September 9, 2016

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE SECRETARY OF THE COMMISSION

In the Matter of: :
:
TENNESSEE VALLEY AUTHORITY :
(Browns Ferry Nuclear Plant Units 1, 2, and 3;:
Docket Nos. 50-259, 50-260, and 50-296;:
NRC-2016-0118) :

BELLEFONTE EFFICIENCY & SUSTAINABILITY TEAM/ MOTHERS AGAINST TENNESSEE RIVER RADIATION'S HEARING REQUEST AND PETITION TO INTERVENE REGARDING TENNESSEE VALLEY AUTHORITY'S LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATES FOR BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.309(f) and a notice published by the U.S. Nuclear Regulatory Commission ("NRC") in the Federal Register Vol. 81, No. 128, July 5, 2016, pages 43661-43669, the Bellefonte Efficiency & Sustainability Team/ Mothers Against Tennessee River Radiation ("BEST/MATRR" or "Petitioner") requests a hearing and seeks leave to intervene in the Tennessee Valley Authority's ("TVA") license amendment request ("LAR") for extended power uprates ("EPU") for Browns Ferry Nuclear Plant ("BFN") Units 1, 2, and 3; NRC-2016-0118. BEST/MATRR opposes the granting of the license amendment. This petition sets forth BEST/MATRR's interests in this proceeding, the reasons this intervention should be granted, and specific contentions we seek to have addressed. As demonstrated below, BEST/MATRR has representational standing, through its members, to make this request.

II. DEMONSTRATION OF STANDING

BEST/MATRR has numerous members who reside and/or own property within a 50 mile radius of the Browns Ferry Nuclear Plant. Some members have been active participants in TVA and NRC Browns Ferry meetings since the 1970s. BEST/MATRR has recorded high radiation levels downwind of Browns Ferry, and produced a report on the Browns Ferry reactors, emissions and area health issues in 2013.[1]

Bellefonte Efficiency & Sustainability Team ("BEST"), based in Scottsboro, Alabama, downwind of Browns Ferry, 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 TVA's 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. In February 2008, BEST became the Tennessee Valley Chapter of the 501©(3) nonprofit environmental organization Blue Ridge Environmental Defense League (BREDL). 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 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 beautiful mountains and an unpolluted, healthy river. 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. MATRR provides research, educational materials, presentations and a website for the group. BEST members voted to add MATRR to the organizational name in June 2014, becoming BEST/MATRR.

BEST/MATRR is a local chapter of Blue Ridge Environmental Defense League (BREDL), which is a regional, community-based 501(c)(3) nonprofit environmental

[1] Joseph Mangano and Gretel Johnston, Radioactive Emissions and Health Hazards Surrounding Browns Ferry Nuclear Power Plant in Alabama , BEST/MATRR, June 4, 2013. http://best-matrr.org/pdfs/AL_BFN_Report_2013-final-dig2.pdf

organization working in Virginia, North Carolina, South Carolina, Tennessee, Alabama and Georgia. BREDL's founding principles are earth stewardship, environmental democracy, social justice, and community empowerment. BREDL encourages government agencies and citizens to take responsibility for conserving and protecting our natural resources and protecting public health. BREDL also functions as a "watchdog" of the environment, monitoring issues and holding government officials accountable for their actions. BREDL is a league of community groups called "chapters." BREDL and its chapters are unitary, with a common incorporation, financial structure, board of directors and executive officer.

Under 10 CFR § 2.309(d), a request for hearing or petition for leave to intervene must state: 1) the name and address of petitioner; 2) the nature of the petitioner's right under the Atomic Energy Act to be made a party to the proceeding; 3) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and 4) the possible effect of any order that may be entered in the proceeding on the petitioner's interest. Other standing requirements are found in NRC case law. See Pacific Gas and Electric Co. (Diablo Canyon Power Plant Independent Spent Fuel Storage Installation), LBP-02-23, 56 NRC 413, 426 (2002).[2]

BEST/MATRR seeks admission of its contention in order to protect its members' interest in a clean and healthy environment, including protection from the health and

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[2] In determining whether a petitioner has sufficient interest to intervene in a proceeding, the Commission has traditionally applied judicial concepts of standing. See Metropolitan Edison Co. (Three Mile Island Nuclear station, Unit 1), CLI-83-25, 18 NRC 327, 332 (1983) (citing Portland General Electric Co. (Pebble Springs Nuclear Plant, Units 1 and 2), CLI-76-27, 4 NRC 610 (1976)). Contemporaneous judicial standards for standing require a petitioner to demonstrate that (1) it has suffered or will suffer a distinct and palpable harm that constitutes injury-in-fact within the zone of interests arguably protected by the governing statutes (e.g., the Atomic Energy Act of 1954 (AEA), the National Environmental Policy Act of 1969 (NEPA)); (2) the injury can be fairly traced to the challenged action; and (3) the injury is likely to be redressed by a favorable decision. See Carolina Power & Light Co. (Shearon Harris Nuclear Power Plants), LBP-99-25, 50 NRC 25, 29 (1999). An organization that wishes to intervene in a proceeding may do so either in its own right by demonstrating harm to its organizational interests, or in a representational capacity by demonstrating harm to its members. See Hydro Resources, Inc. (2929 Coors Road, Suite 101, Albuquerque, NM 87120), LBP-98-9, 47 NRC 261, 271 (1998). To intervene in a representational capacity, an organization must show not only that at least one of its members would fulfill the standing requirements, but also that he or she has authorized the organization to represent his or her interests. See Private Fuel Storage, L.L.C. (Independent Fuel Storage Installation), LBP-98-7, 47 NRC 142, 168, aff'd on other grounds, CLI-98-13, 48 NRC 26 (1998).

environmental hazards posed by TVA's LAR for EPUs for BFN Units 1, 2, and 3. BEST/MATRR has standing to intervene through members who live, work, and/or own property within 50 miles of BFN Units 1, 2, and 3, located in Limestone County, and their interests may be affected by the results of the proposal for EPUs for BFN Units 1, 2, and 3. Furthermore, BEST/MATRR has standing to intervene, because "dose consequences can increase by the percentage change in the power level of a license amendment request for an extended power uprate, [so] the 50-mile presumption for standing applies to hearings on such amendment requests."[3] See Houston Lighting and Power Co. (South Texas Project, Units 1 and 2), LBP-79-10, 9 NRC 439, 443 (1979); see also Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-522, 9 NRC 54, 56 (1979); additionally see PPL Susquehanna LLC (Susquehanna Steam Electric Station, Units 1 & 2), LBP-07-10, 66 NRC 1, 19 (2007).[4] Given that dose consequences can increase by the percentage change in the power level of a LAR for an extended power uprate, BEST/MATRR members health, safety, property value, and means of livelihood could be adversely affected if the NRC grants TVA's LAR for EPUs for BFN Units 1, 2, and 3. BEST/MATRR has attached declarations from members **Tom Moss, Nancy Muse, Cheryl Carlson, Roy Simmons, Juliana Larson, Lynn Larson, Albert Bates, Gretel Johnston, Paige Winslett and Steve Johnston** who have authorized BEST/MATRR to bring this legal action on their behalves.

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[3]NRC, "Commission, Appeal Board and Licensing Board Decisions: July 1972 – September 2010," NUREG-0386, Digest 16, June 2011 (located at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0386/d16/sr0386d16.pdf: last visited on August 27, 2016), Prehearing Matters, p. 72.

[4] "A petitioner may base its standing upon a showing that its residence, or that of its members, is within the geographical zone that might be affected by an accidental release of fission products." Houston Lighting and Power Co. (South Texas Project, Units 1 and 2), LBP-79-10, 9 NRC 439, 443 (1979). "Distances of as much as 50 miles have been held to fall within this zone." Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-522, 9 NRC 54, 56 (1979). "Given that dose consequences can increase by the percentage change in the power level of a license amendment request for an extended power uprate, the 50-mile presumption for standing applies to hearings on such amendment requests." PPL Susquehanna LLC (Susquehanna Steam Electric Station, Units 1 and 2), LBP-07-10, 66 NRC 1, 19 (2007). See NRC, "Commission, Appeal Board and Licensing Board Decisions: July 1972 – September 2010," NUREG-0386, Digest 16, June 2011 (located at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0386/d16/sr0386d16.pdf: last visited on August 27, 2016), Prehearing Matters, pp. 62, 72.

III. FACTUAL BACKGROUND

On July 5, 2016, the NRC published a notice in the Federal Register notice soliciting public comments on the proposed EPUs for BFN Units 1, 2, and 3.[5] The same notice also stated that "[a] request for a hearing must be filed by September 6, 2016."[6]

The July 5, 2016 Federal Register notice 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 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.[7]

The AREVA loss-of-coolant accident ("LOCA") analyses that were conducted to help justify the LAR for the EPUs for BFN Units 1, 2, and 3 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 EPUs for BFN Units 1, 2, and 3 discussed in ANP-

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[5] 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.

[6] *Id*., p. 43661.

[7] 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.

3377NP, ANP-3378NP, and ANP-3384NP, predicted PCTs of 2030°F,[8] 2008°F,[9] and 2086°F,[10] respectively.

The overall predicted PCT is 2030°F for ATRIUM 10XM fuel, which is used at BFN Units 1, 2, and 3.[11] AREVA's analyses "were performed for a [reactor] core composed entirely of ATRIUM 10XM fuel at beginning-of-life (BOL) conditions. 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)."[12]

And the overall predicted PCT is 2086°F for ATRIUM 10 fuel, which is used at BFN Units 1, 2, and 3.[13] 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."[14]

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[8] 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.

[9] 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.

[10] 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.

[11] 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.

[12] *Id*., p. 1.1.

[13] 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.

[14] *Id*., p. 1.2.

AREVA has stated that "[t]he models and computer codes used by AREVA for LOCA analyses [regarding the EPUs for BFN Units 1, 2, and 3] 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.[15]

The EXEM BWR-2000 Evaluation Model LOCA calculations for the EPUs for BFN Units 1, 2, and 3 "were performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46."[16] 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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[17]

IV. 10 C.F.R. 50 APPENDIX K I.A.5: HISTORICAL BACKGROUND

A. Experimental Data Demonstrates that 10 C.F.R. 50 Appendix K, I.A.5 Is Non-Conservative

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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[18]

10 C.F.R. 50 Appendix K, I.A.5 is non-conservative. As documented in an affidavit submitted on behalf of the Petitioner by Mark Leyse ("Leyse Declaration") (Attachment A), experimental data, along with appropriate citations, demonstrates that the Baker-Just correlation is inadequate for use in computer safety models like AREVA's EXEM BWR-2000 Evaluation Model.

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15 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.

16 *Id*.

17 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 09/02/16).

18 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 09/02/16).

The Baker-Just correlation—used in Appendix K to Part 50 ECCS evaluation calculations—dates back to 1962.[19] The Baker-Just correlation is primarily based on data from Alexis Lemmon and W. A. Bostrom's experiments,[20] which were conducted in the 1950s.[21] 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.[22] (Lemmon's specimens were Zircaloy-2 cylinders that were 2.0 inches long and 0.5 inches in diameter.[23]) There are radiative heat losses in experiments conducted with inductive heating, which affect a specimen's oxidation kinetics.[24]

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[25]] 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

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[19] 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).

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

[21] 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).

[22] 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.

[23] 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), p. C-4.

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

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

> homogeneity by only optimizing the geometry of the specimen and *registered considerably increased reaction rates*[26] [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.

Results of larger scale experiments discussed in the Leyse Declaration, along with appropriate citations, present far more conclusive evidence that the Baker-Just correlation is indeed inadequate for use in computer safety models like AREVA's EXEM BWR-2000 Evaluation Model. For example, as discussed in the Leyse Declaration, on November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, disclosed to Leyse 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.[27] A computer safety model is supposed to *over-predict* temperatures in order to ensure an adequate margin of safety.

If a reactor's power level is set too high after being "qualified" by LOCA analyses that do not ensure an adequate margin of safety, a real-life LOCA would lead to a beyond design-basis accident. In other words, if a reactor's power level is set too high after being "qualified" by LOCA analyses that do not ensure an adequate margin of safety, in the event of a LOCA, the criteria set forth in 10 C.F.R. § 50.46(b) would be violated: 1) the

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[26] *Id*.

[27] 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).

PCT would exceed 2200°F; 2) the maximum cladding oxidation would locally exceed 0.17 times the total cladding thickness before oxidation; 3) the total amount of hydrogen generated from the chemical reaction of the cladding with steam would exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; 4) the reactor core geometry would not remain amenable to cooling; 5) there would not be long-term cooling of the reactor core; the core temperature would not be maintained at an acceptably low value and decay heat would not be removed for the extended period of time required, as a consequence of the long-lived decay of fission products that remain in the core.

B. In 1971, in the Licensing Hearings for Indian Point Nuclear Plant Unit 2, Union of Concerned Scientists Alleged that the Baker-Just Correlation Is Inadequate for Use in Computer Safety Models that Simulate Loss-of-Coolant Accidents

The Indian Point Nuclear Plant Unit 2 licensing hearings were held because the Citizens Committee for the Protection of the Environment and other intervenors opposed the licensing of Indian Point's Unit 2 reactor. The Union of Concerned Scientists ("UCS") provided technical expertise for the Citizens Committee. UCS alleged that Westinghouse lacked a foundation for its claim that its emergency systems would prevent a meltdown in the event of a loss-of-coolant accident.[28] Prior to the hearings, UCS had stated that until an independent third party reviewed and assured the performance of emergency systems they couldn't "support the licensing and operation of any additional power reactors in the United States."[29]

In the Indian Point Unit 2 licensing hearings, UCS contended that results of the First Transient Experiment of a Zircaloy Fuel Rod Cluster ("FRF-1") experiment, which was conducted at the Transient Reactor Test Facility ("TREAT"), a nuclear reactor in

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[28] Daniel F. Ford, Henry W. Kendall, James J. MacKenzie, "A Critique of the New A. E. C. Design Criteria for Reactor Safety Systems," Union of Concerned Scientists, October 1971; I.A. Forbes, D.F. Ford, H.W. Kendall, J.J. MacKenzie, "Nuclear Reactor Safety: An Evaluation of New Evidence," *Nuclear News*, 14, No. 9, September 1971.

[29] Henry W. Kendall's *A Distant Light: Scientists and Public Policy* has reprinted I.A. Forbes, D.F. Ford, H.W. Kendall, J.J. MacKenzie, "Nuclear Reactor Safety: An Evaluation of New Evidence," *Nuclear News*, 14, No. 9, September 1971. For the quoted passage see Henry W. Kendall, *A Distant Light: Scientists and Public Policy*, (New York: Springer-Verlag, 2000), p. 35.

Idaho, indicated that the zirconium-steam reaction is more severe than industry claimed.[30] According to scientists at Oak Ridge National Laboratory, as of 1971, the FRF-1 experiment "was conducted under the most realistic loss-of-coolant accident conditions of any experiment to date."[31] UCS believed that results of the FRF-1 experiment indicated Con Edison's license application should be re-evaluated.[32]

It was reported that industry's computer safety model (using the Baker-Just correlation) vastly under-predicted the extent of the zirconium-steam reaction that occurred in the FRF-1 experiment,[33] indicating the model was unfit for simulating the type of LOCAs that could occur at Indian Point.

In fact, data from the FRF-1 experiment indicates that computer safety models (using the Baker-Just correlation) 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 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 (using the Baker-Just correlation) predicted that there would be no zirconium-steam reaction at 1800°F—that no hydrogen would be produced in a LOCA if local temperatures of the fuel rods were to reach 1800°F.[34]

However, in the Indian Point Unit 2 licensing hearings, Dr. John Bernard Roll, a manager in Westinghouse's Nuclear Fuel Division, testified that data hadn't been accurately recorded in the FRF-1 experiment. He claimed the test results didn't prove

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[30] 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.

[31] 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, p. 75.

[32] 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. 2297-2298.

[33] 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, 2166-2167.

[34] 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.

anything.[35] Unfortunately, the AEC wasn't too eager to replicate the test with accurate data measurements in order to investigate whether its results were valid or not. It decided to kill funding for the TREAT reactor's LOCA test program.[36]

In the Indian Point Unit 2 licensing hearings, Daniel F. Ford of UCS asked a number of questions about the Baker-Just correlation. Ford questioned whether or not the Baker-Just correlation was valid; that is, he questioned whether or not the Baker-Just correlation is adequate for use in computer safety models that simulate LOCAs. Regarding Ford's questions, Dr. Roll stated: "The line of questioning between Mr. Ford and myself really questioned the validity and applicability of the assumptions which Baker and Just made, and whether or not the validity of these assumptions in any way through a question or use of the equation [the Baker-Just correlation] in the analysis of the loss of coolant accident."[37]

By alleging that the Baker-Just correlation is inadequate for use in computer safety models that simulate LOCAs, UCS also alleged that 10 C.F.R. 50 Appendix K, I.A.5 is non-conservative. 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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[38]

C. In 1971, in the Licensing Hearings for Indian Point Nuclear Plant Unit 2, Dr. John Bernard Roll, a Manager in Westinghouse's Nuclear Fuel Division, Made False Statements Under Oath, Defending the Baker-Just Correlation

In the Indian Point Unit 2 licensing hearings, after Dr. Roll testified that data hadn't been accurately recorded in the FRF-1 experiment, he testified that Westinghouse's FLECHT

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[35] 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.

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

[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 16, 1971, (ADAMS Accession No. ML100350625), p. 3863.

[38] 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 09/02/16).

tests were superior to the FRF-1 experiment in terms of replicating how fuel rods would perform in an accident. He claimed that the FLECHT results reaffirmed the validity of the industry's computer safety model (using the Baker-Just correlation) for simulating the extent of the zirconium-steam reaction that would occur in the event of a LOCA.[39] In fact, some of the FLECHT results did just the opposite.

In 1971, the year after Westinghouse's FLECHT tests had been completed, employees of Westinghouse, including Dr. Roll, testified in the licensing hearings for Indian Point Unit 2, because the Unit 2 reactor is a Westinghouse design. The testimony of Dr. Roll served to counter charges that a reactor accident would be worse than industry claimed. Dr. Roll's job was to review, interpret, and model data from experiments that simulated LOCAs.[40] When he was under oath, Dr. Roll made false statements, defending the Baker-Just correlation's use in the industry's computer safety model for simulating the extent of the zirconium-steam reaction that would occur in the event of a LOCA.

After Dr. Roll was sworn in, Leonard Trosten, the attorney representing the Indian Point Unit 2 license applicant, Con Edison, asked him to respond to UCS's allegation that industry's computer safety model (using the Baker-Just correlation) vastly under-predicted the extent of the zirconium-steam reaction that occurred in the FRF-1 experiment,[41] indicating the model was unfit for simulating the type of LOCAs that could occur at Indian Point.

After Dr. Roll testified that data hadn't been accurately recorded in the FRF-1 experiment and that the test results didn't prove anything,[42] he stated:

> I'd like to add further that we [Westinghouse] have, as a part of our work, in particular under the FLECHT program, reviewed the extent of zirc-water reaction, under what we considered to be much more representative

[39] 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. 2297-2299.
[40] 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), p. 2297.
[41] 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, 2166-2167.
[42] 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. 2298-2299.

> conditions [than those of the FRF-1 experiment], that is zircalloy clad fuel rods with our particular time and temperature histories and our particular coolant content, that is our particular water conditions, and *I believe as reported in the documentation summarized in the FLECHT reports we find very good agreement with the Baker-just equation* [correlation], and so we believe in summary that the Oak Ridge report[43] [on the FRF-1 experiment] presents a single data point to germaneness to our specific application must be questioned inasmuch as the data point was not, the test was not run to substantiate the Baker-Just equation [correlation] [emphasis added].
>
> And secondly, in summary, *the work that we have done under the FLECHT program and reported in the FLECHT reports we believe reaffirms our use of the Baker-Just equations in evaluating zirc-water reaction under our conditions of loss of coolant accident* [emphasis added].[44]

In the Indian Point Unit 2 licensing hearings, Dr. Roll failed to mention that in the FLECHT program, part of the FLECHT Run 9573 test bundle incurred thermal runaway, as a result of the heat generated by the zirconium-steam reaction.

As stated above in Section IV.A, on November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, disclosed to Leyse that an NRC (TRACE code) computer simulation (using the Baker-Just correlation) of FLECHT Run 9573, *under-predicted* cladding and steam temperatures at the elevation of the hottest section of the test's fuel rod simulators.[45] More than four decades after the Indian Point Unit 2 licensing hearings, the truth has been revealed: Dr. Roll made false statements, defending the Baker-Just correlation's use in the industry's computer safety model for simulating the extent of the zirconium-steam reaction that would occur in the event of a LOCA.

After Dr. Roll testified that the work Westinghouse had "done under the FLECHT program and reported in the FLECHT reports...reaffirms our use of the Baker-Just equations [correlation] in evaluating zirc-water reaction under our conditions of loss of

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[43] 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, p. 75.
[44] 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), p. 2299.
[45] 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).

coolant accident,"[46] Daniel F. Ford of UCS asked him a number of questions about FLECHT results and the Baker-Just correlation. Ford asked Dr. Roll to "describe the techniques of FLECHT measurement of zircalloy-water reaction that were used in your FLECHT tests."[47]

Dr. Roll answered Ford, explaining:

> The measurement that we took in evaluating the result of our FLECHT test with regard to extent of zirc-water reaction were in fact metalographic cross-sections at various enlargements from which the experienced metalographers can infer [the] nature of the phases in the cross-section. That is they can determine the portion of the original zircalloy which remains as original zircalloy. That portion which is oxygen saturated, that portion which is in fact converted to zirconium oxide. With these direct measurements at a number of cross-sections, one can then calculate explicitly the quantity of zirconium which has been converted to zirconium dioxide and the quantity of zirconium which is oxygen saturated from which you can then determine the total quantity of zirconium which has in fact reacted in some way with the oxygen.[48]

Dr. Roll further explained:

> I believe the technique of looking at zirconium and zirconium oxide is in itself a primary source of data and need not be substantiated somewhere else. The question is, how do we know what is the extent of [the] zirconium and oxygen reaction. The answer is, you know this by looking at the quantity of zirconium which has been converted to zirconium oxide.[49]

Attempting to explain that the FLECHT program data affirmed that the Baker-Just correlation is adequate for use in computer safety models that simulate LOCAs, Dr. Roll concluded:

> Let me refer to WCAP-7665 figures on page B-20, in particular and on B-23.[50] … I believe the answer to your question, does the prediction; *i.e.*,

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[46] 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), p. 2299.
[47] *Id*., p. 2300.
[48] *Id*., p. 2302.
[49] *Id*., p. 2303.
[50] See F. D. Kingsbury, J. F. Mellor, and A. P. Suda, "Materials Evaluation," Appendix B of WCAP-7665. 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), Appendix B, pp. B-20, B-23.

Baker-Just, go over the top of the data? I think the answer is essentially yes, looking particularly at the figure on page B-20.[51]

(**The figure on page B-20 of WCAP-7665 is copied below.**)

The document Dr. Roll referred to is "Materials Evaluation," Appendix B of WCAP-7665. "Materials Evaluation" explains that; "In order to properly analyze FLECHT [tests conducted with zirconium test bundles], information regarding the amount of energy released during a test by the metal-water reaction was required. The purpose of the materials evaluation portion of the FLECHT program was to determine the extent of metal-water reaction in the…tests [conducted with zirconium test bundles] and compare it with the predictions of an analytical model."[52] Westinghouse's analytical model used the Baker-Just correlation.

To conduct the materials evaluation, metallographic specimens were selected from the FLECHT program's zirconium test bundles.[53] Then the thicknesses of the oxide layers of those specimens were compared to predicted (calculated) oxide layer thicknesses that were simulated (generated under the same temperature conditions that generated the real-life selected specimens). Westinghouse's analytical model, using the Baker-Just correlation, predicted (calculated) the oxide layer thicknesses. "Materials Evaluation," Appendix B of WCAP-7665 explains that "[t]he calculated oxide thickness data…were obtained using the Baker and Just parabolic rate equation [Baker-Just correlation] and the detailed temperature-time output of the thermocouples located at the sections examined."[54]

Supporting Dr. Roll's conclusion, "Materials Evaluation," Appendix B of WCAP-7665, concluded that "[t]he Baker-Just parabolic rate equation [Baker-Just correlation]

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[51] 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), p. 2303.
[52] See F. D. Kingsbury, J. F. Mellor, and A. P. Suda, "Materials Evaluation," Appendix B of WCAP-7665. 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), Appendix B, p. B-1.
[53] *Id.*, Appendix B, pp. B-2, B-3.
[54] *Id.*, Appendix B, p. B-19.

appears to provide a satisfactory basis for determining the overall extent of metal-water reaction."[55]

Also, supporting Dr. Roll's conclusion, "Materials Evaluation," Appendix B of WCAP-7665, explains that, as shown in Figure B-12 (**see below**), "[i]t is evident that the calculated thicknesses are consistently high, with the error increasing with increasing oxide thickness. … The calculated oxide thickness data…were obtained using the Baker and Just [correlation]."[56]

[55] *Id*., Appendix B, p. B-24.
[56] *Id*., Appendix B, p. B-19.

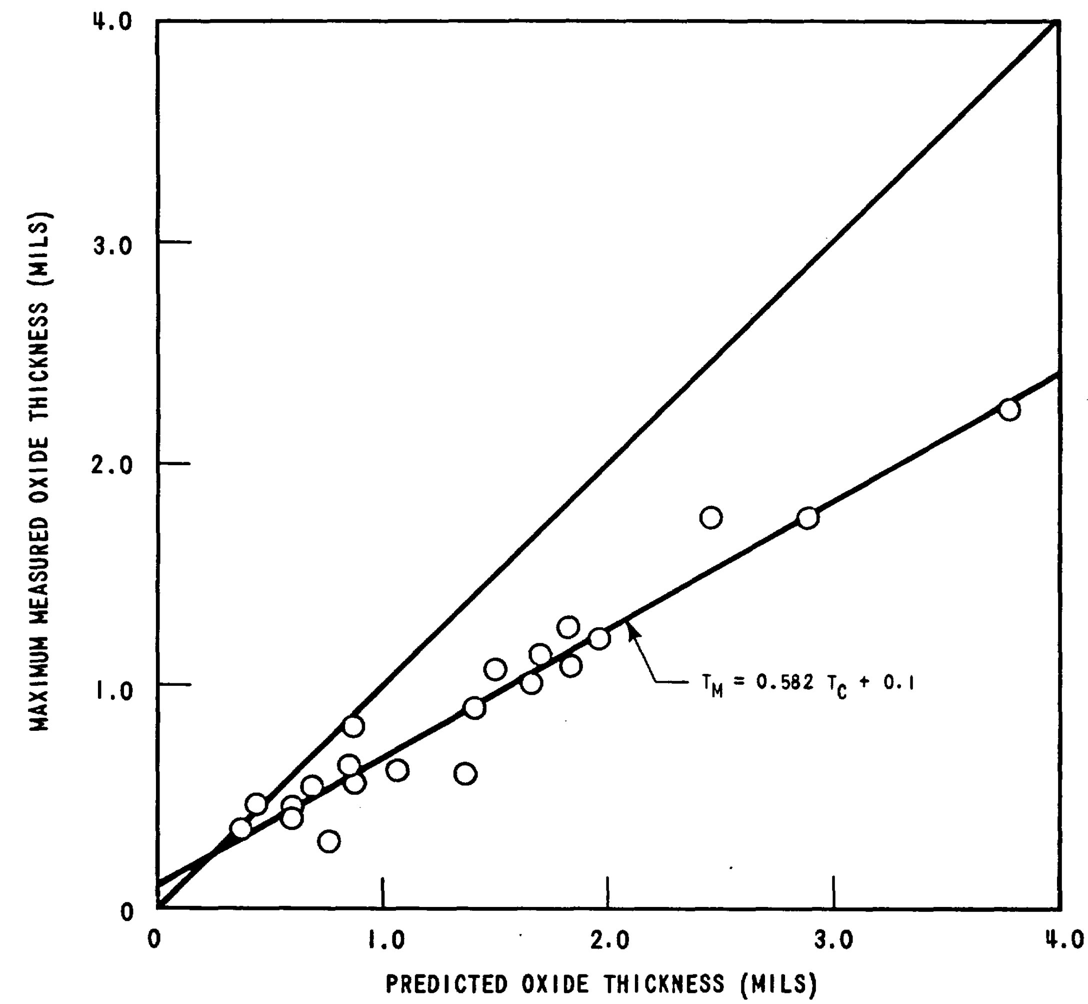


Figure B-12. Comparison of Measured versus Predicted Values of Oxide Thickness

The figure on page B-20 of WCAP-7665 that Dr. Roll referred to in his testimony in the Indian Point Unit 2 licensing hearings

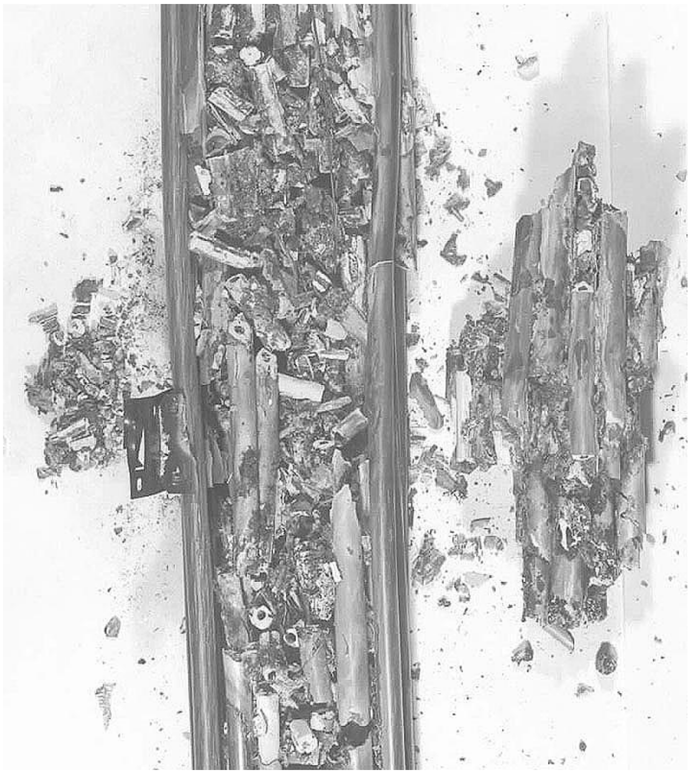
Cherry-Picking Experimental Data: The Main Problem with Dr. John Bernard Roll's Testimony

As stated above, in the Indian Point Unit 2 licensing hearings, Dr. Roll failed to mention that in the FLECHT program, part of the FLECHT Run 9573 test bundle incurred thermal runaway, as a result of the heat that was generated by the zirconium-steam reaction. Dr. Roll also failed to mention that in the materials evaluation of the FLECHT program, samples were *not taken* from the section of the FLECHT Run 9573 test bundle that incurred thermal runaway.[57] In other words, there was cherry-picking of experimental data in the FLECHT program.

A section of the FLECHT Run 9573 test bundle's zirconium cladding essentially caught on fire. The cladding burned in steam—then, when cooled, shattered like overheated glass doused with cold water. (**A photograph of the destroyed test bundle is depicted below.**) In WCAP-7665, Westinghouse referred to the severely burnt, shattered section of the FLECHT Run 9573 test bundle as the "severe damage zone" and noted that "the remainder of the [test] bundle was in excellent condition."[58]

[57] See F. D. Kingsbury, J. F. Mellor, and A. P. Suda, "Materials Evaluation," Appendix B of WCAP-7665. 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), Appendix B, p. B-4.

[58] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, (ADAMS Accession No. ML070780083), p. 3.97.



The severe-damage zone of the FLECHT test bundle from Run 9573

Metal experts at Idaho Nuclear Corporation examined metallographic specimens that were selected from the FLECHT Run 9573 test bundle as well as three other zirconium test bundles from the FLECHT program. They wanted to determine the extent of the zirconium-steam reaction that had occurred at different locations of the test bundles. However, they did *not* examine any metallographic specimens from the FLECHT Run 9573 test bundle's severe damage zone.[59] Metallographic specimens were not taken from the severe damage zone. By way of an analogy what they did would be like trying to determine how severely trees burned in a forest fire by ignoring trees reduced to ash and only examining those that had been singed.

Westinghouse is likely the company responsible for the cherry-picking. They likely only sent Idaho Nuclear Corporation sections of the test bundle that were in decent shape—or metallographic specimens that were taken sections of the test bundle that were

[59] F. D. Kingsbury, J. F. Mellor, A. P. Suda, "Materials Evaluation," Appendix B of Westinghouse's "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, (ADAMS Accession No. ML070780083), pp. B.1-B.25.

in decent shape. But who knows what actually happened? The main thing is that there *was* cherry-picking. The identity of the culprit is less important.

Regardless of who cherry-picked the "singed" cladding samples, Westinghouse was enabled to downplay the extent of the zirconium-steam reaction. This is a serious problem. In a reactor accident, the reaction between zirconium and steam generates a lot of heat and leads to a meltdown. One thing is certain, in the Indian Point Unit 2 licensing hearings, Dr. Roll, a Westinghouse employee, made false statements, defending the Baker-Just correlation. His false statements were intended to support the claim that the Baker-Just correlation is adequate for use in computer safety models that simulate LOCAs.

<u>Problems with the Metallurgical Data from the FLECHT Program</u>

Four of the FLECHT tests were conducted with bundles of heater rods sheathed in zirconium alloy (Zircaloy) cladding. Those tests are FLECHT Runs 2443, 2544, 8874, and 9573.

<u>FLECHT Runs 8874 and 9573</u>

There are significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from FLECHT Run 9573, because Westinghouse did not obtain metallurgical data from the locations of the rods from Run 9573 that incurred thermal runaway—the severe-damage zone. FLECHT run 8874 had also incurred thermal runaway. And Westinghouse did not obtain metallurgical data from the locations of the rods from Run 8874 that incurred runaway oxidation.[60] It is probable that the locations of the test bundles from Runs 8874 and 9573 that Westinghouse did examine were steam starved: the examined locations had limited oxidation because they were only exposed to a limited amount of steam.

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[60] See F. D. Kingsbury, J. F. Mellor, and A. P. Suda, "Materials Evaluation," Appendix B of WCAP-7665. 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), Appendix B, p. B-4.

It is reasonable to assume that—as in the CORA-2 experiment, in which local steam starvation conditions are postulated to have occurred[61]—in FLECHT Runs 8874 and 9573, violent oxidation essentially consumed much of the available steam, so that time-limited and local steam starvation conditions, which cannot be detected in a post-test investigation, would have occurred.

Therefore, Westinghouse's application of the Baker-Just zirconium-steam correlation (used in computer safety models) to the oxide layers on the test bundles from FLECHT Runs 8874 and 9573 were to locations that most likely were steam starved or partly steam starved (hydrogen produced by the zirconium-steam reaction would have also diluted the available steam). Clearly, that is not a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models.

Subsequently, the NRC applied the Baker-Just and Cathcart-Pawel correlations to the metallurgical data from the four FLECHT Zircaloy experiments:[62] unfortunately, the NRC did not apply the Baker-Just and Cathcart-Pawel correlations to metallurgical data from the locations of FLECHT Runs 8874 and 9573 that incurred thermal runaway. Hence, NRC's analyses are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.[63]

<u>FLECHT Runs 2443 and 2544</u>

There are also significant problems with Westinghouse's examinations of the metallographic cross-sections that were taken from test rods from FLECHT Runs 2443 and 2544.

A Westinghouse report states that two of the FLECHT experiments—Runs 2443 and 2544—with Zircaloy test bundles had unintended internal gas pressure increases, at the middle sections of the bundles, which caused the Zircaloy cladding to balloon and move away from the heat source of the internally heated rods and from the location of the

[61] 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.
[62] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, (ADAMS Accession No: ML050250359), pp. 21-22.
[63] *Id*.

thermocouples.[64] The actual temperatures of the Zircaloy cladding of the test bundles at the middle section were lower than the temperatures Westinghouse recorded. Therefore, the quantity of oxidation which occurred at the middle sections of the test bundles from FLECHT Runs 2443 and 2544, occurred at lower temperatures than Westinghouse claimed.

The thickness of each oxide layer would have been accurately measured; however, the examiners concluded that the thicknesses of the oxide layers from the middle sections of the test bundles from FLECHT runs 2443 and 2544 had been produced at higher temperatures than they were actually produced at. Hence, the metallurgical data was erroneously associated with cladding temperatures that were too high. Clearly, Westinghouse's metallurgical data from FLECHT Runs 2443 and 2544 is not valid for performing a legitimate verification of the adequacy of the Baker-Just correlation for use in computer safety models.

The NRC's subsequent analyses—which used data from FLECHT Runs 2443 and 2544—are also not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in computer safety models.[65]

(Interestingly, in Westinghouse's comparison of eight metallurgical samples from run 2443, taken from two feet above and below the midplane location, *all* of the measured oxide thicknesses *exceeded* the predicted oxide thicknesses.[66])

Problems with the Analysis of FLECHT Run 9573 Continue in the Post-Fukushima Era

45 years after the Indian Point licensing hearings, the NRC does not seem concerned that industry's computer safety models still under-predict the extent of the zirconium-steam reaction. On November 17, 2009, Mark Leyse submitted a 10 C.F.R. §

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[64] 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, (ADAMS Accession No: ML070780083), p. 3-95.

[65] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, (ADAMS Accession No: ML050250359), pp. 21-22.

[66] In all eight cases measured oxide thicknesses were less than 0.1×10^{-3} inches thick; however, all the predicted thicknesses were zero inches. See F. D. Kingsbury, J. F. Mellor, A. P. Suda, Westinghouse Electric Corporation, Appendix B, "Materials Evaluation," of "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," p. B-9.

2.802 petition for rulemaking, PRM-50-93,[67] to the NRC that discusses the section of the FLECHT Run 9573 test bundle that incurred thermal runaway—the severe-damage zone.

As part of its technical analysis of PRM-50-93, the NRC did a computer simulation of what occurred in FLECHT Run 9573. They wanted to compare the results of their simulation to the data that Westinghouse reported on FLECHT Run 9573. However, there was a big problem with the NRC's simulation. They did *not* simulate the section of the test bundle that incurred thermal runaway—the severe-damage zone.[68] (Or if they did simulate that section, they decided not to release their findings.)

By way of an analogy: what the NRC did would be like simulating a forest fire and omitting trees reduced to ash and only simulating those that had been singed. After doing such a bogus simulation one might try to argue that trees actually do not burn down in forest fires. The NRC basically did just that. They used the results of their simulation to argue that the Baker-Just correlation is adequate for use in computer safety models that simulate LOCAs.[69]

On January 31, 2013, Leyse gave a presentation to NRC Chairwoman Allison M. Macfarlane and the four NRC Commissioners. They invited Leyse to present his views on a panel addressing public participation in the NRC's rulemaking process.[70] In his presentation, Leyse discussed the NRC's computer simulation of FLECHT Run 9573. He stated: "You cannot do legitimate computer simulations of an experiment that incurred runaway oxidation by not actually modeling the section of the test bundle that incurred runaway oxidation. So, the staff's…simulations were frankly a waste of money." Leyse

[67] Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No. ML093290250).
[68] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, (ADAMS Accession No. ML12265A277), pp. 7-9.
[69] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Conservatism of 2200 degrees F, Metal-Water Reaction Rate Correlations, and 'The Impression Left from [FLECHT] Run 9573' ," October 16, 2012, (ADAMS Accession No. ML12265A277), pp. 7-9.
[70] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf).

offered to meet with the NRC staff members who were (and still are) reviewing PRM-50-93, to discuss it, "try to sort things out, expedite things."[71]

After everyone on the panel concluded their presentations, Chairwoman Macfarlane stated: "Let me first note that I think Mr. Leyse demonstrated and has been and is continuing to be in the process of demonstrating that the public actually has a lot of valuable input. The public actually knows things that people at government agencies don't know and may not be aware of, and actually, the social science literature is ripe with this information as well, confirming this is true."[72] Later on, Commissioner William Magwood assured Leyse that he and the other Commissioners would instruct their staff "to follow up on" his criticism of the NRC's computer simulation of FLECHT Run 9573.[73]

The NRC Commissioners seemed receptive to Leyse's allegation that the computer simulation of FLECHT Run 9573 was inadequate. However, a couple of months after the meeting on public participation, the NRC staff released yet more of its technical analysis of PRM-50-93, including a statement that their simulation of FLECHT Run 9573 *over*-predicted the extent of the zirconium-steam reaction.[74] The NRC staff simply reiterated their claim that the results of their simulation of FLECHT Run 9573 show that the Baker-Just correlation is adequate for use in computer safety models that simulate LOCAs.

In November 2015, after Leyse made a series of additional complaints, the NRC finally disclosed the results of a computer simulation of FLECHT Run 9573 that included the section of the test bundle that incurred thermal runaway—the severe-

[71] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf), pp. 55-56.
[72] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf), pp. 65-66.
[73] NRC, Public Participation in NRC Regulatory Decision-Making, Transcript of Proceedings, January 31, 2013, (available at: http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2013/20130131b.pdf), p. 83.
[74] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to Minimum Allowable Core Reflood Rate," March 8, 2013, (ADAMS Accession No. ML13067A261), p. 4.

damage zone. And the simulation *under*-predicted temperatures Westinghouse had reported for that section.[75]

<u>The NRC's Computer Simulation of FLECHT Run 9573 that Included the Section of the Test Bundle that Incurred Thermal Runaway—the Severe-Damage Zone</u>

The FLECHT Run 9573 test bundle incurred thermal runaway 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 foot (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)."[76]

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 Leyse 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)."[77]

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)

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[75] 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).
[76] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, April 1971, (ADAMS Accession No. ML070780083), p. 3.97.
[77] 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).

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.[78]

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)."[79] (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 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.

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[78] *Id*.

[79] 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.

This is powerful evidence that the Baker-Just correlation is inadequate for use in computer safety models that simulate LOCAs. This also means that 10 C.F.R. 50 Appendix K, I.A.5 is non-conservative. 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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[80]

V. CONTENTION

Based on BEST/MATRR's vested interest in the safe operation of BFN, BEST/MATRR members are personally affected and aggrieved by the EPUs that are proposed for all three of BFN'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 EPUs for BFN Units 1, 2, and 3 must be denied. The proposed EPUs 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 EPUs would increase BFN's original licensed thermal power level of 3,293 MWt for each unit by approximately 20 percent for all three units.[81]

Petitioner's requests for leave to intervene and a hearing are supported by the Leyse Declaration (Attachment A). BEST/MATRR alleges that *non-conservative* computer safety model analyses were performed in order to justify the EPUs for BFN

[80] 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 09/02/16).

[81] 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.

Units 1, 2, and 3. As explained in the Leyse Declaration, experimental data, along with appropriate citations, 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 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.

AREVA has stated that its EXEM BWR-2000 Evaluation Model's LOCA calculations for the EPUs for BFN Units 1, 2, and 3 "were performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46."[82] 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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[83]

As discussed above, in Section IV (of BEST/MATRR's hearing request and petition to intervene regarding the LAR for the EPUs for BFN Units 1, 2, and 3), 10 C.F.R. 50 Appendix K, I.A.5 is *non-conservative*.

BEST/MATRR's members must not be subjected to the LAR for the EPUs for BFN Units 1, 2, and 3, because the proposed LAR is "justified" by *non-conservative* Appendix K computer safety model evaluations.

Contention: The EPUs for BFN Units 1, 2, and 3 must not be granted because the EXEM BWR-2000 Evaluation Model's LOCA calculations for "qualifying" the EPUs for BFN Units 1, 2, and 3 are scientifically indefensible.

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[82] 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.

[83] 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 09/02/16).

Contention: TVA has not scientifically demonstrated that at higher power levels (3,952 MWt) that in the event of a LOCA, at any of the BFN units, the PCT would not exceed the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.[84]

Contention: The health and safety of BEST/MATRR's members, as well as that of the general public, must not be threatened by scientifically indefensible EPUs for BFN Units 1, 2, and 3.

The health and safety of BEST/MATRR's members must not be threatened because the requirements of 10 C.F.R. 50 Appendix K, I.A.5 were defended by an industry professional, Dr. John Bernard Roll, a manager in Westinghouse's Nuclear Fuel Division, who made false statements when he was under oath in the Indian Point Unit 2 licensing hearings.

The health and safety of BEST/MATRR's members must not be threatened because FLECHT program data was cherry-picked. In order to defend the Baker-Just correlation, Dr. Roll discussed the cherry-picked FLECHT program data when he was under oath in the Indian Point Unit 2 licensing hearings. Besides the fact that FLECHT program data was cherry-picked, there were problems with the metallurgical data from the FLECHT program, as explained in the Leyse Declaration.

The health and safety of BEST/MATRR's members must not be threatened because the NRC is considering an amendment request for EPUs for BFN Units 1, 2, and 3, which is dependent on Appendix K LOCA analyses, after the NRC disclosed that a computer simulation of FLECHT Run 9573, including the section of the test bundle that incurred thermal runaway, *under*-predicted temperatures Westinghouse had reported for that section.[85]

The health and safety of BEST/MATRR's members must not be threatened after the NRC has disclosed powerful evidence that the Baker-Just correlation is inadequate for

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[84] NRC, § 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html : last visited on 09/04/16).

[85] 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).

use in computer safety models that simulate LOCAs, which means that 10 C.F.R. 50 Appendix K, I.A.5 is non-conservative.

The LAR for the EPUs for BFN Units 1, 2, and 3 must be denied.

AREVA's analyses (conducted to help justify the EPUs for BFN Units 1, 2, and 3) also under-predict the PCTs that would occur in the event of a LOCA for ATRIUM 10XM fuel and ATRIUM 10 fuel, respectively. If the EPUs for BFN Units 1, 2, and 3 were granted and power levels of the BFN reactors were set too high, in the event of a LOCA at one of the BFN units, the PCT would exceed the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.[86] And if the PCT were to exceed the 2200°F limit, the LOCA would (by definition) become a beyond design-basis accident. If one of the Browns Ferry reactors were to melt down, hundreds of kilograms of explosive hydrogen gas would be generated. It is likely that the hydrogen would then explode and destroy a reactor building, releasing large quantities of harmful radioactive material into the environment, as occurred in the Fukushima Daiichi accident.

It is unacceptable to subject BEST/MATRR's members to the dangers of granting EPUs for BFN Units 1, 2, and 3. AREVA's *Appendix K* LOCA calculations are "supported" by *false statements* that were made by a manager in Westinghouse's Nuclear Fuel Division, Dr. Roll, when he was under oath in the Indian Point Unit 2 licensing hearings. AREVA's *Appendix K* LOCA calculations are "supported" by the *cherry-picked* FLECHT program data that Dr. Roll discussed when he was under oath in the Indian Point Unit 2 licensing hearings in order to defend the Baker-Just correlation.

It was over four decades ago that Dr. Roll, a manager in Westinghouse's Nuclear Fuel Division, made false statements when he was under oath in the Indian Point Unit 2 licensing hearings. That was in an Atomic Energy Commission, the NRC's predecessor, licensing hearing. In a contemporary licensing hearing, if Dr. Roll were to make false

[86] NRC, § 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html : last visited on 09/04/16).

statements and not disclose important experimental data when he was under oath, he would be in violation of 10 C.F.R. § 52.4, "Deliberate misconduct."[87]

10 C.F.R. § 52.4(b) states: "Deliberate misconduct means an intentional act or omission that a person or entity knows: (i) Would cause a licensee or an applicant for a license, standard design certification, or standard design approval to be in violation of any rule, regulation, or order; or any term, condition, or limitation, of any license, standard design certification, or standard design approval."[88]

10 C.F.R. § 50.46(b) is the regulation that would be violated, as a consequence of Dr. Roll's false statements and failure to disclose important experimental data when he was under oath in the Indian Point Unit 2 licensing hearings.

It is unacceptable to subject BEST/MATRR's members to the consequences of Dr. Roll's violation of 10 C.F.R. § 52.4, "Deliberate misconduct."

On November 17, 2009, Mark Leyse submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-93,[89] which addresses issues similar to those raised by BEST/MATRR in this Contention and in the Leyse Declaration. 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 the Leyse Declaration and Section IV of this hearing request, on November 24, 2015, Aby Mohseni, Deputy Director of the NRC's Division of Policy and Rulemaking, disclosed to Leyse that an NRC (TRACE code) computer simulation (using the Baker-Just correlation) of a Westinghouse design-basis accident experiment (FLECHT Run 9573),

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[87] 10 C.F.R. § 52.4, "Deliberate misconduct," (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part052/part052-0004.html: last visited on 09/05/16).
[88] *Id*.
[89] Mark Leyse, PRM-50-93, November 17, 2009, (ADAMS Accession No. ML093290250).

under-predicted cladding and steam temperatures at the elevation of the hottest section of the test's fuel rod simulators.[90]

A computer safety model is supposed to *over-predict* temperatures in order to ensure an adequate margin of safety. If a reactor's power level is set too high after being "qualified" by LOCA analyses that do not ensure an adequate margin of safety, a real-life LOCA would lead to a beyond design-basis accident in violation of criteria set forth in 10 C.F.R. § 50.46(b).

BEST/MATRR and Leyse are 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, at the elevation of the hottest section of the test's fuel rod simulators.[91]

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 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)*"[92] [emphasis added].

Furthermore, the TRACE computer safety model simulation that *under-predicted* cladding and steam temperatures for FLECHT Run 9573 demonstrates that the Baker-Just correlation is inadequate for use in computer safety models that simulate LOCAs. As explained in the Leyse Declaration, there is additional data from other experiments, along with appropriate citations, that also demonstrates that the Baker-Just correlation is inadequate for use in computer safety models that simulate LOCAs. That means that 10 C.F.R. 50 Appendix K, I.A.5 is non-conservative. In regard to the zirconium-steam

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[90] 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).
[91] 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).
[92] NRC, "Information Notice No. 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998, (ADAMS Accession No: ML003730714), p. 1.

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 cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just [correlation]."[93]

Nonetheless, the NRC is considering an LAR for EPUs for BFN Units 1, 2, and 3, which is dependent on non-conservative *Appendix K* 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.[94]

The health and safety of BEST/MATRR's members must not be threatened by the fact that the NRC has not concluded its review of PRM-50-93—after more than six years. The fact alone that, on November 24, 2015, Aby Mohseni disclosed to Leyse that an NRC computer simulation (using the Baker-Just correlation) of FLECHT Run 9573, *under-predicted* cladding and steam temperatures at the elevation of the hottest section of the test's fuel rod simulators,[95] is reason enough to deny the LAR for EPUs for BFN Units 1, 2, and 3.

Furthermore, the NRC must not grant the LAR for EPUs for BFN Units 1, 2, and 3, because AREVA's *Appendix K* LOCA calculations are "supported" by *false statements* that were made by a manager in Westinghouse's Nuclear Fuel Division, Dr. Roll, when he was under oath in the Indian Point Unit 2 licensing hearings. AREVA's *Appendix K* LOCA calculations are "supported" by the *cherry-picked* FLECHT program data that Dr. Roll discussed when he was under oath in the Indian Point Unit 2 licensing hearings in order to defend the Baker-Just correlation. That was over four decades ago; however, in a contemporary licensing hearing, if Dr. Roll were to make false statements and not

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[93] 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 09/02/16).

[94] 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.

[95] 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).

disclose important experimental data when he was under oath, he would be in violation of 10 C.F.R. § 52.4, "Deliberate misconduct."[96]

The health and safety of BEST/MATRR members must not be compromised by Dr. Roll's violation of 10 C.F.R. § 52.4. And the health and safety of BEST/MATRR members must not be compromised by the application of the non-conservative Appendix K model that has been employed to help qualify the proposed EPUs for BFN Units 1, 2, and 3. AREVA's LOCA analyses regarding the EPUs for BFN Units 1, 2, and 3 predicted PCTs of 2030°F for ATRIUM 10XM fuel[97] and 2086°F for ATRIUM 10 fuel;[98] however, those PCTs were calculated with a non-conservative Appendix K model, which used the Baker-Just correlation.[99] By definition, a non-conservative model does not ensure an adequate margin of safety. And, if a reactor's power level is set too high after being "qualified" by LOCA analyses that do not ensure an adequate margin of safety, a real-life LOCA would have a PCT that exceeded the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.[100]

If the PCT were to exceed 2200°F, it would be a beyond design-basis accident. If one of the Browns Ferry reactors were to melt down, hundreds of kilograms of explosive hydrogen gas would be generated. It is likely that the hydrogen would then explode and destroy a reactor building, releasing large quantities of harmful radioactive material into the environment, as occurred in the Fukushima Daiichi accident. Clearly, the health and

[96] 10 C.F.R. § 52.4, "Deliberate misconduct," (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part052/part052-0004.html: last visited on 09/05/16).

[97] 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.

[98] 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.

[99] 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.

[100] NRC, § 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, (This information is available at: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html : last visited on 09/02/16).

safety of BEST/MATRR members must not be compromised by EPUs for BFN Units 1, 2, and 3.

VI. CONCLUSION

For the reasons stated, BEST/MATRR respectfully requests that its contention be admitted.

Respectfully submitted,

/s/

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September 9, 2016

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE SECRETARY OF THE COMMISSION

In the Matter of: :
:
TENNESSEE VALLEY AUTHORITY :
(Browns Ferry Nuclear Plant Units 1, 2, and 3;:
Docket Nos. 50-259, 50-260, and 50-296;:
NRC-2016-0118) :

CERTIFICATE OF SERVICE

I hereby certify that on September 9, 2016, I posted the foregoing BELLEFONTE EFFICIENCY & SUSTAINABILITY TEAM/ MOTHERS AGAINST TENNESSEE RIVER RADIATION'S HEARING REQUEST AND PETITION TO INTERVENE REGARDING TENNESSEE VALLEY AUTHORITY'S LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATES FOR BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 This has been filed on the NRC's Electronic Information Exchange system this 9th day of September, 2016. It is my understanding that as a result, the NRC Commission, Atomic Safety and Licensing Board, and parties were served.

Respectfully submitted,

/s/

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September 9, 2016