PRM-50-93 (75FR03876)
PRM-50-95 (75FR66007)

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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

April 7, 2011

Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 AND PRM-50-95; NRC-2009-0554

Note:

Appendix A to this letter: Report by Nuclear Energy Agency Groups of Experts entitled "In-Vessel and Ex-Vessel Hydrogen Sources" is currently a separate document in ADAMS and non-publically available pending NRC determination that there are no legal restrictions precluding the NRC from making the document available to the public.

Office of the Secretary

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I. Statement of Petitioner's Interest

On November 17, 2009, Mark Edward Leyse, Petitioner (in these comments "Petitioner" means Petitioner for PRM-50-93 and sole author of PRM-50-95), submitted a petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the Nuclear Regulatory Commission ("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;[1] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[2, 3]

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that

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

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

[3] It is noteworthy that in 1975, Fred C. Finlayson stated, "[r]ecommendations are made for improvements in criteria conservatism, especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates);" see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, Abstract, p. iii.

the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction considered in emergency core cooling system ("ECCS") evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] 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.[5]

On June 7, 2010, Petitioner, submitted an enforcement action 10 C.F.R. § 2.206 petition on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a LOCA.

On October 27, 2010, NRC published in the Federal Register a notice stating that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner submitted on behalf of NEC, meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[6]

When Petitioner wrote the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner did not foresee that NRC would docket it as PRM-50-95. PRM-50-95 was written and framed as a 10 C.F.R. § 2.206 petition, not as a 10 C.F.R. § 2.802 petition; however, it is laudable that NRC is reviewing the issues Petitioner raised in PRM-50-95.

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

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

[6] 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.

II. Supplementary Information to PRM-50-93 and PRM-50-95

A. NRC Does Not Acknowledge the Existence of Reports which Explicitly State that Analyses Using the Baker-Just and Cathcart-Pawel Correlations Under-Predict Hydrogen Production in Multi-Rod Bundle Severe Fuel Damage Experiments

"In-Vessel and Ex-Vessel Hydrogen Sources," Part I, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," published in 2001, explicitly states that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[7]

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

> Reflooding and quenching of the uncovered core is the most important accident management measure to terminate a severe accident transient. If the core is overheated, this measure can lead to increased oxidation of the Zircaloy cladding which in turn can trigger a temperature escalation. Relatively short flooding and quenching times can thereby lead to high hydrogen source rates which must be taken into account in risk analysis and in the design of hydrogen mitigation systems.
>
> Until recently, the experimental database on quenching phenomena was rather scarce. The available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the few available tests (CORA, LOFT LP-FP-2).[8]

This indicates that available Zircaloy-steam oxidation correlations—including the legally-required Baker-Just and Cathcart-Pawel correlations—are not adequate for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

The LOFT LP-FP-2 experiment, conducted in 1985, is considered "particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe fuel damage experiments and the TMI-2 accident."[9] In the LOFT LP-FP-2 experiment, "[t]he first recorded and qualified rapid temperature rise

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

[8] *Id.*

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

associated with the rapid reaction between Zircaloy and water occurred at about…[2060°F]"[10]—approximately 140°F lower than the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

(It is noteworthy that " "GAMA Perspective Statement on In-Vessel Hydrogen Sources," [was] prepared by B. Clément (IPSN), K. Trambauer (GRS), W. Scholtyssek (FZK), on the basis of information collected from GAMA [Working Group on the Analysis and Management of Accidents] members and the previous Principal Working Group on Coolant System Behaviour (PWG2). It was endorsed by GAMA in April 2001 and approved for publication by CSNI [Committee on the Safety of Nuclear Installations] in June 2001."[11])

(It is also noteworthy that "[GAMA] is mainly composed of technical specialists in the areas of coolant system thermal-hydraulics, in-vessel protection, containment protection, and fission product retention. Its general functions include the exchange of information on national and international activities in these areas, the exchange of detailed technical information, and the discussion of progress achieved in respect of specific technical issues. Severe accident management is one of the important tasks of the group."[12])

In 2005, NRC denied PRM-50-76,[13] which addressed the fact that the Baker-Just and Cathcart-Pawel correlations are deficient because they were not developed to consider how heat transfer would affect Zircaloy-steam reaction kinetics in the event of a LOCA.[14]

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

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

[12] *Id.*, p. 3.

[13] NRC, "Denial of a Petition for Rulemaking to Revise Appendix K to 10 CFR Part 50 and Associated Guidance Documents (PRM-50-76)," Attachment 1, Federal Register Notice, June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359 (hereinafter "Denial of PRM-50-76," Attachment 1).

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

In 2005, regarding the fact that data from isothermal tests are used for the development of oxidation correlations, NRC stated:

> For the development of oxidation correlations, limited by oxygen diffusion into the metal, well-characterized isothermal tests are more important than the complex thermal hydraulics suggested by [Robert H. Leyse]. [Robert H. Leyse's] suggested use of complex thermal-hydraulic conditions would be counter-productive in reaction kinetics tests because temperature control is required to develop a consistent set of data for correlation development. Isothermal tests allow this needed temperature control. *It is more appropriate to apply the developed correlations to more prototypic transients (including complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations are indeed limiting.* This is what was done by Westinghouse in WCAP-7665, by Cathcart and Pawel in NUREG-17 and by the NRC in its technical safety analysis of PRM-50-76[15] [emphasis added].

"Denial of PRM-50-76," Attachment 1 states that the Baker-Just and Cathcart-Pawel correlations were used in analyses of prototypic transients (including those with complex thermal hydraulic conditions) to verify that the proposed phenomena embodied in the correlations were limiting. Obviously, NRC overlooked the fact that it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[16]

Regarding Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to all four of the FLECHT Zircaloy-clad experiments reported in "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,"[17] "Denial of PRM-50-76," Attachment 1, states:

> The Baker-Just correlation using the current range of parameter inputs is conservative and adequate to assess Appendix K ECCS performance. Virtually every data set published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F.

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[15] NRC, "Denial of PRM-50-76," Attachment 1, pp. 21-22.

[16] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

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

> [Robert H. Leyse] did not take into account Westinghouse's metallurgical analyses performed on the cladding for all four FLECHT Zircaloy-clad experiments reported in ["PWR FLECHT Final Report"]. [Robert H. Leyse] also ignored the Westinghouse application of the Baker-Just correlation to these experiments, which had the "complex thermal hydraulic phenomena" deemed important by the petitioner. This application of the correlation to the metallurgical data clearly demonstrates the conservatism of the Baker-Just correlation for 21 typical temperature transients. The NRC also applied the Baker-Just correlation to the FLECHT Zircaloy experiments with nearly identical results, confirming the ["PWR FLECHT Final Report"] results. ...
>
> The NRC applied the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to the four FLECHT Zircaloy experiments, confirming the best-estimate behavior of the Cathcart-Pawel equations for large-break LOCA reflood transients. The NRC applied the Cathcart-Pawel oxide thickness equation to 15 of their transient temperature experiments. The equation was conservative or best-estimate for 13 experiments and non-conservative for the remaining two. This result is consistent with the application of the Cathcart-Pawel equations, which are intended for use in best-estimate LOCA calculations in accordance with [Regulatory Guide] 1.157.[18]

First, as mentioned in PRM-50-93, there is no metallurgical data from the locations of run 9573 that incurred runaway oxidation, because Westinghouse did not obtain such data. So neither Westinghouse nor the NRC applied the Baker-Just correlation to metallurgical data from the locations of run 9573 that incurred runaway oxidation; furthermore, the NRC did not apply the Cathcart-Pawel oxygen uptake and ZrO_2 thickness equations to metallurgical data from the locations of run 9573 that incurred runaway oxidation.

Second, as discussed in Petitioner's comments on PRM-50-93, dated March 15, 2010, it is reasonable to assume that—as in the CORA-2 and CORA-3 experiments, in which local steam starvation conditions are postulated to have occurred[19]—during FLECHT run 9573, the violent oxidation essentially consumed the available steam, so

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[18] NRC, "Denial of PRM-50-76," pp. 20-22.

[19] 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.

that time-limited and local steam starvation conditions, which cannot be detected in the post-test investigation, would have occurred.

So Westinghouse and NRC's application of the Baker-Just correlation as well as NRC's application of the Cathcart-Pawel correlation to oxide layers on the bundle from FLECHT run 9573 were to locations that most likely were steam starved: those are not legitimate verifications of the adequacy of the Baker-Just and Cathcart-Pawel correlations for use in ECCS evaluation calculations.

It is unfortunate that NRC performed such an inadequate technical analysis of PRM-50-76. NRC ignored data from multi-rod bundle severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) that indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA. And, as stated above, NRC overlooked the fact it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments."[20]

Furthermore, NRC ignored ORNL reports from 1990 and 1991, discussing the CORA-16 experiment, which explicitly state that "[c]ladding oxidation was not accurately predicted by available correlations"[21] and that "[t]he predicted and observed cladding thermal response are in excellent agreement until application of the available Zircaloy oxidation kinetics models causes the low-temperature (900-1200°C) [(1652-2192°F)] oxidation to be underpredicted."[22]

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[20] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

[21] 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.

[22] 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.

1. Brief Overview of the Isothermal Experiments Used for the Development of the Baker-Just and Cathcart-Pawel Correlations

The Baker-Just correlation—used in Appendix K to Part 50 ECCS evaluation calculations—is primarily based on data from Lemmon and Bostrom's experiments,[23] conducted with inductively heated Zircaloy-2 specimens. In Lemmon's experiments, "Lemmon measured the rates of reaction between Zircaloy-2 and steam in the temperature range 1000-1700°C by inductively heating specimens in steam at 50 psia and measuring the rate of hydrogen evolution."[24] (Bostrom's experiments were conducted in a temperature range above that of design basis accidents: 1300-1860°C.[25]) Lemmon's specimen was a Zircaloy-2 cylinder that was 2 inches long and 0.5 inches in diameter.[26]

(It is noteworthy that in the course of producing his public comments on Petitioner's PRM-50-93, Robert H. Leyse became aware that NRC staff had never studied the basic references of ANL-6548,[27] the report regarding the Baker-Just correlation. In NRC's technical review of PRM-50-76, NRC staff did not review the basic references of ANL-6548. Robert H. Leyse's actions in prompting NRC to acquire the basic references[28] of ANL-6548 are well documented: see the letter from T. J. McGinty to Robert H. Leyse, dated April 16, 2010 (ADAMS Accession Number: ML100950085). Robert H. Leyse submitted Comment 13 on PRM-50-93 (ADAMS Accession Number: ML101020563), emphasizing that PRM-50-93 is based on sound science and that NRC staff had not had access to the reports (discussing experiments that the Baker-Just correlation is primarily based on) cited in ANL-6548, until March 2010.

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

[24] 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.

[25] *Id.*

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

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

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

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

The Cathcart-Pawel correlation—used in best-estimate ECCS evaluation calculations—is based on data from "Zirconium Metal-Water Oxidation Kinetics: IV Reaction Rate Studies."[29] Cathcart and Pawel's experiments were conducted in two different furnaces with Zircaloy-4 PWR tube specimens. In the MaxiZWOK furnace, the specimen was 18 inches long (only a small segment of that tube—in close proximity to the thermocouple stations—served as the specimen); in the MiniZWOK furnace, the specimen was about 1.2 inches long.[30]

III. CONCLUSION

It is unfortunate that NRC has overlooked the fact it was reported in 2001 that "[t]he available Zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the [CORA and LOFT LP-FP-2] experiments,"[31] and overlooked the fact that ORNL reports from 1990 and 1991 explicitly state that analyses using the available Zircaloy oxidation kinetics models under-predicted the low-temperature (1652-2192°F) oxidation in the CORA-16 experiment.

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 would help improve public and plant-worker safety.

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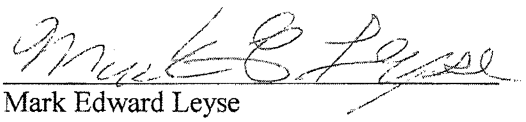
[29] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079.

[30] *Id.*, pp. 12, 15.

[31] Report by NEA Groups of Experts, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

Respectfully submitted,



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Dated: April 7, 2011

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Thursday, April 07, 2011 10:24 PM
To: Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul
Cc: Robert H. Leyse; Dave Lochbaum; Deborah Brancato; Phillip Musegaas; Raymond Shadis; necnp@necnp.org; Powers, Dana A; Ed Lyman
Subject: NRC-2009-0554 (Fourth)
Attachments: NRC-2009-0554 (Fourth).pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Edward Leyse's, Petitioner's, fourth response, dated April 7, 2011, to the NRC's notice of solicitation of public comments on PRM-50-93 and PRM-50-95, NRC-2009-0554, published in the Federal Register on October 27, 2010.

Sincerely,

Mark Edward Leyse