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Comment On: NRC-2016-0118-0001
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RULES AND DIRECTIVES BRANCH USNRC

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General Comment

August 2, 2016

Cindy Bladey
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: Bellefonte Efficiency and Sustainability Team/Mothers Against Tennessee River Radiation's (BEST/MATRR) Comments on the Amendment Request for the Extended Power Uprate for Browns Ferry Nuclear Plant Units 1, 2, and 3; Docket Nos. 50-259, 50-260, and 50-296, NRC-2016-0118

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Add= L. Ronewicz (LMR3)

Dear Ms. Bladley;

Please accept the enclosed PDF file of Comments written by Mark Leyse on behalf of BEST/MATRR.

Garry Morgan, Director - Radiation and Public Health Monitoring Project
BEST/MATRR
P.O. Box 241
Scottsboro, Al. 35768

PDF Attachment as stated, by Mark Leyse, Nuclear Safety Consultant

Attachments

Leyse's Browns Ferry Comments 8-2-16

August 3, 2016

Cindy Bladey
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Bellefonte Efficiency and Sustainability Team/Mothers Against Tennessee River Radiation's Comments on the Amendment Request for the Extended Power Uprate for Browns Ferry Nuclear Plant Units 1, 2, and 3; Docket Nos. 50-259, 50-260, and 50-296, NRC-2016-0118

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August 3, 2016

Cindy Bladey
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to the U.S. Nuclear Regulatory Commission's ("NRC") notice of solicitation of public comments on the amendment request for the extended power uprate ("EPU") for Browns Ferry Nuclear Plant ("BFN") Units 1, 2, and 3; Docket Nos. 50-259, 50-260, and 50-296; NRC-2016-0118 (hereinafter: "EPU for BFN"); NRC-2016-0118.

Bellefonte Efficiency and Sustainability Team/Mothers Against Tennessee River Radiation's Comments on the Amendment Request for the EPU for BFN; NRC-2016-0118

I, Mark Leyse ("Commenter"), am responding to the Federal Register notice that the NRC published on July 5, 2016 soliciting public comments on the proposed EPU for BFN.[1] Commenter is writing these comments on behalf of the Bellefonte Efficiency and Sustainability Team/Mothers Against Tennessee River Radiation ("BEST/MATRR").

I. Statement of the Bellefonte Efficiency and Sustainability Team/Mothers Against Tennessee River Radiation's Interest

The Bellefonte Efficiency and Sustainability Team ("BEST") is based in Scottsboro, Alabama, downwind of the Browns Ferry Nuclear Plant. BEST is a member-supported organization dedicated to protecting Tennessee Valley residents against the hazards of ionizing radiation. BEST was formed by citizens concerned about the proposed start-up of Tennessee Valley Authority's ("TVA") Bellefonte Nuclear Power Plant. Construction of Bellefonte, begun in 1974, had already been canceled twice; it has subsequently been cancelled yet again and is now up for auction as surplus. BEST is presently concerned with the safety of three TVA nuclear plants: Browns Ferry Nuclear Plant, located near Athens, Alabama, Sequoyah Nuclear Plant, located in Soddy-Daisy, Tennessee, and

[1] NRC, "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information," NRC-2016-0118, Federal Register, Vol. 81, No. 128, July 5, 2016.

Watts Bar Nuclear Plant, located near Spring City, Tennessee, which also produces tritium for nuclear weapons.

Mothers Against Tennessee River Radiation ("MATRR") is an educational project of BEST, inspired by parents and grandparents who love the Tennessee Valley and want their offspring to inherit the same beautiful mountains and river that they have been able to enjoy. MATRR sees no benefit strong enough to justify knowingly increasing the incidence of cancer in children. MATRR sees nuclear power as an unnecessary, short-term "solution" for a very long term, 100,000 year legacy of radioactive waste that will be left to future generations.

In February 2008, BEST became a local chapter of a league of environmental groups from seven Southeastern states, the Blue Ridge Environmental Defense League ("BREDL"). BREDL is a 501(c)(3) non-profit organization.

Based on BEST/MATRR's vested interest in the safe operation of Browns Ferry, BEST/MATRR members are personally affected and aggrieved by the extended power uprates that are proposed for all three of Browns Ferry's General Electric ("GE") Mark I boiling water reactors ("BWR"). The defective, antiquated BWR Mark I design performed poorly in the Fukushima Daiichi accident. In the accident, three BWR Mark I reactors melted down, generating hundreds of kilograms of explosive hydrogen gas. Hydrogen then detonated at different times, destroying three reactor buildings, which released large quantities of harmful radioactive material into the environment.

BEST/MATRR believes that the amendment request for the EPU for BFN should be denied. The proposed EPU would increase BFN's current licensed "steady-state reactor core power level for each unit from 3,458 megawatt thermal (MWt) to 3,952 MWt," constituting a thermal power level increase of approximately 14.3 percent for all three units. The proposed EPU would increase BFN's original licensed thermal power level of 3,293 MWt for each unit by approximately 20 percent for all three units.[2]

[2] NRC, "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information," NRC-2016-0118, Federal Register, Vol. 81, No. 128, July 5, 2016, p. 43666.

BEST/MATRR alleges that non-conservative computer safety model analyses were performed in order to justify the EPU for BFN. As explained in these comments, experimental data indicates that the EPU analyses under-predict the rates of the chemical reaction between zirconium and steam that would occur in the event of a loss-of-coolant accident ("LOCA"). This means that the analyses under-predict the rates in which energy (heat) is released, hydrogen generated, and zirconium fuel-cladding oxidized by the zirconium-steam reaction.

On November 17, 2009, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-93,[3] which addresses issues similar to those raised by BEST/MATRR in these comments. However, the NRC is still reviewing PRM-50-93, more than six years after it was submitted. It is difficult to know how long the NRC will continue reviewing PRM-50-93. But there is ample evidence that the Browns Ferry EPU analyses under-predict the zirconium-steam reaction rates that would occur in the event of a LOCA. For example, as discussed in Sections III.B and III.C of these comments, on November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, disclosed to Commenter that an NRC (TRACE code) computer simulation (using the Baker-Just correlation) of a Westinghouse design-basis accident experiment (FLECHT Run 9573), *under-predicted* cladding and steam temperatures at the elevation of the hottest section of the test's fuel rod simulators.[4] A computer safety model is supposed to *over-predict* temperatures in order to ensure an adequate margin of safety.

Commenter is not aware of any actions that the NRC has taken or of any information notices that the NRC has sent licensees, after finding that its TRACE computer safety model *under-predicted* cladding and steam temperatures for FLECHT Run 9573.

The NRC has sent out information notices in other instances in which a computer safety model's simulations indicated that NRC regulations could be violated. For example, the NRC sent out "Information Notice No. 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," after Westinghouse notified the NRC that one of its computer

[3] Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No. ML093290250).

[4] Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, e-mail to Mark Leyse, regarding the NRC's TRACE computer simulation of the FLECHT Run 9573 test bundle, November 24, 2015, (ADAMS Accession No: ML15341A160).

safety models "may predict higher fuel temperatures and internal pressures at high burnup conditions. This, in turn, may lead to code [computer simulation] results...*that do not meet the loss-of-coolant accident (LOCA) criterion in 10 CFR 50.46(b)(2)*"[5] [emphasis added].

By contrast, in November 2015, the NRC found that a TRACE code computer simulation (using the Baker-Just correlation) of Westinghouse's FLECHT Run 9573 *under-predicted* cladding and steam temperatures at the elevation of the hottest section of the test's fuel rod simulators,[6] and the NRC has done nothing. Instead, the NRC is considering an amendment request for an EPU for BFN, which is dependent on design-basis accident (LOCA) analyses.

By overlooking the deficiencies of computer safety models, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[7] The health and safety of BEST/MATRR members should not be compromised by the application of the *non-conservative* models that have been employed to help qualify the proposed EPU for BFN. The health and safety of BEST/MATRR members should not be compromised by an EPU for BFN.

I.A. Information Regarding Mark Leyse, Commenter

On March 15, 2007, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-84,[8] to the NRC. PRM-50-84 was summarized briefly in American Nuclear Society's *Nuclear News*'s June 2007 issue[9] and commented on and deemed "a well-

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[5] NRC, "Information Notice No. 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998, (ADAMS Accession No: ML003730714), p. 1.

[6] Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, e-mail to Mark Leyse, regarding the NRC's TRACE computer simulation of the FLECHT Run 9573 test bundle, November 24, 2015, (ADAMS Accession No: ML15341A160).

[7] Charles Miller et al., NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident," SECY-11-0093, July 12, 2011, (ADAMS Accession No: ML111861807), p. 3.

[8] Mark Leyse, PRM-50-84, March 15, 2007 (ADAMS Accession No. ML070871368).

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

documented justification for...recommended changes to the [NRC's] regulations"[10] by the Union of Concerned Scientists ("UCS").

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

Additionally, PRM-50-84 requested 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. (Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.)

In 2008, the NRC decided to consider the safety issues raised in PRM-50-84 in its rulemaking process.[11] And in 2009, the NRC published "Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of a proposed rulemaking, addressing four objectives: the fourth being the issues raised in PRM-50-84.[12] In 2012, the NRC Commissioners voted unanimously to approve a

[10] David Lochbaum, Union of Concerned Scientists, "Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84)," July 31, 2007, (ADAMS Accession No. ML072130342), p. 2.
[11] NRC, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," Docket No. PRM–50–84; NRC–2007–0013, Federal Register, Vol. 73, No. 228, November 25, 2008, pp. 71564-71569.
[12] NRC, "Performance-Based Emergency Core Cooling System Acceptance Criteria," NRC–2008–0332, Federal Register, Vol. 74, No. 155, August 13, 2009, pp. 40765-40776.

proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that is partly based on the safety issues Commenter raised in PRM-50-84.[13]

Commenter also coauthored a paper, "Considering the Thermal Resistance of Crud in LOCA Analysis," that was presented at the American Nuclear Society's 2009 Winter Meeting.[14]

On November 17, 2009, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-93.[15] 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; and 2) to stipulate minimum allowable core reflood rates, in the event of a LOCA.

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

PRM-50-93 was discussed briefly in the American Nuclear Society's March 2010 issue of *Nuclear News*.[16] 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

[13] NRC, Commission Voting Record, Decision Item: SECY-12-0034, Proposed Rulemaking—10 CFR 50.46(c): Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42), January 7, 2013, (ADAMS Accession No. ML13008A368).
[14] Rui Hu, Mujid S. Kazimi, Mark Leyse, "Considering the Thermal Resistance of Crud in LOCA Analysis," American Nuclear Society, 2009 Winter Meeting, Washington, D.C., November 15-19, 2009.
[15] Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No. ML093290250).
[16] American Nuclear Society, *Nuclear News*, March 2010, p. 36.

PRM-50-93]—that would better ensure safety in event of a loss of coolant accident.[17]

On October 27, 2010, the NRC published in the Federal Register that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Commenter authored and submitted on behalf of New England Coalition—requesting that the NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature of VYNPS—meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802.[18] The NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95.[19] PRM-50-95 was discussed briefly in the July 30, 2010 issue of Platts's *Inside NRC*.[20]

Commenter has also written a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-103,[21] and reports for Natural Resources Defense Council, including "Preventing Hydrogen Explosions In Severe Nuclear Accidents: Unresolved Safety Issues Involving Hydrogen Generation And Mitigation."[22]

II. The LOCA Analyses AREVA Conducted to Help Justify the Amendment Request for the EPU for BFN

The Federal Register notice that the NRC published on July 5, 2016 soliciting public comments on the proposed EPU for BFN states:

> The Power Uprate Safety Analysis Report (PUSAR) summarizes the results of safety evaluations performed that justify uprating the licensed thermal power at BFN. The PUSAR uses GEH [General Electric-Hitachi] GE14 fuel as the principal reference fuel type for the evaluation of the impact of EPU [extended power uprate]. However, the BFN units will utilize AREVA ATRIUM 10XM fuel, with some legacy ATRIUM 10 fuel, under EPU conditions. Therefore, the AREVA Fuel Uprate Safety

[17] 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, (ADAMS Accession No. ML101180175), p. 1.
[18] 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.
[19] Mark Leyse, PRM-50-95, June 7, 2010, (ADAMS Accession No. ML101610121).
[20] Suzanne McElligott, *Inside NRC*, July 30, 2010.
[21] Mark Leyse, PRM-50-103, October 14, 2011, (ADAMS Accession No. ML11301A094).
[22] Mark Leyse, Author, and Christopher Paine, Contributing Editor, "Preventing Hydrogen Explosions In Severe Nuclear Accidents: Unresolved Safety Issues Involving Hydrogen Generation And Mitigation," NRDC Report, R:14-02-B, March 2014.

> Analysis Report (FUSAR) for Browns Ferry Units 1, 2, and 3 and fuel related reports are provided to supplement the PUSAR and address the impact of EPU conditions on the AREVA fuel in the BFN units. The AREVA analyses contained in the FUSAR have provided disposition of the critical characteristics of the GE14 fuel and have been shown to bound ATRIUM 10XM and ATRIUM 10 fuel.[23]

Commenter now discusses the AREVA LOCA analyses that were conducted to help justify the amendment request for the EPU for BFN.

AREVA's LOCA analyses regarding the EPU for BFN are discussed in three AREVA reports: ANP-3377NP (regarding ATRIUM 10XM fuel), ANP-3378NP (regarding ATRIUM 10XM fuel), and ANP-3384NP (regarding ATRIUM 10 fuel). An important result of a LOCA analysis is the value that the maximum temperature the cladding of the fuel rods is predicted to reach: the peak cladding temperature ("PCT"). The LOCA analyses regarding the EPU for BFN discussed in ANP-3377NP, ANP-3378NP, and ANP-3384NP, predicted PCTs of 2030°F,[24] 2008°F,[25] and 2086°F,[26] respectively.

So the overall predicted PCT is 2030°F for ATRIUM 10XM fuel, which is used at BFN (Units 1, 2, and 3).[27] AREVA's analyses "were performed for a [reactor] core composed entirely of ATRIUM 10XM fuel at beginning-of-life (BOL) conditions.

[23] NRC, "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information," NRC-2016-0118, Federal Register, Vol. 81, No. 128, July 5, 2016, p. 43666.

[24] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)," ANP-3377NP, Revision 3, Attachment 11 "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A184), pp. 6.1, 6.3, 6.9, 8.6.

[25] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU),"ANP-3378NP, Revision 3, Attachment 13 "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A185), pp. 2.3, 5.1, 5.4, 6.1.

[26] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU)," ANP-3384NP, Revision 3, Attachment 15 "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A187), pp. 2.2, 5.1, 5.4, 6.1.

[27] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)," ANP-3377NP, Revision 3, Attachment 11 "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A184), p. 1.2.

Calculations assumed an initial core power of 102% of 3952 MWt, providing an analysis licensing basis power of 4031 MWt. The 2.0% increase reflects the maximum uncertainty in monitoring reactor power, as per NRC requirements. 3952 MWt corresponds to 120% of the original licensed thermal power (OLTP) and is referred to as extended power uprate (EPU)."[28]

And the overall predicted PCT is 2086°F for ATRIUM 10 fuel, which is used at BFN (Units 1, 2, and 3).[29] Apparently, the plan for BFN is that all three reactors will primarily use ATRIUM 10XM fuel after the EPU is implemented. The plan is to *maybe* include some ATRIUM 10 fuel "in a transition cycle" along with ATRIUM 10XM fuel after the EPU is implemented. "At EPU power, any ATRIUM-10 fuel would be in its third cycle of operation."[30]

II.A. The Computer Safety Model that AREVA Used to Conduct LOCA Analyses for the Amendment Request for the EPU for BFN

AREVA has stated that "[t]he models and computer codes used by AREVA for LOCA analyses [regarding the EPU for BFN] are collectively referred to as the EXEM BWR-2000 Evaluation Model." The EXEM BWR-2000 Evaluation Model has been approved for reactor licensing analyses by the NRC.[31]

The EXEM BWR-2000 Evaluation Model LOCA calculations for the EPU for BFN "were performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46."[32] In regard to the zirconium-steam reaction that would occur in the event of a LOCA, 10 C.F.R. 50 Appendix K, I.A.5 requires that "[t]he rate of energy release, hydrogen generation, and

[28] *Id.*, p. 1.1.
[29] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU)," ANP-3384NP, Revision 3, Attachment 15 "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A187), p. 1.1.
[30] *Id.*, p. 1.2.
[31] AREVA, "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)," ANP-3377NP, Revision 3, Attachment 11 "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU) (Non-Proprietary)," August 2015, (ADAMS Accession No: ML15282A184), p. 1.1.
[32] *Id.*

cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[33]

II.B. The Experiments Behind the Baker-Just Correlation

The Baker-Just correlation—used in Appendix K to Part 50 ECCS evaluation calculations—dates back to 1962.[34] In order to develop the Baker-Just correlation, Louis Baker, Jr. and Louis C. Just partly relied on data from their own experiments. Their experiments were conducted at "the melting temperature of zirconium, (in which fine [zirconium] wires were directly heated in water and the hydrogen evolution from the resulting molten droplets was measured to calculate the reaction rate)."[35] The melting temperature of zirconium—approximately 3362°F (1850°C)—is far greater than the temperature range of design-basis accidents, which have a maximum temperature of 2200°F (1204.4°C).

The Baker-Just correlation is primarily based on data from Alexis Lemmon and W. A. Bostrom's experiments,[36] which were conducted in the 1950s.[37] Bostrom's experiments were conducted above the temperature range of design-basis accidents. Lemmon and Bostrom's experiments were conducted with tiny *inductively* heated Zircaloy-2 specimens. There are radiative heat losses in experiments conducted with inductive heating, which affect a specimen's oxidation kinetics.

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[33] NRC, "Appendix K to Part 50—ECCS Evaluation Models," (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appk.html : last visited on 07/23/16).

[34] Louis Baker, Jr. and Louis C. Just, "Studies of Metal-Water Reactions at High Temperatures: III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, May 1962, (ADAMS Accession No: ML050550198).

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

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

[37] W. A. Bostrom, "The High Temperature Oxidation of Ziracloy in Water," WAPD-104, March 1954, (ADAMS Accession No: ML100900446) and Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, (ADAMS Accession No: ML100570218).

Regarding the Bostrom and Lemmon experiments that were used to help develop the Baker-Just Correlation, a 1978 Journal of Nuclear Materials paper states:

> Bostrom inductively heated specimens of Zircaloy-2 in water (with a steam bubble enveloping the specimen) under isothermal conditions and determined K_p in the temperature range 1300-1860°C by the hydrogen evolution method. Lemmon measured the rates of reaction between Zircaloy-2 and steam in the temperature range 1000-1700°C by inductively heating specimens in steam at 50 psia [pounds per square inch absolute] and measuring the rate of hydrogen evolution.[38]

Describing Lemmon's experiments in more detail, Lemmon's own 1957 report, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures" states:

> The reaction between solid Zircaloy 2 and steam at 50 psia was measured over the temperature range 1000 to 1690°C. ... The Zircaloy 2 specimens were heated by electrical induction and reacted with flowing steam at a pressure of 50 psia. ... The [Zircaloy 2] specimen was supported on a thermocouple protection tube and enclosed inside a Vycor tube; it was inductively heated to the reaction temperature by power applied through the induction coil.[39]

Lemmon's specimens were Zircaloy-2 cylinders that were 2.0 inches long and 0.5 inches in diameter.[40]

Regarding radiative heat losses experienced in Lemmon's experiments, Lemmon's own 1957 report states:

> The passage of steam through the reactor [unit] greatly increased the heat losses from the samples; and a large increase in power to the induction coil was required. Sample temperatures dropped as much as 100 or 200°C below the desired temperature before the power adjustment was effective. This sometimes took as long as [five] min.[41]

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[38] 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.
[39] Alexis W. Lemmon, "Studies Relating to the Reaction Between Zirconium and Water at High Temperatures," Battelle Memorial Institute, BMI-1154, January 1957, (ADAMS Accession No: ML100570218), pp. C-1, C-2, C-3.
[40] *Id.*, p. C-4.
[41] *Id.*, p. C-7.

Regarding how radiative heat losses in inductive specimen heating experiments affect oxidation kinetics, a 2003 paper by G. Schanz states:

> [Ocken] stated that [the] advantage [of experiments with inductive (Urbanic and Heidrick) and direct electrical heating (Biederman, *et al.*) of a specimen in a cool environment[42]] would be the temperature gradient from heated specimen to cool surrounding[s], leading to temperature gradients in the cladding wall in the same sense as in a reactor. In total disagreement with the argument of Ocken, the author of this paper stresses the advantage of a constant cladding wall temperature and thus of a better defined specimen temperature, as provided in furnace experiments! ... This argument was already used by Sawatzky, *et al.*, who performed similar studies with inductive specimen heating as Urbanic and Heidrick. Sawatzky reached an important improvement of the specimen temperature homogeneity by only optimizing the geometry of the specimen and *registered considerably increased reaction rates*[43] [emphasis added].

Radiative heat losses in an experiment conducted with inductive heating cause a specimen's zirconium-steam reaction rates to *decrease* below what they would be if there were no radiative heat losses. The very experiments that the Baker-Just correlation is primarily based on would have had radiative heat losses that decreased zirconium-steam reaction rates. Lemmon and Bostrom's experiments certainly did not replicate the oxidation kinetics that would occur in a nuclear reactor's core, in the event of a LOCA. Yet the Baker-Just correlation—required by Appendix K to Part 50 I.A.5—is almost entirely based on the results of their experiments. This fact alone is evidence that the Baker-Just correlation is *likely* inadequate for use in computer safety models like AREVA's EXEM BWR-2000 Evaluation Model. Section III of these comments presents far more conclusive evidence that the Baker-Just correlation is indeed inadequate for use in computer safety models.

III. Experiments in which Zirconium-Steam Reaction Rates Occurred that Exceed the Rates Predicted by Computer Safety Models

In Section III, Commenter provides information about experimental results that indicate the currently used zirconium-steam reaction correlations, such as the Baker-Just

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

correlation, are inadequate for use in computer safety models like the NRC's TRACE code and AREVA's EXEM BWR-2000 Evaluation Model. When using the currently used zirconium-steam reaction correlations, computer safety models *under-predict* the zirconium-steam reaction rates that occurred in the experiments discussed in Section III. Computer safety models are supposed to *over-predict* reaction rates in order to ensure an adequate margin of safety. The experimental results discussed in Section III are evidence that the NRC and nuclear industry's computer safety models under-predict the zirconium-steam reaction rates that would occur in the event of a design-basis accident (LOCA), meaning that the amendment request for the EPU for BFN should be denied.

III.A. Oxidation Models Are Unable to Predict the Fuel-Cladding Temperature Escalation that Commenced at "Low Temperatures" in the PHEBUS B9R-2 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 uranium dioxide (UO_2) fuel rods. The B9R test was conducted in two parts: the B9R-1 test and the B9R-2 test.[44] A 1996 European Commission report states that the B9R-2 test had an unexpected fuel-cladding temperature escalation in the mid-bundle region (see Figure 1); the highest temperature escalation rates were from 20°C/sec (36°F/sec) to 30°C/sec (54/°C/sec).[45]

Discussing PHEBUS B9R-2, the 1996 European Commission report states:

> The B9R-2 test (second part of B9R) illustrates the oxidation in different cladding conditions representative of a pre-oxidized and fractured state. This state results from a first oxidation phase (first part name B9R-1, of the B9R test) terminated by a rapid cooling-down phase. During B9R-2, an unexpected strong escalation of the oxidation of the remaining Zr occurred when the bundle flow injection was switched from helium to steam while the maximum clad temperature was equal to 1300 K [1027°C (1880°F)]. *The current oxidation model was not able to predict the strong heat-up rate observed* even taking into account the measured large clad deformation and the double-sided oxidation (final state of the cladding from macro-photographs).

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[44] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 311.

[45] T.J. Haste *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents," European Commission, Report EUR 16695 EN, 1996, p. 33.

... No mechanistic model is currently available to account for enhanced oxidation of pre-oxidized and cracked cladding[46] [emphasis added].

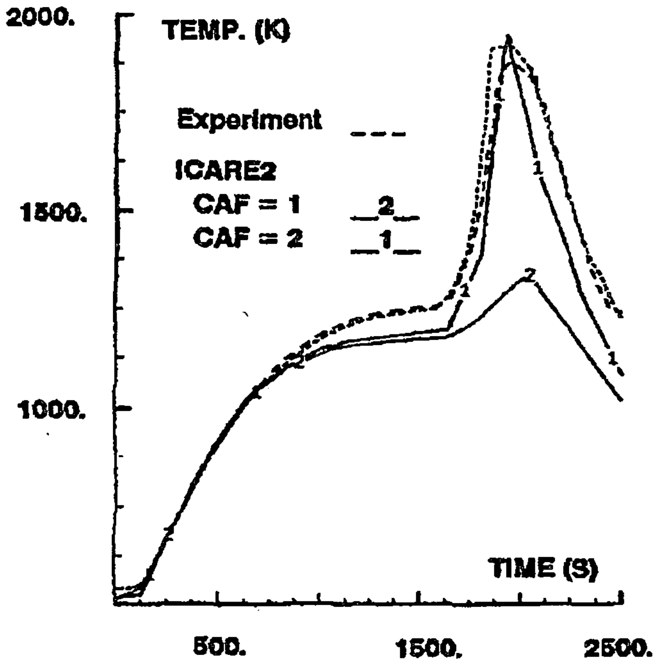

Figure 1. Local Cladding Temperature vs. Time in the PHEBUS B9R-2 Test[47]

Today, in 2016, oxidation models still cannot accurately predict the local fuel-cladding temperature escalation that commenced in PHEBUS B9R-2 in steam when local fuel-cladding temperatures were 1027°C (1880°F). The PHEBUS B9R-2 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just correlation, are inadequate for use in computer safety models like AREVA's EXEM BWR-2000 Evaluation Model.

The fact that PHEBUS B9R-2 was conducted with a pre-oxidized test bundle makes its results particularly applicable to high burnup fuel. High burnup fuel rods would also be "pre-oxidized": when high burnup (and other) fuel rods are discharged from the reactor core and loaded into the spent fuel pool, the fuel cladding can have local zirconium dioxide (ZrO_2) "oxide" layers that are up to 100 μm thick (or greater); there

[46] *Id.*, p. 126.
[47] G. Hache, R. Gonzalez, B. Adroguer, Institute for Protection and Nuclear Safety, "Status of ICARE Code Development and Assessment," in NRC "Proceedings of the Twentieth Water Reactor Safety Information Meeting," NUREG/CP-0126, Vol. 2, 1992, (ADAMS Accession No: ML042230126), p. 312.

can also be local crud layers on top of the oxide layers, which can sometimes also be up to 100 μm thick.

III.B. "Low Temperature" Oxidation Rates Are Under-Predicted for FLECHT Run 9573

Westinghouse's "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report" (hereinafter: "WCAP-7665") 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."[48] The FLECHT tests were conducted with bundles of heater rods sheathed in zirconium alloy (Zircaloy) cladding. Runaway oxidation was not expected to occur in any of the tests; however, the FLECHT Run 9573 test bundle incurred runaway oxidation (see Figure 2).

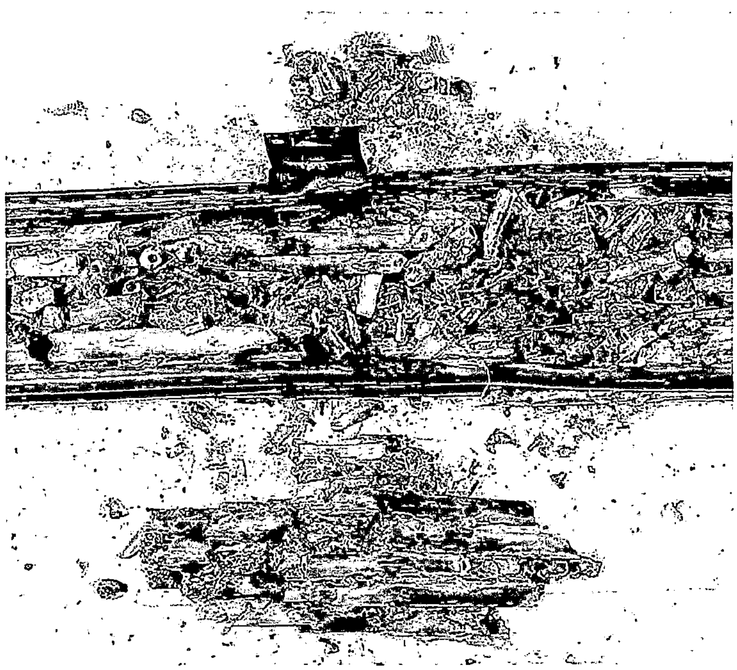

Figure 2. Section of the FLECHT Run 9573 Test Bundle that Incurred Runaway Oxidation

[48] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, (ADAMS Accession No: ML070780083), p. 1.1.

The FLECHT Run 9573 test bundle incurred runaway oxidation around its seven foot elevation. WCAP-7665 states: "Post-test bundle inspection indicated a locally severe damage zone within approximately ±8 inches of a Zircaloy grid at the 7 ft elevation. The heater rod failures were apparently caused by localized temperatures in excess of 2500°F." WCAP-7665 also states: "During the test, heater element failures started at 18.2 seconds... At the time of the initial failures, midplane [at the 6 foot elevation] 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)."[49]

The NRC conducted TRACE code computer simulations of FLECHT Run 9573 and found that TRACE *under-predicted* temperatures that were reported by Westinghouse at the 7 ft elevation of the test bundle. On November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, sent Commenter an e-mail regarding the NRC's TRACE computer simulation of FLECHT Run 9573. In his e-mail, Mr. Mohseni disclosed findings of "the completed simulation [for] the cladding and steam temperatures at the 7-ft elevation (at 18 seconds)."[50]

TRACE *under-predicted* cladding and steam temperatures at the 7-foot elevation of the FLECHT Run 9573 test bundle. TRACE is supposed to *over-predict* temperatures in order to ensure an adequate margin of safety. The Baker-Just and Cathcart-Pawel zirconium-steam reaction correlations were used for the TRACE simulations. The TRACE simulations need to be considered as evidence that the NRC and nuclear industry's computer safety models under-predict the zirconium-steam reaction rates that would occur in the event of a design-basis accident (LOCA).

[49] *Id.*, p. 3.97.

[50] Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, e-mail to Mark Leyse, regarding the NRC's TRACE computer simulation of the FLECHT Run 9573 test bundle, November 24, 2015, (ADAMS Accession No: ML15341A160).

III.C. FLECHT Run 9573—a Comparison between Computer Safety Model Predictions and the Results Westinghouse Reported

According to Mr. Mohseni's e-mail, when the TRACE code used the Cathcart-Pawel and Baker-Just correlations, it predicted *cladding* temperatures of 1526 K (2287°F) and 1561 K (2350°F), respectively. And, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *steam* temperatures of 1370 K (2006°F) and 1397 K (2055°F), respectively. Those are predicted cladding and steam temperatures for the FLECHT Run 9573 test bundle at the 7-ft elevation, at 18 seconds.[51]

Westinghouse reported that at 18.2 seconds, heater rod failures occurred around the 7 foot elevation when *cladding* temperatures were in excess of 1644 K (2500°F). (Who knows how high the cladding temperatures actually were; they could have been hundreds of degrees Fahrenheit higher than 1644 K (2500°F).)

And Westinghouse reported that at 16.0 seconds, a steam probe at the 7 foot elevation recorded *steam* temperatures that exceeded 1644 K (2500°F). And a Westinghouse memorandum stated that after 12 seconds, the steam-probe thermocouple recorded "an extremely rapid rate of temperature rise (over 300°F/sec)."[52] (Who knows how high the steam temperatures actually were at 18 seconds; they were likely hundreds of degrees Fahrenheit higher than 1644 K (2500°F).)

Taking the time difference of 0.2 seconds (between 18 and 18.2 seconds) into account, when TRACE used the Cathcart-Pawel and Baker-Just correlations, it predicted *cladding* temperatures that were at least 200°F and 140°F lower, respectively, than the temperatures Westinghouse reported. That is *non-conservative*.

When TRACE used the Cathcart-Pawel and Baker-Just correlations, at 18 seconds it predicted *steam* temperatures that were about 500°F and 450°F lower, respectively, than the temperatures Westinghouse measured at 16 seconds. Westinghouse also reported that after 12 seconds, steam temperatures were increasing at a rate greater than

[51] *Id.*

[52] Robert H. Leyse, Westinghouse, Nuclear Energy Systems, Test Engineering, Memorandum RD-TE-70-616, "FLECHT Monthly Report," December 14, 1970. This Memorandum is available at Appendix I of PRM-50-93. See Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No: ML093290250), Appendix I.

300°F/sec. So steam temperatures were even greater at 18 seconds than they were at 16 seconds. Hence, the TRACE predictions for steam temperatures are *non-conservative.*

The FLECHT Run 9573 results indicate that the currently used zirconium-steam reaction correlations, such as the Cathcart-Pawel and Baker-Just correlations, are inadequate for use in computer safety models like the NRC's TRACE code and AREVA's EXEM BWR-2000 Evaluation Model.

III.D. "Low Temperature" Oxidation Rates Are Under-Predicted for the CORA-16 Experiment

When Oak Ridge National Laboratory ("ORNL") investigators compared the results of the CORA-16 experiment—a BWR severe fuel damage test, simulating a meltdown, conducted with a multi-rod zirconium alloy bundle—with the predictions of computer safety models, they found that the zirconium-steam reaction rates that occurred in the experiment were under-predicted. The investigators concluded that the "application of the available Zircaloy oxidation kinetics models [zirconium-steam reaction correlations] causes the low-temperature [1652-2192°F] oxidation to be underpredicted."[53]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16.[54] However, it is *unsubstantiated* that cladding strain actually increased reaction rates.

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, the NRC has pointed out that an NRC report, NUREG/CR-4412,[55] "explain[s] that under *certain* conditions ballooning

[53] 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.
[54] 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.
[55] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, (ADAMS Accession No: ML083400371).

and deformation of the cladding can increase the available surface area for oxidation, thus enhancing the apparent oxidation rate"[56] [emphasis not added].

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA or [severe accident] will permit [fuel] cladding deformation (ballooning and possibly rupture) to occur because the fuel rod internal pressure may be greater than the external (coolant) pressure. In this case, oxidation and deformation can occur simultaneously. This in turn may result in an apparent enhancement of oxidation rates because: 1) ballooning increases the surface area of the cladding and permits more oxide to form per unit volume of Zircaloy and 2) the deformation may crack the oxide and provide increased accessibility of the oxygen to the metal. However deformation generally occurs before oxidation rates become significant; *i.e.*, below 1000°C [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[57]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[58] NUREG/CR-4412 also explains that "it is possible to make a very crude estimate of the expected average enhancement of oxidation kinetics by deformation;"[59] the report provides a graph of the "rather sparse"[60] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[61]

NUREG/CR-4412 has a brief description of the rather sparse data; in one case, two investigators (Furuta and Kawasaki), who heated specimens up to temperatures between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[62]

In fact, NUREG/CR-4412 states that only one pair of investigators (Bradhurst and Heuer) conducted tests that encompassed the temperature range—1652°F to 2192°F—in

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[56] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, (ADAMS Accession No: ML112211930), p. 3.
[57] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 27.
[58] *Id.*, pp. 27, 30.
[59] *Id.*, p. 30.
[60] *Id.*
[61] *Id.*, p. 29.
[62] *Id.*, p. 30.

which zirconium-steam reaction rates were under-predicted for CORA-16. Bradhurst and Heuer reported that "[m]aximum enhancements occurred at slower strain rates. ... However, the overall weight gain or average oxide thickness in [the Zircaloy-2 specimens] was only minimally increased because of the localization effects of cracks in the oxide layer." [63] A second report states that "Bradhurst and Heuer...found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[64] (In CORA-16, in the temperature range from 1652°F to 2192°F, cladding strain would have occurred over a brief period of time, tens of seconds, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain may only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater.[65]

Furthermore, ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F temperature range for CORA-16.

There is also one phenomenon the NRC did not consider in its 2011 analysis of CORA-16: "[t]he swelling of the [fuel] cladding...alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,"[66] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA

[63] *Id.*
[64] F. J. Erbacher, S. Leistikow, "A Review of Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident," Kernforschungszentrum Karlsruhe, KfK 3973, September 1985, p. 6.
[65] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, p. 30.
[66] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, (ADAMS Accession No: ML041950059), p. 43.

experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

CORA-16 is an example of an experiment that had zirconium-steam reaction rates that were under-predicted in the "low temperature" range from 1652°F to 2192°F by computer safety models. The CORA-16 results indicate that the currently used zirconium-steam reaction correlations, such as the Baker-Just correlation, are inadequate for use in computer safety models like AREVA's EXEM BWR-2000 Evaluation Model.

III.E. Computer Safety Models Fail to Accurately Predict the Onset of the Fuel-Cladding Temperature Escalation that Commenced in the LOFT LP-FP-2 Experiment (in the Design-Basis Accident Temperature Range)

In the LOFT LP-FP-2 experiment, there was a fuel-cladding temperature escalation that commenced when fuel-cladding temperatures were lower than the 2200°F PCT limit.

Computer safety models have failed to accurately predict the onset of the fuel-cladding temperature escalation that occurred in the LOFT LP-FP-2 experiment.

Regarding a fairly recent computer safety model (ASTEC V1.3 code) simulation of the LOFT LP-FP-2 experiment, a 2010 paper, "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation" states:

> The onset of core uncovery and heat-up was very well reproduced by ASTEC (fig. 17), but the onset of temperature escalation in the upper part of the CFM [center fuel module] was delayed.[67]

In "Recent Advances in ASTEC Validation on Circuit Thermal-Hydraulic and Core Degradation," in figure 17 (see Figure 3 of these comments), the graph of the cladding-temperature values in the ASTEC V1.3 simulation of the LOFT LP-FP-2 experiment depicts that the onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature greater than 1700 K (2600°F); figure 17 (see Figure 3 of these comments) also shows that in the experiment the actual onset of the temperature escalation (at the 1.067 m elevation) commenced at a temperature well below 1500 K

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

(2240°F)—definitely below 2200°F.[68] Hence, the difference between the calculated and actual experimental value for the onset of the temperature escalation (at the 1.067 m elevation) is greater than 200 K (360°F)—a significant difference.

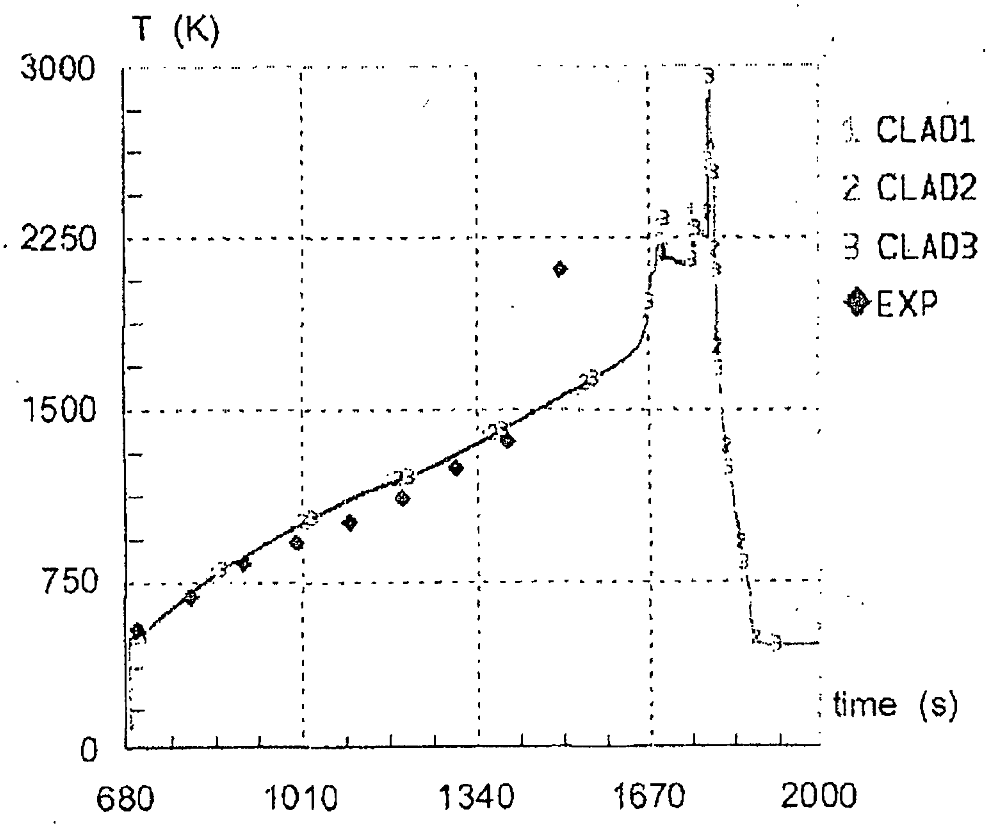

Fig. 17. LOFT-LP-FP-2 CFM clad temperature at elevation 1.067 m.

Figure 3. Onset of the Temperature Escalation that Occurred in the LOFT LP-FP-2 Experiment (at the 1.067 m Elevation)[69]

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[68] *Id.*
[69] *Id.*

III.F. An Experiment for which the Quantity of Hydrogen Produced by the Zirconium-Steam Reaction at about 1800°F Is Under-Predicted by Computer Safety Models: The FRF-1 Experiment

The First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") experiment—conducted in the Transient Reactor Test Facility ("TREAT") facility—was not a large-scale experiment yet UCS and the authors of a report on the FRF-1 experiment[70] claimed that, as of 1971, it simulated "the most realistic loss-of-coolant accident conditions of any experiment to date."[71]

Data from the FRF-1 experiment indicates that computer safety models under predict the quantity of hydrogen produced by the Zircaloy-steam reaction. In the experiment, at fuel rod temperatures of about 1800°F, the Zircaloy-steam reaction generated 1.2 ± 0.6 liters of hydrogen. In the Indian Point Unit 2 ("IP-2") licensing hearing, Westinghouse, which had performed experimental simulations of loss-of-coolant accidents, and conducted computer simulations of such accidents, testified that their computer safety models predicted that there would be no zirconium-steam reaction at 1800°F—that no hydrogen would be produced in a loss-of-coolant accident if local temperatures of the fuel rods were to reach 1800°F.[72]

In the IP-2 licensing hearing, Dr. Jack Roll of Westinghouse contended that data from the FRF-1 experiment was not reliable, because "the measurement of the extent of [zirconium-steam] reaction was in fact by an inferred route, and there were no direct measurements taken," that "[t]here was a large uncertainty in the measurement of total hydrogen evolution during the experiment," and that there was "an uncertainty in the temperatures of the fuel [rods] during the experiment."[73] Westinghouse concluded that it is not possible to know if the data from the FRF-1 experiment actually demonstrated that

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[70] R. A. Lorenz, D. O. Hobson, G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," ORNL-4635, March 1971.

[71] Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, Springer-Verlag, New York, 2000, p. 43.

[72] 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 1, 1971, (ADAMS Accession No. ML100350644), pp. 2152-2153.

[73] 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 2, 1971, (ADAMS Accession No. ML100350642), pp. 2297-2299.

the extent of the zirconium-steam reaction was higher (or much higher) than would be predicted by computer safety models.

Unfortunately, there was not a means to confirm if Westinghouse's claims were correct or not, because the Atomic Energy Commission decided to discontinue funding for the TREAT facility loss-of-coolant accident experimental program.[74] The FRF-1 experiment could not be replicated; its results could not be confirmed.

IV. Conclusion: The Amendment Request for the EPU for BFN Should Be Denied

BEST/MATRR alleges that non-conservative computer safety model analyses were performed in order to justify the EPU for BFN. When using the Baker-Just correlation, computer safety models like AREVA's EXEM BWR-2000 Evaluation Model *under-predict* the zirconium-steam reaction rates that occurred in experiments discussed in Section III. Computer safety models are supposed to *over-predict* reaction rates in order to ensure an adequate margin of safety. The experimental results discussed in Section III are evidence that AREVA's EXEM BWR-2000 Evaluation Model under-predict the zirconium-steam reaction rates that would occur in the event of a design-basis accident (LOCA), meaning that the amendment request for the EPU for BFN should be denied.

Respectfully submitted on behalf of BEST/MATRR,

/s/

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[74] W. B. Cottrell, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971," ORNL-TM-3411, July 1971, p. x.