]

And offering yet another account of the rapid cladding temperature increase 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;[169] 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."[170]

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…"[171] 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.

(It is noteworthy that "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, published in 2001, states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[172])

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

[169] 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.

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

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

[172] Report by Nuclear Energy Agency Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

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."[173] 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. …
>
> 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.[174]

And regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities" states:

> From the analyses of core temperature measurements in [the LOFT] LP-FP-2 [experiment], the rapid increase in temperature shown in fig 14.[175] 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,"[176] 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

[173] 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.

[174] 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 (hereinafter: "Instrumentation Capabilities").

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

[176] 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.

> 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.[177, 178]

Additionally, regarding core temperature measurements in the LOFT-LP-FP-2 experiment, "Instrumentation Capabilities" 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,"[179] t]his event is believed to be the rupture of the control rod cladding.[180]

And regarding the durability of pressure sensors, thermocouples, and radiation monitors in the LOFT-LP-FP-2 experiment and TMI-2 accident, "Instrumentation Capabilities" 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.[181]

It is significant that "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment" and "Instrumentation Capabilities" state that the rapid temperature increase in the LOFT LP-FP-2 experiment, as a result of the autocatalytic oxidation reaction of Zircaloy cladding, commenced at approximately 1400 K (2060°F)—well below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

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[177] 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.

[178] A. B. Wahba, "Instrumentation Capabilities," p. 135.

[179] 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.

[180] A. B. Wahba, "Instrumentation Capabilities," p. 136.

[181] *Id.*, p. 147.

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

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

> Oxidation of Zircaloy cladding materials by steam becomes a significant heat source which increases with temperature; *if the heat removal capability is lost*, it determines a feedback between temperature increase and cladding oxidation [emphasis added].[182]

Furthermore, Figure 1[183] 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),[184] and Figure 13[185] of the same paper depicts that if the initial heat up rate is 1 K/sec. or greater, a rapid cladding temperature increase would commence at 1200°C (2192°F), in which the rate of increase would be 10 K/sec. or greater.[186]

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

In AEC's ECCS rulemaking hearing, conducted in the early '70s, Henry Kendall and Daniel Ford of Union of Concerned Scientists, on behalf of Consolidated National Intervenors ("CNI"),[187] dedicated the largest portion of their direct testimony to criticizing the BWR FLECHT Zr2K test,[188] conducted with a Zircaloy assembly. Among

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[182] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 195.

[183] See Appendix F Fig. 1. LWR Severe Accident-Relevant Melting and Chemical Interaction Temperatures which Result in the Formation of Liquid Phases.

[184] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 196.

[185] See Appendix F Fig. 13. Dependence of the Temperature Regimes on Liquid Phase Formation on the Initial Heat-Up Rate of the Core.

[186] Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 205.

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

[188] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-17 (hereinafter: "Assessment of Emergency Core Cooling System Effectiveness"); this paper cites Union of Concerned Scientists, "An Evaluation of

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."[189]

(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].[190]

And discussing the use of stainless steel heater-rod assemblies in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness" 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].[191]

Nuclear Reactor Safety," Direct Testimony Prepared on Behalf of Consolidated National Intervenors, USAEC Docket RM-50-1, March 23, 1972, as the source of this information.

[189] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness" p. A8-18.

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

[191] *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" 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…[192] 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.[193]

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

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[192] 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.

[193] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," pp. A8-18, A8-19.

rate of 15 K/sec."[194] 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.[195] 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%,[196] of the total energy input.[197]

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."[198])

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" 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.' "[199, 200] It is significant that "in the CORA test facility, [cladding] temperature escalation start[ed] between 1100 and 1200°C [(2012 to 2192°F)], giving rise to a

[194] 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.
[195] L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of BWR-Type Fuel Elements with B_4C/Steel Absorber," FZKA 7447, p. 5.
[196] 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.
[197] *Id.*, p. 5.
[198] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," p. A8-19.
[199] 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.
[200] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," p. A8-24.

maximum heating rate of 15 K/sec:"[201] "a rapid [cladding] temperature escalation, [greater than 10°C/sec. (18°F/sec.)], signal[s] the onset of an autocatalytic oxidation reaction."[202]

Furthermore, the graphs of "Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies"[203] and "Analysis of Zr2K Thermal Response"[204] depict thermocouple measurements taken during the Zr2K test that resemble thermocouple measurements taken during severe fuel damage experiments: the graphs depict rapid temperature increases 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 rapid temperature increases, 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 rapid cladding temperature increases taken during the Zr2K test were not valid. GE stated "that the 'erratic thermocouple outputs do not represent actual cladding

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[201] 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.
[202] 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.
[203] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," 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-13l97, June 1971, Figures A-11 and A-12, as the source of this information.
[204] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," 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.

temperatures, but are the result of equipment malfunctions'[205] associated with the Zr2K test."[206] 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 rapid cladding temperature increases taken during the Zr2K test resemble thermocouple measurements of rapid cladding temperature increases taken during severe fuel damage experiments.

In its analysis of the rapid cladding temperature increase that occurred during the Zr2K test, "Assessment of Emergency Core Cooling System Effectiveness" states:

> One of the more difficult aspects of evaluation of Zr2K test results is associated with the fundamental data for the tests, the recorded thermocouple…responses. *GE has been very liberal with their accreditation of observed [thermocouple] responses as erratic.* However, several proffered examples of erratic response seem to show well defined inter-rod correlations. Under such circumstances, "unexplained" might be a better description for the observed [thermocouple] behavior than "erratic" [emphasis added].[207]

Discussing the "well defined inter-rod correlations"[208] that occurred during "the extreme temperature excursion,"[209] "Assessment of Emergency Core Cooling System Effectiveness" 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,

[205] 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.
[206] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," pp. A8-24, A8-27.
[207] *Id.*, p. A8-19.
[208] *Id.*
[209] *Id.*, p. A8-21.

> thermocouples…on rod 16 indicate an apparent sharp temperature rise. Because of the anomalous electrical activity of rod 16 at this time, experimental analysts have been inclined to discount this [thermocouple] response as anomalous also. *However, it is interesting to note that the extreme temperature excursion… (adjacent to rod 16) occurred at the same time the rod 16 [thermocouple] excursion occurred and is matched by [the] nearly identical temperature excursion in rod 9, the other rod diametrically adjacent to rod 16. Moreover, it seems entirely too coincidental that temperature turnaround should be achieved in rod 24 at essentially the same time that the actual failure (rod current going to zero) for both rods 16 and 24 occurred.* Under those circumstances, it does not seem surprising that rod 17, still being driven by "normal" electric current and in direct view of the three hottest rods in the test (rods 16, 23. and 24) should then become the highest temperature rod for most of [the] remaining significant portion of the temperature transient. During this period, rods 17 and 23 both underwent electrical anomalies in which excessive currents were delivered to them. It was not until the current to both of these rods actually went to zero, approximately 12 minutes after the thermal transient began, that rod 17 relinquished its role as the highest temperature rod for the test.
>
> The relationships described above seem to indicate a systematic correlation between the electrical anomalies of the "failed" rods and temperature extremes for the bundle [emphasis added].[210]

So, as "Assessment of Emergency Core Cooling System Effectiveness" states, the observed thermocouple measurements were not erratic. And, as stated above, the thermocouple measurements of the rapid cladding temperature increases taken during the Zr2K test resemble thermocouple measurements of rapid cladding temperature increases taken during severe fuel damage experiments.

In the conclusion of its analysis of the rapid cladding temperature increase that occurred during the Zr2K test "Assessment of Emergency Core Cooling System Effectiveness" 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-

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

> 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].[211]

Indeed, "a more thorough analysis and interpretation of the Zr2K-[thermocouple] data would have been desirable."[212] 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" 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.' "[213, 214]

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.

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[211] *Id.*, p. A8-27.
[212] *Id.*
[213] 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.
[214] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," p. A8-24.

Furthermore, it is significant that in 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].[215]

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

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

> CNI's direct testimony concluded that a near thermal runaway condition existed in Test ZR-2.[216] 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].[217, 218]

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

[215] 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.

[216] 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.

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

[218] 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.

> 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]."[219]

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*

[219] *Id*.

or the issue of the existence of a near thermal runaway condition [emphasis added].[220]

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]."[221]

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.[222]

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[223]—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-

[220] *Id.*, pp. 5.12, 5.14.
[221] *Id.*
[222] *Id.*, p. 5.41.
[223] 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, p. 3-3.

of-coolant accident."[224] 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.[225] 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.).[226]

The FLHT-1 test is reported on in "Full-Length High-Temperature Severe Fuel Damage Test 1." 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.[227]

"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 increase; 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 rapid cladding temperature increase 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;"[228] *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.[229] The

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[224] 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 (hereinafter: "FLHT-1 Test Report").
[225] *Id.*, p. 3.1.
[226] *Id.*, pp. 4.1-4.2.
[227] *Id.*, p. v.
[228] *Id.*, p. 4.6.
[229] *Id.*, p. 4.11.

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" states that during the FLHT-1, -2, -4, and -5 tests that "[t]he heatup phase of the tests culminated near 1700 K in a rapid [cladding] temperature escalation, [greater than] 10 K/sec., signaling the onset of an autocatalytic oxidation reaction."[230] 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)."[231] According to a different account, in the LOFT LP-FP-2 experiment, the onset of rapid oxidation occurred at approximately 1500 K (2240°F).[232] 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.[233]

Furthermore, although the graphs of "Typical Cladding Temperature Behavior"[234] and "Pseudo Sensor Readings for Fuel Peak Temperature Region"[235, 236] 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

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[230] F. E. Panisko, N. J. Lombardo, "Results from In-Reactor Severe Fuel Damage Tests that used Full-Length Fuel Rods," in "Proceedings of the U.S. Nuclear Regulatory Commission: Twentieth Water Reactor Safety Information Meeting," p. 282.
[231] J. J. Pena, S. Enciso, F. Reventos, "Thermal-Hydraulic Post-Test Analysis of OECD LOFT LP-FP-2 Experiment," p. 33.
[232] 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.
[233] 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.
[234] W. N. Rausch, *et al.*, "FLHT-1 Test Report," p. 4.7.
[235] *Id.*, p. 5.3.
[236] Pseudo sensor readings are the averages of the readings of two or more thermocouples.

that the rapid temperature increase began at a temperature far lower than 1700 K, at a temperature closer to 1520 K (see Appendix H Figure 4.1. Typical Cladding Temperature Behavior and Figure 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region). The slopes of the lines of the cladding-temperature value plots in the graphs become nearly vertical, when the cladding-temperature values reach approximately 1520 K, indicating the onset of the rapid temperature increase, 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 rapid temperature increase 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.[237]

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…"[238]

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

[237] W. N. Rausch, *et al.*, "FLHT-1 Test Report," p. 4.6.
[238] *Id.*

oxidation reaction would have begun at 1700 K (2600°F). How can it be explained that after the coolant flow rate was increased—when cladding temperatures reached approximately 1475 K (2200°F)—that the cladding temperatures were able to increase by 225 K (400°F)? Why would the test conductors have not been able to terminate the cladding-temperature rise, as they did earlier in the test when cladding temperatures reached 1450 K (2150°F)? And how can it be explained that the test conductors did not have enough time to increase the coolant flow rate back up to 18 kg/hr. (40 lb/hr.), as they did when cladding temperatures reached 1450 K (2150°F), earlier in the test?

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

In "Compendium of ECCS Research for Realistic LOCA Analysis," discussing an earlier NRU reactor test, 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."[239] 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)."[240]

(It is noteworthy that other reports state that the MT-6B test had a PCT of 1400 K (2060°F)[241] and 1280°C (2336°F) (1553 K).[242] 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

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

> consist of a series of preplanned, discrete flow-reduction steps. The size and duration of each reduction is selected to *control the steam-Zircaloy reaction*—and hence the temperature ramps and hydrogen generation rate. …
>
> The bundle [coolant] flow rate will then be decreased in a series of precalculated flow steps… The duration of the time between steps is dictated by the time needed to reach near steady state and also by *the requirement that the Zircaloy-steam reaction be limited*. About 14 steps, each of about 1/2 hr. duration, are expected. *The last flow reduction step will be calculated to give a peak cladding temperature of about 2150 K (3400°F).* …
>
> The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150 K (3400°F) [emphasis added].[243]

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."[244]

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"[245] and "Pseudo Sensor Readings for Fuel Peak Temperature Region,"[246] 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).

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[243] W. N. Rausch, *et al.*, "FLHT-1 Test Report," pp. 4.3-4.5.
[244] *Id.*, p. 4.6.
[245] *Id.*, p. 4.7.
[246] *Id.*, p. 5.3.

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

> The reactor power was decreased at approximately 17:11:07, 85 sec. after the start of the excursion (approximately 131 minutes in Figure 4.1). The reactor reached 10% of the initial power approximately 35 sec. later and reached low neutron level in another 30 sec.
>
> There were two Indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.[247]

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.[248]

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."[249]

The FLHT-1 test is highly significant precisely because, once cladding temperatures reached as high as approximately 1475 K (2200°F), the test conductors

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[247] *Id.*, pp. 4.6-4.7.
[248] *Id.*, p. 4.7.
[249] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 8-2.

could not prevent the cladding-temperature rise by increasing the coolant flow rate. Increasing the coolant flow rate did not prevent the onset of an autocatalytic oxidation reaction—which occurred at cladding temperatures of approximately 1520 K (~2275°F) or lower.

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

The PHEBUS B9R test was conducted in a light water reactor—as part of the PHEBUS severe fuel damage program—with an assembly of 21 UO_2 fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[250]

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].[251]

And offering a different account of the elevation at which the rapid temperature increase 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.)"[252] "Degraded Core Quench: A Status Report" states that the rapid temperature increase 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.[253]

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[250] 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.
[251] *Id*.
[252] 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.
[253] 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

Neither paper states what peak cladding temperatures were at the outset of the autocatalytic oxidation reaction; however, a graph of the cladding-temperature values at the 0.6 meter "hot-level" indicates that the autocatalytic oxidation reaction began when cladding temperatures were below 1477 K (2200°F)[254] (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)).

E. The Damage PWR Fuel Assembly Components Would Incur and 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 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…[255]—IP-2 licensing hearing, November 1971

In this section, Petitioner will discuss papers that report that, in the event of a LOCA, PWR core component damage could commence at relatively low temperatures.

It is significant that the alleged conservatism of IP-2 and -3 LBPCTs of 1937°F and 1961°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.

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and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," p. 311.

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

[255] 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.

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.

If the fuel cladding were to incur runaway oxidation between 1832°F and 2192°F, peak cladding temperatures would begin rapidly increasing at tens of degrees Fahrenheit per second. And within tens of seconds peak cladding temperatures would increase to over approximately 2600°F and temperatures of the (Ag,In,Cd) control rods would also increase to temperatures over approximately 2192°F (1200°C): temperatures at which the control rods would incur significant damage.

In Dr. Robert E. Henry's presentation slides from "TMI-2," one of the presentation slides states that "the core damage was generally caused by the cladding oxidation."[256] And another one of Dr. Henry's presentation slides 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."[257]

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

"Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents" states that "[e]xperiments were performed at several laboratories to investigate the behavior of (Ag,In,Cd) control rods during severe reactor accidents"[258, 259] and that the

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[256] Robert E. Henry, presentation slides from "TMI-2," 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.

[257] *Id*.

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

[259] P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," Kernforschungszentrum Karlsruhe, KfK 4670, 1989, p. 1.

essential results of the experiments are summarized in a paper titled "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents."[260]

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

> The (Ag,In,Cd) alloy melts at about 800°C, but will not affect core degradation as long as the molten material is contained within the stainless steel cladding. As the temperature increases, some of the control rod constituents will vaporize within the cladding until failure occurs, either from internal pressurization or from melting of the cladding. At low pressures of the primary system and when no Zircaloy is present, the control rod fails between 1350 and 1450°C. Failure of the control rods with the Zircaloy guide tubes occurs at about 1200°C as a result of thermal expansion, physical contact, and eutectic chemical interactions between the stainless steel cladding and the Zircaloy guide tube. The high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the cladding fails. The ejected material results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy. Due to the high vapor pressure of Cd it vaporizes. Liquid control rod material continues to vaporize if it remains at high temperatures. The control rod material which will flow out of the hot regions of the core freezes and may inhibit steam and/or water flow. At high system pressure, over-pressurization of the rod does not occur. Instead, upon failure, the alloy flows to cooler regions of the reactor core. In all cases the resulting reaction products melt at low temperatures and enhance by this the degradation of the reactor core [emphasis added].[261, 262]

So "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," states that "[f]ailure of the control rods with the Zircaloy guide tubes occurs at about 1200°C"[263] and that "[t]he high internal pressure in the control rod will result in a violent ejection of vapor, aerosol and molten material when the cladding fails"[264] that "results in the formation of low-temperature melting alloys consisting of the (Ag,In) constituents and the surrounding Zircaloy."[265]

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260 David A. Petti, "Silver-Indium-Cadmium Control Rod Behavior and Aerosol Formation in Severe Reactor Accidents," NUREG/CR-4876, EG + E-2501, 1987.
261 *Id*.
262 P. Hofmann, M. Markiewicz, "Chemical Behavior of (Ag,In,Cd) Absorber Rods in Severe LWR Accidents," KfK 4670, pp. 1-2.
263 *Id*., p. 1.
264 *Id*.
265 *Id*., pp. 1-2.

And regarding eutectic interactions of the absorber rod's steel cladding tube and the Zircaloy guide tube that can cause liquefaction to occur locally at approximately 1200°C, "Behavior of AgInCd Absorber Material" states:

> The absorber rod should fail; *i.e.*, melt down, upon attainment of the melting point of the steel cladding tube (~1400°C) at the latest. *On account of eutectic interactions of the steel cladding tube and the zircaloy guide tube, liquefaction can take place locally as early as from 1200°C on.* The (Ag,In,Cd) absorber melt contributes essentially to the propagation of damage in the bundle which is an unambiguous finding of chemical-analytical studies of the reaction products by means of the scanning electron microscope [emphasis added].[266]

It is significant that "when no Zircaloy is present, the control rod fails between 1350°C and 1450°C"[267] or that the control rod fails at ~1400°C, at the latest.[268] So when Zircaloy is present, the control rod fails at a temperature—approximately 1200°C—that is between 150°C and 250°C lower—a substantial temperature difference.

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

> - *The (Ag,In,Cd) absorber alloy starts to melt at about 800°C, but this will not affect core degradation as long as the molten material is contained within the stainless steel (AISI 316) cladding.* The chemical interaction between the absorber alloy and stainless steel is negligible.
>
> - Failure of the stainless steel absorber rod cladding takes place as a result of either internal pressurization (high Cd vapor pressure) or eutectic interactions with the Zircaloy guide tube (bowing of the rods at high temperatures). The released (Ag,In,Cd) melt can then interact with the Zircaloy guide tube.
>
> - The Zircaloy will be chemically dissolved by the absorber alloy. The dissolution of the Zircaloy can be described by a parabolic rate law. *The dissolution rate is very fast; at 1200°C, it takes only about 50 [seconds] to dissolve 1 mm Zircaloy and about 4 minutes to destroy the entire 2.25 mm thick Zircaloy crucible wall.*

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266 L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material," FZKA 7448, 2008, p. 14.

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

268 L. Sepold, S. Hagen, P. Hofmann, G. Schanz, "Behavior of AgInCd Absorber Material," FZKA 7448, p. 14.

> - As soon as solid state contact occurs between the stainless steel cladding and the Zircaloy guide tube, eutectic interactions take place which can be described by parabolic rate laws. *Liquid phases form at around 1000°C, and a fast and complete liquefaction of both components takes place above 1250°C. Only small amounts of stainless steel are necessary to dissolve great amounts of Zircaloy, and it takes only a little more than 2 minutes to destroy the 2.25 mm thick Zircaloy crucible wall at 1200°C.*
>
> - Thin ZrO_2 layers (~10 µm) on the Zircaloy surface delay the chemical interactions of Zircaloy with the (Ag,In,Cd) alloy or the stainless steel, but cannot prevent them. The ZrO_2 layer must be dissolved by the Zry before chemical interactions can take place. The required incubation period depends on temperature and time. Dissolved oxygen in the Zircaloy, forming oxygen-stabilized α-Zr(O), reduces the reaction rates and shifts the liquefaction temperature to slightly higher levels.
>
> - With respect to the chemical behavior of (Ag,In,Cd) absorber rods during severe reactor accidents, meltdown and relocation must be assumed to occur at temperatures around 1250°C. The resulting melt destroys the Zircaloy cladding of the fuel rods and dissolves a part of the UO_2, contributing substantially to fuel element degradation. *Since UO_2 fuel can be liquefied at temperatures as low as 1250°C, this process has a strong impact on the release of volatile fission products*.
>
> - The premature low-temperature failure of the PWR absorber rods and the localized relocation of (Ag,In,Cd) alloy within the reactor core may cause criticality problems during flooding of the destroyed core [emphasis added].[269]

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

> The absorber rod alloy (80 wt% silver, 15% indium, 5% cadmium) is thermodynamically stable with its stainless steel cladding, even in the liquid state (>800°C). *However, the absorber rod guide tube is made from Zircaloy, which will chemically interact with the stainless steel cladding of the absorber rod. During a severe reactor accident, localized contact between stainless steel and Zircaloy exists at many places. This solid-state contact results in chemical interactions with the formation of liquid phases around 1150°C.* After failure of the absorber rod cladding, the molten Ag-In-Cd alloy (melting point ~800°C) comes into contact with the Zircaloy guide tube and chemically destroys it. Then, the molten Ag-

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

> In-Cd can even attack and chemically dissolve the Zircaloy cladding of the fuel rods well below the melting point of Zircaloy (~1760°C). The relocating Ag-In-Cd alloy is therefore able to propagate and accelerate the core-melt progression at rather low temperatures.
>
> The chemical interactions between Ag-In-Cd and Zircaloy were studied in separate-effects tests which are described in [reference 19].[270] The reaction zone growth rate (decrease in Zircaloy wall thickness) is plotted in an Arrhenius diagram against the reciprocal temperature in Fig. 10. *At temperatures >1200°C, the chemical interactions result in a sudden and complete liquefaction of the compatibility specimens.* As a consequence, the Zircaloy cladding can be chemically dissolved ~600 K below its melting point and may even result in a low-temperature UO_2 fuel dissolution. For phase considerations of melting reactions, the quaternary U-Zr-Fe-O system may be regarded as a model system for the complicated multi-component system of a beginning core melt; iron represents the stainless steel. A detailed description of the phase relations is given in [reference 4].[271]
>
> The chemical interaction between the Ag-In-Cd alloy and Zircaloy is theoretically described by a model under conditions of convective mixing in the (Zr, Ag, In) liquid phase in [reference 20].[272] Homogeneous bulk saturation of the liquid phase with Zr takes place in the course of the Zircaloy dissolution by the absorber melt resulting in a gradual decrease of the interaction process. Two main parameters of the model are calculated: Zr concentration in the saturated melt and convective mass transfer coefficient in the liquid phase[273] [emphasis added].[274]

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

> The CORA-7 test[275] indicated that the reaction between [Inconel] grid spacer and [Zircaloy] cladding was not symmetrical and that control rod material (Ag-In-Cd) may influence the interaction between grid spacer and cladding.[276]

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[270] P. Hofmann, M. Markiewicz, Journal of Nuclear Materials, 209, 1994, p. 92.
[271] P. Hofmann, *et al*., Nuclear Technology 87, 1989, p. 146.
[272] M.S. Veshchunov, P. Hofmann, Journal of Nuclear Materials, 228, 1996, p. 318.
[273] *Id*.
[274] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, pp. 201-202.
[275] P. Hofmann, *et al*., "Material Behavior in the Large PWR Bundle Experiment CORA-7," International CORA Workshop 1991, September 23-26, 1991, Karlsruhe, Germany.
[276] L.J. Siefken, M.V. Olsen, "A Model for the Effect of Inconel Grid Spacers on Progression of Damage in Reactor Core," Nuclear Engineering and Design 146, 1994, p. 436.

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

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

It is significant that "[t]he chemical reaction between Inconel and Zircaloy influences the meltdown of the reactor core in the vicinity of Inconel grid spacers."[277]

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[278] and Japan,[279] in-pile experiments at the PBF facility in Idaho,[280] and by examinations of the damaged Three Mile Island (TMI-2) core.[281] *The experiments in Germany and Japan have revealed the existence of chemical interactions between Inconel and Zircaloy that take place at temperatures as low as 1273 K [(1832°F)]*, more than 200 K lower than the melting temperature of Inconel. *Thus in a reactor core with Inconel grid spacers the meltdown of the core may begin at the location of the grid spacers* [emphasis added].[282]

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[277] 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.

[278] 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.

[279] 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.

[280] 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.

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

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

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;[283] and significant that experiments have revealed chemical interactions between Inconel and Zircaloy occur at temperatures as low as 1832°F.

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.*[284]
>
> 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).*[285]
>
> In a first approach, the reaction behavior of Zircaloy with Inconel 718 is comparable to that with Type 316 stainless steel.[286] *At temperatures <1100°C, Inconel attacks the Zircaloy faster than stainless steel; above*

[283] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[284] P. Hofmann, *et al.*, Nuclear Technology 118, 1997, p. 200.
[285] P. Hofmann, M. Markiewicz,, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Stainless Steel at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 5106, 1994.
[286] P. Hofmann, M. Markiewicz, "Chemical Interactions between As-Received and Pre-Oxidized Zircaloy and Inconel 718 at High Temperatures," Kernforschungszentrum Karlsruhe, KfK 4729, 1994.

> *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* [287] [emphasis added].[288]

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"[289] and that "[l]ocalized liquefaction of these components started around 1200°C [(2192°F)]."[290] 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)]."[291]

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

F. The Heat Transfer Coefficients Used in ECCS Evaluation Calculations for PWR Cores are Based on Data from Thermal Hydraulic Experiments Conducted with Stainless Steel and Inconel 600 Heater-Rod Bundles

The heat transfer coefficients used in ECCS evaluation calculations for PWR cores—which have fuel assemblies with zirconium alloy cladding—are based on data from thermal hydraulic experiments conducted with stainless steel and Inconel 600 heater-rod bundles. Heat transfer coefficients based on data from thermal hydraulic experiments conducted with stainless steel and Inconel 600 bundles are not adequate for ECCS evaluation calculations, modeling heat transfer for fuel assemblies with zirconium alloy cladding in a LOCA environment.

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[287] P. Hofmann, *et al*., Nuclear Technology 118, 1997, p. 200.
[288] P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," p. 202.
[289] *Id*.
[290] *Id*.
[291] *Id*.

Interpretations of the results of experiments conducted with stainless steel and Inconel 600 multi-rod bundles would most likely lead the interpreters to false conclusions. For example, a stainless steel or Inconel 600 multi-rod bundle heated up to peak cladding temperatures between 1832°F and 2200°F would not incur runaway oxidation; however, a zirconium alloy multi-rod bundle heated up to peak cladding temperatures between 1832°F and 2200°F would (with high probability) incur runaway oxidation.

It is significant that NRC states that "[h]eat transfer coefficients are not directly measurable quantities. They must be calculated from *measured temperatures,* known heat sources, and known thermal properties" [emphasis added].[292] Petitioner would add that heat transfer coefficients used for LOCA analyses of real reactor cores with zirconium alloy fuel assemblies must *also* be calculated from thermal hydraulic experiments conducted with zirconium alloy multi-rod bundles.

Unfortunately, most thermal hydraulic experiments have been conducted with stainless steel and Inconel 600 multi-rod bundles. And it is significant that some of the thermal hydraulic experiments that have been conducted with zirconium alloy multi-rod bundles have had results that do not conclusively demonstrate the effectiveness of ECCS in cooling the fuel cladding; *e.g.*, in cases in which there would be reflood rates of one inch or less per second.

In the rulemaking hearings, conducted in the early 1970s, the AEC Commissioners concluded that "the heat transfer mechanism is [not] different for zircaloy and stainless steel, and…that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods."[293]

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[292] NRC, "Technical Safety Analysis of PRM-50-76, A Petition for Rulemaking to Amend Appendix K to 10 C.F.R. Part 50 and Regulatory Guide 1.157," April 29, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041210109, p. 7 (hereinafter "Technical Safety Analysis of PRM-50-76").

[293] 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 (hereinafter "Commission Decision"). 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.

The AEC Commissioners also concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients; they maintained that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside the rod.[294]

However, it would be reasonable to claim that the AEC Commissioners—like others in the early 1970s—did not realize that the heat generated by the exothermic Zircaloy-steam reaction is very substantial. It has been pointed out in recent years that investigators (David O. Hobson and Philip L. Rittenhouse) conducting furnace experiments with Zircaloy tubes—in the same time period—did not fully recognize "the effect of the large exothermic heat of oxidation of [Zircaloy]."[295]

Furthermore, in 1971, AEC claimed that the Zircaloy-steam reaction was "negligible below 1900°F."[296]

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

> The basic model used for [the] metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[297]

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

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[294] *Id.*, pp. 1123-1124.

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

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

[297] *Id.*

G. The Reflood Heat Transfer Coefficients Required by Appendix K to Part 50 are Based on Data from Thermal Hydraulic Experiments Conducted with Stainless Steel Heater-Rod Bundles

> [D]uring the questioning of the Hanauer task force,[298] one witness, G. Norman Lauben, an [Atomic Energy Commission] regulatory staff engineer, had dissented from one item in the task force's prepared testimony. He had raised his hand in response to a general question put to the task force: did any of them think that there were parts of the policy statement [on ECCS] that were not conservative enough? Practically in tears, he acknowledged in a quaking voice that he did not think certain heat-transfer coefficients used in the industry computer methods were conservative enough.[299] He did not make any broad challenge to the overall adequacy of the computer methods, merely this single objection. Still, it was obvious that it required considerable courage to take even this one minor[300] exception to the official policy.[301]—Daniel Ford

The reflood heat transfer coefficients required by Appendix K to Part 50 are based on data from thermal hydraulic experiments conducted with stainless steel heater-rod bundles. (IP-2 and -3 use NRC-approved best-estimate ECCS evaluation models in lieu of Appendix K to Part 50 calculations.[302] However, Petitioner is discussing Appendix K reflood heat transfer coefficients to provide more background information on thermal hydraulic experiments conducted with stainless steel heater-rod bundles.)

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

[299] "Norman Lauben of the regulatory staff felt that the FLECHT heat transfer results were not demonstrably conservative;" 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. 5.33.

[300] In Petitioner's opinion, the quoted passage would have been more accurate if the word "minor" had not been included: non-conservative heat transfer coefficients are not a minor problem.

[301] Daniel F. Ford, *Meltdown: The Secret Papers of the Atomic Energy Commission*, Simon & Schuster, New York, 1986, pp. 122-123.

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

Appendix K to Part 50—ECCS Evaluation Models I(D)(5), *Required and Acceptable Features of the Evaluation Models*, *Post-Blowdown Phenomena*, *Refill and Reflood Heat Transfer for Pressurized Water Reactors*, states:

> For reflood rates of one inch per second or higher, *reflood heat transfer coefficients shall be based on applicable experimental data* for unblocked cores, *including [the] FLECHT results [reported in] "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,"* Westinghouse Report WCAP-7665, April 1971. … Westinghouse Report WCAP-7665 has been approved for incorporation by reference by the Director of the Federal Register. … New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.
>
> b. During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer [emphasis added].

It is significant that the "applicable experimental data" that Appendix K reflood heat transfer coefficients are based on, is data from experiments conducted with stainless steel heater-rod bundles, not Zircaloy bundles.

"PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" states:

> Properly used, PWR FLECHT test results can improve the accuracy of reactor LOCA analysis. The heat transfer correlations which were developed are *conservative* in that they do not take any credit for the effects of "fallback" or borated coolant and are *based on stainless steel clad data* [emphasis added].[303]

So Appendix K to Part 50—ECCS Evaluation Models I(D)(5) is based on the premise that stainless steel cladding heat transfer coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

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[303] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 5-4 (hereinafter: "PWR FLECHT Final Report").

H. The Fallacy of the AEC Commissioners' Conclusion that "the Heat Transfer Mechanism is [Not] Different for Zircaloy and Stainless Steel": "that the Heat Transfer Correlations Derived from Stainless Steel Clad Heater Rods are Suitable for Use with Zircaloy Clad Fuel Rods"

The practice of using heat transfer correlations derived from stainless steel clad heater rods for ECCS evaluation calculations dates back to the AEC rulemaking hearings, conducted in the early 1970s, when the requirements of the ECCS evaluation models of Appendix K to Part 50 were written.

In the rulemaking hearings, the AEC Commissioners concluded that "the heat transfer mechanism is [not] different for zircaloy and stainless steel, and…that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods."[304]

To discuss the fallacy of the AEC Commissioners' conclusion regarding heat transfer correlations derived from stainless steel clad heater 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 bundle used in run 9573 incurred runaway oxidation.[305]

(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. A Zircaloy bundle used in the BWR FLECHT program—the Zr2K test bundle—also incurred runaway oxidation.)

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,

[304] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," p. 1124.
[305] See Appendix J for photographs of the bundle from FLECHT Run 9573.

> that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[306]

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT Final Report"] when compared to the equivalent stainless steel tests."[307] 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."[308]

Opining that the zirconium-water reaction would not affect heat transfer coefficients, "Commission Decision" 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*.[309]

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

(It is noteworthy that in AEC responses to questions submitted by Anthony Z. Roisman, AEC stated:

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

[306] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," p. 1124.
[307] 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.
[308] *Id.*, p. 17.
[309] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," pp. 1123-1124.

1800°F in LOCTA [a computer code], but the calculated reaction is negligible below 1900°F.[310]

Indeed, computer codes using the Baker-Just correlation may calculate that the Zircaloy-steam reaction is negligible below 1900°F; however, experimental data from multi-rod experiments demonstrates that the Zircaloy-steam reaction is very substantial below 1900°F. For example, in the CORA-2 and CORA-3 experiments, runaway oxidation commenced at 1832°F: peak cladding temperatures started increasing at several tens of degrees Fahrenheit per second.[311])

It is significant that within the first 18.2 seconds of FLECHT run 9573,[312] "negative heat transfer coefficients were observed at the bundle midplane for 5…thermocouples;"[313] *i.e.*, more heat was transferred into the bundle midplane than was removed from that location. In 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."[314]

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

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310 AEC, AEC responses to questions submitted by Anthony Z. Roisman, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, October 29, 1971, Question: Page 12.

311 S. Hagen, P. Hofmann, G. Schanz, L. Sepold, "Posttest Results of Severe Fuel Damage Experiments CORA-2 and CORA-3," KfK 4378, p. 41.

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

313 *Id.*, p. 3-98.

314 Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.

inward. Indeed, the Zircaloy-cladding heater rods were very hot internally, where the thermocouples were located; yet, nonetheless, the heater rods became a heat sink.[315]

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."[316]

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).[317]

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"[318]—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.

Clearly, the exothermic zirconium-water reaction affects the coolant outside the cladding by heating a mixture of steam and hydrogen, and entrained water droplets;

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[315] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.
[316] 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.
[317] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[318] *Id*., p. 3-98.

therefore, the zirconium-water reaction cannot legitimately be separated from cladding heat transfer mechanisms.

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,"[319] 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"[320] is erroneous.

I. Stainless Steel Cladding Heat Transfer Coefficients are Not Always a Conservative Representation of Zircaloy Cladding Behavior, for Equivalent LOCA Conditions

It is significant that for PWR FLECHT run 9573 the "[a]nalysis of the test results showed that heat transfer coefficients for the first eighteen seconds were generally lower than for a comparable stainless steel test"[321] and that "negative heat transfer coefficients were observed at the bundle midplane for 5…thermocouples."[322] Yet the data from run 9573 is *not* considered valid. And "PWR FLECHT Final Report" states:

> Properly used, PWR FLECHT test results can improve the accuracy of reactor LOCA analysis. The heat transfer correlations which were developed are *conservative* in that they do not take any credit for the effects of "fallback" or borated coolant and are *based on stainless steel clad data* [emphasis added].[323]

So Appendix K to Part 50—ECCS Evaluation Models I(D)(5)—which states that "reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores, including [the] FLECHT results [reported in "PWR FLECHT Final Report"]"—is erroneously based on the premise that stainless steel cladding heat transfer

[319] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.
[320] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," p. 1124.
[321] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.
[322] *Id.*, p. 3-98.
[323] *Id.*, p. 5-4.

coefficients are *always* a conservative representation of Zircaloy cladding behavior, for equivalent LOCA conditions.

Indeed, stainless steel cladding heat transfer coefficients are not a conservative representation of representation of Zircaloy cladding behavior, for some of the conditions that would occur in the event of a LOCA.

1. The Rate of Stainless Steel Oxidation is Small Relative to the Oxidation of Zircaloy at Temperatures Below 1400 K but the Rate of Reaction for Stainless Steel Exceeds that of Zircaloy above 1425 K; However, the Heat of Reaction is about One-Tenth that of Zircaloy, for a Given Mass Gain

It is significant that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400 K. At higher temperatures and near the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[324] 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].[325]

2. Criticisms of Thermal Hydraulic Experiments Conducted with Stainless Steel Bundles, Dating Back to the Early 1970s

Discussing one of Henry Kendall and Daniel Ford's, of Consolidated National Intervenors ("CNI"),[326] criticisms of the BWR FLECHT tests (which would also apply to other thermal hydraulic experiments conducted with stainless steel bundles, like the PWR FLECHT tests), "Assessment of Emergency Core Cooling System Effectiveness" 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 [BWR] FLECHT tests utilized [Zircaloy] rods. The

[324] 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.
[325] *Id.*, p. 4.4.
[326] 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.

> 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].[327]

And discussing the use of stainless steel heater-rod bundles in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness" 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].[328]

3. After the PWR-FLECHT Test Program (Cited in Appendix K to Part 50) was Concluded, the Subsequent PWR FLECHT and FLECHT-SEASET Programs were Conducted with Stainless Steel Bundles

The subsequent PWR FLECHT programs—like the FLECHT Low Flooding Rate Test Series—were conducted with stainless steel bundles. And the FLECHT-SEASET program was conducted with stainless steel bundles.

(It is noteworthy that the rig of safety assessment IV ("ROSA-IV") facility, which conducted PWR thermal hydraulic experiments, used Inconel 600 bundles.[329] And the Rod Bundle Heat Transfer ("RBHT") facility at Penn State University—currently investigating PWR-related problems—conducts tests with Inconel 600 bundles.[330])

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[327] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness," EQL Report No. 9, pp. A8-2, A8-6.
[328] *Id.*, p. A8-6.
[329] Yasuo Koizumi, Yoshinari Anoda, Hiroshige Kumamaru, Taisuke Yonomoto, Kanji Tasaka**,** "High-Pressure Reflooding Experiments of Multi-Rod Bundle at ROSA-IV TPTF," Nuclear Engineering and Design**,** Volume 120, Issues 2-3, June 1990, pp. 301-310.
[330] 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.

J. A Thermal Hydraulic Experiment Conducted with a Stainless Steel Bundle that is Commonly Included as a Benchmark Test in the Validation Matrix of Several Computer Codes

Regarding a FLECHT-SEASET test conducted with a stainless steel multi-rod bundle, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer" states:

> The FLECHT-SEASET test 31504 is commonly included as a benchmark test in the validation matrix of several computer codes. Run 31504 is a forced reflood test with 2.5 cm./sec. [(~1.0 in./sec.)] flooding rate. … In the experiment the reflood is initiated when the PCT [peak cladding temperature] reaches 1144 K (1600°F). Subcooled liquid at 323 K is injected at the bottom of the test section at 2.5 cm./sec. The pressure (272 kPa) is set at the outlet of the bundle.[331]

The report, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," Volume 2, states that in the FLECHT-SEASET test 31504, the peak cladding temperature ("PCT") at the onset of reflood was 1585°F, that the rod peak power was 0.70 kW/ft, and that the PCT during reflood, remained under the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F; it states that the PCT was approximately 2100°F.[332]

(It is noteworthy that according to Appendix A of "Rod Bundle Heat Transfer Test Facility Test Plan and Design," the FLECHT-SEASET test 31504 had a reflood rate of 0.97 in./sec., a PCT at the onset of reflood of 1585°F, and an overall PCT of 2101°F (an increase of 516°F).[333])

(It is also noteworthy that "A Moving Subgrid Model for Simulation of Reflood Heat Transfer" states that the original COBRA-TF and the new COBRA-TF/FHMG

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[331] Cesare Frepoli, John H. Mahaffy, and Lawrence E. Hochreiter, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer," Nuclear Engineering and Design, 224, 2003, pp. 139, 140.
[332] M. J. Loftus, *et al.*, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," Volume 2, NUREG/CR-1532, June 1980, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070740185, pp. 31504-1, 31504-2.
[333] NRC, L. E. Hochreiter *et al.*, Appendix A, Literature Review, of "Rod Bundle Heat Transfer Test Facility Test Plan and Design," NUREG/CR-6975, July 2010, p. A-29 (hereinafter "RBHT Test Facility Test Plan").

codes are used to simulate the FLECHT-SEASET test 31504 and that code predictions are compared with test data.[334])

Regarding FLECHT-SEASET tests 31504 and 32753, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data" states:

> This report presents the results of an assessment of the capabilities of the TRAC-M(F90), Version 3.580, and TRAC-M(F77), Version 5.5.2A, codes to calculate reflood and steam cooling phenomena for pressurized-water reactors (PWRs). The reflood assessment was performed using test data from FLECHT-SEASET Run 31504, while the steam cooling assessment was performed using test data from FLECHT-SEASET Run 32753. These tests simulate unblocked bundle forced reflood and steam cooling conditions in PWRs.[335]

And, regarding the assessment of the capabilities of the TRAC-M(F90), Version 3.580, and TRAC-M(F77), Version 5.5.2A, codes to calculate reflood and steam cooling phenomena, "Assessment of the TRAC-M Codes" states:

> The assessment shows that predictions of the reflood phenomena derived using both codes are inaccurate; however, it is judged that *they can conservatively predict peak clad temperatures in heated rods* since the code model expels more water from the test section than measured. The predictions of steam cooling in single-phase flow conditions are acceptable [emphasis added].[336]

It is significant that the FLECHT-SEASET test 31504 was conducted with a stainless steel bundle, not with a zirconium alloy bundle. Therefore, it would seem that the TRAC-M codes conservatively predict PCTs for heated stainless steel rods; however, the TRAC-M codes certainly do not conservatively predict PCTs for the zirconium alloy fuel rods used in PWRs.

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

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[334] Cesare Frepoli, John H. Mahaffy, and Lawrence E. Hochreiter, "A Moving Subgrid Model for Simulation of Reflood Heat Transfer," Nuclear Engineering and Design, 224, 2003, p. 139.

[335] NRC, "Assessment of the TRAC-M Codes Using FLECHT-SEASET Reflood and Steam Cooling Data," NUREG-1744, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011520327, p. 1 (hereinafter "Assessment of the TRAC-M Codes").

[336] *Id*., p. iii.

reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.74 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F).[337]

(It is noteworthy that, in 2005, the NRC stated that it was "reviewing…data from [the early '80s, from the program the TH-1 tests were part of,] to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[338])

If the FLECHT-SEASET test 31504 had been conducted with a zirconium alloy bundle instead of a stainless steel bundle, the test results would have been different: with high probability, the zirconium alloy bundle would have had a PCT that exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F and it would have incurred runaway oxidation, like FLECHT run 9573 (conducted with a Zircaloy bundle).

It is unfortunate that NRC has never conducted a thermal hydraulic experiment (with a Zircaloy multi-rod bundle) in which the PCT at the onset of reflood was 1600°F and there was a reflood rate of 1.0 in./sec. or lower.

(It is also noteworthy that NRC's document, "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction," from 2002, states that "good core quenching rates are achieved even for flooding rates of one inch per second;"[339] however, it is important to recognize that NRC's claim is based on the results of tests conducted with stainless steel bundles, not zirconium alloy bundles.)

[337] For all of the values of reflood rates and PCTs in the TH-1 tests see C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, "Prototypic Thermal-Hydraulic Experiment," p. 13.
[338] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.
[339] "Return to Nucleate Boiling during Blowdown and Steam Cooling Restriction," Attachment 3 of "Research Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K," June 20, 2002, p. 2 (hereinafter "Return to Nucleate Boiling"); Attachment 3 is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021720713; the letter's Accession Number: ML021720690.

1. The Significance of Peak Cladding Temperatures at the Onset of Reflood and Low Reflood Rates

It is significant that in Nuclear Energy Institute's ("NEI") comments on PRM-50-93, dated April 12, 2010, NEI states:

> Depending on the plant design, core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F; these are significantly lower than in FLECHT Run 9573 [1970°F] and at flooding rates substantially above the 1.1 inch/second of this test. Flooding rates as low as [1.1 in./sec.] are possible only after significant cooling is established within the core.[340]

(It is noteworthy that NEI makes the claim that "[f]looding rates as low as [1.1 in./sec.] are possible only after significant cooling is established within the core,"[341] yet NEI provides no experimental data to substantiate its claim. NEI also does not provide any experimental data that indicates what initial reflood rates would be or what the time duration of the initial reflood rates would be before the effects of steam binding set in, causing reflood rates to decrease. Additionally, NEI does not provide any experimental data from tests conducted with full-length zirconium alloy multi-rod bundles that indicates that there would in fact be significant cooling in the core when reflood rates dropped to 1 in./sec. or lower.)

It is likely that NEI is basing its claim that "[f]looding rates as low as [1.1 in./sec.] are possible only after significant cooling is established within the core,"[342] on the data of thermal hydraulic experiments conducted with stainless steel multi-rod bundles, *not* zirconium alloy bundles.

(It is also noteworthy that, regarding cladding temperature increases during blowdown, "Assessment of the TRAC-M Codes" states:

During a large-break LOCA, cladding temperature changes as follows:

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[340] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," April 12, 2010, Attachment, p. 3.
[341] *Id*.
[342] *Id*.

> Cladding temperature increases during blowdown from normal operating conditions of approximately 325°C *to approximately 550-800°C (roughly 1000-1500°F)* [emphasis added].[343])

And regarding bottom reflood, "Return to Nucleate Boiling" states:

> Bottom reflood progresses very quickly during the onset of reflood. However, the intense steam generation soon retards the overall progression of the quench front to a relatively uniform progression. Nevertheless, good core quenching rates are achieved even for flooding rates of one inch per second.
>
> … During reflood, the flow regime, cladding temperature rise and quench behavior is strongly dependant on the flooding rate.[344]

It is important to recognize that when "Return to Nucleate Boiling" states that "good core quenching rates are achieved even for flooding rates of one inch per second," this claim is based on the results of tests conducted with stainless steel bundles, *not* zirconium alloy bundles.

(In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.)

As stated above, the TH-1 tests demonstrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126 (reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.74 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F).[345]

If indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F,"[346] this is highly problematic, because it means that, with high probability, reflood rates of 1 in./sec. or lower would not be sufficient to quench the core.

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[343] NRC, "Assessment of the TRAC-M Codes," p. 3.
[344] "Return to Nucleate Boiling," p. 2.
[345] For all of the values of reflood rates and PCTs in the TH-1 tests see C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, "Prototypic Thermal-Hydraulic Experiment," p. 13.
[346] *Id*.

K. FLECHT Run 9573 (Conducted with a Zircaloy Bundle) Incurred Runaway Oxidation

In 1973, the AEC Commissioners stated, "[i]t is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[347] Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the bundle used in run 9573 incurred runaway oxidation.

"PWR FLECHT Final Report" states that, "[t]he objective of the PWR FLECHT…test program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of PWR emergency core cooling systems."[348] A runaway oxidation reaction was not expected to occur in any of the FLECHT tests.[349]

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT Final Report"] when compared to the equivalent stainless steel tests."[350] 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."[351]

1. The Low Flood Rate of FLECHT Run 9573

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

> At this time [2004] we know that high temperature tests similar to [FLECHT] run 9573 would require rod bundle powers well outside the range of operation of any current or proposed PWRs. Also, no realistic transient experiments or analyses have indicated cladding temperatures at the beginning of reflood anywhere near the 1970°F achieved in [FLECHT] run 9573. If [FLECHT] run 9573 were repeated the results

[347] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," p. 1124.
[348] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 1-1.
[349] "PWR FLECHT Final Report" does not mention that an autocatalytic oxidation reaction occurred during FLECHT run 9573.
[350] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," pp. 16-17.
[351] *Id*., p. 17.

> would probably be the same, the high temperatures and high power would quickly catapult the cladding into the severe metal-water reaction regime, destroying the bundle and producing very little useful heat transfer information.[352]

Indeed, it is reasonable to postulate that if FLECHT run 9573 were repeated that the bundle would once again be destroyed by runaway oxidation; however, this would be as a consequence of the low flood rate of the coolant (1.1 in./sec.) as well as the high initial cladding temperatures and high power of the bundle. In "Technical Safety Analysis of PRM-50-76," the NRC neglected to mention the fact that FLECHT run 9573 had a low coolant flood rate. Regarding the significance that coolant flood rates played in the PWR FLECHT test program, "PWR FLECHT Final Report" states, "[i]n general, the effect on heat transfer coefficient[s] of varying system parameters was clearly discernable, *with flooding rate being by far the most influential parameter investigated*" [emphasis added].[353] The NRC's "Compendium of ECCS Research for Realistic LOCA Analysis" reiterates that in the PWR FLECHT test program, flooding rates were the most influential parameter for affecting heat transfer coefficients.[354]

It is significant that run 9573 had a lower initial cladding temperature than, and the same power level as, other Zircaloy tests conducted in the PWR FLECHT test program that did not incur runaway oxidation. It is also significant that FLECHT run 9573 had the lowest flood rate of the four Zircaloy tests.[355] Additionally, it is noteworthy that "Consolidated National Intervenors pointed out that most of [the Zircaloy] runs were made at unreasonably high flooding rates, and that a different result was obtained from run 9573 where the flooding rate was about one inch per second."[356]

It would be reasonable to postulate that if FLECHT run 9573 were repeated—with the same or a lower coolant flood rate, yet with lower initial cladding temperatures (that in the event of a LOCA, would occur at the beginning of reflood at current and/or

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[352] NRC, "Technical Safety Analysis of PRM-50-76," p. 8.
[353] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 5-1.
[354] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.4-14.
[355] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. B-2.
[356] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision," p. 1124.

proposed PWRs) and a lower power level (within the operational range of current and/or proposed PWRs)—that the bundle would still incur runaway oxidation and be destroyed, because FLECHT run 9573 had the lowest flood rate of the four Zircaloy tests. Furthermore, it is likely that such a test would produce valuable heat transfer information.

2. A Comparison of FLECHT Run 9573 (Conducted with a Zircaloy Bundle) and Two FLECHT Runs Conducted with a Stainless Steel Bundle

PWR FLECHT stainless steel runs 6553 and 9278 (with the same peak power levels as Zircaloy run 9573), at the hot rod midplane elevation, at the onset of flood, had cladding temperatures of 2012°F and 2028°F, respectively, flood rates of 1 in./sec., and peak cladding temperatures of 2290°F and 2286°F, respectively.[357] In contrast to Zircaloy run 9573—with a slightly lower clad temperature at the onset of flood and a slightly higher flood rate—runs 6553 and 9278 did not incur runaway oxidation. In fact, runs 6553 and 9278 were conducted with the same stainless steel bundle, and after run 9278 was conducted, the bundle was reused for more tests, because it remained intact.

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

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

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

> Daniel Ford: Mr. Moore, is it correct that in the [PWR] FLECHT tests[358] negative heat transfer coefficients [calculated as a result of heat transfer from the coolant to the fuel cladding] were observed at axial levels in a number of different instances?

[357] F. F. Cadek, D. P. Dominicis, R. H. Leyse, “PWR FLECHT Final Report,” p. 3-6.

[358] The transcript states “BWR FLECHT tests, not “PWR FLECHT tests”;” however, it is most likely that Daniel Ford was actually asking a question about the PWR FLECHT tests. First, James S. Moore was an employee of Westinghouse Electric, which conducted the PWR FLECHT tests. Second, negative heat transfer coefficients—calculated as a result of heat transfer from the coolant to the fuel cladding—occurred in the PWR FLECHT tests. Third, results from the PWR FLECHT tests would have been used for ECCS evaluation calculations for IP-2, because IP-2 is a PWR.

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

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

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

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

James S. Moore: Yes, yes.

Daniel Ford: Thank you.

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

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

Daniel Ford: Yes.

James S. Moore: Yes.

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

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

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

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

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

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

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

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

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

James S. Moore: Yes.

Daniel Ford: Where are those calculations presented?

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

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

James S. Moore: Yes.[359]

It is rather odd that James S. Moore would answer, "yes," to Daniel Ford's last question in the portion of the IP-2 licensing hearing transcript quoted above, given the results of PWR FLECHT run 9573. Ford had asked Moore, "whether [Westinghouse had] performed parametric calculations that indicate under no circumstances, that is, under no combination of parameters, [would there be] superheated steam at lower than the ten-foot elevations[, where] it was observed at in the FLECHT test[s],"[360] in the event of a LOCA?

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

Additionally, it is rather odd that James S. Moore would claim that "[f]or the hot spot calculation, [negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding would] never…exist."[361] And that "[negative heat transfer coefficients: calculated as a result of heat transfer from the coolant to the fuel cladding occurred] well above the hot spot. … [That t]hey were where the temperature was quite low [on] the [fuel] cladding,"[362] given the results of PWR FLECHT run 9573.

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

This was reported in a Westinghouse document, "PWR FLECHT Final Report," in April 1971, months before James S. Moore's testimony.

Regarding the superheated steam, which exceeded 2500°F, and negative heat transfer coefficients observed at the bundle midplane in FLECHT run 9573, "PWR FLECHT Final Report" states:

> At the time of the initial [heater element: fuel-cladding simulator] failures [in FLECHT run 9573], midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure). …anomalous (negative) heat transfer coefficients were observed at the bundle midplane for 5 of 14 thermocouples during this period. These may have been related to the high steam probe temperatures measured at the 7 ft elevation.[363]

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

> The high fluid [superheated steam] temperature [that occurred in FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[364]

[361] *Id.*, p. 2923.
[362] *Id*.
[363] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," pp. 3-97, 3-98.
[364] 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.

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

> The final FLECHT test (Bundle Z-10) was completed on December 11, 1970. The test was run with flooding of 1 in./sec. beginning at 2000°F. Several heaters failed approximately 18 seconds after flooding when the peak indicated midplane temperature was 2325°F. Heater failure at this temperature is unlikely, particularly under conditions of decay heat and increasing temperature. *The steam probe thermocouple located one foot above midplane in close proximity to a Zircaloy grid indicated an extremely rapid rate of temperature rise (over 300°F/sec.) beginning approximately 12 seconds after flooding and reaching 2450°F by 16 seconds after flooding.* It appears likely that *ignition of the Zircaloy grids led to high rates of heat input* at the elevation one foot above (and below) midplane* and this caused over-temperature and failure of the heaters. Test results are currently being studied.
>
> The temperature measuring system in FLECHT was the object of a complete audit by Idaho Nuclear Corporation prior to the final FLECHT test. The audit was very thorough and required approximately seven days. Idaho Nuclear Corporation found that the total temperature measurement system was highly reliable and the final Zircaloy test was run with no changes to the system.
>
> *The ratio of surface area to heat capacity for a Zircaloy grid is approximately 15 times that of a heater rod; hence, Zircaloy-steam reactions can lead [to] steeper temperature ramps in the vicinity of a Zircaloy grid [emphasis added].[365]

It is significant that "PWR FLECHT Final Report" states that the negative heat transfer coefficients that were observed at the bundle midplane in FLECHT run 9573 were "anomalous."[366] Perhaps that is why James S. Moore claimed that negative heat transfer coefficients would never occur at the hot spot of the fuel cladding, in the event of a LOCA. However, Westinghouse's conclusion that the negative heat transfer coefficients observed in FLECHT run 9573 were anomalous had no scientific basis. For example, Westinghouse did not conduct any subsequent tests with Zircaloy bundles after FLECHT run 9573 (with similar test parameters) to confirm that the negative heat

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[365] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970.
[366] F. F. Cadek, D. P. Dominicis, R. H. Leyse "PWR FLECHT Final Report," p. 3-98.

transfer coefficients were in fact anomalous. This is unfortunate, given the importance of the safety issues involved.

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

L. A Current Heat Transfer Experiment Program that Conducts Tests with Inconel 600 Bundles: the Rod Bundle Heat Transfer Facility Program

Regarding the Rod Bundle Heat Transfer ("RBHT") facility program, NRC's Advisory Committee on Reactor Safeguards ("ACRS") states:

> The Rod Bundle Heat Transfer (RBHT) facility at Penn State University was developed to address issues related to emergency core cooling, including the development of a better understanding of reflood and rewetting in realistic, bundled geometries. Currently, this facility is being used to conduct oscillating reflood tests to determine the effect of the inlet flow rate, magnitude, and frequency on peak clad temperature (PCT).[367]

So according to NRC's ACRS, the RBHT facility was developed to address issues related to emergency core cooling, including phenomena that would affect PCTs. Unfortunately, as the RBHT test plan states, "[o]xidation [is] not simulated in [the RBHT facility] tests, since [the] cladding is Inconel."[368] And, as the RBHT test plan acknowledges, "Inconel will not oxidize, while Zircaloy will [oxidize] and create a secondary heart source at very high PCTs; Zircaloy reaction can be significant at high temperature[s]."[369]

It is unfortunate that NRC's ACRS believes that "the development of a better understanding of reflood and rewetting in realistic, bundled geometries"[370] can be achieved through experiments that do not simulate the heat that would be generated by the exothermic Zircaloy-steam reaction in the event of a LOCA.

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[367] Advisory Committee on Reactor Safeguards, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," NUREG-1635, Vol. 9, Draft Final, March 2010, p. 68.
[368] NRC, L. E. Hochreiter *et al.*, "RBHT Test Facility Test Plan," p. 250.
[369] *Id*.
[370] Advisory Committee on Reactor Safeguards, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," p. 68.

And regarding the fact that the RBHT facility program would provide new data for best-estimate computer code models, "RBHT Test Facility Test Plan" states:

> The RBHT facility is a unique facility which will provide new data for the fundamental development of best-estimate computer code models. This effort will reduce the uncertainty in the NRC's thermal-hydraulic computer codes which will enhance the understanding of the complex two-phase phenomena which is modeled for the reflood transient.[371]

Clearly, the RBHT facility program goals of "[providing] new data for the fundamental development of best-estimate computer code models," cannot be achieved, because the RBHT facility experiments do not simulate the heat that would be generated by the exothermic Zircaloy-steam reaction in the event of a LOCA.

It is unfortunate that NRC's ACRS and the test planers of the RBHT facility program have not studied the data from the NRU TH-1 tests (thermal hydraulic experiments conducted with a Zircaloy bundle).

1. A Comparison between the FLECHT-SEASET Test 31504 (Conducted with a Stainless Steel Bundle) and Two NRU TH-1 Tests (Conducted with a Zircaloy Bundle)

It is significant that "RBHT Test Facility Test Plan" states:

> A number of important rod bundle experiments were reviewed to determine the availability of data, test facility design, types of tests, instrumentation, and data from tests."[372]

Among the experiments listed in the RBHT test plan are the FLECHT Low Flooding Rate Cosine Test Series, FLECHT-SEASET 161-Rod Unblocked Bundle Tests, and the NRU Rod Bundle Tests, conducted at Chalk River Canada. These experiments are also discussed in Appendix A of the RBHT test plan.

(It is noteworthy that when Appendix A discusses the NRU Rod Bundle Tests, it only discusses Thermal-Hydraulic Experiment 2 and 3 ("TH-2 and -3"). Appendix A does not discuss the NRU TH-1 tests, which had higher PCTs than the TH-2 and -3 tests:

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371 NRC, L. E. Hochreiter *et al.*, "RBHT Test Facility Test Plan," Abstract, p. iii.
372 *Id.*, p. 39.

the highest PCTs in the TH-2 and -3 tests were 1834°F[373] and 1912°F,[374] respectively. The highest three PCTs in the TH-1 tests were 1991°F, 1991°F, and 2040°F.[375])

Regarding the FLECHT-SEASET 161-Rod Unblocked Bundle tests, (conducted with stainless steel), Appendix A of "RBHT Test Facility Test Plan" states:

> The FLECHT-SEASET [161-Rod] Unblocked Bundle tests were the first publicly available reflood experiments on the newer 17x17 fuel array which was adopted by the utilities in the 1980s. [376]

And also regarding the FLECHT-SEASET 161-Rod Unblocked Bundle tests, Appendix A of the RBHT test plan states:

> The FLECHT-SEASET 161-rod unblocked bundle experiments represent the best reflood experiments which were performed. It was recognized in the test planning that data was needed for advanced reflood computer code development such that an effort was made to obtain additional local heat transfer and fluid flow data in addition to the total heater rod heat transfer data for an empirical correlation. …
>
> The analysis for the test data is the most complete of all the FLECHT test series. Test 31504 was analyzed in detail with several plots given in the reports which can be used for computer code validation. …
>
> Two of the FLECHT-SEASET tests were used as US Standard Problem 9 for the purposes of code validation. It is strongly recommended that these data be used for validating the NRC merged code.[377]

And regarding the TH-2 and -3 tests, Appendix A of the RBHT test plan states that "[the Zircaloy reaction should] exist [in the TH-2 and -3 tests] because Zircaloy is used."[378] The RBHT test plan also acknowledges that "Inconel will not react, while

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[373] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 2 (TH-2)," NUREG/CR-2526, 1982, located in ADAMS Public Legacy, Accession Number: 8212220265, pp. v, 17.
[374] C. L. Mohr, *et al.*, Pacific Northwest Laboratory, "LOCA Simulation in NRU Program: Data Report for Thermal-Hydraulic Experiment 3 (TH-3)," NUREG/CR-2527, 1983, located in ADAMS Public Legacy, Accession Number: 8304120660, p. 14.
[375] 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," p. 13.
[376] NRC, L. E. Hochreiter *et al.*, Appendix A, Literature Review, of "RBHT Test Facility Test Plan," p. A-22.
[377] *Id.*, pp. A-25, A-26.
[378] *Id.*, p. A-87.

Zircaloy will react and create a secondary heart source at very high PCTs; [Zircaloy] reaction can be significant."[379]

It is significant that a comparison between the FLECHT-SEASET test 31504 (conducted with a stainless steel bundle) and two NRU TH-1 tests (conducted with a Zircaloy bundle, driven by low-level fission heat: an amount to simulate decay heat), indicates that if the FLECHT-SEASET test 31504 had been conducted with the same parameters, yet with a Zircaloy bundle, that its PCT would have (with high probability) exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

First, the FLECHT-SEASET test 31504 had a peak power of 0.70 kW/ft, reflood rate of 0.97 in./sec., PCT at the onset of reflood of 1585°F, and an overall PCT of 2101°F (an increase of 516°F).[380]

Second, the TH-1 test no. 127 had an average fuel rod power of 0.37 kW/ft,[381] reflood rate of 1.0 in./sec., a PCT at the start of reflood of 966°F, and an overall PCT of 1991°F (an increase of 1025°F); and the TH-1 test no. 130 had an average fuel rod power of 0.37 kW/ft, reflood rate of 0.74 in./sec., a PCT at the start of reflood of 998°F, and an overall PCT of 2040°F (an increase of 1042°F).[382]

It is significant that 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.[383] So because of the heat generated from the metal-water reaction, the peak cladding temperature increased by 190°F, after the reactor shutdown.

It can be extrapolated that if the FLECHT-SEASET test 31504 had been conducted with a zirconium alloy bundle, so that it would have had additional heat,

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[379] *Id.*
[380] *Id.*, p. A-29.
[381] C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," p. 10.
[382] For all of the values of reflood rates and PCTs in the TH-1 tests see C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, "Prototypic Thermal-Hydraulic Experiment" NUREG/CR-1882, p. 13.
[383] *Id.*

generated from the metal-water reaction, that its overall PCT would have (with high probability) exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

M. 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 TH-1 tests are reported on in "Prototypic Thermal-Hydraulic Experiment."[384]

Describing the TH-1 test bundle, "Prototypic Thermal-Hydraulic Experiment" states:

> The [TH-1] fuel bundle consists of a 6x6 segment of a 17x17 PWR design with the four corner rods removed for easier insertion in the shroud. This provides a basic test array of 6x6-4 or 32 rods. The outer row of rods, including the corner rods of the next inner ring, will not be pressurized and will serve as guard rod heaters during the test. The test section consists of 11 fuel rods and [one] instrument thimble tube arranged in a cruciform pattern. The test rods in the test section were unpressurized for the thermal-hydraulic test series.[385]

(In the pre-transient phase of the TH-1 tests, the average fuel rod power was 0.37 kW/ft[386] and the test loop inlet pressure was planned to be approximately 0.28 MPa (40 psia):[387] "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."[388])

As discussed above, the TH-1 tests[389] demonstrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases: test no. 126

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[384] C. L. Mohr, G. M. Hesson, G. E. Russcher, R. K. Marshall, L. L. King, N. J. Wildung, W. N. Rausch, W. D. Bennett, "Prototypic Thermal-Hydraulic Experiment."
[385] *Id.*, p. 4.
[386] *Id.*, p. 10.
[387] 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, p. 6-5.
[388] *Id.*
[389] For all of the values of reflood rates and PCTs in the TH-1 tests see C. L. Mohr, *et al.*, "Prototypic Thermal-Hydraulic Experiment," p. 13.

(reflood rate of 1.2 in./sec.) had a PCT at the start of reflood of 800°F and an overall PCT of 1644°F (an increase of 844°F), test no. 127 (reflood rate of 1.0 in./sec.) had a PCT at the start of reflood of 966°F and an overall PCT of 1991°F (an increase of 1025°F), test no. 130 (reflood rate of 0.74 in./sec.) had a PCT at the start of reflood of 998°F and an overall PCT of 2040°F (an increase of 1042°F).

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

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

It is significant that in NEI's comments, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," dated April 12, 2010, NEI states:

> Depending on the plant design, core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F…[390]

If indeed, "core reflood starts at cladding temperatures of between 1300°F (or less) and 1600°F…"[391] it is highly problematic, because it means that, with high probability, reflood rates of 1 in./sec. or lower would not be sufficient to quench the core.

[390] NEI, "Industry Comments on Petition for Rulemaking (PRM-50-93); Multi-Rod (Assembly) Severe Fuel Damage Experiments. Docket ID NRC-2009-0554," April 12, 2010, Attachment, p. 3.
[391] *Id.*

(In the event of a LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.)

1. TH-1 Test No. 130

As discussed above in section III.D.1.c., 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.[392] 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 runaway oxidation if the reactor had not shutdown when the PCT was approximately 1850°F.

(It is noteworthy that, in 2005, the NRC stated that it was "reviewing…data from [the early '80s, from the program the TH-1 tests were part of,] to determine its value for assessing the current generation of codes such as TRAC-M (now renamed TRACE)."[393])

IV. CONCLUSION

Petitioner requests that NRC order the licensee of IP-2 and -3 to lower the LBPCTs of IP-2 and -3 in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs. Experimental data demonstrates that IP-2 and -3's LBPCTs of 1937°F and 1961°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 IP-2 and -3's LBPCTs need to both be decreased to temperatures lower than 1832°F in order to provide necessary margins of safety.

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[392] *Id.*
[393] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," p. 19.

Petitioner also requests that NRC order the licensee of IP-2 and -3 to determine how far below 1832°F the LBPCT values of IP-2 and -3 need to be lowered to in order to provide necessary margins of safety—to help prevent partial or complete meltdowns—in the event of LOCAs.

Additionally, Petitioner requests that NRC order the licensee of IP-2 and -3 to lower both of IP-2 and -3's LBPCTs to 1600°F—in the interim—before conservative LBPCT values for IP-2 and -3 are determined.

To uphold its congressional mandate to protect the lives, property, and environment of the people of New York, NRC must not allow IP-2 and -3'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
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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Respectfully submitted,

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Dated: March 28, 2011

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.

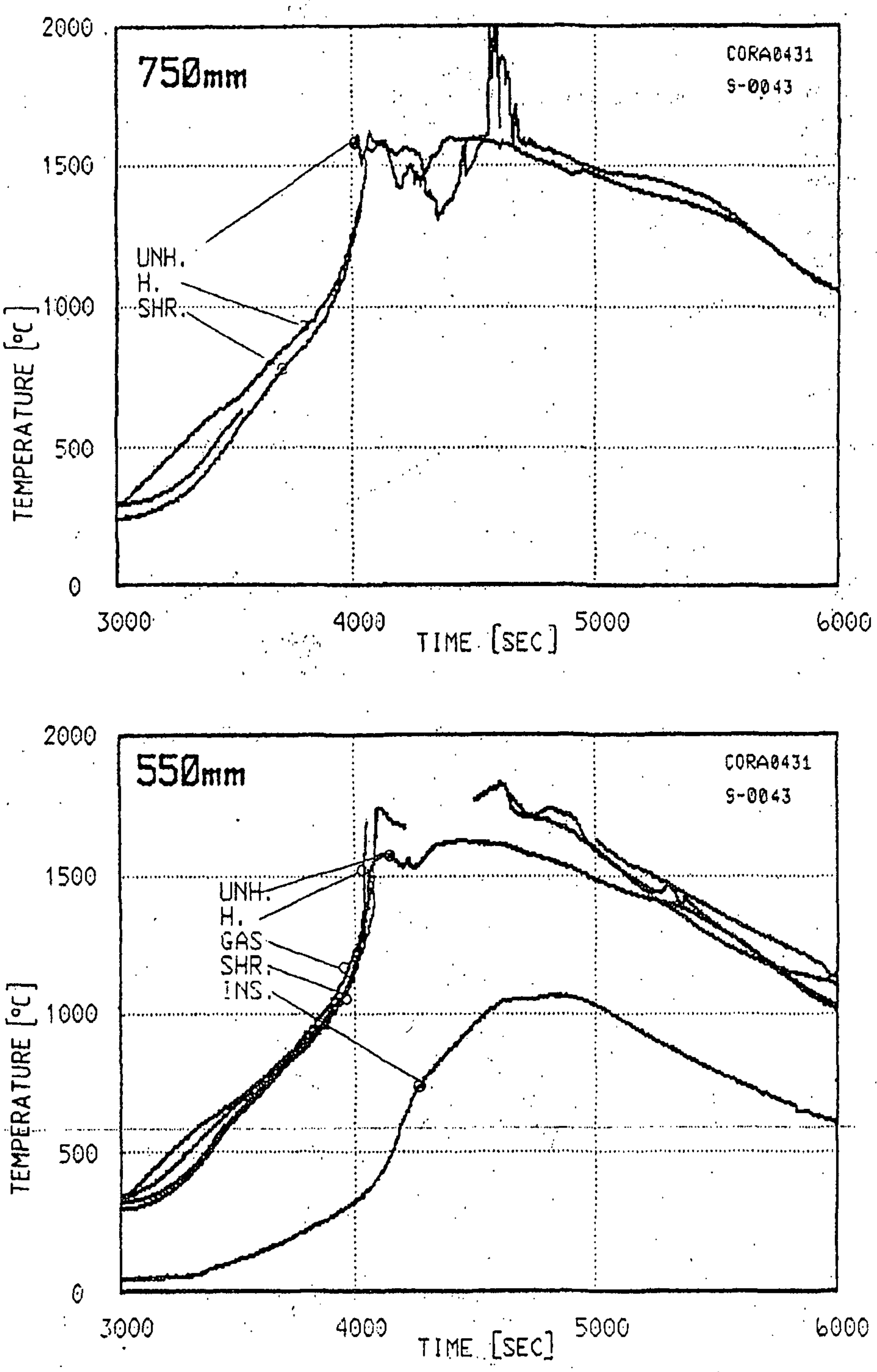


Fig. 12. Temperatures during test CORA-2 at 500 mm and 750 mm elevation. Temperatures of heated (H) and unheated rod (UNH), atmosphere (gas), shroud (SHR), and outer surface of shroud insulation (INS)

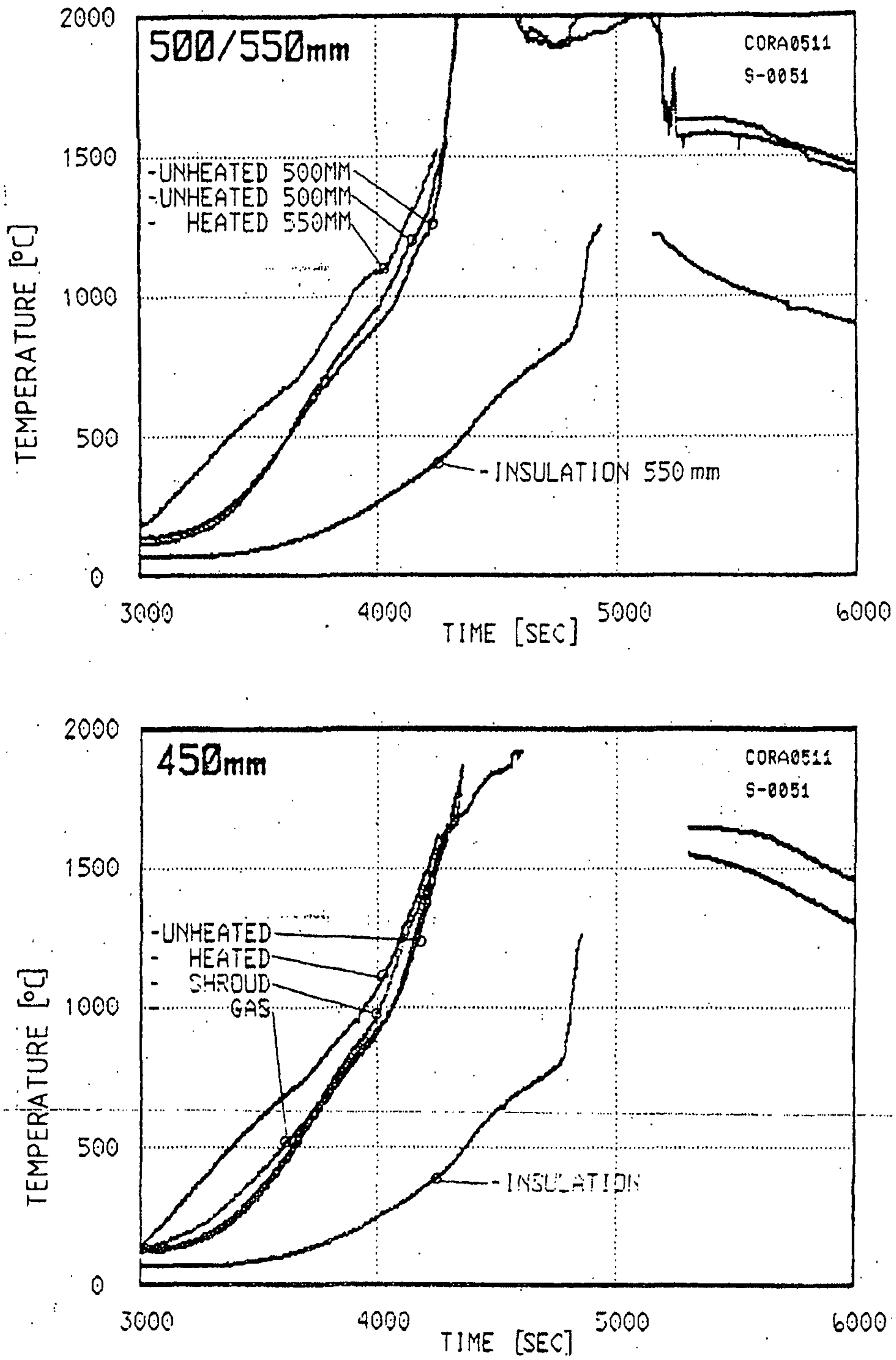


Fig. 13. Temperatures measured during test CORA-3 at 450 mm and 550 mm elevation

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.

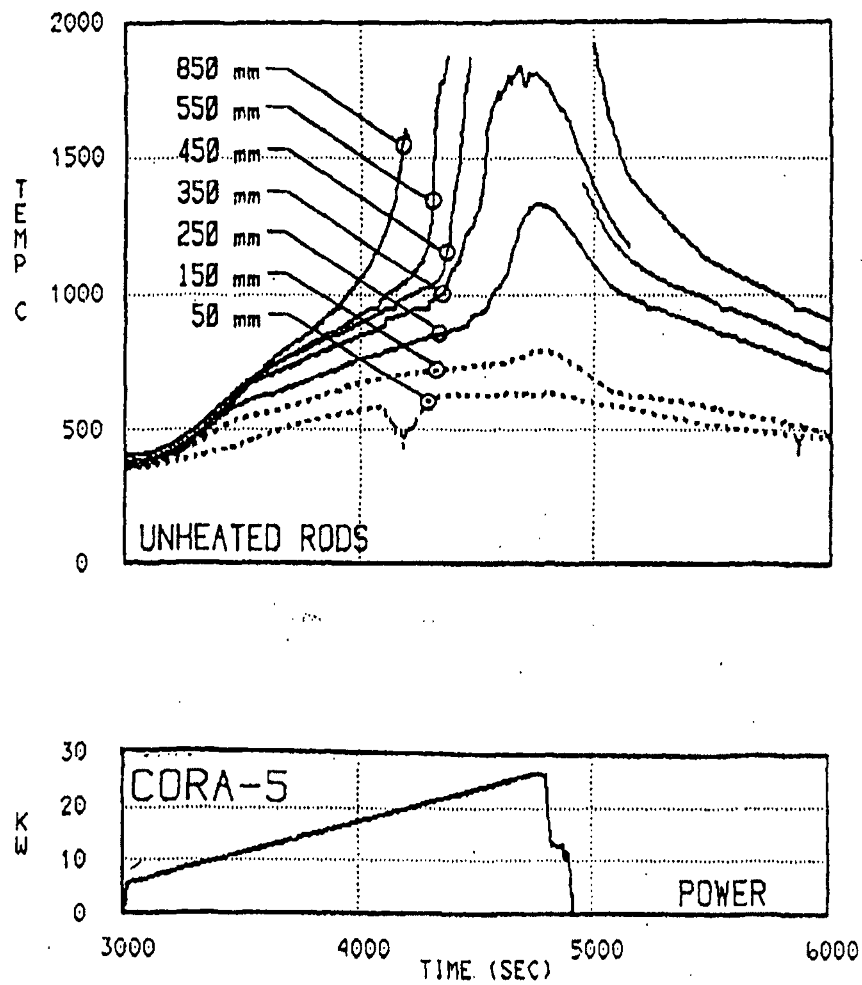


Fig.15: **Temperatures of unheated rods and power history of CORA-5**

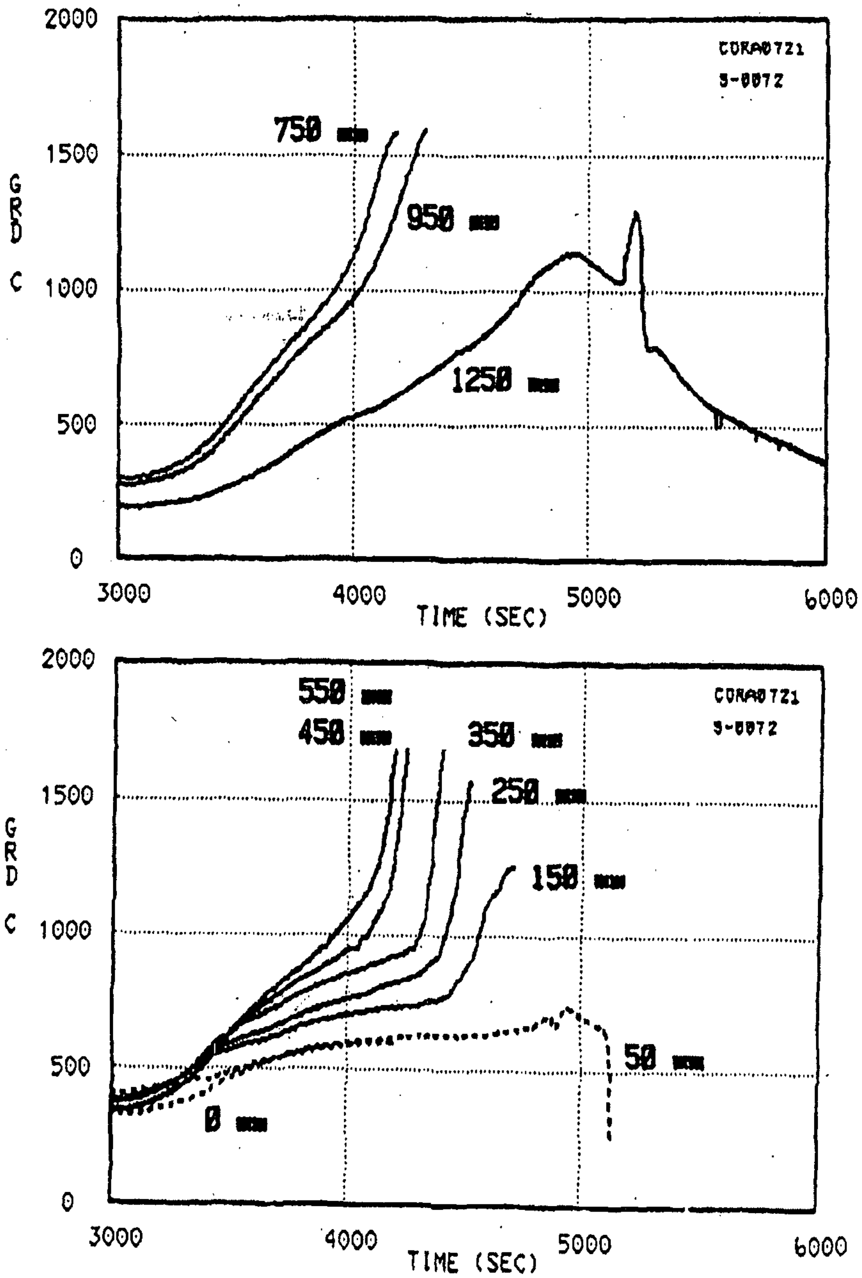


Fig.16: Temperatures of unheated rods during CORA-12

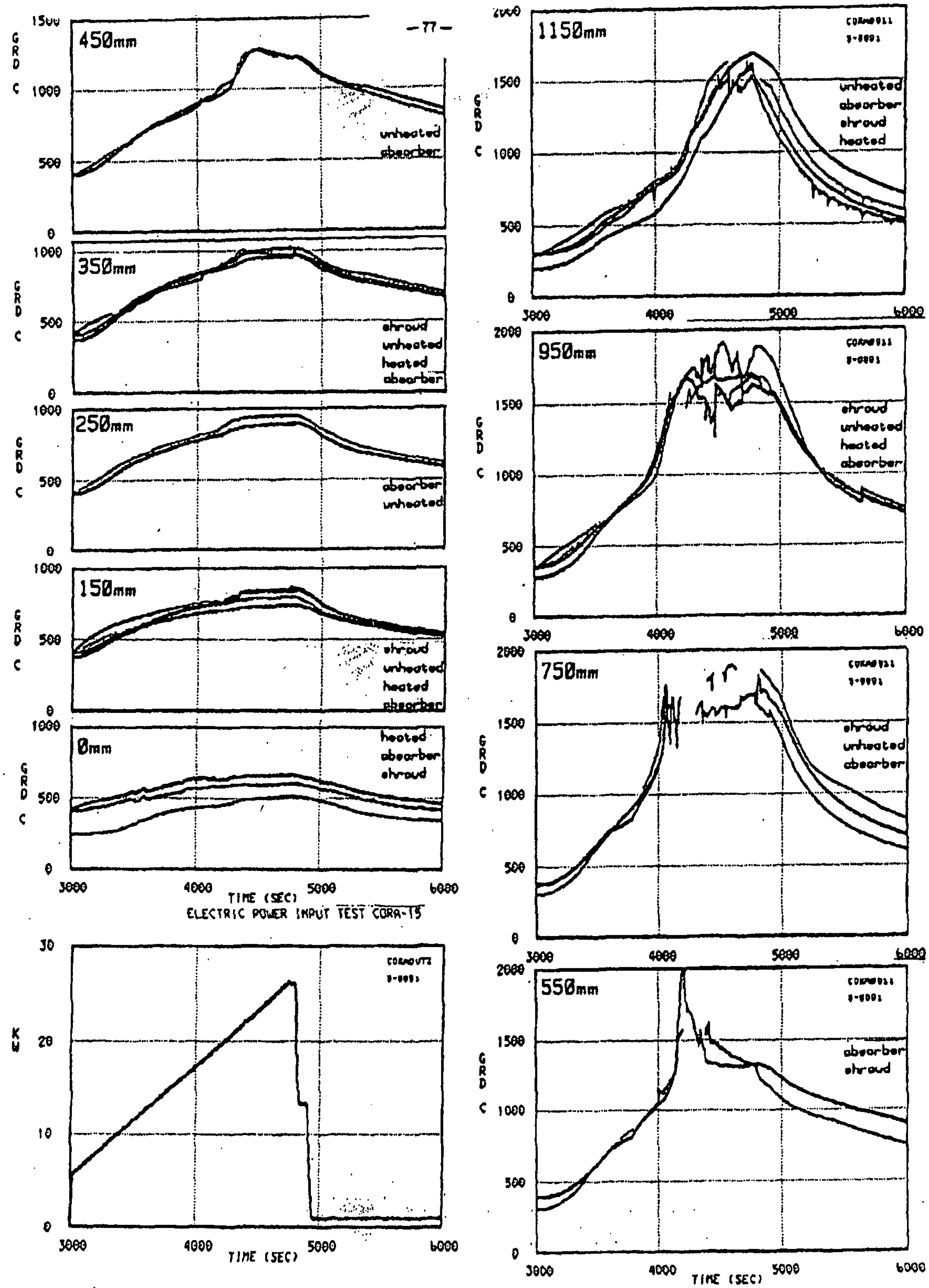


Fig.17: Temperatures at different elevations during CORA-15

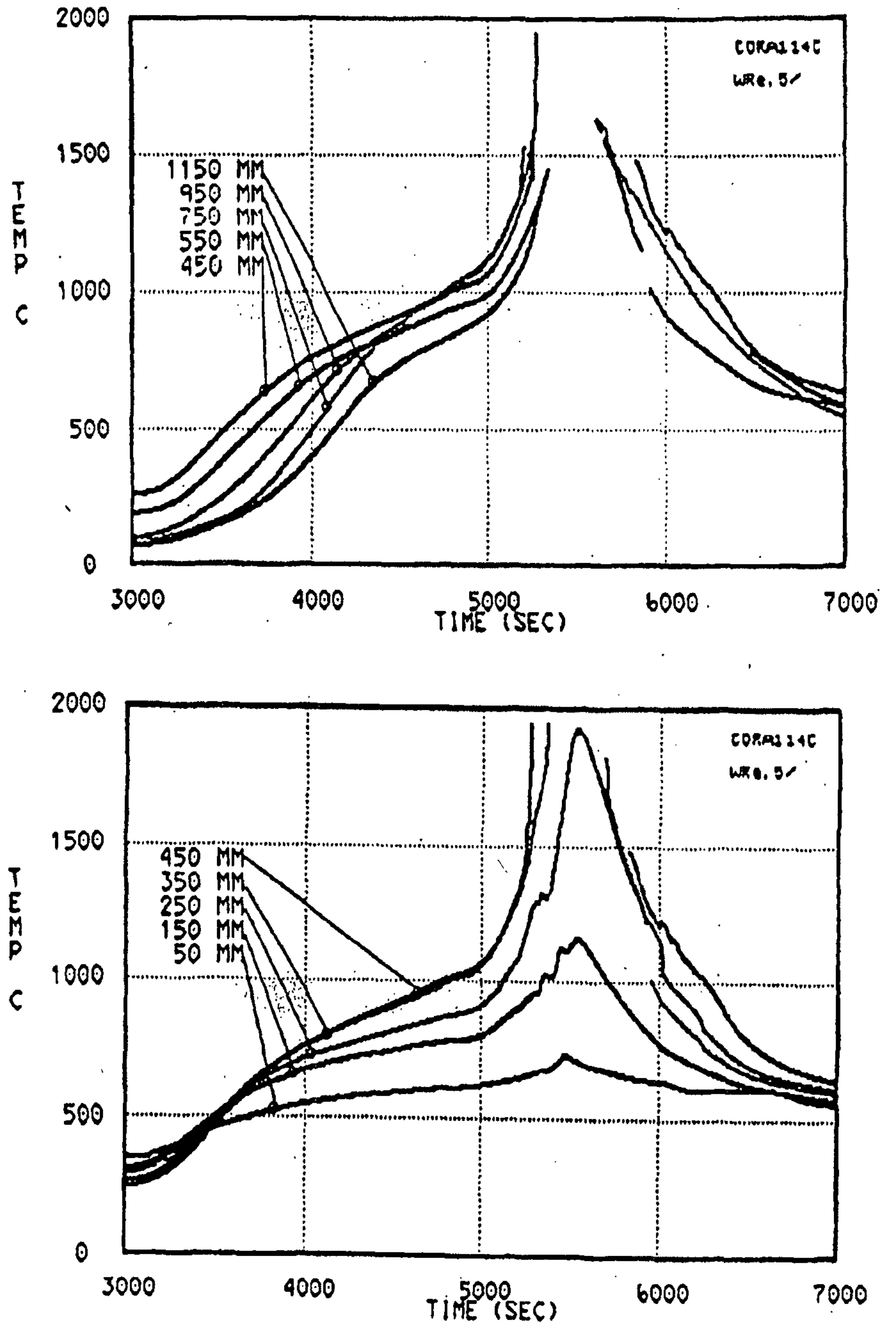
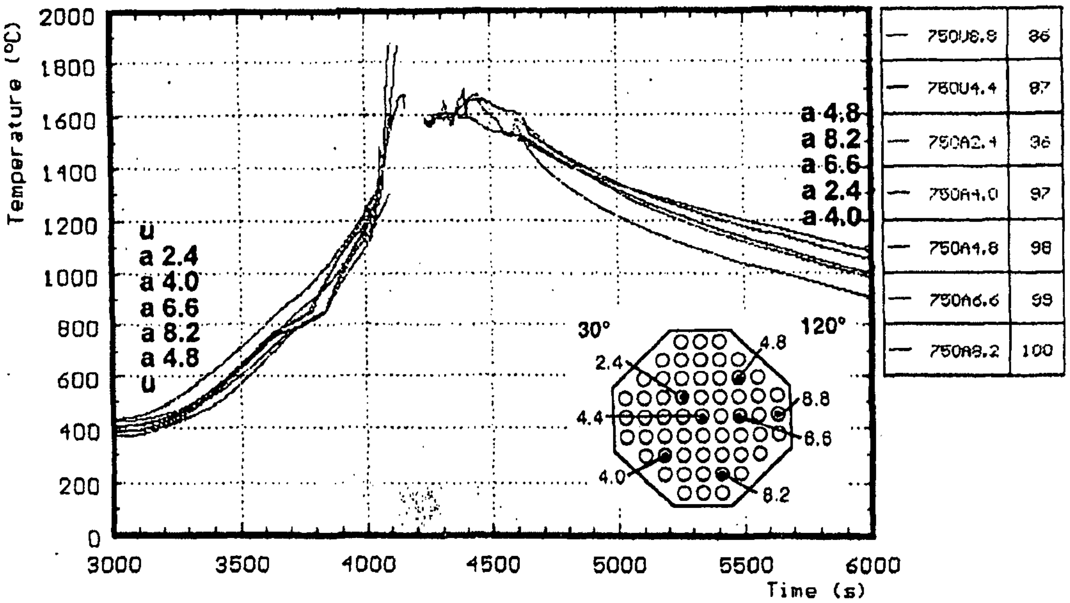
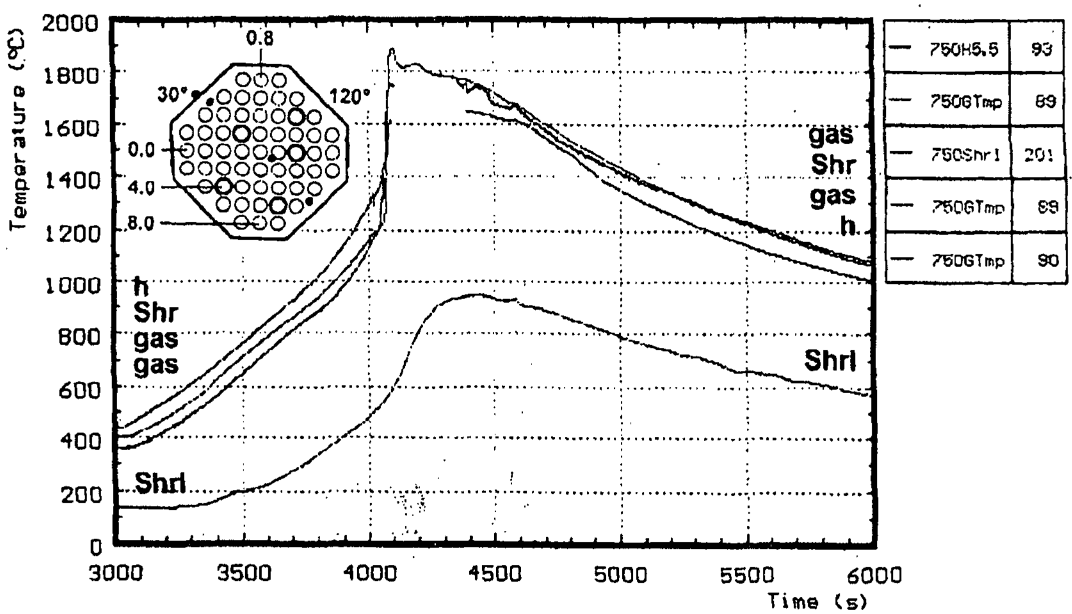


Fig.18: Temperatures of unheated rods during CORA-9





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)

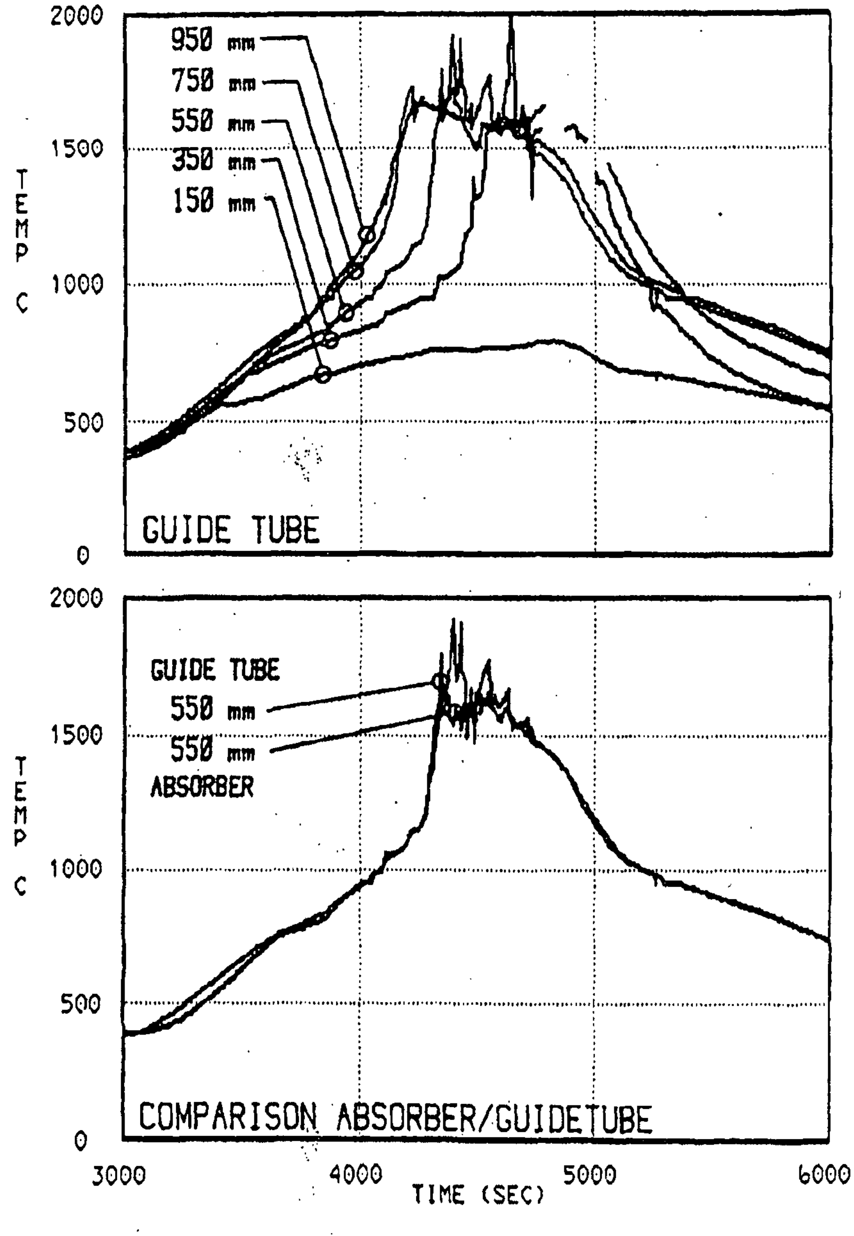


Fig. 20: Temperatures of guide tube and absorber rod during test CORA-5

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.

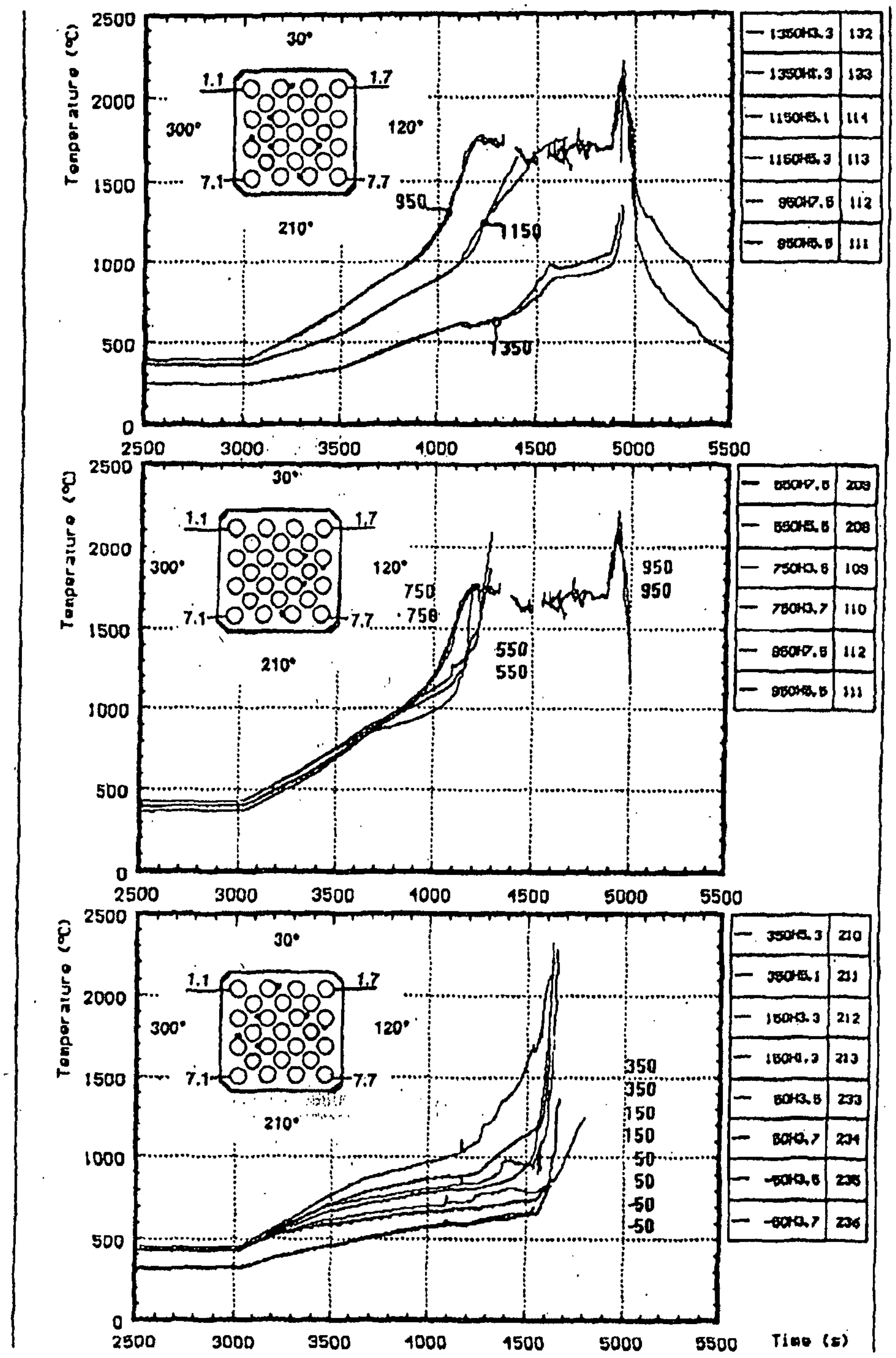


Fig. 37: Temperatures of the heated rods (CORA-13)

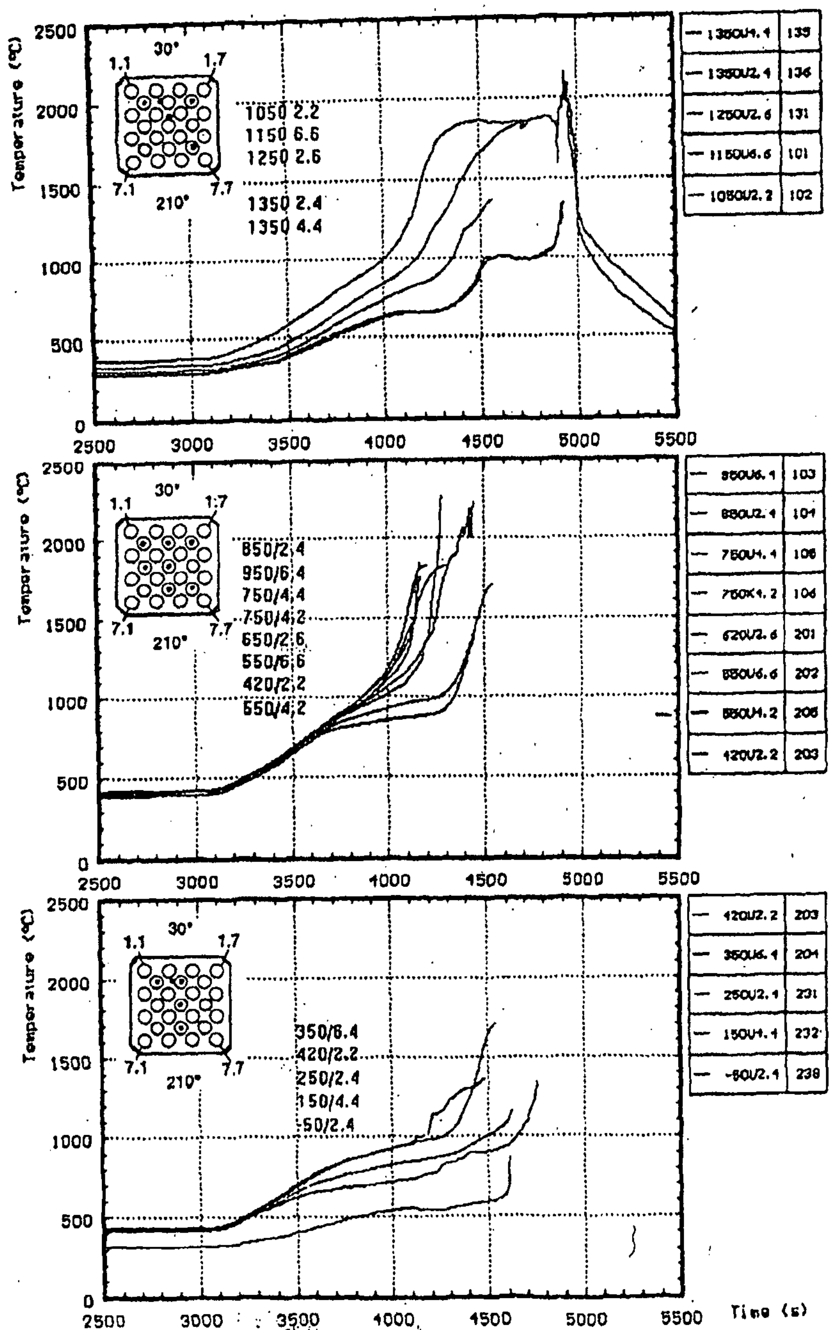


Fig. 39: Temperatures of the unheated rods (CORA-13)

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.

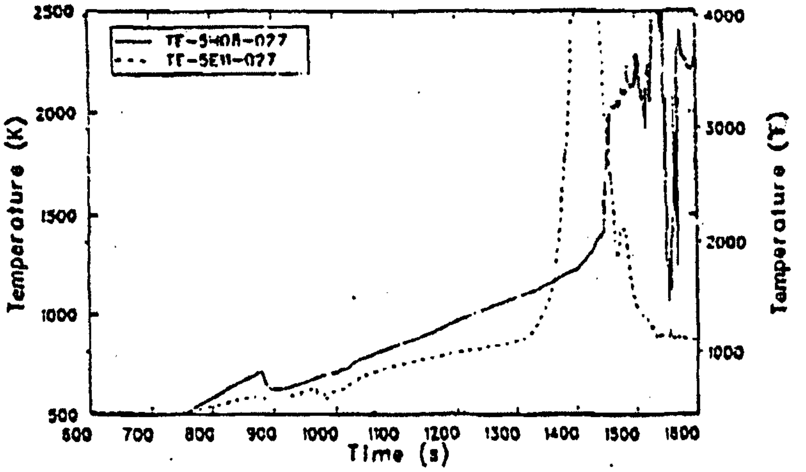


Figure 3.7 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5.

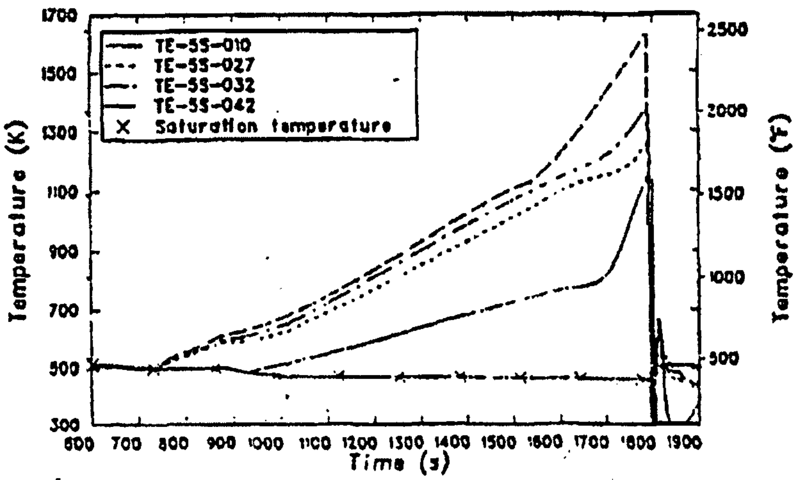


Figure 3.8 Comparison of four external wall temperatures at the 1.07-, 0.81-, 0.69-, and 0.25-m (42-, 32-, 27-, and 10-in.) elevations on the south side of the flow shroud.

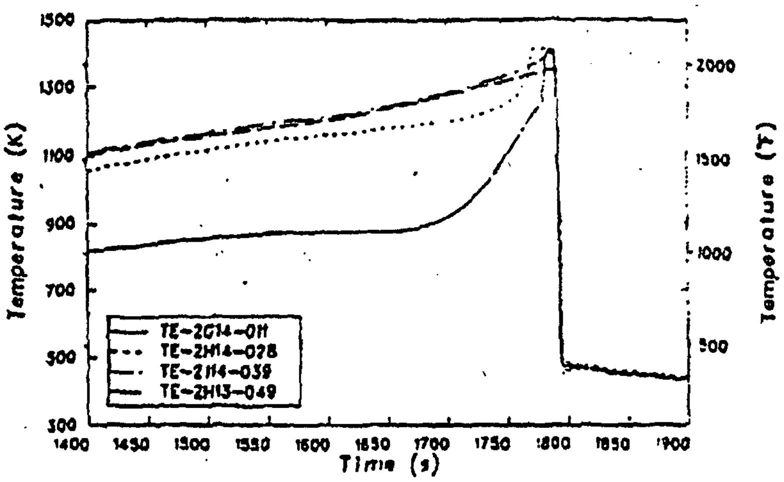


Figure 3.9 Comparison of cladding temperatures at the 1.24-, 0.99-, 0.71-, and 0.28-m (49-, 39-, 28-, and 11-in.) elevations in Fuel Assembly 2.

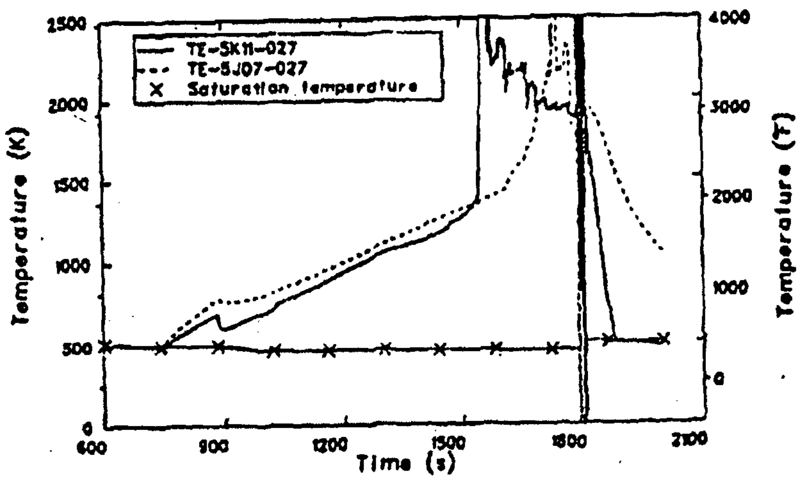


Figure 3.10 Comparison of two cladding temperatures at the 0.69-m (27-in.) elevation in Fuel Assembly 5 with saturation temperature.

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.

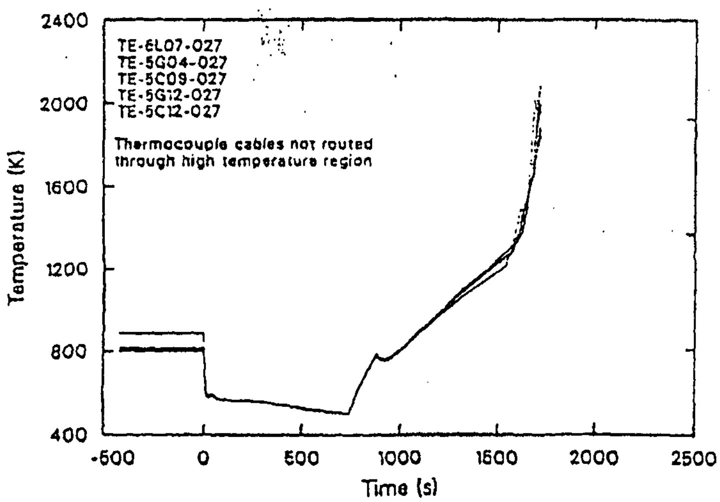


Fig. 14: CFM fuel cladding temperature at the 0.686 m (27 in) elevation

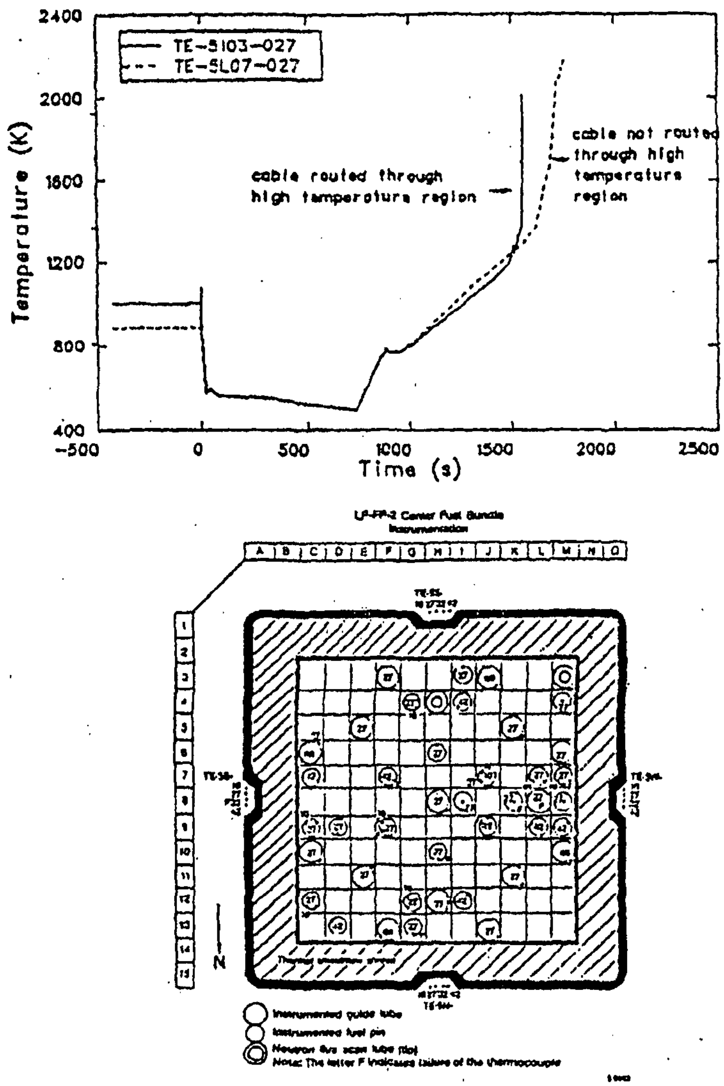


Fig. 15: Comparison of temperature data with and without cable shunting effects at the 0.686 m (27 in.) elevation in the CFM

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.

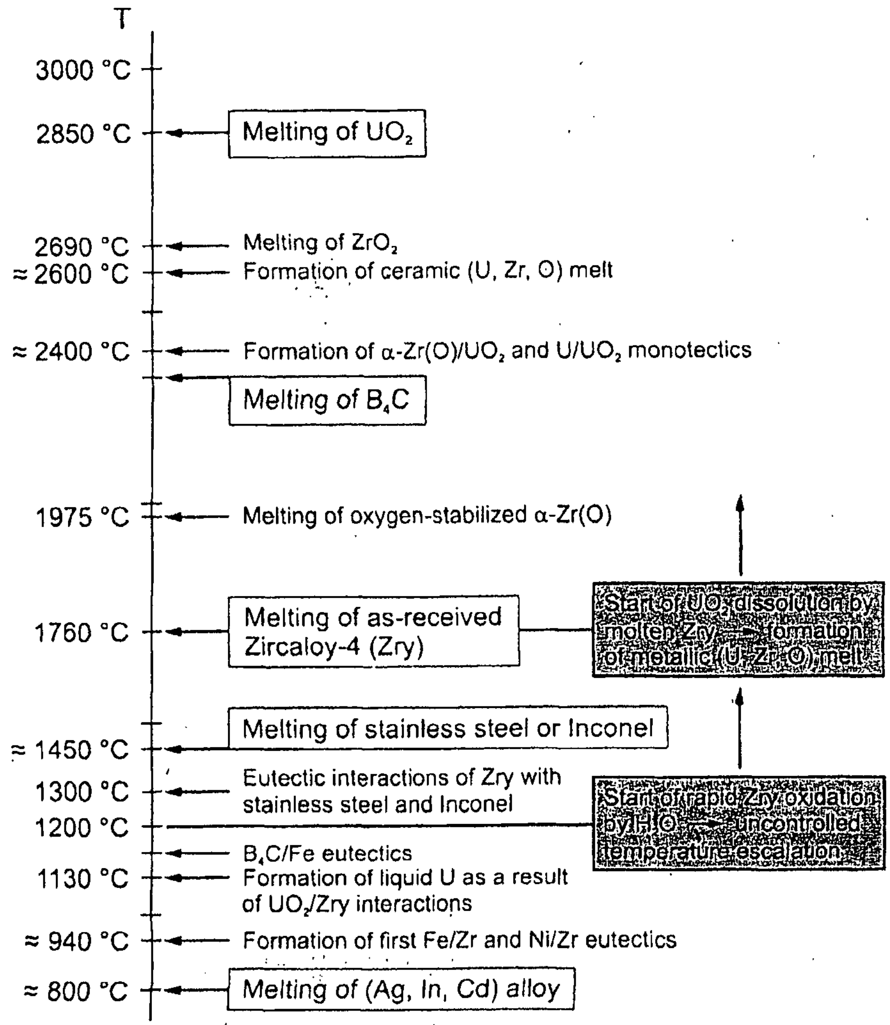


Fig. 1. LWR severe accident-relevant melting and chemical interaction temperatures which result in the formation of liquid phases.

- 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

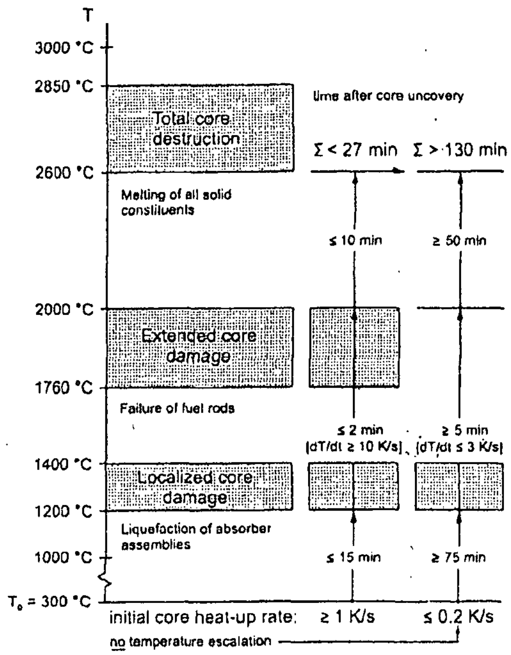


Fig. 13. Dependence of the temperature regimes on liquid phase formation on the initial heat-up rate of the core. Small heat-up rates drastically reduce the amount of molten Zircaloy (1800–2000°C) and give more time for possible accident management measures.

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].

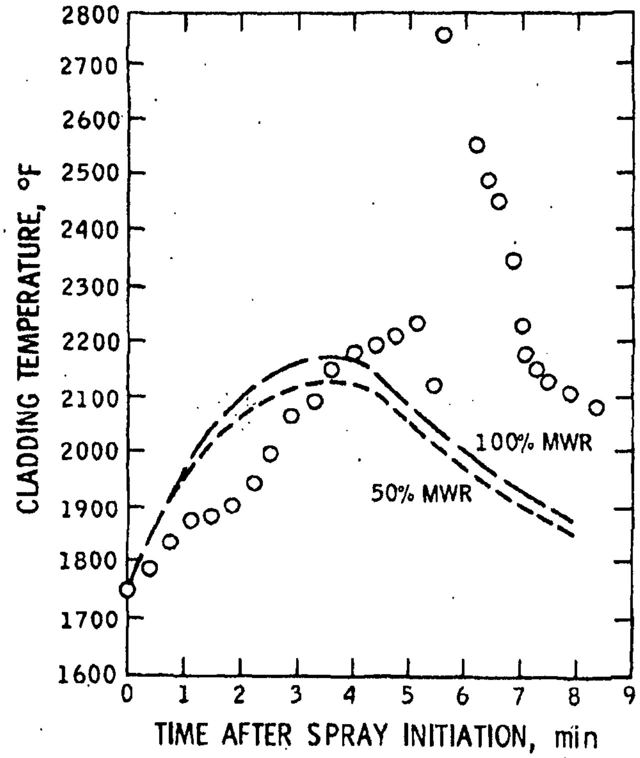
The CORA-10 test simulated the behavior of a rod bundle with additional cooling at its lower end (TMI-2 conditions) [34]. Fig. 14 depicts the axial bundle temperature profile at different times and the material relocation. One can recognize the influence of the higher heat losses at the lower end (30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

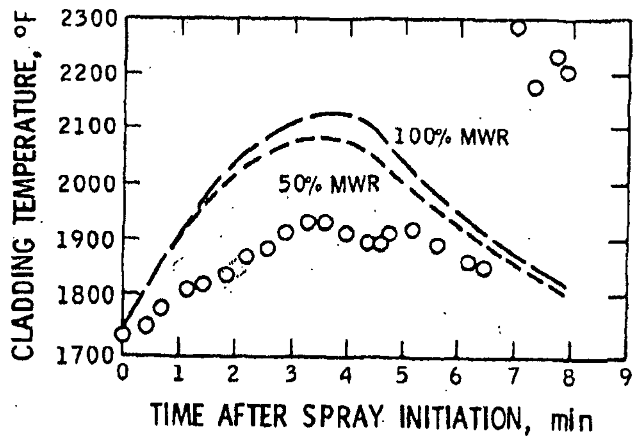
Appendix G Figure A8.9 Comparison of Predicted and Measured Thermal Histories for Zr2K Rods with TC Anomalies[7] and Figure A8.10 Analysis of Zr2K Thermal Response[8]

[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

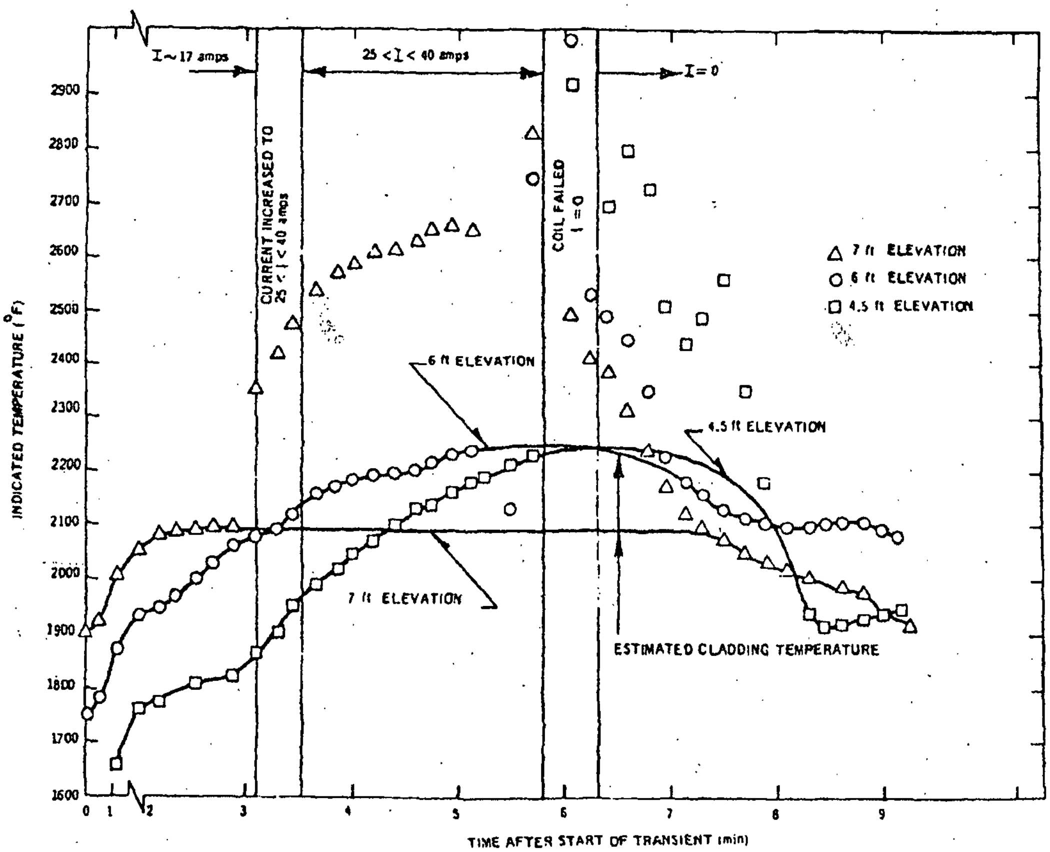


Bundle Zr2K Rod 31 Midplane Thermal Response Prediction

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

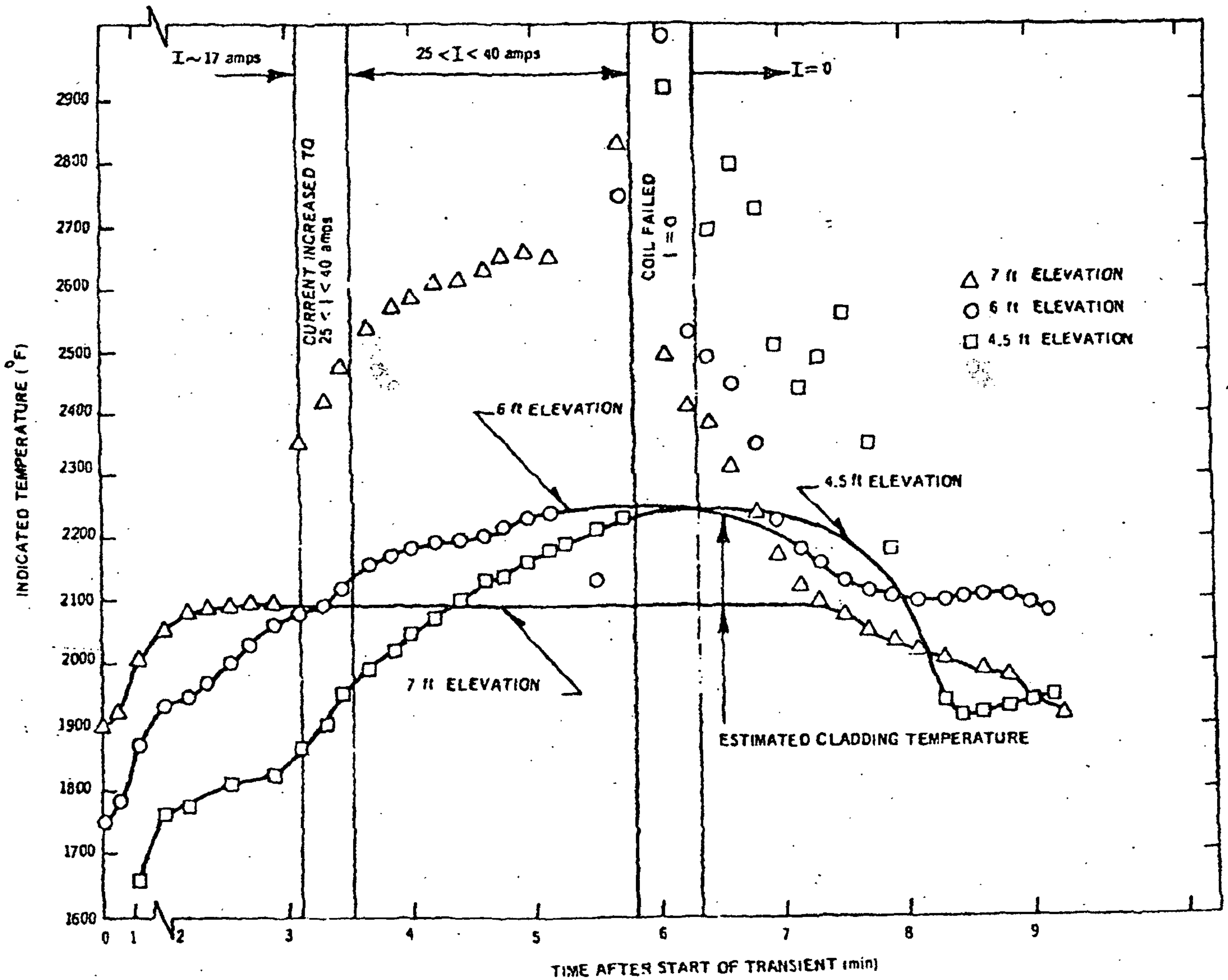
Analysis of Zr2K Thermal Response



(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.

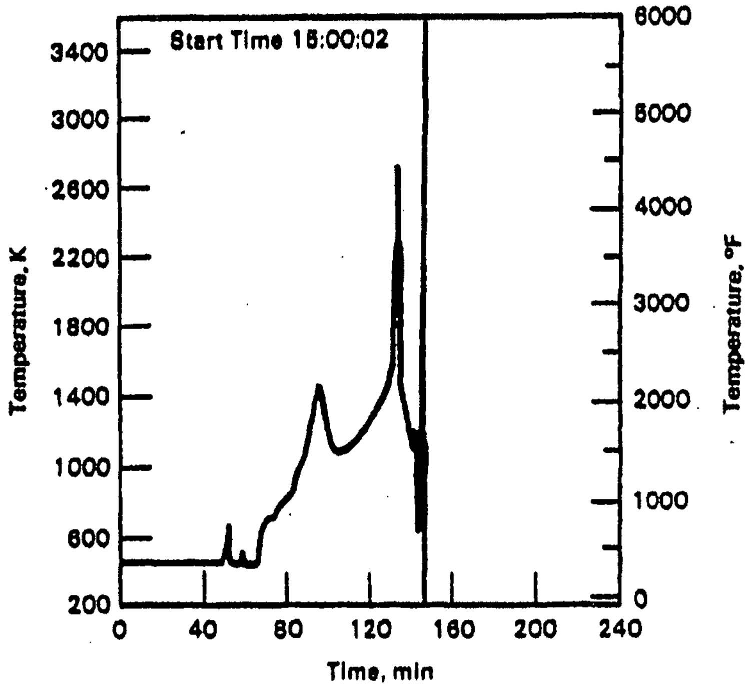


FIGURE 4.1. Typical Cladding Temperature Behavior

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.

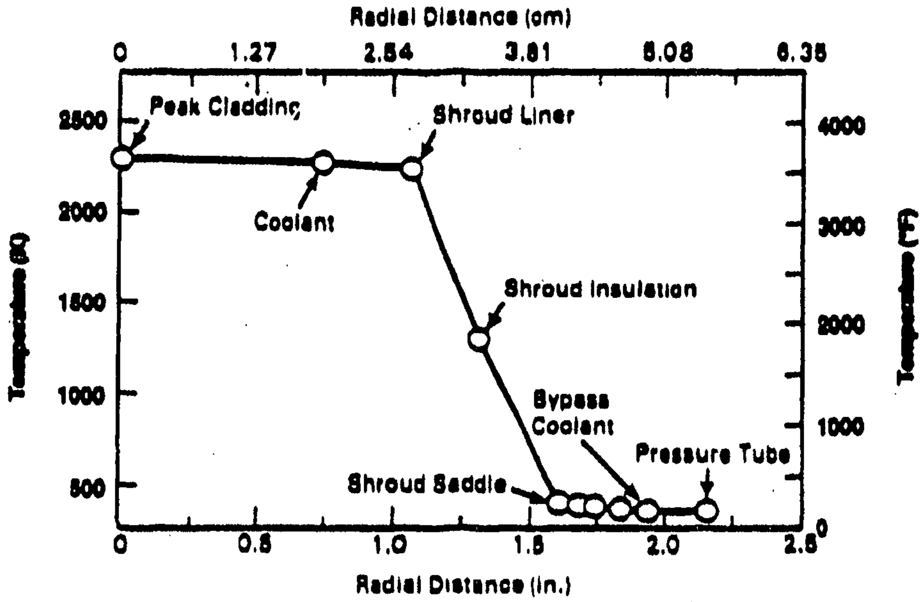


FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

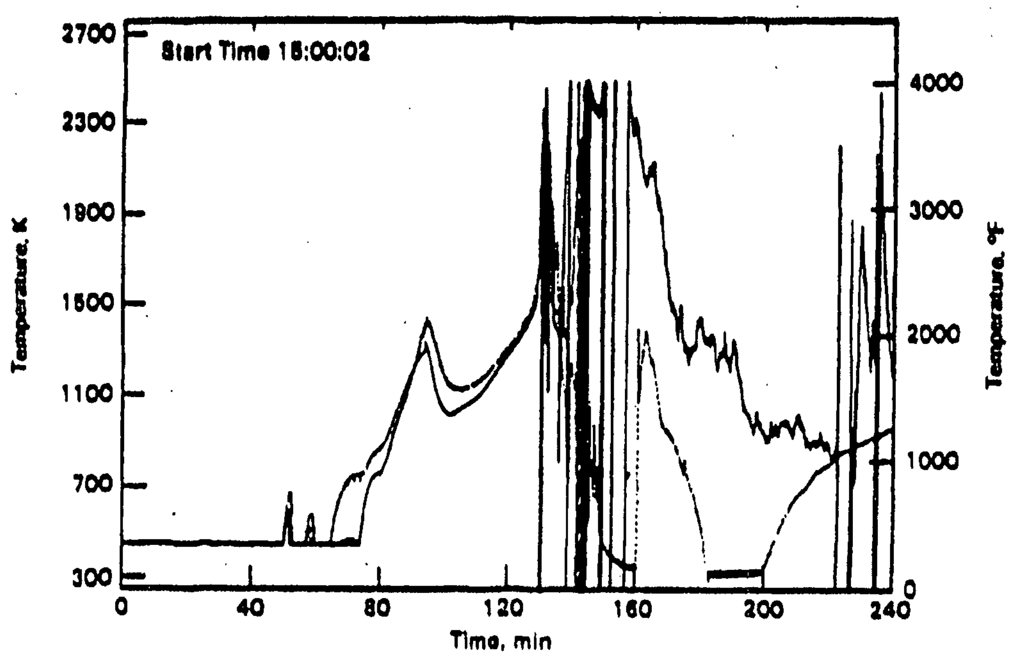


FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

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.

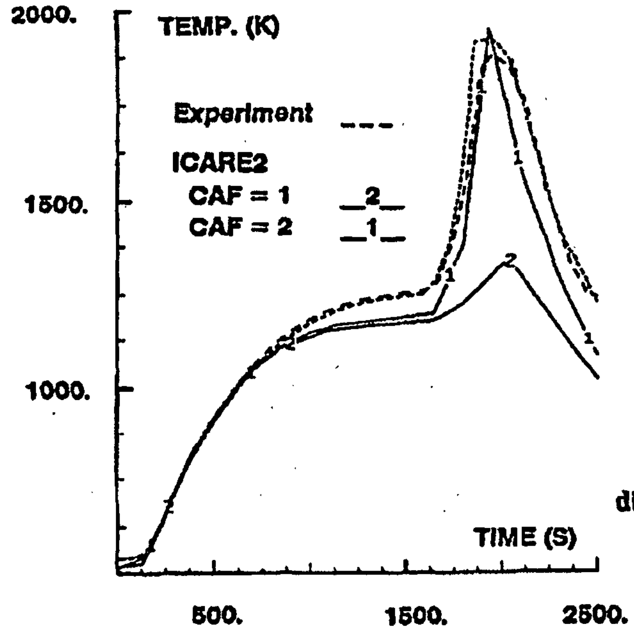


Fig.1 :
Sensitivity calculation on the B9R test. Temperature escalation at the hot level (0.6 m) with different Contact Area Factors (CAF)

3.2.2 PHEBUS C3 + test

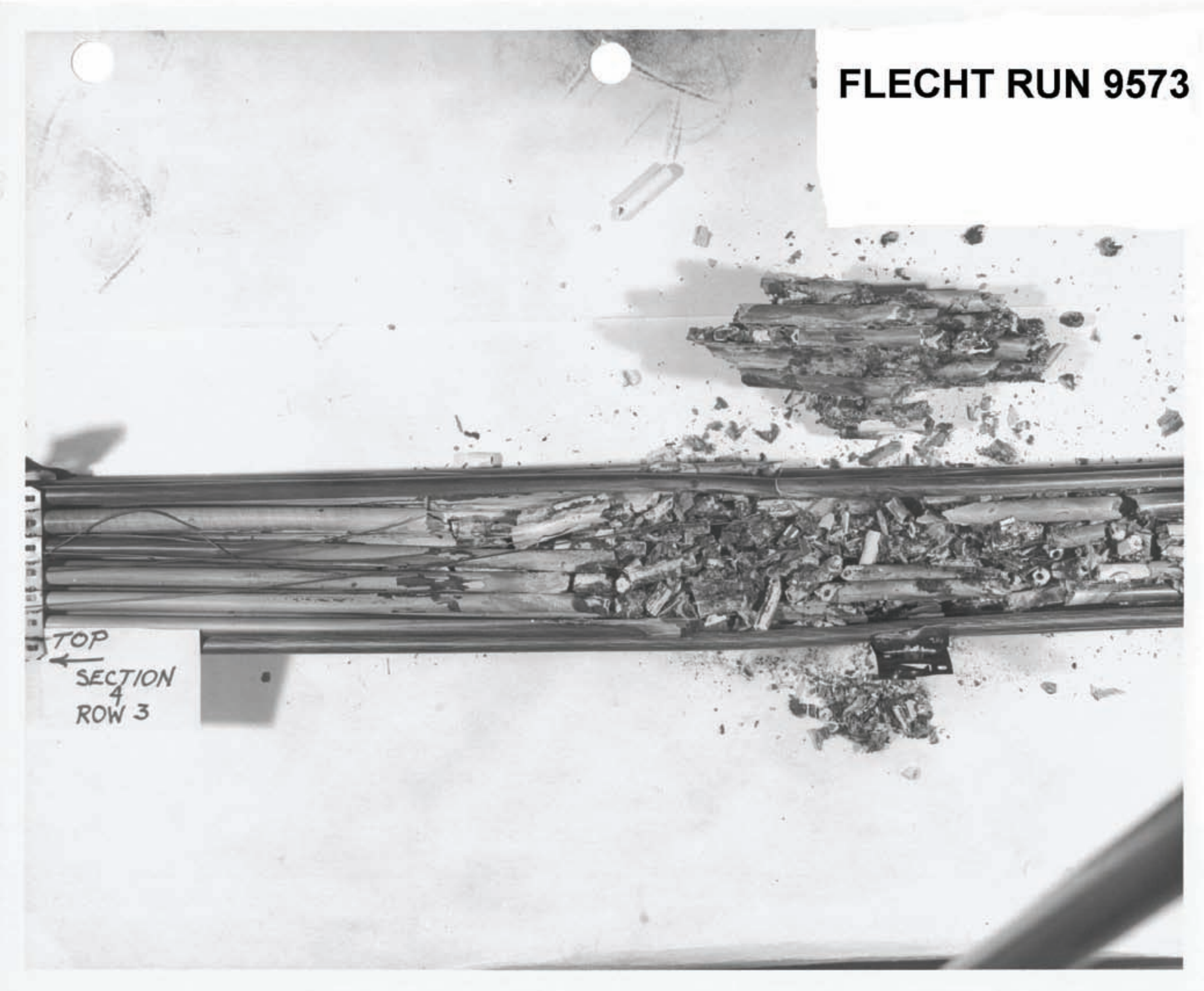
The main objective of this test was to study UO_2 dissolution by chemical interaction with solid Zr in a first stage and with liquid Zr in a second stage in the case of limited cladding oxidation. The first low temperature oxidation phase was performed during 3000 s with pure steam at 0.6 MPa so as to 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.

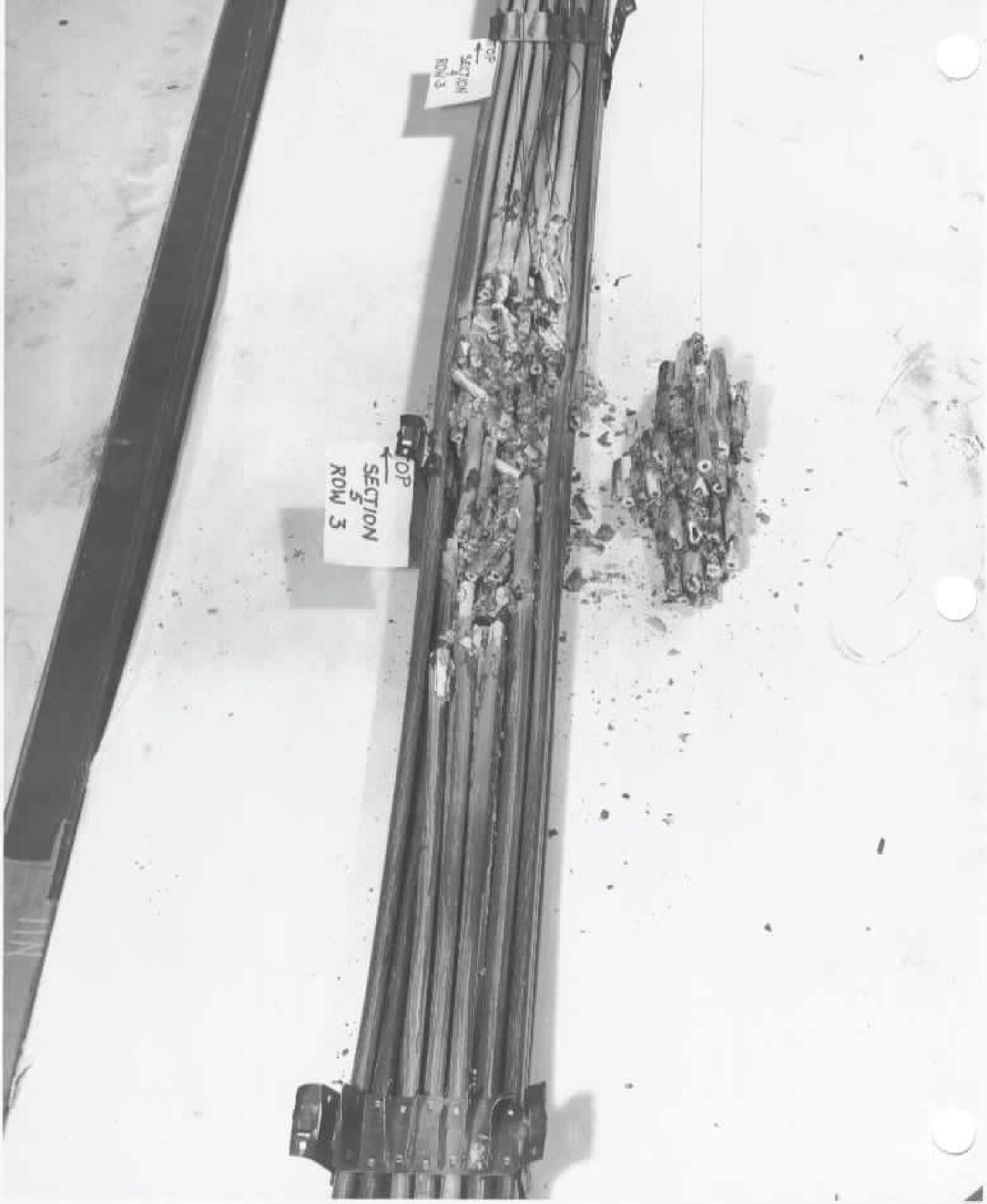
After adjusting the shroud heat losses in the first steam phase (see next section), the calculated and measured inner fuel rod temperatures at the 0.10, 0.40 and 0.60 m elevations agree well, until the thermocouple failures shown in Fig. 2 by arrows. Above 2200 K the calculation agrees with the fuel 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

Appendix J Photographs of the Bundle from FLECHT Run 9573

FLECHT RUN 9573





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Appendix K Edwin S. Lyman, PhD, *Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack at the Indian Point Nuclear Plant* (Union of Concerned Scientists, September 2004)

CHERNOBYL ON THE HUDSON?

THE HEALTH AND ECONOMIC IMPACTS OF A TERRORIST ATTACK AT THE INDIAN POINT NUCLEAR PLANT

Edwin S. Lyman, PhD
Union of Concerned Scientists
September 2004

Commissioned by Riverkeeper, Inc.

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TABLES

EXECUTIVE SUMMARY

Since 9/11, the specter of a terrorist attack at the Indian Point nuclear power plant, thirty-five miles upwind from midtown Manhattan, has caused great concern for residents of the New York metropolitan area. Although the Nuclear Regulatory Commission (NRC) ordered modest security upgrades at Indian Point and other nuclear power plants in response to the 9/11 attacks, the plants remain vulnerable, both to air attacks and to ground assaults by large terrorist teams with paramilitary training and advanced weaponry. Many question whether the NRC's security and emergency planning requirements at Indian Point are adequate, given its attractiveness as a terrorist target and the grave consequences for the region of a successful attack.

This report presents the results of an independent analysis of the health and economic impacts of a terrorist attack at Indian Point that results in a core meltdown and a large radiological release to the environment. We find that, depending on the weather conditions, an attack could result in as many as 44,000 near-term deaths from acute radiation syndrome or as many as 518,000 long-term deaths from cancer among individuals within fifty miles of the plant. These findings confirm that Indian Point poses a severe threat to the entire New York metropolitan area. The scope of emergency planning measures should be promptly expanded to provide some protection from the fallout from an attack at Indian Point to those New York area residents who currently have none. Security at Indian Point should also be upgraded to a level commensurate with the threat it poses to the region.

A 1982 study by Sandia National Laboratories found that a core meltdown and radiological release at one of the two operating Indian Point reactors could cause 50,000 near-term deaths from acute radiation syndrome and 14,000 long-term deaths from cancer. When these results were originally disclosed to the press, an NRC official tried to reassure the public by saying that the kind of accident the study considered would be less likely than "a jumbo jet crashing into a football stadium during the Superbowl."

In the post-9/11 era, the possibility of a jumbo jet crashing into the Superbowl --- or even a nuclear power plant --- no longer seems as remote as it did in 1982. Nonetheless, NRC continues to argue that the 1982 Sandia report is unrealistic because it focused on "worst-case" accidents involving the simultaneous failure of multiple safety systems, which are highly unlikely to occur by chance. But when the potential for terrorist attacks is considered, this argument no longer applies. "Worst-case" scenarios are precisely the ones that terrorists have in mind when planning attacks.

Both NRC and Entergy, the owner of Indian Point, assert that even for the most severe terrorist attack, current emergency plans will be adequate to protect residents who live in the evacuation zone within 10 miles of the plant. They also say that there will be no significant radiological impact on New York City or any other location outside of the 10-mile zone. Accordingly, NRC has opposed proposals made after 9/11 to extend the emergency planning zone around Indian Point. However, NRC and Entergy have not

provided the public with any documentation of the assumptions and calculations underlying these claims.

In view of the lack of public information available on these controversial issues, we carried out an independent technical analysis to help inform the debate. Our calculations were performed with the same state-of-the-art computer code that NRC uses to assess accident consequences. We used the NRC's guidance on the radiological release from a core meltdown, current estimates of radiation risk, population data from the 2000 census, and the most recent evacuation time estimate for the 10-mile Indian Point emergency planning zone. Following the format of the 1982 Sandia report, we calculated the numbers of near-term deaths from acute radiation syndrome, the numbers of long-term deaths from cancer, and the maximum distance at which near-term deaths can occur. We evaluated the impact of both evacuation and sheltering on these outcomes. We also estimated the economic damages due to the long-term relocation of individuals from contaminated areas, and the cost of cleanup or condemnation of those areas.

The health and environmental impacts of a large radiological release at Indian Point depend strongly on the weather conditions. We have carried out calculations for over 140,000 combinations of weather conditions for the New York area and wind directions for the Indian Point site, based on a year's worth of weather data. For this data set, we have determined the average consequences, the peak consequences, and the consequences for '95th percentile" weather conditions (in other words, only 5% of the weather sequences analyzed resulted in greater consequences).

We believe that the 95th percentile results, rather than the average values, represent a reasonable assessment of the likely outcome of a successful terrorist attack, since such attacks would most likely not occur at random, but would be timed to coincide with weather conditions that favor greater casualties. Attacks capable of causing the peak consequences that we calculate would be difficult to achieve because of inaccuracies in weather forecasts, restricted windows of opportunity and other factors, but remain within the realm of possibility.

For a successful attack at one of the two operating Indian Point reactors, we find that

- The number of near-term deaths within 50 miles, due to lethal radiation exposures received within 7 days after the attack, is approximately 3,500 for 95th percentile weather conditions, and approximately 44,000 for the worst case evaluated. Although we assumed that the 10-mile emergency planning zone was entirely evacuated in these cases, this effort was inadequate because (according to Entergy's own estimate) it would take nearly 9.5 hours to fully evacuate the 10-mile zone, whereas in our model the first radiological release occurs about two hours after the attack.

- Near-term deaths can occur among individuals living as far as 18 miles from Indian Point for the 95th percentile case, and as far as 60 miles away in the worst case evaluated. Timely sheltering could be effective in reducing the number of

near-term deaths among people residing outside of the 10-mile emergency planning zone, but currently no formal emergency plan is required for these individuals.

- The number of long-term cancer deaths within 50 miles, due to non-acutely lethal radiation exposures within 7 days after the attack, is almost 100,000 for 95th percentile weather conditions and more than 500,000 for the worst weather case evaluated. The peak value corresponds to an attack timed to coincide with weather conditions that maximize radioactive fallout over New York City.

- Based on the 95th percentile case, Food and Drug Administration guidance would recommend that many New York City residents under 40, and children in particular, take potassium iodide (KI) to block absorption for radioactive iodine in the thyroid. However, there is no requirement that KI be stockpiled for use in New York City.

- The economic damages within 100 miles would exceed $1.1 trillion for the 95th percentile case, and could be as great as $2.1 trillion for the worst case evaluated, based on Environmental Protection Agency guidance for population relocation and cleanup. Millions of people would require permanent relocation.

We hope that this information will be useful to Federal, State and local homeland security officials as they continue to develop plans to protect all those at risk from terrorist attacks in the post-9/11 world.

INTRODUCTION

(a) The terrorist threat to nuclear power plants

Public concern about the vulnerability of nuclear power plants to catastrophic acts of sabotage soared in the aftermath of the September 11 terrorist attacks. There is ample justification for this concern.

Soon after the 9/11 attacks, the Nuclear Regulatory Commission conceded that U.S. nuclear power plants were not designed to withstand the high-speed impact of a fully fueled, modern passenger jet. The report of the 9/11 Commission has revealed that al Qaeda considered attacks on nuclear plants as part of their original plan, but declined to do so primarily because of their mistaken belief that the airspace around nuclear power plants in the U.S. was "restricted," and that planes that violated this airspace would likely be shot down before impact.[1]

But al Qaeda is surely now aware that no such restrictions were in place on 9/11. And it is clear from press reports that even today, no-fly zones around nuclear plants are imposed only at times of elevated threat level, and are limited in scope to minimize their economic impact on the aviation industry. This policy reflects a confidence in the ability of the intelligence community to provide timely advance warning of a surprise attack that --- given the 9/11 example --- is not entirely warranted. Moreover, even when no-fly zones are in place around nuclear plants, they are not likely to be effectively enforced. For instance, the U.S. government does not require that surface-to-air anti-aircraft protection be provided at nuclear plants, although such defenses have been routinely employed in Washington, D.C. since the 9/11 attacks.

In addition to the aircraft threat, many have begun to question the adequacy of physical security at nuclear plants to protect against ground-based, paramilitary assaults, in view of revelations that thousands of individuals received sophisticated training in military tactics at al Qaeda camps in Afghanistan. Press reports have documented many security failures at nuclear plants around the country, and have called attention to the troubling statistic that during a series of security performance tests in the 1990s, guard forces at nearly 50% of US plants failed to prevent mock terrorist teams from simulating damage that would have caused meltdowns had they been real attacks. This information, which was widely available but largely ignored before 9/11, suddenly became far more alarming in the new threat environment.

Today, the danger of a terrorist attack at a nuclear power plant in the United States --- either from the air or from the ground --- is apparently as great as ever. According to a January 14, 2004 speech by Robert L. Hutchings, Chairman of the National Intelligence Council (NIC),[2]

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[1] *The 9/11 Commission Report, Authorized Edition*, W.W. Norton, New York, 2004, p. 245.
[2] Robert L. Hutchings, "Terrorism and Economic Security," speech to the International Security Management Organization, Scottsdale, AZ, January 14, 2004.

> "targets such as nuclear power plants … are high on al Qa'ida's targeting list as a way to sow panic and hurt our economy … The group has continued to hone its use of transportation assets as weapons … although we have disrupted several airline plots, we have not eliminated the threat to airplanes. There are still al Qa'ida operatives who we believe have been deployed to hijack planes and fly them into key targets … Al Qa'ida's intent is clear. Its capabilities are circumscribed but still substantial. And our vulnerabilities are still great."

More recently, the 9/11 Commission concluded that "major vulnerabilities still exist in cargo and general aviation security. These, together with inadequate screening and access controls, continue to present aviation security challenges."[3]

(b) The Nuclear Regulatory Commission: an agency in denial

Since 9/11, members of the public, non-profit groups and lawmakers across the United States have been calling for major security upgrades at nuclear power plants, including consideration of measures such as military protection against ground assault and anti-aircraft defenses against jet attack. Yet the response of the Nuclear Regulatory Commission (NRC), the agency that regulates both the safety and security of US nuclear reactors, has not been commensurate with the magnitude of the threat.[4] And the Department of Homeland Security, the agency charged with coordinating the defense of the entire US critical infrastructure against terrorist attacks, appears to be merely following NRC's lead.[5]

Notwithstanding a steady stream of FBI warnings citing nuclear power plants as potential terrorist targets, NRC continues to maintain that there is no need to consider measures that could reduce the vulnerability of nuclear plants to air attack. NRC's position is that "the best approach to dealing with threats from aircraft is through strengthening airport and airline security measures."[6]

As it became clear that NRC was not going to require the nuclear industry to protect nuclear plants from attacks on the scale of September 11, some groups began calling for plants to be shut permanently. Because many of the most dangerous fission products in a nuclear reactor core decay rapidly after shutdown, the health consequences of a terrorist attack on a shutdown nuclear reactor would be significantly lower than those of an attack on an operating reactor.[7]

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[3] *9/11 Commission Report* (2004), op cit., p. 391.
[4] D. Hirsch, D. Lochbaum and E. Lyman, "NRC's Dirty Little Secret," *Bulletin of the Atomic Scientists*, May/June 2003.
[5] E. Lyman, "Nuclear Plant Protection and the Homeland Security Mandate," Proceedings of the 44th Annual Meeting of the Institute of Nuclear Materials Management, Phoenix, Arizona, July 2003.
[6] US Nuclear Regulatory Commission, "Frequently Asked Questions About NRC's Response to the 9/11/01 Events," revised March 15, 2004. On the NRC web site: http://www.nrc.gov/what-we-do/safeguards/911/faq.html#3.
[7] Calculations by the author, using the computer code MACCS2, indicate that for an attack occurring at twenty days after reactor shutdown and resulting in core melt and loss of containment, the number of early fatalities from acute radiation sickness would be reduced by 80% and the number of latent cancer fatalities

Public concern has been greatest for those plants seen as prime terrorist targets because of their symbolic importance or location near large population and commercial centers, such as the Indian Point nuclear power plant in Westchester County, New York, whose two operating reactors are situated only 24 miles from the New York City limits, 35 miles from midtown Manhattan and in close proximity to the reservoir system that supplies drinking water to nine million people. The post-9/11 movement to shut down Indian Point has attracted a level of support from the public and elected officials not seen since the early 1980s, including calls for shutdown by over 400 elected officials and over 50 municipalities.

In response to this challenge, NRC, Entergy (the owner of Indian Point), other nuclear utilities, and their trade group in Washington, the Nuclear Energy Institute (NEI), have undertaken a massive public relations campaign to assuage public fears about the risk of terrorism at Indian Point. First, they assert that a combination of robust nuclear plant design, physical security and redundant safety measures would be able to stop any terrorist attack from causing significant damage to the reactor core. Second, they argue that even if terrorists were to successfully attack Indian Point and cause a large radiological release, the public health consequences could be successfully mitigated by execution of the emergency plans already in place for residents within the 10-mile-radius "emergency planning zone" (EPZ). And third, they claim that outside of the 10-mile EPZ, exposures would be so low that no special precautions would be necessary to adequately protect the public from radiation, other than possible interdiction of contaminated produce and water.[8]

A typical example of the third argument can be found in a recent letter the NRC sent to Alex Matthiessen, Executive Director of Riverkeeper:[9]

> "Outside of 10 miles, direct exposure is expected to be sufficiently low that evacuation or sheltering would not be necessary. Exposure to a radioactive plume would not likely result in immediate or serious long-term health effects. Consideration of public sheltering and evacuation in emergency plans is very conservative and recommended at very low dose levels, well below the levels where health effects would be expected to occur."

resulting from lower exposures would be reduced by 50%, compared to an attack when the reactor is operating at full power. This calculation does not consider an attack on the storage pools for the highly radioactive spent fuel, which could result in significant long-term radiological contamination over a wide area and enormous economic consequences. For an extensive discussion of this threat, as well as an analysis of approaches for mitigating it, see R. Alvarez et al., "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States," *Science and Global Security* **11** (2003) 1-51.

[8] The NRC defines two "emergency planning zones," or EPZs. The 10-mile "plume exposure" EPZ is the region where evacuation or other actions could be ordered to protect the public from coming into contact with an atmospheric release of radioactivity. The 50-mile "ingestion" EPZ is the region where interdiction of agricultural products and water supplies could be ordered to prevent the consumption of contaminated produce. No evacuation planning is required for individuals residing within the ingestion EPZ but outside of the plume exposure EPZ.

[9] Letter from Cornelius F. Holden, Jr., Office of Nuclear Reactor Regulation, US NRC, to Alex Matthiessen, Riverkeeper, September 30, 2003.

The purpose of this report is to address these three claims, with an emphasis on the second and third, by conducting a quantitative assessment of the potential consequences of a terrorist-induced radiological release at Indian Point for individuals both within and without the 10-mile EPZ, including residents of New York City.

There is a considerable need today for an independent study of these questions. At a time when the importance of rigorous emergency planning for catastrophic terrorist attacks is obvious, it is essential that responsible officials be fully apprised of the facts, especially if they contradict long-held assumptions and biases. The lives of many people could be put at jeopardy if emergency plans are not designed with the most accurate information at hand.

This means, in particular, that the emergency planning process should be designed to account for the full spectrum of potential consequences, including so-called "fast-breaking" release scenarios in which radioactive releases to the environment would begin within about thirty minutes after an attack. This was one of the major conclusions of the report carried out for the government of New York State by James Lee Witt Associates.[10] Certain terrorist attack scenarios could be capable of causing such rapid releases.

But NRC and the Federal Emergency Management Agency (FEMA) continue to be reluctant to require testing of fast-breaking radiological releases in emergency planning exercises, asserting that such events are highly unlikely to occur.[11] However, this argument is no longer relevant in an age when terrorists have acquired unprecedented levels of technical expertise, and are actively targeting critical infrastructure facilities with the intent to maximize casualties and economic damages. If current emergency plans cannot successfully cope with all credible terrorist-induced events, they should be upgraded. If upgrading to a sufficiently protective level is so cumbersome as to be practically impossible, then other options, including plant shutdown, should not be ruled out.

Members of the public deserve to be fully informed of the potential consequences for their health and property of a successful terrorist attack at Indian Point, so that they can prepare for an attack in accordance with their own judgment and willingness to accept risk. This principle is consistent with the guidance of the Department of Homeland Security, whose Web site www.ready.gov advises that "all Americans should begin a process of learning about potential threats so we are better prepared to react during an attack." Sources of technical information other than NRC and the nuclear industry are

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[10] James Lee Witt Associates, *Review of Emergency Preparedness of Areas Adjacent to Indian Point and Millstone*, March 2003, Executive Summary, pg. x.

[11] Although it was anticipated that the widely publicized June 8, 2004 emergency planning exercise at Indian Point would involve a "fast-breaking" release, NRC in fact chose a scenario in which no release at all occurred. It was assumed that terrorists attacked the plant with a jet aircraft but missed the reactor and only managed to crash into the switchyard, causing a loss of off-site power but not enough damage to result in a radiological release. Thus the exercise provided no information as to the effectiveness of the Indian Point emergency plan in protecting residents of the EPZ from injury had the plane actually hit its target and initiated the damage scenario that is assessed in this report.

also essential to facilitate a factually accurate and honest discussion of the risks and benefits of continued operation of Indian Point in the post-9/11 era.

Some observers may criticize the public release of this report as irresponsible because they believe it (1) could assist terrorists in planning attacks, or (2) could interfere with the successful execution of emergency plans by unnecessarily frightening members of the public who the authorities claim are not at risk.

We are acutely aware of such concerns and, after careful consideration, have concluded that they do not have merit. We have reviewed this report carefully and omitted any information specific enough to be useful to terrorists seeking to attack Indian Point. Unfortunately, far more detailed information about nuclear plant design, operation and vulnerabilities than this report contains has already been --- and continues to be --- widely disseminated. For example, a paper written by staff of the Oak Ridge National Laboratory (ORNL) and the Defense Threat Reduction Agency (DTRA), published in 2004 in a technical journal and available on the Internet, contains a diagram of a generic nuclear power plant indicating where truck bombs of various sizes could be detonated in order to stage an attack with a 100% probability of core damage.

There can be little doubt that al Qaeda and other terrorist organizations are already well aware of the severity of the consequences that could result from an attack at Indian Point. It is NRC and FEMA that seem not to appreciate this risk, and it is to them above all that we direct this study. We also believe that there is a considerable cost, but no apparent benefit, to withholding information that could help people to protect themselves in the event of a terrorist attack at Indian Point. Better information will enable better coordination of all populations at risk and help to avoid situations where some individuals take inappropriate actions that endanger others.

This report would not have been necessary had we seen any indication that NRC and other government authorities fully appreciate the seriousness of the risk to the public from radiological sabotage, or if certain members of the Nuclear Regulatory Commission had not made statements regarding severe accident consequences and risks that contradicted the results of quantitative analyses developed and refined over several decades by NRC's own technical staff and contractors.

For instance, at a recent briefing on NRC's emergency preparedness program, NRC Commissioner Edward McGaffigan, comparing the radiological exposure from a reactor accident to air travel, radon and other sources of exposure to natural radioactivity, said that[12]

> "..the order of magnitude of the release is similar to all of these other things in people's lives and they should not panic over a few hundred millirem or even a couple of rem …but it's this radiation phobia, absolutely inflamed by these anti -

[12] US NRC, *Briefing on Emergency Preparedness Program Status*, Public Meeting, September 24, 2003, transcript, p. 73.

nuclear groups putting out their misinformation that actually hurts emergency planning …"

Commissioner McGaffigan's statement is misleading on at least three counts:

(1) Current emergency planning guidance is already based on the principle that exposures of "a couple of rem" would be acceptable following a large radiological release;

(2) The potential doses from a large radiological release can greatly exceed "a few hundred millirem or even a couple of rem" far downwind of the release site, and for many individuals could result in a significant increase in their lifetime risk of cancer (10% or greater) or even pose a risk of severe injury or death from acute radiation exposure;

(3) Even if the average dose resulting from a large release were on the order of "a couple of rem," the total collective detriment (latent cancer fatalities and economic damages) could be very high if a large number of people in a densely populated area were so affected.

We believe that misinformation originating within NRC itself is the biggest obstacle to development of the robust radiological emergency planning strategies needed to cope with today's heightened threat. Statements like those cited above raise the concern that those responsible for regulating the nuclear industry and protecting it from terrorist attack are either in a chronic state of denial or actually believe the propaganda generated by the nuclear industry for public consumption. If this is indeed the case, then one cannot have confidence that emergency planning officials are basing their decisions on accurate and unbiased information. Since the departure of NRC Commissioner Greta Dicus a few years ago, the current Commission does not have any members with backgrounds in radiation protection and health issues. One wonders whether the NRC Commissioners truly understand and appreciate the full extent of the dangers posed by the facilities that they regulate.

(c) The CRAC2 Report

Given the lack of credible information from public officials on the potential consequences of a terrorist attack at Indian Point, concerned neighbors of the plant turned to one of the few sources on this subject in the public domain --- the so-called "CRAC2 Report," carried out by Sandia National Laboratories (SNL) under contract for NRC in 1981. This study, formally entitled "Technical Guidance for Siting Criteria Development," used a computer code developed by SNL known as CRAC2 ("Calculation of Reactor Accident Consequences") to analyze the consequences of severe nuclear plant accidents and to study their dependence on population density, meteorological conditions and other characteristics. The version of the CRAC2 Report that had been submitted to NRC for eventual public release only contained average values of consequence results,

but the "peak" values for worst -case weather conditions were obtained by Congressman Edward Markey in 1982 and provided to the Washington Post.[13]

At many reactor sites, the CRAC2 Report predicted that for unfavorable weather conditions, a severe nuclear reactor accident could cause tens of thousands of early fatalities as a result of severe radiation exposure, and comparable numbers of latent cancer fatalities from smaller exposures. For Indian Point 3 (which at the time operated at a significantly lower power than it now does), CRAC2 predicted peak values of 50,000 early fatalities and 14,000 latent cancer fatalities, with early fatalities occurring as far as 17.5 miles downwind of the site.

The CRAC2 Report only considered accidents affecting operating nuclear reactors, and did not evaluate the consequences of accidents also involving spent fuel storage pools. Spent fuel pool loss-of-coolant accidents could themselves result in large numbers of latent cancer fatalities, widespread radiological contamination and huge cleanup bills, even if only a fraction of the fuel in the pool were damaged.

The release of the CRAC2 figures caused a great deal of consternation, but NRC was able to defuse the controversy by claiming that the peak results corresponded to accidents with extremely low probabilities (said to be one in a billion), and hence were not a cause for concern. In fact, Robert Bernero, director of the NRC's risk analysis division at the time, said (in a moment of unfortunate prescience) that such severe accidents would be less likely than "a jumbo jet crashing into a football stadium during the Super bowl."[14]

When Riverkeeper and other groups dusted off and called attention to the CRAC2 Report following the September 11 attacks, the NRC appeared unable to appreciate the new relevance of the study in a world where the possibility of a jumbo jet crashing into the Superbowl was no longer so remote. For example, in rejecting a 2001 petition filed by Riverkeeper to shut down the Indian Point plant until Entergy implemented a number of prudent security-related measures, the NRC merely repeated its old probability-based arguments, saying that[15]

> "...the reactor siting studies in the CRAC2 Report …used generic postulated releases of radioactivity from a spectrum of severe (core melt) accidents, independent of the probabilities of the event occurring or the impact of the mitigation mechanisms. The studies were never intended to be realistic assessments of accident consequences. The estimated deaths and injuries resulted from assuming the most adverse condition for each parameter in the analytical code. In the cited studies, the number of resulting deaths and injuries also reflected the assumption that no protective actions were taken for the first 24

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[13] Subcommittee on Oversight & Investigations, Committee on Interior and Insular Affairs, U.S. House of Representatives, "Calculation of Reactor Accident Consequences (CRAC2) For U.S. Nuclear Power Plants Conditional on an 'SST1' Release," November 1, 1982.
[14] Robert J. McCloskey, "The Odds of the Worst Case," *Washington Post*, November 17, 1982.
[15] US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Notice of Director's Decision Under 10 CFR 2.206, November 18, 2002.

> hours. The studies did not, and were never intended to, reflect reality or serve as a basis for emergency planning. The CRAC2 Report analyses used more simplistic models than current technologies."

Earlier in 2002, in a letter to the New York City Council, the NRC also said that[16]

> "The Sandia study does not factor in the numerous probabilistic risk studies that have been performed since 1982. More realistic, current inputs, assumptions, and modeling techniques would be expected to result in much smaller health consequences."

In a more recent "point paper" on homeland protection and preparedness, NRC continued to repeat these themes, although its conclusions were somewhat more equivocal: [17]

> "The Sandia Siting Study ["CRAC2"] …was performed to develop technical guidance to support the formulation of new regulations for siting nuclear power reactors. A very large radiation release and delayed evacuation, among other factors, accounts for the more severe consequences …As an overall conclusion, that report does not present an up-to-date picture of risk at nuclear plants and does not reflect current knowledge in probabilistic or phenomenological modeling.
>
> "Since September 11, 2001, the NRC has been performing assessments of the consequences of a terrorist attack on a nuclear power plant. These assessments are much more detailed than past analyses and reflect our improved understanding of severe accident phenomena. The more recent analyses have involved a more realistic assessment of the radiation release, emergency planning capabilities, radiation spreading, and health effects. More recent analysis indicates a general finding that public health effects from terrorist attacks at most sites are likely to be relatively small."

Although NRC continues to harshly criticize the CRAC2 Report and anyone who cites its results, it has not publicly identified the "more realistic, current inputs, assumptions and modeling techniques that would be expected to result in much smaller health consequences," much less demonstrated the validity of these results by providing the public with its calculations for independent review. In fact, NRC now considers that these analyses are too sensitive for public release, making it impossible for the public to verify its claims.

NRC's unwillingness to share this kind of information with the public is not unexpected. NRC (like its predecessor, the Atomic Energy Commission) has worked over its history to shield the public from estimates of the consequences of severe accidents without simultaneous consideration of the low probabilities of such accidents. By multiplying

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[16] Hubert Miller, Region I Administrator, US NRC, letter to Donna De Constanzo, Legislative Attorney, New York City Council, July 24, 2002.

[17] US Nuclear Regulatory Commission, "Point Paper on Current Homeland Protection and Preparedness Issues," November 2003, on the NRC Web site, www.nrc.gov.

high consequence values with very low probability numbers, the consequence figures appear less startling to the layman but are obscured in meaning. For instance, a release that could cause 100,000 cancer fatalities would only appear to cause 1 cancer fatality per year if the associated probability of the release were 1/100,000 per year.

This issue was central to the so-called Indian Point Special Proceeding, a 1983 review conducted by a panel of NRC administrative judges that examined whether Indian Point posed unusually high risks because of its location in the densely populated New York metropolitan area. Before this proceeding, the NRC ruled that all testimony on accident consequences must also contain a discussion of accident probabilities. However, in its decision, the three-judge Atomic Safety and Licensing Board panel concluded that "the Commission should not ignore the potential consequences of severe-consequence accidents by always multiplying those consequences by low probability values."[18] One of the judges dissented from this majority opinion, insisting that singling out Indian Point "to the exclusion of many other sites similarly situated in effect raises again the question of considering consequences without their associated probabilities. This we have been restricted from doing by the Commission."[19] Today, it appears that this minority opinion ultimately prevailed at NRC.

The results of the CRAC2 Report are indeed of questionable applicability today. But the reasons for this are not the ones that NRC has identified, but include, for example, the fact that the CRAC2 Report

- used census data from 1970, at a time before rampant suburban sprawl greatly increased the population densities in formerly rural areas close to some nuclear reactor sites;

- assumed that the entire 10-mile emergency planning zone would be completely evacuated within at most six hours after issuance of a warning (contrary to NRC's assertion that the CRAC2 peak results reflect the assumption that "no protective actions were taken for the first 24 hours"), whereas the current evacuation time estimate for the Indian Point EPZ, based on updated assessments of likely road congestion, is nearly ten hours;

- assumed aggressive medical treatment for all victims of acute radiation exposure in developing estimates of the number of early fatalities, and employed a now-obsolete correlation between radiation dose and cancer risk that underestimated the risk by a factor of 4 relative to current models;

- sampled only 100 weather sequences out of 8760 (an entire year's worth), a method which we find underestimates the peak value occurring over the course of a year by 30%.

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[18] US Nuclear Regulatory Commission, Atomic Safety and Licensing Board, Indian Point Special Proceeding, Recommendations to the Commission, October 24, 1983, p. 107.
[19] Ibid, "Dissenting Views of Judge Gleason," p. 433.

In 1990, the CRAC2 code was retired in favor of a new code known as MACCS ("MELCOR Accident Consequence Code System"), which was updated to MACCS2 in 1997. The MACCS2 code, also developed by Sandia National Laboratories, is the state-of-the-art consequence code employed by both NRC and DOE in conducting dose assessments of radiological releases to the atmosphere. It includes numerous improvements over the CRAC2 code.[20]

However, the fundamental physics models that form the basis for both the CRAC2 and MACCS2 codes have not changed in the past two decades. Nor has evidence arisen since the CRAC2 Report was issued that would suggest that the CRAC2 "source term" --- that is, the fraction of the radioactive contents of the reactor core assumed to be released to the environment during a severe accident --- significantly overestimated potential releases. On the contrary, the Chernobyl disaster in 1986 demonstrated that such large releases were possible.[21] The state-of-the art revised source term developed by NRC, as defined in the NRC report NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," is little different from the source terms used in the CRAC2 Report.[22] Recent experimental work, including the Phébus tests in France, have provided further confirmation of the NUREG-1465 source term.[23] Other tests, such as the VERCORS experiments in France, have found that NUREG-1465 actually underestimates the releases of some significant radionuclides.

The NRC continues to stress the absence of consideration of accident probabilities in dismissing the results of the CRAC2 Report. However, this criticism is invalid in the post-9/11 era. Accident probabilities are not relevant for scenarios that are intentionally caused by sabotage. Severe releases resulting from the simultaneous failure of multiple safety systems, while very unlikely if left up to chance, are precisely the outcomes sought by terrorists seeking to maximize the impact of their attack. Thus the most unlikely accident sequences may well be the most likely sabotage sequences.

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[20] D.I. Chanin and M.L. Young, *Code Manual for MACCS2: Volume 1, User's Guide*, SAND97-0594, Sandia National Laboratories, March 1997.

[21] The nuclear industry often argues that a Chernobyl-type accident could not happen in the United States because the reactor was of a different and inferior type to US plants and lacked a robust containment structure. While it is true that the specific accident sequence that led to the destruction of the Chernobyl-4 reactor and the resulting radiological release was characteristic of graphite-moderated reactors like Chernobyl and would not likely occur at a US light-water reactor (LWR), it is simply false to claim that there are no possible accident sequences that could result in consequences similar to those of Chernobyl --- namely, core melt, loss or bypass of containment, and large radiological release to the environment. In fact, because such an event is not as likely to be as energetic as the Chernobyl explosion, and the plume is not likely to be as hot as the Chernobyl plume (which was fed by the burning of a large mass of graphite), the radiological release from a severe accident at a US LWR will not rise as high or disperse as far. Therefore, radiological exposure to the public near a US LWR could be far greater than was the case at Chernobyl, because the plume would be more concentrated closer to the plant.

[22] L. Soffer, et al., *Accident Source Terms for Light-Water Nuclear Power Plants, Final Report,* NUREG-1465, US NRC, February 1995.

[23] US NRC, Memorandum from Ashok Thadani to Samuel J. Collins, "Use of Results from Phébus-FP Tests to Validate Severe Accident Codes and the NRC's Revised Accident Source Term (NUREG-1465)," Research Information Letter RIL-0004, August 21, 2000.

Other aspects that add an element of randomness to accident scenarios, such as meteorological conditions, can also be controlled through the advance planning and timing of a terrorist attack. Therefore, even if NRC were correct in claiming that the CRAC2 Report assumes the "most adverse condition" for each accident -related parameter, such an approach would still be appropriate for analyzing the potential maximum consequences of a sophisticated terrorist attack.

<u>We have not been able to identify any issues that would suggest the consequence estimates provided in the CRAC2 Report were significantly overstated.</u> But in light of the problems with the CRAC2 Report discussed earlier, we have conducted our own analysis of the consequences of a sophisticated terrorist attack at the Indian Point plant, using the MACCS2 code and the most up-to-date information available. This included the NUREG-1465 revised source term, the most current dose conversion and cancer risk coefficients recommended by the International Commission on Radiological Protection (ICRP), and the most recent evacuation time estimate (ETE) for Indian Point developed by consultants for Entergy Nuclear, the plant operator. We used the SECPOP2000 code, developed for NRC by Sandia National Laboratories, to generate a high-resolution MACCS2 site data file that includes a regional population distribution based on 2000 Census data and an economic data distribution based on 1997 government statistics.

<u>For Indian Point, we find that the MACCS2 results for peak early fatalities are generally consistent with the CRAC2 Report, but that the CRAC2 Report significantly underestimates the peak number of latent cancer fatalities that could occur.</u>

Moreover, the consequence estimates in this report are based on a number of optimistic assumptions, or "conservatisms," that tend to underes timate the true consequences of a terrorist attack at Indian Point. For example:

1. We use an evacuation time estimate that assumes the attack takes place in the summer in good weather, and does not take into account the possibility that terrorists may time their attack when evacuation is more difficult or actively interfere with the evacuation.

2. We only consider the permanent resident population of the 10-mile plume exposure EPZ, and not the daily transient population, which would increase the total population of the EPZ by about 25%.

3. We use values for the rated power of the Indian Point reactors from 2002 that are about 5% lower than the current values.

4. The only health consequences we consider are early fatalities from acute radiation syndrome and latent fatalities from cancer. We do not assess the excess mortality associated with the occurrence of other well-documented health effects of radiation such as cardiovascular disease. We also do not consider non-fatal effects of radiation, such as the reduction in intelligence quotient (IQ) of children irradiated in utero or other birth defects.

5. The NUREG-1465 source term does not represent the maximum possible radiological release from a core melt. Also, the assumed delay time between the attack and the start of the radiological release is nearly two hours, which is not nearly as short as the minimum of 30 minutes that is contemplated in NRC's emergency planning regulations.

6. The calculations assume only that the reactors itself are attacked and that the large quantity of spent fuel in the wet storage pools remains undamaged.

In the following sections, we discuss some technical issues related to severe accident and sabotage phenomena. Then we describe the methodology, tools and input parameters used to carry out the calculation. Finally, we present our results and conclusions.

ACCIDENTS: DESIGN-BASIS, BEYOND-DESIGN-BASIS, AND DELIBERATE

The NRC has traditionally grouped nuclear reactor accidents into two main categories: "design-basis" accidents, and "beyond-design-basis" or "severe" accidents.

(a) Design-basis accidents

Design-basis accidents are accidents that nuclear plants must be able to withstand without experiencing unacceptable damage or resulting in radiological releases that exceed the regulatory limits known as "Part 100" releases (because of where they can be found in the NRC regulations).

One of the more challenging design-basis accidents for pressurized-water reactors (PWRs) like those at Indian Point is a loss-of-coolant accident (LOCA). In the "primary" system of a PWR, the reactor core, which is contained in a steel vessel, is directly cooled by the flow of high-pressure water forced through pipes. In a LOCA, a pipe break or other breach of the primary system results in a loss of the water essential for removing heat from the reactor fuel elements. Even if the nuclear reactor is immediately shut down or "scrammed," an enormous quantity of heat is still present in the fuel, and cooling water must be restored before a significant number of fuel elements reach temperatures above a critical limit. If heated beyond this limit, the fuel element cladding can become brittle and shatter upon contact with cooling water. Eventually, the core geometry can become "uncoolable" and the fuel pellets themselves will reach temperatures at which they start to melt.

In a design-basis LOCA, it is assumed that the emergency core cooling system (ECCS) works as designed to provide makeup coolant water to the nuclear fuel, terminating the event before it becomes impossible to control. Even in this case, however, a significant fraction of the radioactive inventory in the core could be released into the coolant and transported out of the primary system through the pipe break. The primary system therefore must be enclosed in a leak-tight containment building to ensure that Part 100 limits are not exceeded in the event of a design-basis LOCA. To demonstrate compliance with Part 100, dose calculations at the site boundary are carried out by specifying a so-called "source term" --- the radioactive contents of the gases within the containment following the LOCA --- and assuming that the containment building leaks at its maximum design leak rate, typically about 0.1% per day. Such an event was historically considered a "maximum credible accident."

(b) Beyond-design-basis accidents

In contrast to design-basis accidents, "beyond-design-basis" accidents (also known as "severe" accidents) are those in which multiple failures occur, backup safety systems do not work as designed, the core experiences a total "meltdown" and radiological releases far greater than the Part 100 limits become possible. For example, if the ECCS does not work properly after a LOCA, the core will continue to overheat, eventually forming a

molten mass that will breach the bottom of the steel reactor vessel and drop onto the containment floor. It will then react violently with any water that is present and with concrete and other materials in the containment. At this point, there is little hope that the event can be terminated before much of the radioactive material within the fuel is released in the form of gases and aerosols into the containment building.

Even worse is the potential for mechanisms such as steam or hydrogen explosions to rupture the containment building, releasing its radioactive contents into the environment. Although not the only distinguishing feature, a major distinction between design-basis and severe accidents is whether containment integrity is maintained. Even a small rupture in the containment building --- no more than a foot in diameter --- would be sufficient to depressurize it and to vent the gases and aerosols it contains into the environment in less than half an hour.[24] This would result in a catastrophic release of radioactivity on the scale of Chernobyl, and Part 100 radiation exposure limits would be greatly exceeded.

The containment building can also be "bypassed" if there is a ru pture in one of the interfaces between the primary coolant system and other systems that are outside of containment, such as the "secondary" coolant system (the fluid that drives the turbine generators) or the low-pressure safety injection system. For instance, the rupture in the steam generator that occurred at Indian Point 2 in February 2000 created a pathway in which radioactive steam from the primary system was able to pass into the secondary system, which is not enclosed in a leak-tight boundary. If that event had coincided with significant fuel damage, the radiological release to the environment could have been far greater.

NRC has always had an uncomfortable relationship with beyond-design-basis accidents. By their very definition, they are accidents that were not considered in the original design basis for the plant. In fact, according to NRC, "the technical basis for containment design was intended to ensure very low leakage under postulated loss-of-coolant accidents. No explicit consideration was given to performance under severe accidents."[25] Indeed, NRC has never instituted a formal regulatory requirement that severe accidents be prevented. In 1985, the Commission ruled by fiat in its Severe Accident Policy Statement that "existing plants pose no undue risk to health and safety" and that no regulatory changes were required to reduce severe accident risk. NRC's basic assumption is that if a plant meets design basis requirements, then it will have sufficient resistance against severe accidents, and it has devoted considerable resources to the task of "confirmatory research" to justify this assumption. NRC believes that this approach provides "adequate protection" of public health and safety because the probability of a

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[24] US Nuclear Regulatory Commission, *Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects, Analysis of Containment Building Failure Modes, Preliminary Report*, NUREG-0850, Vol. 1, November 1981, p. 3-2.
[25] US Nuclear Regulatory Commission, *Reactor Risk Reference Document (Appendices J-0),* NUREG-1150, Draft for Comment, February 1987, p. J.10-1.

severe accident capable of rupturing or bypassing the containment prior to effective evacuation of the EPZ is so low in most cases as to be below regulatory concern.[26]

(c) "Deliberate accidents"

It is true that a spontaneous occurrence of the multiple system failures necessary to cause a severe accident and large radiological release is typically a very improbable event. However, if one considers the possibility of sabotage or "deliberate" accidents, the low - probability argument that NRC uses to justify the continued operation of nuclear plants completely breaks down. Terrorists with basic and readily available knowledge of how nuclear plants operate can design their attack to maximize the chance of achieving a core melt and large radiological release. With modest inside assistance, as contemplated by NRC in its regulations and practices, saboteurs would be able to identify a plant-specific set of components known as a "target set." If all elements of a target set are disabled or destroyed, significant core damage would result. Thus, by deliberately disrupting all redundant safety systems, saboteurs can cause a severe event that would have had only a very low probability of occurrence if left to chance.

The likelihood of a successful attack is enhanced for plants with "common-cause" failure modes. A common-cause failure is a single event that can lead to the failure of multiple redundant systems. For example, if the diesel fuel supplied to a nuclear plant with two independent emergency diesel generators from the same distributor is impure, then both generators may fail to start for the same reason if off-site power is lost and emergency power is needed. This would result in a station blackout, one of the most serious challenges to pressurized-water reactors like Indian Point. While some common-cause failure modes can be corrected, others are intrinsic to the design of currently operating nuclear plants. Common-cause failure modes make the saboteurs' job easier, as fewer targets would have to be disabled to achieve the desired goal.

In addition to causing a core meltdown, terrorists also have the means to ensure that the radioactive materials released from the melting fuel can escape into the environment by breaching, severely weakening or bypassing the containment.[27] Finally, saboteurs can maximize the harm caused by a radiological release by staging their attack when the meteorological conditions favor a significant dispersal over densely populated areas, and even interfering with the execution of emergency plans.

NRC has formally maintained for at least two decades that it does not make sense to assign probabilities to terrorist attacks. In a 2002 memorandum, NRC stated that[28]

> "the horrors of September 11 notwithstanding, it remains true that the likelihood of a terrorist attack being directed at a particular nuclear facility is not

[26] There have been situations where NRC concluded that "adequate protection" was not met at certain nuclear plants and required additional safety measures. However, such instances are rare.
[27] We have decided not to describe such means in greater detail, although we have little doubt that terrorists are already familiar with them.
[28] US NRC, Memorandum and Order, CLI-02-025, December 18, 2002, p. 17.

> quantifiable. Any attempt at quantification or even qualitative assessment would be highly speculative. In fact, the likelihood of attack cannot be ascertained with confidence by any state-of-the-art methodology …we have no way to calculate the probability portion of the [risk] equation, except in such general terms as to be nearly meaningless."

Yet at other times, NRC does not hesitate to invoke probabilities when arguing that the public has nothing to fear from terrorist attacks on nuclear plants. For example, here is what NRC has to say about the CRAC2 study in its recent "point paper" on homeland protection and preparedness:[29]

> "Over the years, the NRC has performed a number of consequence evaluations to address regulatory issues …We have considered the extent to which past analyses, often the subject of public statements by advocacy groups and the media, can be superceded [sic] by more recent analysis …Past studies usually have considered …a number of scenarios, which resulted in only minor consequences. The most limiting severe scenarios, which comprise a minority of the calculations and represent *very low probability events* [emphasis added], are the predictions typically cited in press accounts. These scenarios have assumed …very large radiation releases, bounding emergency response assumptions or bounding conditions (including weather) for the spread of the radiation. The combination of these factors produces large and highly unlikely results."

These two excerpts are inconsistent. If it is meaningless to quantify the likelihood of a terrorist attack, then one cannot dismiss the possibility of terrorist attacks causing the most severe consequences by claiming they are "highly unlikely." Therefore, in order to base emergency planning on the best possible information, NRC must accept the fact that the growing threat of domestic terrorism has forever altered the delicate risk calculus that underlies its approach to safety regulation. NRC can no longer shy away from confronting the worst-case consequences of terrorist attacks on nuclear power plants. And perhaps the most attractive target in the country, where the consequences are likely to be the greatest, is Indian Point.

[29] US NRC, "Point Paper on Current Homeland Protection and Preparedness Issues" (2003), op cit.

THE HEALTH CONSEQUENCES OF A RADIOLOGICAL RELEASE FROM INDIAN POINT

The Indian Point power plant is located on 239 acres on the Hudson River in the village of Buchanan in Westchester County, New York. There are two operating pressurized-water reactors (PWRs) on site, Indian Point 2, rated at 971 MWe, and Indian Point 3, rated at 984 MWe. Both reactors are operated by Entergy Nuclear.

Indian Point is located in one of the most densely populated metropolitan areas in the United States, situated about 24 miles from the New York City limits and 35 miles from midtown Manhattan. Extrapolating from 2000 Census data, in 2003 over 305,000 persons resided within the roughly ten-mile radius plume exposure emergency planning zone for Indian Point, and over 17 million lived within 50 miles of the site.[30]

The types of injury that may occur following a catastrophic release of radioactive material resulting from a terrorist attack at Indian Point fall into two broad categories. The first category, "early" injuries and fatalities, are those that are caused by short-term whole-body exposures to doses of radiation high enough to cause cell death. Early injuries include the constellation of symptoms known as **acute radiation syndrome** that should be familiar to anyone who has read *Hiroshima* by John Hersey --- gastrointestinal disturbance, epilation (hair loss) and bone marrow damage. Other early injuries include severe skin damage, cataracts and sterility. For sufficiently high doses, early fatalities --- death within days or weeks --- can occur. These so-called "deterministic" effects are induced only when levels of radiation exposure exceed certain thresholds.

Another class of injury caused by ionizing radiation exposure is genetic damage that is insufficient to cause cell death. At doses below the thresholds for deterministic effects, radiation may cause damage to DNA that interferes with the normal process of cell reproduction. This damage can eventually lead to cancer, which may not appear for years or even decades, depending on the type. Because a single radiation-induced DNA lesion is believed to be capable of progressing to cancer, there is no threshold for these so-called "stochastic" effects.[31]

The clinical response of individuals to ionizing radiation exposure is highly variable from person to person. Some individuals have a lower capability of DNA repair and thus are more susceptible to the carcinogenic effects of radiation --- a condition that is most severe in people with certain genetic diseases like ataxia telangiectasia. Children are particularly vulnerable to radiation exposure. For the same degree of exposure to a

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[30] A figure of 20 million people within 50 miles of Indian Point has often been quoted. This value may have been obtained by summing the populations of all counties that are either totally or partially within the 50-mile zone.

[31] A small but vocal group of pro-nuclear activists continue to maintain, in the face of overwhelming scientific evidence to the contrary, that a threshold dose exists below which ionizing radiation may have no effect or even may provide health benefits. However, there is a growing body of experimental data that indicates that low-dose radiation may actually be a more potent carcinogen than high-dose radiation because of low-dose "bystander effects."

radioactive plume, children will receive a greater absorbed dose than adults because of their lower body weight and higher respiration rate, even though their lung capacity is smaller. And because children and fetuses have much higher growth rates than adults, the same radiation dose has a greater chance of causing cancer in children and fetuses than in adults.

Exposure to low-dose ionizing radiation has also been associated with excess mortality from diseases other than cancer, such as cardiovascular disease, possibly as a result of radiation-induced inflammation. There is growing evidence that the effect of low-dose radiation exposure on mortality from diseases other than cancer may be as great as its effect on mortality from cancer, implying that current, cancer-based risk estimates may be too low by a factor of two.[32]

A radiological release from a nuclear plant accident would consist of many different types of radioactive materials. Some isotopes, such as cesium-137, emit penetrating gamma rays and can cause radiation injury from outside of the body. Other isotopes do not emit radiation that can penetrate skin but are most dangerous when inhaled or ingested, where they can concentrate in internal organs and deliver high doses to surrounding tissue. Iodine-131, which concentrates in the thyroid gland, and strontium-90, which concentrates in teeth and bones, are in this category. Some isotopes have short half-lives and do not persist in the environment, while others are long-lived and can result in long-term contamination.

NRC requires that evacuation planning in the event of a radiological emergency take place only within the so-called "plume exposure" emergency planning zone (EPZ), a roughly circular area with a radius of approximately ten miles. The choice of this distance was based in part on NRC analyses indicating that in the event of a severe accident, dose rates high enough to cause early fatalities from acute radiation syndrome would be confined to a region within about ten miles of the release point. However, dose rates outside of this region, although on average not high enough to cause early fatalities, could be high enough to result in a significant risk of cancer unless effective protective measures are taken. NRC's emergency planning regulations were never designed to limit such exposures in the event of the "worst core melt sequences," for which the protection goal is that "immediate life threatening doses would generally not occur outside the zone."[33]

Thus the current emergency planning basis is not now, and never was, intended to protect the public from significant but not immediately lethal exposures in the event of the "worst core melt sequences," such as those that could result from a well-planned terrorist attack. It should therefore be no surprise that NRC's emergency planning procedures

[32] A. MacLachlan, "UNSCEAR Probes Low-Dose Radiation Link to Non-Cancer Death Rate," *Nucleonics Week*, June 17, 2004.
[33] US NRC, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Plants*, NUREG-0654, 1980, p. 12.

would not protect individuals either inside or outside the EPZ from such exposures in the event of an attack.

The proximity of Indian Point to New York City, its populous suburbs and its watershed, given the potential hazard it represents, has long been an issue of concern and controversy. Following the Three Mile Island accident in March 1979, the Union of Concerned Scientists (UCS) unsuccessfully petitioned the NRC to suspend operations at Indian Point, in part because of its location in a densely populated area. At the same time, the NRC formed two task forces to examine the risks posed by Indian Point and the Zion plant near Chicago "because of the high population densities surrounding those units" and initiated a formal adjudication, the Indian Point Special Proceeding, to review the issues raised in the UCS petition and others.[34]

During the Special Proceeding, three NRC administrative judges heard testimony regarding the potential impacts of a severe accident at Indian Point on New York City residents. For instance, the director of New York City's Bureau of Radiation Control testified that potassium iodide (KI), which can block the uptake of radioactive iodine by the thyroid if taken near the time of exposure, should be stockpiled for "possible immediate use in New York City," at a time when NRC did not recommend that KI be provided even for residents of the 10-mile EPZ.

The administrative judges reached some disturbing conclusions in the proceeding. They stated that "under certain meteorological conditions, delayed fatalities from cancer appear to be possible almost anywhere in the city" and that "a severe release at Indian Point could have more serious consequences than that same release at virtually any other site licensed by the Commission." And they urged the Commission "to give serious consideration to the potential costs to society of dangerous, low probability accidents. Such accidents could, as Staff testimony has shown, result in fatalities that number in the hundreds or thousands."

The Commission appears to have essentially forgotten these conclusions. Many of the technical issues resolved during the course of the Special Proceeding are being debated all over again today.

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[34] US NRC, Indian Point Special Proceeding, 1983, p. 5.

THE MACCS2 CODE

MACCS2 is a computer code that was developed by Sandia National Laboratories under NRC sponsorship as a successor to CRAC2.[35] It is designed to estimate the health, environmental and economic consequences of radiation dispersal accidents, and is widely used by NRC and DOE for various safety applications. It utilizes a standard straight-line Gaussian plume model to estimate the atmospheric dispersion of a point release of radionuclides, consisting of up to four distinct plumes, and well-established models to predict the deposition of radioactive particles on the ground from both gravitational settling ("dry deposition") and precipitation ("wet deposition"). [36] From the dispersion and deposition patterns, the code can then estimate the radiation doses to individuals as a result of external and inhalation exposures to the radioactive plume and to external radiation from radionuclides deposited on the ground ("groundshine"). The code also has the capability to model long-term exposures resulting from groundshine, food contamination, water contamination and inhalation of resuspended radioactive dust.

The code also can evaluate the impact of various protective actions on the health and environmental consequences of the release, including evacuation, sheltering and, in the long term, remediation or condemnation of contaminated areas. Most parameters, such as the average evacuation speed, decontamination costs, and the dose criteria for temporary relocation and long-term habitation, can be specified by the user.

MACCS2 requires a large number of user-specified input parameters. A given release is characterized by a "source term," which is defined by its radionuclide content, duration and heat content, among other factors. The shape of the Gaussian plume is determined by the wind speed, the release duration, the atmospheric stability (Pasquill) class and the height of the mixing layer at the time of the release.

MACCS2 requires the user to supply population and meteorological data, which can range from a uniform population density to a site-specific population distribution on a high-resolution polar grid. The meteorological data can range from constant weather conditions to a 120-hour weather sequence. The code can process up to 8760 weather sequences --- a year's worth --- and generate a frequency distribution of the results.

The code allows the user to define the dose-response models for early fatalities (EFs) and latent cancer fatalities (LCFs). We use the MACCS2 default models. For EFs, MACCS2 uses a 2-parameter hazard function, with a default LD_{50} dose (the dose associated with a 50% chance of death) of 380 rem. LCFs, MACCS2 uses the standard linear, no-threshold model, with a dose-response coefficient of 0.1 LCF/person-Sievert and a dose-dependent reduction factor of 2, per the 1991 recommendations of the International Committee on

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[35] Chanin and Young (1997), op cit.

[36] Much of the following section is based on a recent comprehensive review of MACCS2 by the Department of Energy, which we would recommend to readers interested in a more in-depth discussion of the capabilities and limitations of the code. See Office of Environment, Safety and Health, U.S. Department of Energy, *MACCS2 Computer Code Application Guidance for Documented Safety Analysis: Interim Report*, DOE-EH-4.2.1.4-Interim-MACCS2, September 2003.

Radiological Protection (ICRP) in ICRP 60.[37] The corresponding coefficients used in the CRAC2 model, based on now-antiquated estimates, were lower by a factor of 4.

For the calculation of the committed effective dose equivalent (CEDE) resulting from inhalation and ingestion of radionuclides, we have replaced the default MACCS2 input file with one based on the more recent dose conversion factors in ICRP 72.[38] We have shown previously that this substitution reduces the projected number of latent cancer fatalities from a severe nuclear reactor accident by about one-third.[39] (The default MACCS2 file incorporates EPA guidance based on ICRP 30, which although out of date continues to be the basis for regulatory analyses in the United States.)

When using MACCS2 several years ago, we discovered an error that resulted in an overcounting of latent cancer fatalities in the case of very large releases. After pointing this out to the code manager, SNL sent us a revised version of the code with the error corrected, which we have used for the analysis in this report.

Like most radiological consequence codes in common use, MACCS2 has a number of limitations. First of all, because it incorporates a Gaussian plume model, the speed and direction of the plume are determined by the initial wind speed and direction at the time of release, and cannot change in response to changing atmospheric conditions (either in time or in space). Consequently, the code becomes less reliable when predicting dispersion patterns over long distances and long time periods, given the increasing likelihood of wind shifts. Also, the Gaussian plume model does not take into account terrain effects, which can have a highly complex impact on wind field patterns and plume dispersion. And finally, MACCS2 cannot be used for estimating dispersion less than 100 meters from the source.

However, MACCS2 is adequate for the purpose of this report, which is to develop order-of-magnitude estimates of the radiological consequences of a catastrophic attack at Indian Point for residents of New York City and the entire New York metropolitan area, and to assess the impact of different protective actions on these consequences. We restrict our evaluations to a circular area with a radius of 50 miles centered on Indian Point, except for the calculation of long-term doses and economic impacts, which we assess out to 100 miles.

In the next section, we discuss the basis for the MACCS2 input parameters that we use in our evaluation.

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[37] MACCS2 does not allow the user to specify different dose-response models for different radionuclides. We use a model with a dose-dependent reduction factor of 2, even though this assumption likely underestimates the carcinogenic potential of alpha-emitters, which is not reduced in effectiveness at low doses or dose rates.

[38] International Commission on Radiological Protection (ICRP*), Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 5, Compilation of Ingestion and Inhalation Dose Coefficients*, ICRP Publication 72, Pergamon Press, Oxford, 1996.

[39] E. Lyman, "Public Health Risks of Substituting Mixed -Oxide for Uranium Fuel in Pressurized-Water Reactors," *Science and Global Security* **9** (2001), pgs. 33-79. See Footnote 48.

THE SABOTAGE SCENARIO

The scenario that we analyze is based on the so-called "revised source term" that NRC defined in 1995 in NUREG-1465. The revised source term was developed as a more realistic characterization of the magnitude and timing of radionuclide releases during a core-melt accident than the source term originally specified for use in Part 100 siting analyses. In its entirety, the PWR revised source term presented in NUREG-1465 corresponds to a severe accident in which the primary coolant system is depressurized early in the accident sequence. An example is a "large break loss-of-coolant accident" (LBLOCA), in which primary coolant is rapidly lost and the low-pressure safety injection system fails to operate properly, resulting in core melt and vessel failure. This scenario is one of the most severe events that can occur at PWRs like Indian Point, and could result in a relatively rapid release of radioactivity.

(a) The source term

A severe accident of this type would progress through four distinct phases. As the water level in the core decreases and the fuel becomes uncovered, the zirconium cladding tubes encasing the fuel rods overheat, swell, oxidize and rupture. When that occurs, radionuclides that have accumulated in the "gap" between the fuel and the cladding will be released into the reactor coolant system. If there is a break in the reactor coolant system (as would be the case in a LBLOCA), then these radionuclides would be released into the atmosphere of the containment building. These so-called "gap" releases consist of the more volatile radionuclides contained in irradiated fuel, such as isotopes of krypton, xenon, iodine and cesium. This period is known as the "gap release" phase, and is predicted to last about 30 minutes. The oxidation of the zirconium cladding by water also generates hydrogen, which is a flammable gas.

As the core continues to heat up, the ceramic fuel pellets themselves begin to melt, releasing greater quantities of radionuclides into the reactor vessel and through the breach in the reactor coolant system into the containment building atmosphere. The molten fuel mass then collapses and drops to the bottom of the reactor vessel, where it aggressively attacks the steel, melts through the bottom and spills onto the floor of the containment building.[40] The period between the start of fuel melting and breach of the reactor vessel is known as the "early in-vessel" phase, and typically would last about an hour.

When the molten fuel breaches the reactor vessel and drops to the containment building floor, it violently reacts with any water that has accumulated in the cavity and with the concrete floor itself. This "core-concrete interaction" causes further releases of radionuclides from the molten fuel into the containment building. This period is known as the "ex-vessel" phase, and would last for several hours.

[40] This scenario is not theoretical. During the 1979 accident at Three Mile Island Unit 2, part of the melted core relocated to the bottom of the reactor vessel where it began melting through the steel. The re-introduction of forced cooling water flow terminated this sequence before vessel failure.

At the same time, some portion of the molten core may remain in the reactor vessel, where it would continue to degrade in the presence of air and release radionuclides. Also, radionuclides released during the in-vessel phase that deposit on structures within the primary coolant system may be re-released into the containment building. These releases take place during the "late in-vessel" phase and could continue for many hours.

At the time when the molten core falls to the floor of the reactor vessel, steam explosions may occur that could blow apart the reactor vessel, creating high-velocity "missiles" that could rupture the containment building and violently expel the radioactive gases and aerosols it contains into the environment. This would result in a shorter in-vessel phase. If the vessel remains intact until melt-through, hydrogen or steam explosions are also possible when the molten fuel spills onto the concrete below the vessel, providing another opportunity for containment failure.

The complete revised source term (all four phases) is a general characterization of a low-pressure severe accident sequence, such as a large-break loss of coolant accident with failure of emergency core cooling systems. According to the timing of the accident phases in the revised source term, the "gap release" phase would begin within a few minutes after the initiation of the event and lasts for 30 minutes. At that time, the early in-vessel phase begins as the fuel pellets start to melt. This phase is assumed to last for 1.3 hours, and ends when the vessel is breached.

In our scenario, we assume that the attackers have weakened but not fully breached the containment, so that there is a high probability that the containment building will be ruptured by a steam or hydrogen explosion at the time of vessel breach. This results in a rapid purge of the radionuclide content of the containment building atmosphere into the environment, followed by a longer-duration release due to core-concrete interactions and late in-vessel releases.

We do not wish to discuss in detail how saboteurs could initiate this type of accident sequence. However, since NRC asserts that even in a terrorist attack these events are unlikely to occur, we need to present some evidence of the plausibility of these scenarios. One such scenario would involve a 9/11-type jet aircraft attack on the containment building, possibly accompanied by a ground attack on the on-site emergency power supplies. (One must also assume that interruption of off-site power takes place during an attack, given that off-site power lines are not under the control of the licensee and are not protected.)

The Nuclear Energy Institute (NEI) issued a press release in 2002 describing some of the conclusions of a study conducted by the Electric Power Research Institute (EPRI) that purported to show that penetration of a PWR containment by a jet aircraft attack was impossible. A study participant later acknowledged that (1) the justification for limiting the impact speed to 350 mph was based on pilot interviews and not on the results of simulator testing, and (2) even at 350 mph, their analysis actually found that the 42-inch

thick reinforced concrete containment dome of a PWR suffered "substantial damage" and the steel liner was deformed.[41]

However, even if penetration of the containment does not occur, the vibrations induced by the impact could well disrupt the supports of the coolant pumps or the steam generators, causing a LBLOCA. The emergency core cooling system pumps, which require electrical power, would not be available under blackout conditions caused by the disabling of both off-site and on-site power supplies. Thus makeup coolant would not be provided, the core would rapidly become uncovered and the NUREG-1465 sequence would begin. Other engineered safety features such as containment sprays and recirculation cooling would not be available in the absence of electrical power. The damaged containment building would then be far less resistant to the pressure pulse caused by a steam spike or hydrogen explosion, and would have a much higher probability of rupture at vessel breach. We note that the steel liner of a reinforced concrete containment structure like that at Indian Point only carries 10 to 20% of the internal pressure load, and therefore may fail well before the design containment failure pressure is reached if the concrete shell is damaged.

Because the emergency diesel generators are themselves quite sensitive to vibration, a ground assault may not even be necessary to disable them, since the aircraft impact itself, followed by a fuel-air explosion, could cause them to fail.

One can find support for the credibility of this scenario in the recently leaked summary of a report prepared for the German Environment Ministry by the nuclear safety consultant GRS on the vulnerability of German nuclear reactors to aircraft attacks.[42] In the summary, GRS defined a series of credible damage scenarios and then determined whether or not the resulting accident sequence would be controllable The report considered an attack on the Biblis B PWR by a small jet (Airbus A320) or medium-sized jet (Airbus A300) travelling at speeds from 225 to 394 miles per hour, where the peak speed of 394 mph was determined through the use of simulators. GRS concluded that for an event in which the jet did not penetrate the containment, but the resulting vibrations caused a primary coolant leak, and the control room was destroyed by debris and fire (a condition similar to a station blackout), then control of the sequence of events would be "uncertain."[43] Biblis B was designed for protection against the crash of a 1960s-era Starfighter jet and as a result is equipped, like most German reactors, with a double containment. In contrast, Indian Point 2 and 3, while of the same 1970s vintage as Biblis B, were not designed to be resistant to airplane crashes, and do not have double containments.

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[41] R. Nickell, "Nuclear Plant Structures: Resistance to Aircraft Impact," 44[th] Annual Meeting of the Institute of Nuclear Materials Management, Phoenix, AZ, July 13-17, 2003.
[42] Mark Hibbs, "Utilities Expect Showdown with Trittin over Air Terror Threat," *Nucleonics Week* **45**, February 12, 2004.
[43] Gesellschaft für Anlagen und Reaktorsicherheit, *Schutz der deutschen Kernkraftwerke vor dem Hintergrund der terroristischen Anschläge in den USA vom 11. September 2001*, (*Protection of German Nuclear Power Plants in the Context of the September 11, 2001 Terrorist Attacks in the US*), November 27, 2002.

The NUREG-1465 revised source term is shown in Table 1. The source term is characterized by grouping together fission products with similar chemical properties and for each group specifying a "release fraction"; that is, the fraction of the core radionuclide inventory released from the damaged fuel into the containment building atmosphere. Noble gases include krypton (Kr); halogens include iodine (I); alkali metals include cesium (Cs); noble metals include ruthenium (Ru); the cerium (Ce) group includes actinides such as plutonium (Pu) and the lanthanide (La) group includes actinides such as curium (Cm).

TABLE 1: NUREG-1465 radionuclide releases into containment for PWRs

	Gap	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (hrs)	0.5	1.3	2.0	10.0
Release fractions (%):				
Noble Gases (Kr)	0.05	0.95	0	0
Halogens (I)	0.05	0.35	0.25	0.1
Alkali Metals (Cs)	0.05	0.25	0.35	0.1
Tellurium group (Te)	0	0.05	0.25	0.005
Barium, Strontium (Ba, Sr)	0	0.02	0.1	0
Noble Metals (Ru)	0	0.0025	0.0025	0
Cerium group (Ce)	0	0.0005	0.005	0
Lanthanides (La)	0	0.0002	0.005	0

It is important to note that NUREG-1465 is not intended to be a "worst-case" source term. The accompanying guidance specifically states that "it is emphasized that the release fractions for the source terms presented in this report are intended to be representative or typical, rather than conservative or bounding values…"[44] In fact, the release fractions for tellurium, the cerium group and the lanthanides were significantly lowered in response to industry comments. Upper-bound estimates, which are provided in a table in the back of NUREG-1465, indicate that "virtually all the iodine and cesium could enter the containment."[45] And experimental evidence obtained since NUREG-1465 was published in 1995 suggests that the tellurium, ruthenium, cerium and lanthanide release fractions in the revised source term may significantly underestimate actual releases of these radionuclide groups.[46] Thus our use of the NUREG-1465 source term is far from the worst possible case and may underestimate the impacts of credible scenarios.

[44] NUREG-1465, p. 13.
[45] NUREG-1465, p. 17.
[46] Energy Research, Inc., Expert Panel Report on Source Terms for High-Burnup and MOX Fuels, 2002.

We model this scenario in MACCS2 as a two-plume release. The first release begins at the time of vessel breach and containment failure, 1.8 hours after initiation of the accident, and continues over a period of 200 seconds as the containment atmosphere is rapidly vented. The second plume lasts for two hours as core-concrete interactions occur. For simplicity, only the first two hours of the late in-vessel release are included; the last eight hours are omitted, although this late release would likely make a significant contribution to public exposures, given the nearly ten-hour evacuation time estimate for the 10-mile EPZ.

We further assume that the entire radionuclide inventory released from the damaged fuel into the containment atmosphere escapes into the environment through the rupture in the containment. There is little information in the literature about realistic values for the fraction of the containment inventory that is released to the environment. In NUREG-1150, NRC states that "in some early failure cases, the [containment to environment] transmission fraction is quite high for the entire range of uncertainty. In an early containment failure case for the Sequoyah plant …the fractional release of radioactive material ranges from 25 percent to 90 percent of the material released from the reactor coolant system."[47] A review of the default values of this fraction for the Sequoyah and Surry plants used in supporting analyses for NUREG-1150 indicates that environmental releases ranging from 80 to 98% of the radionuclides in the containment atmosphere were typically assumed. The only case in which significant retention within the containment building occurs is when there is a delay of several hours between the initiation of core degradation and the time of containment failure, which is not the case for the scenario we are considering. Given that we are using only the first three phases of the NUREG-1465 source term, which may underestimate the maximum release of radionuclides like iodine and cesium by 35%, we believe it is reasonable to neglect the retention within the containment building of at most 20% of the radionuclide inventory.

Another plume characteristic that is very important for determining the distribution and magnitude of consequences is the heat energy that it contains. The oxidation of zirconium cladding during core degradation generates a large amount of heat in a short period of time, which can cause the plume to become buoyant and rise. Greater initial plume heights result in lower radionuclide concentrations close to the plant, but wider dispersal of the plume.

It is unlikely that a radiological release at any US PWR would produce a plume as high as the one released during the Chernobyl disaster. Because of the large mass of graphite moderator in the Chernobyl-4 reactor, a hot and long-duration graphite fire caused a very high plume that was responsible for dispersing radionuclides over vast distances. However, at the same time, the exposure and contamination within 50 miles of the Chernobyl site was much lower than it would have been if the plume had not risen so high. This means that the cooler plume that would be characteristic of a core meltdown at Indian Point could actually be a greater threat to the New York metropolitan area than the contamination pattern resulting from the Chernobyl accident might suggest.

[47] US NRC, *Severe Accident Risks: An Assessment for Five Nuclear Power Plants*, NUREG-1150, Volume 2, December 1990, p. C-108.

Table 2 shows the two-plume source term for input into MACCS2, adapted from the NUREG-1465 source term in Table 1. The first plume consists of the containment radionuclide inventory at the time of vessel breach (the sum of the first and second columns in Table 1). The second plume consists of the releases generated by core-concrete interactions and a fraction of the late-in-vessel releases (the sum of the third column and one-fifth of the fourth column in Table 1).

TABLE 2: Source term used in MACCS2 model

Plume	Release time (hrs)	Duration(hrs)	Energy release (MW)	Kr	I	Cs	Te	Ba	Ru	Ce	La
1	1.8	0.06	2.8	1	0.4	0.3	0.05	0.02	0.0025	0.0005	0.0002
2	1.86	2	1.6	0	0.27	0.37	0.25	0.1	0.0025	0.005	0.005

The reactor core inventory used was calculated for a representative 3565 MWt PWR at the end of an equilibrium 18-month cycle using the SCALE code, and was then scaled to the Indian Point 2 power rating of 3071 MWt.[48] Since Indian Point 2 operates on a 24-month cycle, the inventory we use here does not represent the peak inventory of the reactor core, which occurs just before refueling.

(b) Meteorology

The calculation of radiological consequences from a severe accident is strongly dependent on the meteorological conditions at the time of the release and for several days afterward. Relevant factors include the wind speed, the wind direction, the atmospheric stability, the height of the mixing layer and the occurrence of precipitation.

The MACCS2 code can utilize a weather sequence of hourly data for a 120-hour period following the initial release. The user has the option to supply a file with an entire year's worth of hourly meteorological data (8760 entries), consisting of wind speed, atmospheric stability class, and precipitation. The program can then calculate up to 8760 results, each corresponding to a release beginning at a different hour of the year. For each set of weather data, MACCS2 can also generate sixteen results by rotating the plume direction into each sector of the compass, repeating the calculation for each plume direction, and then weighting the results with the fraction of the time that the wind blows in that direction (as specified by the user-supplied "wind rose," or set of probabilities that the wind will be blowing in a certain direction at the site). Finally, the code can tabulate the results in a frequency distribution.

[48] Lyman (2001), op cit., pp. 64-66.

The MACCS2 code, like the CRAC2 code before it, has the option to sample a reduced number of weather sequences, based on a semi-random sampling method. The reason for employing a sampling scheme in the past was no doubt the length of computing time needed for each calculation; however, the program runs quickly on modern machines, so there is no need to employ the MACCS2 sampling scheme. In fact, a comparison of the results obtained from sampling, which utilizes about 100 weather sequences, and the results obtained from an entire year's worth of sequences, finds that the peak consequence values in the sampling distribution are 30% or more below the peak consequences over the entire year, if the plume rotation option is not utilized. Thus there is a significant sampling error for peak values associated with the MACCS2 sampling scheme (and presumably the CRAC2 sampling scheme as well).

We were unable to obtain the meteorological data for the Indian Point site needed for input into MACCS2. Instead, we used a meteorological data file for New York City, the location of the nearest National Weather Service weather monitoring station, that was supplied with the original CRAC2 code. This is the same approach that was taken in the CRAC2 Report, which was ostensibly a site-specific study of the 91 sites where nuclear reactors were located or planned, but did not use meteorological data files specific to those sites. Instead, the study used data derived from 29 National Weather Service stations that were "chosen as a representative set of the nation's meteorological conditions."[49] NRC later had to adopt the same approach, using the New York City meteorological data file as a surrogate for Indian Point-specific data in a CRAC2 benchmark exercise, because it was unable to obtain the Indian Point data.[50]

Use of the New York City meteorological data file in lieu of Indian Point site data is a reasonable approximation for the purposes of this report. Two of the most important factors in determining the radiological consequences of a terrorist attack at Indian Point are the wind direction and the precipitation. With regard to the first factor, we use the Indian Point site wind rose to take into account the effect of the variation in wind direction.[51] With regard to precipitation data, since the MACCS2 code only allows for uniform precipitation over the entire evaluation area, the precipitation data set from New York City is just as relevant as data from the Indian Point site for determining the consequences for the New York metropolitan area.

One phenomenon that we cannot fully account for without access to meteorological data specific to the Indian Point site is the coupling between wind direction and wind speed that results from the plant's location in the Hudson River Valley. Wind speeds below a threshold of below 4 meters per second tend to result in plumes that follow the course of the river valley, whereas greater wind speeds produce plumes that are free to travel in any direction and are better approximated by the straight-line Gaussian model. Our use of the

[49] R. Davis, A. Hanson, V. Mubayi and H. Nourbakhsh, *Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants*, NUREG/CR-6295, US Nuclear Regulatory Commission, 1997, p. 3-30.
[50] US Nuclear Regulatory Commission, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, Vol. 1, Sec. 5.3.3.2.3.
[51] James Lee Witt Associates, *Review of Emergency Preparedness of Areas Adjacent to Indian Point and Millstone*, March 2003, Figure 3-1, p. 21.

Indian Point wind rose accounts for this effect, but to the extent that the distribution of wind speeds in the meteorological data file that we use differs from that at the Indian Point site, the calculations may include some cases that involve unrealistic wind patterns. However, any errors in the distribution resulting from this approximation are not likely to be significant in comparison to the uncertainties associated with use of the straight-line Gaussian model in MACCS2. In any event, it is likely that properly accounting for this effect would result in the channeling of a greater number of slow-moving, concentrated plumes directly downriver toward densely populated Manhattan, thereby increasing the overall radiological impact.

We have also run the calculations using the meteorological data file for the Surry site in Virginia to compare the maximum consequences obtained. We find that the values for peak early fatalities differ by less than 1% and the value for peak latent cancer fatalities differs by less than 5%. We interpret this result as an indication that the peak consequences we found for Indian Point are not due to weather conditions unique to the meteorological data file for New York City.

If Entergy were willing to provide us with data from the Indian Point meteorological monitoring station, we would be pleased to use it to assess whether it would have a significant impact on our results. However, we would expect any impact to be minor.

(c) Protective actions

Another crucial factor in determining the consequences associated with a terrorist attack at Indian Point is the effectiveness of the actions taken to protect individuals within the 10-mile emergency planning zone (EPZ).

The MACCS2 emergency planning model requires the user to input the time when notification is given to emergency response officials to initiate protective actions for the surrounding population; the time at which evacuation begins after notification is received; and the effective evacuation speed. Once evacuation begins, each individual then proceeds in a direction radially outward from the release point at a rate given by the effective evacuation speed.

We have assumed that the time at which the off-site alarm is sounded is coincident with the initiation of core melting; that is, 30 minutes after the attack. It is unlikely that the decision to evacuate could be made in much less time. This choice still provides an interval of 78 minutes between the sounding of the alarm and the initiation of the radiological release, consistent with earlier studies such as the CRAC2 Report.

We have assumed that the delay time between receipt of notification by the public within the EPZ and initiation of evacuation is two hours. This is the default parameter in the MACCS2 code, and is consistent both with earlier estimates of the "mobilization time" and with the most recent ones for the Indian Point site, which found that 100% of the public within the EPZ would be mobilized to evacuate by two hours after notification.[52]

[52] James Lee Witt Associates (2003), op cit., Figure 5-6, p. 96.

The effective evacuation speed was obtained from the mobilization time estimate of two hours and the most recent Indian Point evacuation time estimate (ETE) for good summer weather of 9 hours 25 minutes.[53] Subtracting the two-hour mobilization time leaves a maximum time of 7.42 hours for the actual evacuation. Since the maximum travel distance to leave the EPZ is approximately ten miles, this corresponds to an effective evacuation speed of 1.35 miles per hour, or 0.6 meters per second. The high value for the ETE and the correspondingly low effective evacuation speed reflect the severe traffic congestion within the EPZ that is projected to occur in the event that a crisis occurs at Indian Point requiring evacuation.

Outside of the 10-mile EPZ, the baseline dose calculations assume that individuals will take no protective actions.[54] Although this may not be realistic, we believe that it would be inappropriate to assume otherwise. Since NRC and FEMA do not require that any preparation for an emergency be undertaken outside of the 10-mile EPZ, it would not be conservative to assume that individuals outside of the EPZ would receive prompt notification of the event or would know what to do even if they did receive notification. However, to examine the impact of this assumption on the results, we consider a case where the emergency evacuation zone is extended to 25 miles, and the average evacuation speed remains the same as in the 10-mile EPZ case.

(d) Population distribution

In order to accurately calculate the consequences of a terrorist attack at Indian Point, it is necessary to have the correct spatial distribution of population in the vicinity of the site. MACCS2 has the option to use a site population data file, in which the site-specific population is provided on a grid divided into sixteen angular sectors. The user can specify the lengths of sectors in the radial direction.

Most of our analysis is focused on a circular region centered on the Indian Point site with a radius of fifty miles. The ten-mile EPZ is divided into eleven regions, with divisions at the site exclusion zone (about 0.5 miles), at the one-mile point, and nine successive mile-wide intervals. The region between the EPZ and the fifty-mile limit is subdivided into ten intervals (see Figure 1, below).

Permanent resident population data for the ten-mile EPZ was obtained from the estimates for 2003 generated by KLD Associates for the Evacuation Time Estimate study that it prepared for Entergy.[55] The total number of permanent residents within a ten-mile circular zone around Indian Point in 2003, according to KLD, was 267,099. We have not included the transient population in the region in our calculations, even though it would add another 25% to the permanent population estimate, according to KLD data.

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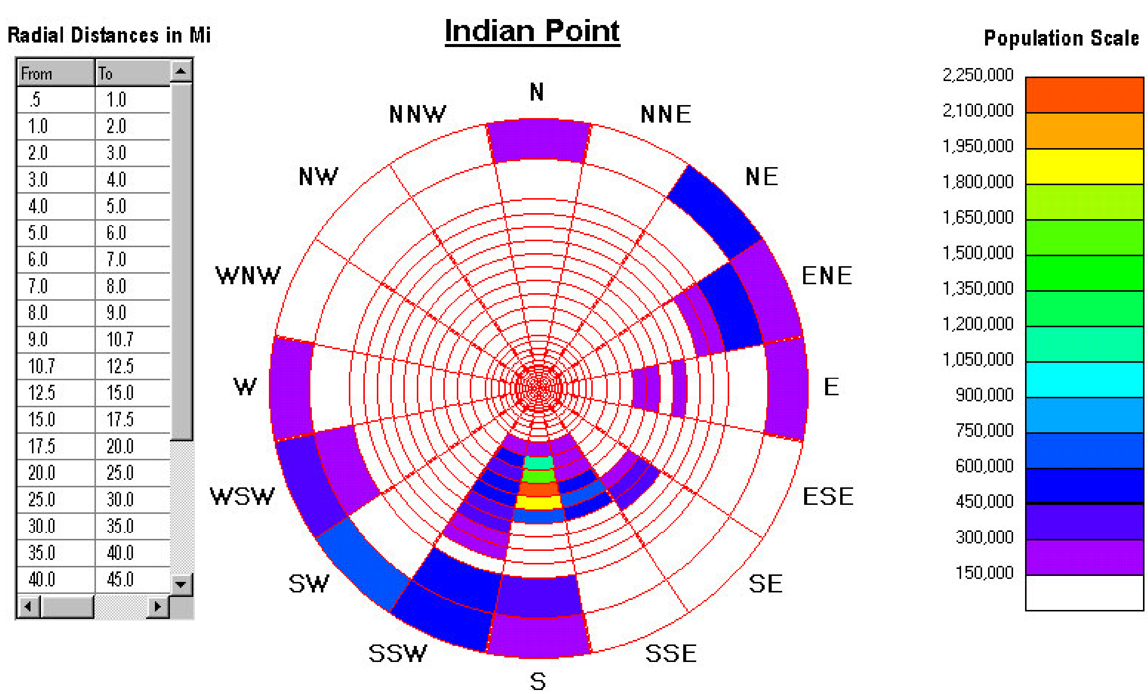
[53] KLD Associates, Inc., *Indian Point Energy Center Evacuation Time Estimate*, Rev. 0 (2003), p. 7-8.
[54] However, the calculation of doses within the EPZ does reflect the impact of "shadow evacuation" of individuals outside of the EPZ, since it uses the KLD Associates evacuation time estimate for the EPZ, which assumes that shadow evacuation occurs.
[55] KLD Associates, Inc. (2003), op cit., p. 3-7.

For the region from 10 to 100 miles from Indian Point, the MACCS2 site data file was generated with the SECPOP2000 code, which is the most recent version of the SECPOP code originally developed by the Environmental Protection Agency and later adopted by NRC for use in regulatory applications.[56] SECPOP2000 utilizes 2000 US Census data to estimate population distributions on a user-specified grid surrounding any location in the United States, drawing on a high-resolution database of over eight million census-blocks. By utilizing the 2000 Census data in SECPOP2000, we have slightly underestimated the population in this region, which appears to have increased by about 1% between 2000 and 2003.

The Indian Point plume exposure EPZ is not in the shape of a perfect circle of ten-mile radius, but includes some regions that are beyond ten miles from the plant. To account for the 38,177 individuals that reside within the EPZ but outside of the 10-mile circular zone (according to KLD estimates for 2003), we used the SECPOP2000 code to determine that an "effective" circular EPZ boundary of 10.68 miles would include the appropriate additional number of permanent residents, and adjusted the MACCS2 grid accordingly.

Figure 1 displays the population rosette generated by SECPOP2000 for Indian Point, out to a distance of 100 miles. The location of New York City is plainly visible on the grid.

FIGURE 1



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[56] N. Bixler et al., *SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program*, NUREG/CR-6525, Rev. 1, Sandia National Laboratories, August 2003.

RESULTS

In this section, we present the results of the MACCS2 simulation of a terrorist attack at IP2, as previously described.

MACCS2 generates results for two distinct periods following a radiological release. First, it calculates the doses to individuals received during the "emergency" phase of the event, defined as the period extending up to the first week following the release. The doses received during this period result from direct exposure to and inhalation of the plume, as well as exposure to plume particles deposited on the ground ("groundshine"). Second, it separately calculates doses received beyond the first week after the release as a result of groundshine, inhalation of resuspended particles, and consumption of contaminated food and water. The first sets of results provided below refer only to the consequences of exposures received during a one-week emergency phase. The economic and long-term health consequences are calculated based on the evaluation of chronic exposures for a period of fifty years following the release, which are dominated by groundshine.

Following the format of the CRAC2 Report summary, our calculation considers several public health and environmental endpoints, including early fatalities, latent cancer fatalities, maximum distance for early fatalities, and total economic costs. The calculations were carried out for each of the 8760 weather sequences in the New York City meteorological data file by rotating the plume direction into each of the 16 sectors of the compass, and then generating a weighted average of the results according to the Indian Point site wind rose. For each endpoint, in addition to the mean of the distribution and the peak value corresponding to the worst-case meteorological conditions encountered during the year, we present the 95th and 99.5th percentile values of the distribution.

The results of the MACCS2 frequency distribution are based on the assumption that the radiological release would occur at random during the year, even though the timing of a terrorist attack most likely would be far from random. As we have previously discussed, one must assume that a terrorist attack intended to cause the maximum number of casualties would be timed to coincide as closely as possible with the most favorable weather conditions. In the case of Indian Point, an attack at night --- the time when a terrorist attack is most likely to be successful --- also happens to be the time when the prevailing winds are blowing toward New York City. Consequently, the mean and other statistical parameters derived from a random distribution are not characteristics of the actual distribution of consequences resulting from a terrorist attack, which would be restricted to a much more limited set of potential release times. A meteorological data set confined to the evening hours would skew the distribution in the direction of increased consequences.

In our judgment, the 95th percentile values of these distributions, rather than the mean values, are reasonable representations of the likely outcome of a well-planned terrorist attack. This choice reflects the fact that the attack time will be largely of the terrorists' choosing, but that some factors will necessarily remain out of their control --- for instance,

the ability to accurately predict precipitation patterns, and the ability to launch an attack exactly as planned.

In the following tables, it is important to note that the peak results in each category do not correspond in general to the same weather sequence. For example, the weather conditions that lead to the maximum number of early fatalities are typically those that involve rainout and substantial deposition of the plume close to the plant, and thus are not the same conditions that lead to peak latent cancer fatalities, which involve rainout of the plume over New York City.

(a) Consequences of radiological exposures during "emergency phase"

Here we consider the consequences of exposures received during the 7-day "emergency phase." We calculate the number of "early fatalities" (EFs) resulting from acute radiation syndrome, both for the residents of the 10-mile EPZ, who are assumed to evacuate according to the scheme described previously, and for the entire population within 50 miles of the plant. Following the CRAC2 Report, we also provide the "early fatality distance," that is, the greatest distance from the Indian Point site at which early fatalities may occur. Finally, we provide an estimate of the number of latent cancer fatalities (LCFs) that will occur over the lifetimes of those who are exposed to doses that are not immediately life-threatening, both for residents of the EPZ and for residents of the 50-mile region.

It is important to note that these estimates are based on dose conversion factors (the radiation doses resulting from internal exposure to unit quantities of radioactive isotopes) appropriate for a uniform population of adults, and do not account for population variations such as age-specific differences. A calculation fully accounting for individual variability of response to radiation exposure is beyond the capability of the MACCS2 code and the scope of this report.

In Table 3, these results are provided for the case in which 100% evacuation of the EPZ occurs, based on the KLD evacuation time estimate and 2-hour mobilization time discussed earlier. Table 4 presents the same information for the case where the EPZ population is sheltered for 24 hours prior to evacuation. Finally, Table 5 presents the results for the extreme case where no special precautions are taken in the EPZ.

In interpreting the results of these tables, one should keep in mind that the MACCS2 code uses different radiation shielding factors for individuals that are evacuating, sheltering or engaged in normal activity. The default MACCS2 parameters (which we adopt in this study) assume that evacuees are not shielded from the radioactive plume by structures, since they are mostly outdoors or in non-airtight vehicles during the evacuation. Individuals who shelter themselves instead of evacuating are shielded to a considerable extent by structures, but may be exposed to higher levels of radiation overall because they remain in areas closer to the site of plume release. The MACCS2 default shielding parameters assume that sheltering reduces doses from direct plume exposure by 40% and doses from plume inhalation by 67%. The relative benefits of sheltering versus

evacuation are obviously quite sensitive to the values of the shielding parameters. Finally, the level of shielding for individuals engaged in "normal activity" falls in between the levels for evacuation and for sheltering, with reductions in doses from direct plume exposure and plume inhalation relative to evacuees of 25% and 59%, respectively.

TABLE 3: Terrorist attack at IP 2, MACCS2 estimates of early fatalities (EFs), latent cancer fatalities (LCFs) and the EF distance resulting from emergency phase exposures, 100% evacuation of EPZ

	Mean	95th percentile	99.5th percentile	Peak
Consequence:				
EFs, within EPZ	527	2,440	11,500	26,200
EFs, 0-50 mi.	696	3,460	16,600	43,700
EF distance (mi.)	5.3	18	24	60
LCFs, within EPZ	9,200	31,600	59,000	89,500
LCFs, 0-50 mi.	28,100	99,400	208,000	518,000

TABLE 4: Terrorist attack at IP 2, MACCS2 estimates of early fatalities (EFs), latent cancer fatalities (LCFs) and the EF distance resulting from emergency phase exposures, 24-hour sheltering in EPZ

	Mean	95th percentile	99.5th percentile	Peak
Consequence:				
EFs, within EPZ	626	2,550	6,370	13,000
EFs, 0-50 mi.	795	3,250	10,200	38,700
EF distance (mi.)	6.2	18	24	60
LCFs, within EPZ	3,770	9,920	12,100	19,400
LCFs, 0-50 mi.	22,700	81,000	192,000	512,000

TABLE 5: Terrorist attack at IP 2, MACCS2 estimates of early fatalities (EFs), latent cancer fatalities (LCFs) and the EF distance resulting from emergency phase exposures, normal activity in EPZ

	Mean	95th percentile	99.5th percentile	Peak
Consequence:				
EFs, within EPZ	4,050	12,600	22,300	38,500
EFs, 0-50 mi.	4,220	13,500	27,300	71,300
EF distance (mi.)	9	18	24	60
LCFs, within EPZ	4,480	10,400	12,500	20,300
LCFs, 0-50 mi.	23,400	82,600	193,000	516,000

A comparison of Tables 3 and 4 indicates that sheltering instead of evacuation results in slightly higher mean early fatalities, but substantially lower 99.5th percentile and peak values. A possible interpretation of this counterintuitive result is that the higher percentile early fatality results for the evacuation case correspond to rare situations in which people evacuate in such a manner as to maximize their radiation exposure (for instance, if they are unfortunate enough to be traveling directly underneath the radioactive plume at the same speed and in the same direction). These situations cannot occur for the sheltering case. Overall, sheltering does appear to substantially reduce the projected number of latent cancer fatalities within the EPZ relative to evacuation, for the default MACCS2 shielding parameters.

A comparison of Table 5 to Tables 3 and 4 indicates that either evacuation or sheltering would substantially reduce the number of early fatalities within the EPZ relative to a case where no protective actions are taken. Also, by comparing Tables 3 and 5, one sees that the number of latent cancer fatalities in the EPZ is considerably lower for the normal activity case than for the evacuation case. There are two reasons for this. First, many evacuees will receive doses that are not high enough to cause early fatalities, yet will contribute to their lifetime cancer risk. In the normal activity case, some of these individuals will receive higher doses and succumb to acute radiation syndrome instead. Second, the MACCS2 default shielding factors give considerable protection to individuals engaged in normal activity compared to evacuees, and may not be realistic.[57]

The peak numbers of latent cancer fatalities for all three cases in the 50-mile zone are disturbingly high, and are more than double the number in the 99.5th percentile. But an examination of the particular weather sequence corresponding to this result indicates that

[57] The protection due to shielding has a bigger impact on the number of latent cancer fatalities, which is a linear function of population dose, than on the number of early fatalities, which is a non-linear function of dose. Shielding would only prevent early fatalities for those individuals whose acute radiation doses would be lowered by sheltering from above to below the early fatality threshold.

the rarity of the event is an artifact of the meteorological data file that we have used, and not a consequence of very extreme or unusual weather conditions for the New York City region. We are not disclosing the details of this weather sequence.

The reader may also notice that the values for the "early fatality distance" for the 95th percentile and above are the same in Tables 3-5, but the mean values are not. This is because the distances for the 95th percentile and above are all greater than 10 miles, so that they are not affected by differences in protective actions that apply only within the 10-mile EPZ.

(b) Doses received by individuals outside of the 10-mile EPZ

It is clear from the previous section that direct exposure to the radioactive plume resulting from a terrorist attack at Indian Point could have severe consequences well beyond the 10-mile EPZ, yet there is no regulatory requirement that local authorities educate residents outside of the EPZ about these risks, or undertake emergency planning to protect these individuals from plume exposures. Therefore, individuals who are now at risk do not have the information that they may need to protect themselves. This is a shortsighted policy, and in fact is inconsistent with government guidelines for protective actions in the event of a radiological emergency.

In this section, we calculate the plume centerline thyroid doses to adults and five-year-old children, and the plume centerline whole-body doses to adults, both at the EPZ boundary and in midtown New York City. (For a given distance downwind of a release, the maximum dose is found at the plume centerline.) We then compare these values to the appropriate protective action recommendations. Thyroid doses are compared to the dose thresholds in the most recent FDA recommendations for potassium iodide administration and whole-body doses are compared to the EPA protective action guides (PAGs) for emergency-phase evacuation. In both cases, the plume centerline doses received to individuals in New York City are well in excess of the projected dose thresholds that would trigger protective actions.

(i) Thyroid doses to children, their consequences, and the need for KI distribution

The statistically significant increase in the incidence of thyroid cancer observed among children exposed to fallout from the Chernobyl disaster leaves little doubt of the causal relationship between the occurrence of these cancers and the massive release of radioactive iodine to the environment resulting from the accident.[58] The effectiveness of widespread distribution of stable iodine in the form of potassium iodide (KI) to block uptake of radioactive iodine in the thyroid was also confirmed in western areas of Poland, where the timely administration of KI was estimated to have reduced peak doses from radioactive iodine by 30%.[59]

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[58] D. Williams, "Cancer After Nuclear Fallout: Lessons from The Chernobyl Accident," *Nature Reviews Cancer* **2** (2002), p. 543-549.
[59] Board on Radiation Effects Research, National Research Council, *Distribution and Administration of Potassium Iodide in the Event of a Nuclear Incident*, National Academies Press, 2003, p. 58.

In the United States, after resisting public demands for many years, the Nuclear Regulatory Commission finally agreed in January 2001 to amend its emergency planning regulations to explicitly consider the use of KI, and to fund the purchase of KI for distribution within the 10-mile plume exposure EPZs of nuclear plants in states that requested it. This effort accelerated after the September 11 attacks, as more states requested the drug, but even today only fewer than two-thirds of the 34 states and tribal governments that qualify for the KI purchase program have actually stockpiled it. New York State is one of the participants.

Despite a few attempts in Congress after September 11 to require the distribution of KI in areas outside of the plume exposure EPZs, the 10-mile limit remains in effect today, and NRC continues to defend it. In a recent Commission meeting on emergency planning, NRC employee Trish Milligan said that[60]

> "...the [NRC] staff has concluded that recommending consideration of potassium iodide distribution out to 10 miles was adequate for protection of the public health and safety."

Earlier in this briefing, Ms. Milligan provided evidence of the NRC staff's thinking that led to this conclusion:[61]

> "When the population is evacuated out of the [10-mile] area and potentially contaminated foodstuffs are interdicted, the risk from further radioactive iodine exposure to the thyroid gland is essentially eliminated."

These statements again show that NRC continues to use design-basis accidents, in which the containment remains intact, as the model for its protective action recommendations. Although NRC claims that its emergency planning requirements take into account all potential releases, including those resulting from terrorist acts, it clearly is not taking into account catastrophic events such as the scenario being analyzed in this report.

These statements also suggest that NRC is committing the fallacy of using the pattern of radioactive iodine exposure that occurred after the Chernobyl accident as the model for the pattern that could occur here. In the Chernobyl event, the majority of the thyroid dose to children occurred through ingestion of contaminated milk and other foodstuffs that were not interdicted due to the failure of the Soviet authorities to act in a timely manner. However, the food pathway dominated in that case primarily because of the extremely high elevation of the Chernobyl plume, which reduced the concentration of radioactive iodine in the plume and therefore the doses received through direct inhalation.
But as pointed out earlier, the plume from a severe accident at a water-moderated PWR like Indian Point would probably not rise as high as the Chernobyl plume, and the associated collective thyroid dose would have a greater contribution from direct plume inhalation and a lower contribution from milk consumption. In this case, the importance

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[60] US NRC, "Briefing on Emergency Preparedness Program Status" (2003), transcript, p. 21.
[61] Ibid, p.19.

of KI prophylaxis would increase relative to that of milk interdiction for controlling overall population exposure to radioactive iodine.

Our calculations clearly indicate that a severe threat to children from exposure to radioactive iodine is present far beyond the 10-mile EPZ where KI is now being made available. In Table 6, we present some results of the distribution for plume centerline thyroid dose to both adults and to five-year-old children at the EPZ boundary and in midtown Manhattan (32.5 miles downwind). In the last column, we provide the projected dose thresholds from the most recent guidelines issued by the FDA for KI prophylaxis.

The thyroid dose to five-year-olds due to I-131 internal exposure was calculated by using the age-dependent coefficients for dose per unit intake provided in ICRP 72, which are approximately a factor of five greater than those for adults. The calculation must also take into account the difference in the rate of intake of air for children and for adults. Children have lower lung capacities than adults, but they have higher metabolic rates and therefore breath more rapidly. The higher breathing rate of children tends to partially offset their lower lung capacity. Data collected by the California Environmental Protection Agency indicates that on average, children consume air at a rate about 75% of that of adults.[62] We have used this figure in our calculation.

TABLE 6: Terrorist attack at IP 2, MACCS2 estimates of centerline thyroid doses to 5-year-olds resulting from emergency phase exposures (all doses in rem)

Location	Age	Mean	95th percentile	99.5th percentile	Peak	FDA KI threshold
Outside EPZ (11.6 mi)	Adult	1,120	3,400	5,850	9,560	10 (ages 18-40) 500 (over 40)
	5 years	3,620	10,900	18,000	32,100	5
Midtown Manhattan (32.5 mi)	Adult	164	429	761	1,270	10 (ages 18-40) 500 (over 40)
	5 years	530	1,310	2,500	4,240	5

The results in Table 6 show that the thyroid doses to 5-year-olds are approximately three times greater than those for adults. This tracks well with information in the World Health Organization's 1999 guidelines for iodine prophylaxis, which states that thyroid doses from inhalation in children around three years old will be increased up to threefold relative to adults.[63]

[62] Air Resources Board, California Environmental Protection Agency, 'How Much Air Do We Breathe?", Research Note #94-11, August 1994. On the Web at www.arb.ca.gov/research/resnotes/notes/94-11.htm.
[63] World Health Organization, *Guidelines for Iodine Prophylaxis Following Nuclear Accidents*, WHO, Geneva, 1999, Sec. 3.3.

These results make clear that both 95th percentile and mean projected thyroid doses can greatly exceed the FDA-recommended threshold for KI prophylaxis administration at locations well outside the 10-mile EPZ, for 5-year-old children and for adults of all ages. In Manhattan, KI would be recommended for children and adults under 40, based on the 95th percentile projection.

The health consequences of doses of this magnitude to the thyroid would be considerable. As the 99.5th percentile is approached, the 5-year-old doses are high enough to cause death of thyroid tissue. In fact, they are on the order of the doses that are applied therapeutically to treat hyperthyroidism and other diseases by destroying the thyroid gland. Children with this condition would require thyroid hormone replacement therapy for their entire lives. At lower doses, in which cells are not killed but DNA is damaged, the risk of thyroid cancer to children would be appreciable. According to estimates obtained from Chernobyl studies, a 95th percentile thyroid dose of 1,310 rem to a 5-year-old child in Manhattan would result in an excess risk of about 0.3% per year of contracting thyroid cancer.[64] Given that the average worldwide rate of incidence of childhood thyroid cancer is about 0.0001% per year, this would represent an impressive increase.

These results directly contradict the reassuring statements by NRC quoted earlier. But it is no secret to NRC that such severe thyroid exposures can occur as the result of a catastrophic release. Results very similar to these were issued by NRC staff in 1998 in the first version of a draft report on the use of KI, NUREG-1633.[65] This draft included a Section VII entitled "Sample Calculations," in which the NRC staff estimated the centerline thyroid doses at the 10-mile EPZ boundary from severe accidents using the RASCAL computer code. Table 5 of the draft report shows that the NRC's calculated dose to the adult thyroid at the 10-mile limit ranged from 1500 to 19,000 rem for severe accidents with iodine release fractions ranging from 6 to 35%, for a single weather sequence.[66] In the introductory section, the report states that "doses in the range of 25,000 rad are used to ablate thyroids as part of a therapeutic procedure. Such thyroid doses are possible during severe accidents."[67] NRC's results are even more severe than ours, which were obtained using the NRC revised source term, with a higher iodine release fraction of 67%.

Given NRC's reluctance to provide information of this type to the public, it is no surprise that the Commission withdrew the draft NUREG-1633 and purged it from its web site, ordering the issuance of a "substantially revised document" taking into account "the many useful public comments" that it received.[68] Lo and behold, the second draft of

[64] The average excess absolute risk per unit thyroid dose for children exposed to Chernobyl fallout has been estimated 2.1 per million children per rad. D. Williams, op cit., p. 544.
[65] F.J. Congel et al., *Assessment of the Use of Potassium Iodide (KI) As A Public Protective Action During Severe Reactor Accidents*, Draft Report for Comment, NUREG-1633, US Nuclear Regulatory Commission, July 1998.
[66] Ibid, p. 26.
[67] Ibid, p. 6.
[68] US NRC, "Staff Requirements --- Federal Register Notice on Potassium Iodide," SRM-COMSECY-98-016, September 30, 1998.

NUREG-1633, which was rewritten by Trish Milligan and reissued four years later, mysteriously failed to include Section VII, "Sample Calculations," as well as all information related to those calculations (such as the clear statement cited earlier that thyroid doses in the range of 25,000 rad are possible during severe accidents).[69] This took place even though the Commission's public direction to the NRC staff on changes to be incorporated into the revision made no explicit reference to this section.[70] However, it is clear that the expurgated information would be inconsistent with NRC's previous rulemaking restricting consideration of KI distribution only to the 10-mile zone. Even after this exercise in censorship, the Commission still voted in 2002 to block release of the revised draft NUREG-1633 as a final document.

Some insight into the level of understanding of the health impacts of a catastrophic release of radioactive iodine of the current Commission can be found in the statement of Commissioner McGaffigan in voting to delay release of the revised NUREG-1633 for public comment. In his comments, McGaffigan wrote[71]

> "Both WHO [the World Health Organization] and FDA set the intervention level on KI prophylaxis for those over 40 at 5 gray (500 rem) to the thyroid …Since we do not expect, *even in the worst circumstances*, any member of the public to receive 500 rem to the thyroid, it would be useful for FDA to clarify whether we should plan for KI prophylaxis for those over 40." [Emphasis added.]

This statement is not consistent with what is known about the potential consequences of a severe nuclear accident. Few experts would claim that such high doses cannot occur "even in the worst circumstances," and the NRC's own emergency planning guidance is not intended to prevent such doses in *all* accidents, but only in *most* accidents. Given that the Commissioner presumably read the first draft of NUREG-1633, he would have seen the results of the staff's thyroid dose calculations and other supporting material. There is no discussion in the public record that provides a rationale for Commissioner McGaffigan's rejection of the informed judgment and quantitative analysis of his technical staff.

In 2003, at the request of Congress a National Research Council committee released a report addressing the issue of distribution and administration of KI in the event of a nuclear incident.[72] Most notably, the committee concluded that[73]

> "1. KI should be available to everyone at risk of significant health consequences from accumulation of radioiodine in the thyroid in the event of a radiological incident…

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[69] US NRC, "Status of Potassium Iodide Activities, SECY-01-0069, Attachment 1 (NUREG-1633, draft for comment; prepared by P.A. Milligan, April 11, 2001).
[70] US NRC, SRM-COMSECY-98-016.
[71] US NRC, Commission Voting Record on SECY-01-0069, "Status of Potassium Iodide Activities," June 29, 2001.
[72] National Research Council (2003), op cit.
[73] Ibid, p. 5.

2. KI distribution programs should consider …local stockpiling outside the emergency planning zone …"

While the committee did not itself take on the politically sensitive question of how to determine the universe of individuals who would be "at risk of significant health consequences," it did recommend that "the decision regarding the geographical area to be covered in a KI distribution program should be based on risk estimates derived from calculations of site-specific averted thyroid doses for the most vulnerable populations."[74] This is the type of information that we provide in Table 6 (and the type that NRC struck from draft NUREG-1633). We hope that the information in our report provides a starting point for state and local municipalities to determine the true extent of areas that could be significantly affected by terrorist attacks at nuclear plants in their jurisdiction and to make provisions for availability of KI in those regions. Our calculations show that New York City should be considered part of such an area.

However, even timely administration of KI to all those at risk can only reduce, but cannot fully mitigate, the consequences of a release of radioactive iodine resulting from a terrorist attack at Indian Point. The projected dose to individuals who undergo timely KI prophylaxis can be reduced by about a factor of 10. A review of the results of Table 6 shows that doses and cancer risks to many children in the affected areas will still be high even after a ten-fold reduction in received dose. And KI can only protect people from exposure to radioactive iodine, and not from exposure to the dozens of other radioactive elements that would be released to the environment in the event of a successful attack.

(ii) Whole-body doses and the need for evacuation or sheltering

In addition to KI distribution, the other major protective action that will be relied on to reduce exposures following a terrorist attack at Indian Point is evacuation of the population at risk. In Table 7, we present the results of our calculation for the projected centerline whole-body "total effective dose equivalents" (TEDEs) just outside the EPZ boundary and in downtown Manhattan, and compare those with the EPA recommended dose threshold for evacuation during the emergency phase following a radiological incident. As in the discussion of projected thyroid doses and KI prophylaxis, we find that projected centerline TEDEs would exceed the EPA Protective Action Guide (PAG) for evacuation of 1-5 rem at distances well outside of the 10-mile plume exposure EPZ within which NRC requires evacuation planning.

[74] Ibid, p. 162.

TABLE 7: Terrorist attack at IP 2, MACCS2 estimates of adult centerline whole-body total effective dose equivalents (TEDEs) resulting from emergency phase exposures (all doses in rem)

	Mean	95th percentile	99.5th percentile	Peak	EPA PAG
Location					
EPZ boundary (11.6 mi)	198	549	926	1,490	1-5
Midtown Manhattan (32.5 mi)	30	77	131	307	1-5

From the results in Table 7, it is clear that according to the EPA early phase PAG for evacuation of 1-5 rem, evacuation would be recommended for individuals in the path of the plume centerline not only outside of the EPZ boundary, but in New York City and beyond. An individual in Manhattan receiving the 95th percentile TEDE of 77 rem during the emergency phase period would have an excess absolute lifetime cancer fatality risk of approximately 8%, which corresponds to a 40% increase in the lifetime individual risk of developing a fatal cancer (which is about one in five in the United States).

We now examine the potential reduction in health consequences that could result from evacuation of a larger region than the current 10-mile EPZ by considering a case in which the boundary of the plume exposure EPZ is expanded from 10.7 to 25 miles. We calculate the impact of different protective actions in this region on the numbers of early fatalities and latent cancer fatalities among the population within the expanded EPZ but outside of the original 10-mile EPZ. The residents of the expanded EPZ are assumed either (1) to evacuate with the same mobilization time and at the same average speed as the residents of the original EPZ, or (2) to shelter in place for 24 hours and then evacuate. The results are provided in Table 8.

TABLE 8: Terrorist attack at IP 2, MACCS2 95th percentile estimates of early fatalities (EFs) and latent cancer fatalities (LCFs) resulting from emergency phase exposures; 25-mile EPZ

	Normal activity	Evacuation	Sheltering for 24 hrs
Consequence:			
EFs, 10.7-25 mi	664	0	0
LCFs, 10.7-25 mi	19,800	45,700	9,020

These results indicate that evacuation and sheltering are equally effective in eliminating the risk of early fatalities among residents of the 10.7-25 mile region for the 95th percentile case. On the other hand, one sees that evacuation also tends to increase the number of latent cancer fatalities relative to normal activity, while sheltering reduces the number. Thus for this scenario, it appears that sheltering of individuals in the 10.7-25 mile region would be preferable to evacuation of this region for the MACCS2 evacuation and sheltering models we use here. This is consistent with the results we obtained earlier when considering the comparative impacts of evacuation and sheltering of residents of the 10-mile EPZ, again indicating that evacuation tends to increase population doses by placing more people in direct contact with the radioactive plume. However, other models and other shielding parameter choices may lead to different conclusions. We would urge emergency planning officials to evaluate an exhaustive set of scenarios, and to conduct a realistic and site-specific assessment of the degrees of shielding that structures in the region may provide, to determine what types of actions would provide the greatest protection for residents of regions outside of the 10-mile EPZ.

(c) Long-term economic and health consequences

In this section we provide MACCS2 order-of-magnitude estimates of the economic costs of the terrorist attack scenario, the numbers of latent cancer fatalities resulting from long-term radiation exposures (primarily as a result of land contamination), and the number of people who will require permanent relocation. NRC has used MACCS2 to estimate the economic damages of reactor accidents for various regulatory applications.[75]

There is no unique definition of the economic damages resulting from a radiological contamination event. In the MACCS2 model, which is a descendant of the CRAC2 model, the total economic costs include the cost of decontamination to a user-specified cleanup standard, the cost of condemnation of property that cannot be cost-effectively decontaminated to the specified standard, and a simple lump-sum compensation payment to all members of the public who are forced to relocate either temporarily or permanently as a result of the attack. Although simplistic, this model does provide a reasonable estimate of the order of magnitude of the direct economic impact of a successful terrorist attack at Indian Point.

(i) EPA Protective Action Guide cleanup standard

We first employ the long-term habitability cleanup standards provided by the EPA protective action guide (PAG) for the "intermediate phase," which is the period that begins after the emergency phase ends, when releases have been brought under control and accurate radiation surveys have been taken of contaminated areas. The EPA intermediate phase PAG recommends temporary relocation of individuals and decontamination if the projected whole-body total effective dose equivalent (TEDE) (not taking into account any shielding from structures) over the first year after a radiological

[75] US NRC, Office of Nuclear Regulatory Research, *Regulatory Analysis Technical Evaluation Handbook*, NUREG/BR-0184, January 1997, p. 5.37.

release would exceed 2 rem. The EPA chose this value with the expectation that if met, then the projected (shielded) TEDE in the second (and any subsequent year) would be below 0.5 rem, and the cumulative TEDE over a fifty-year period would not exceed 5 rem.

The MACCS2 economic consequence model evaluates the cost of restoring contaminated areas to habitability (which we define as reducing the unshielded TEDE during the first year of reoccupancy to below 2 rem), and compares that cost to the cost of condemning the property. All cost parameters, including the costs of decontamination, condemnation and compensation, can be specified by the user. We employ an economic model partly based on parameters developed for a recent study on the consequences of spent fuel pool accidents.[76] The model utilizes the results of a 1996 Sandia National Laboratories report that estimates radiological decontamination costs for mixed-use urban areas.[77] We refer interested readers to these two references for information on the limitations and assumptions of the model.

The SECPOP2000 code, executed for the Indian Point site, provides the required site-specific inputs for this calculation, including the average values of farm and non-farm wealth for each region of the MACCS2 grid, based on 1997 economic data. These values are used to assess the cost-effectiveness of decontaminating a specific element versus simply condemning it.

Table 9 presents the long-term health and economic consequences calculated by MACCS2 for a region 100 miles downwind of the release, considering only costs related to residential and small business relocation, decontamination and compensation. Since the calculation was performed using values from a 1996 study and from 1997 economic data, we have converted the results to 2003 dollars using an inflation adjustment factor of 1.10. Because of significant uncertainties in the assignments of parameters for this calculation, the results in Table 9 should only be regarded as order-of-magnitude estimates. The reader should note that the latent cancer fatality figures in Table 9 result from doses incurred after the one-week emergency phase is over, and therefore are additional to the numbers of latent cancer fatalities resulting from emergency-phase exposures reported previously in Tables 3 to 5.

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[76] J. Beyea, E. Lyman and F. von Hippel, "Damages from a Major Release of ^{137}Cs into the Atmosphere of the United States," *Science and Global Security* 12 (2004) 1-12.
[77] D. Chanin and W. Murfin, *Site Restoration: Estimates of Attributable Costs From Plutonium Dispersal Accidents*, SND96-0057, Sandia National Laboratories, 1996.

TABLE 9: Terrorist attack at IP 2, MACCS2 estimates of long-term economic and health consequences, EPA intermediate phase PAG (< 2 rem in first year; approx. 5 rem in 50 yrs)

	Mean	95th percentile	99.5th percentile	Peak
Consequence				
Total cost, 0-100 mi (2003 $)	$371 billion	$1.17 trillion	$1.39 trillion	$2.12 trillion
People permanently relocated	684,000	3.19 million	7.91 million	11.1 million
LCFs, 0-100 mi	12,000	41,200	57,900	84,900
Plume Centerline 50-year TEDE (rem)	4.57	7.04	7.18	7.42

One can see from Table 9 that imposition of the EPA intermediate phase PAG does result in restricting the mean 50-year cumulative TEDE to below 5 rem, but that this limit is exceeded for the higher percentiles of the distribution. Thus for a terrorist attack at the 95th percentile, the subsidiary goal of the EPA intermediate phase PAG is not met.

(ii) Relaxed cleanup standard

In the recent NRC meeting on emergency planning described earlier, NRC staff and Commissioners questioned claims by activists that a severe nuclear accident would render large areas "permanently uninhabitable," arguing that the radiation protection standard underlying that determination is too stringent compared to levels of natural background radiation to which people are already exposed.

For instance, Trish Milligan said that[78]

> "There's been a concern that a radioactive release as a result of a nuclear power plant accident will render thousands of square miles uninhabitable around a plant. It is true that radioactive materials can travel long distances. But it is simply not true that the mere presence of radioactive materials are [sic] harmful…the standard applied to this particular claim has been a whole body dose of 10 rem over 30 years, or approximately 330 millirem per year. This dose is almost the average background radiation dose in the United States which is about 360 millirem per year. Some parts of the country have a background radiation dose two or more times higher than the national average. So in effect this additional 330 millirem dose is an additional year background dose or the difference in dose

[78] US NRC, Briefing on Emergency Preparedness (2003), op cit., transcript, p. 22.

between someone living in a sandy coastal area or someone living in the Rocky Mountains."

Ms. Milligan does not note that her opinion of an acceptable level of radiation is not consistent with national standards, such as the EPA PAGs. The EPA long-term goal of limiting chronic exposures after a radiological release to 5 rem in 50 years corresponds to an average annual exposure of 100 millirem above background, while she implies that even a standard of 330 millirem per year, which would double the background dose on average, is unnecessarily stringent.

However, we can evaluate the impact of weakening the EPA PAGs for long-term exposure on costs and risks. In Table 10, we assess the impact of adopting a long-term protective action guide of 25 rem in 50 years, or an average annual dose of 500 millirem per year. By comparing the 95th percentile columns in Table 10 and Table 9, one can see that relaxing the standard would modestly reduce the post-release cleanup costs by about 25% and drastically reduce the number of relocated individuals by 90%. However, weakening the standard would nearly triple the number of long-term cancer deaths among residents of the contaminated area. Cost-benefit analyses of proposals to weaken long-term exposure standards should take this consequence into account.

TABLE 10: Long-term economic and health consequences of a terrorist attack at IP 2, relaxed cleanup standard (25 rem in 50 years)

	Mean	95th percentile	99.5th percentile	Peak
Consequence:				
Total cost, 0-100 mi (2003 $)	$249 billion	$886 billion	$1.14 trillion	$1.50 trillion
People permanently relocated	118,000	334,000	1.86 million	7.98 million
LCFs, 0-100 mi	36,300	115,000	169,000	279,000

(d) An even worse case

The previous results were based on the analysis of a terrorist attack that resulted in a catastrophic radiological release from only one of the two operating reactors at the Indian Point site. However, it is plausible that both reactors could be attacked, or that an attack on one could result in the development of an unrecoverable condition at the other. Here we present the results of a scenario in which Indian Point 3 undergoes a similar accident sequence to Indian Point 2 after a time delay of just over two hours. This could occur, for example, if Indian Point 3 experienced a failure of its backup power supplies at the time that Indian Point 2 was attacked. Given the loss of off-site power at the same time, Indian Point 3 could experience a small-break LOCA and eventually a core melt, commencing about two hours after accident initiation. We assume that the attackers

weaken the IP3 containment so that it ruptures at the time of vessel failure. In Table 11, we present the results of this scenario for the case of full evacuation of the EPZ.

As bad as this scenario is, it still does not represent the worst case. If any or all of the three spent fuel pools at the Indian Point site were also damaged during the attack, the impacts would be far greater, especially with regard to long-term health and economic consequences.

TABLE 11: Terrorist attack at IP 2 and 3, MACCS2 estimates of early fatalities (EFs) and latent cancer fatalities (LCFs) resulting from emergency phase exposures, 100% evacuation of EPZ

	Mean	95th percentile	99.5th percentile	Peak
Consequence:				
EFs, within EPZ	925	4,660	18,400	34,100
EFs, 0-50 mi.	1,620	8,580	30,900	78,400
EF, distance (mi.)	9.1	21	29	60
LCFs, within EPZ	14,800	42,900	75,100	122,000
LCFs, 0-50 mi.	53,400	180,000	342,000	701,000

CONCLUSIONS

In conclusion, we make the following observations.

1) The current emergency planning basis for Indian Point provides insufficient protection for the public within the 10-mile emergency planning zone in the event of a successful terrorist attack. Even in the case of a complete evacuation, up to 44,000 early fatalities are possible.

2) The radiological exposure of the population and corresponding long-term health consequences of a successful terrorist attack at Indian Point could be extremely severe, even for individuals well outside of the 10-mile emergency planning zone. We calculate that over 500,000 latent cancer fatalities could occur under certain meteorological conditions. A well-developed emergency plan for these individuals, including comprehensive distribution of potassium iodide throughout the entire area at risk, could significantly mitigate some of the health impacts if promptly and effectively carried out. However, even in the case of 100% evacuation within the 10-mile EPZ and 100% sheltering between 10 and 25 miles, the consequences could be catastrophic for residents of New York City and the entire metropolitan area.

3) The economic impact and disruption for New York City residents resulting from a terrorist attack on Indian Point could be immense, involving damages from hundreds of billions to trillions of dollars, and the permanent displacement of millions of individuals. This would dwarf the impacts of the September 11 attacks.

4) The potential harm from a successful terrorist attack at Indian Point is significant even when only the mean results are considered, and is astonishing when the results for 95th and 99.5th meteorological conditions are considered. Given the immense public policy implications, a public dialogue should immediately be initiated to identify the protective measures desired by the entire affected population to prevent such an attack or effectively mitigate its consequences should prevention fail. As this study makes abundantly clear, this population extends far beyond the 10-mile zone that is the focus of emergency planning efforts today.

We hope that this information will be useful for officials in the Department of Homeland Security as it carries out its statutory requirement to conduct a comprehensive assessment of the terrorist threat to the US critical infrastructure, as well as for health and emergency planning officials in New York City and other areas that are not now currently engaged in emergency preparedness activities related to a terrorist attack at Indian Point.

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