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RULEMAKINGS AND
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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

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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 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.

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

In section II.A. of these comments on PRM-50-93 and PRM-50-95, Petitioner responds to NRC's "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA

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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.

Tests,"[7] regarding the fact that it has been postulated that cladding strain was a factor in increasing the zirconium-steam reaction rates that occurred in the boiling water reactor ("BWR") CORA-16 experiment.[8]

In sections II.A. and II.C. of these comments, Petitioner provides information indicating that cladding strain either had a negligible effect or no effect on increasing the zirconium-steam reaction rates that occurred in the BWR CORA-16 and pressurized water reactor ("PWR") CORA-2 experiments, for which computer safety models using the available zirconium-steam reaction correlations under-predicted zirconium-steam reaction rates.

In section II.B. of these comments, Petitioner provides information indicating that computer safety models using the available zirconium-steam reaction correlations did not accurately predict the oxidation rates that occurred in BWR CORA experiments (in addition to CORA-16), at temperatures above approximately 1922°F and greater.

In section II.D. of these comments, Petitioner provides information from a 2011 IAEA report, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," which states that the zirconium-steam reaction correlations used in computer safety models have limitations; one being that the correlations were derived from experiments that tested zirconium in isothermal conditions—in conditions in which the zirconium specimens were kept at a constant temperature.[9]

A. Response to NRC's Recent Evaluation of the CORA-16 Experiment

There is experimental data that indicates that currently used zirconium-steam reaction correlations are inadequate for predicting the reaction rates that would occur in a LOCA. For example, when investigators compared the results of the CORA-16 experiment—a BWR severe fuel damage test, simulating a meltdown, conducted with a

[7] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," August 23, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML112211930.

[8] 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.

[9] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 11.

multi-rod zirconium alloy bundle—with the predictions of computer safety models, they found that the zirconium-steam reaction rates that occurred in the experiment were under-predicted. The investigators concluded that the "application of the available Zircaloy oxidation kinetics models [zirconium-steam reaction correlations] causes the low-temperature [1652-2192°F] oxidation to be underpredicted."[10]

It has been postulated that cladding strain—ballooning—was a factor in increasing the zirconium-steam reaction rates that occurred in the CORA-16 experiment.[11] In NRC's 2011 evaluation of the CORA-16 experiment, NRC stated that an ORNL paper, "In-Vessel Phenomena—CORA," noted that in CORA-16, "cladding strain could be a factor and that cladding strain and significant oxidation occurred simultaneously."[12]

However, NRC erroneously observed that "In-Vessel Phenomena—CORA" "provided an analytical adjustment that improved the timing prediction with respect to the measured temperatures."[13]

In fact, the ORNL paper's authors employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16.[14] There are three graphs in the ORNL paper depicting cladding temperature plots from different cladding elevations (550 mm, 750 mm, and 950 mm) of "heated rod 5.3" in CORA-16:[15] each plot illustrates that cladding temperatures were greater in the experiment than computer safety models—using the available zirconium-steam reaction correlations—initially predicted (*without employing a multiplicative factor*), indicating that zirconium-steam reaction rates were also under-predicted. Each graph also depicts predicted cladding temperature plots that were computer generated by using a simple *multiplier* to *enhance* the predicted zirconium-steam reaction rates (and the amount of heat the zirconium-steam reaction yielded). By using the multiplier the predicted reaction rates

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[10] 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.
[11] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[12] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[13] *Id.*
[14] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[15] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.

were matched closer to the reaction rates that occurred in the experiment; hence, the multiplier also helped the predicted cladding temperatures match the cladding temperatures that occurred in the experiment.

NRC also erroneously stated that "In-Vessel Phenomena—CORA," did not report that computer safety models under-predicted zirconium-steam reaction rates in CORA-16:[16] a simple glance at the three graphs described above[17] reveals that the paper reported that reaction rates were under-predicted. As explained above, the cladding temperatures were initially under-predicted; hence, the authors of the paper employed a multiplier to enhance the predicted reaction rates. Besides a second ORNL paper explicitly states that the low-temperature (1652°F to 2192°F) oxidation that occurred in CORA-16 was under-predicted.[18] (Petitioner has quoted the second ORNL paper, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," in a number of different comments on PRM-50-93/95 that Petitioner has sent to NRC.)

"In-Vessel Phenomena—CORA" reports that in CORA-16, the *estimated* cladding strain was in the "range of 0.005 to 0.11," at 4200 seconds into the experiment, at locations of the cladding where temperatures were between 1832°F and 2372°F. This certainly does not explain why zirconium-steam reaction rates in the cladding temperature range from 1652°F to 1832°F were under-predicted by computer safety models (as the second ORNL paper reports). It is also unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1832°F to 2192°F.

To help explain how cladding strain could have been a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16, NRC pointed out that an NRC report, NUREG/CR-4412,[19] "explain[s] that under *certain* conditions ballooning and

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[16] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[17] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[18] L. J. Ott, "Report of Foreign Travel of L. J. Ott, Engineering Analysis Section, Engineering Technology Division," p. 3.
[19] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," NUREG/CR-4412, April 1986, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML083400371.

deformation of the cladding can increase the available surface area for oxidation, thus enhancing the apparent oxidation rate" [emphasis not added].[20]

Regarding this phenomenon, NUREG/CR-4412 states:

> Depressurization of the primary coolant during a LB LOCA or [severe accident] will permit [fuel] cladding deformation (ballooning and possibly rupture) to occur because the fuel rod internal pressure may be greater than the external (coolant) pressure. In this case, oxidation and deformation can occur simultaneously. This in turn may result in an apparent enhancement of oxidation rates because: 1) ballooning increases the surface area of the cladding and permits more oxide to form per unit volume of Zircaloy and 2) the deformation may crack the oxide and provide increased accessibility of the oxygen to the metal. However deformation generally occurs before oxidation rates become significant; *i.e.*, below [1832°F]. Consequently, the lesser importance of this phenomenon has resulted in a relatively sparse database.[21]

NUREG/CR-4412 states that there is a *relatively sparse database* on the phenomenon of cladding strain enhancing zirconium-steam reaction rates.[22] NUREG/CR-4412 also explains that "it is possible to make a very crude estimate of the expected average enhancement of oxidation kinetics by deformation;"[23] the report provides a graph of the "rather sparse"[24] data. The graph indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[25]

NUREG/CR-4412 has a brief description of the rather sparse data; in one case, two investigators (Furuta and Kawasaki), who heated specimens up to temperatures between 1292°F and 1832°F, reported that "[v]ery small enhancements [of reaction rates] occurred at about [eight percent] strain at [1832°F]."[26]

NUREG/CR-4412 provides other examples of tests by different investigators: 1) tests were conducted in which pressurized tubes were exposed to steam at 1652°F for

[20] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[21] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 27.
[22] *Id.*, pp. 27, 30.
[23] *Id.*, p. 30.
[24] *Id.*
[25] *Id.*, p. 29.
[26] *Id.*, p. 30.

up to 30 minutes;[27] 2) tests were conducted in which pressurized tubes were exposed to steam at 1310°F and 1472°F for between five to seven hours;[28] 3) tests were conducted in which specimens that were heated up to between 752°F and 887°F;[29] and 4) Zircaloy-2 ring compression tests were conducted in flowing steam between 1292°F and 2372°F.[30]

Only one pair of investigators (Bradhurst and Heuer) conducted tests that encompassed the entire cladding temperature range—1652°F to 2192°F—in which zirconium-steam reaction rates were reported to be under-predicted for CORA-16. Bradhurst and Heuer reported that "[m]aximum enhancements occurred at slower strain rates. ... However, the overall weight gain or average oxide thickness in [the Zircaloy-2 specimens] was only minimally increased because of the localization effects of cracks in the oxide layer." [31] A second report states that "Bradhurst and Heuer...found no direct influence [from cladding strain] on Zircaloy-2 oxidation outside of oxide cracks."[32] (In CORA-16, in the cladding temperature range from 1652°F to 2192°F, cladding strain would have occurred over a very brief period of time, because cladding temperatures were increasing rapidly.)

Clearly, it is unsubstantiated that the estimated cladding strain accurately accounts for why reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1652°F to 2192°F. First, there is a relatively sparse database on how cladding strain enhances reaction rates. Second, the little data that is available indicates that cladding strain *may* only *slightly* enhance reaction rates at cladding temperatures of 1832°F and greater (in a LOCA environment in which local cladding temperatures would be increasing rapidly).[33]

The graphs in "In-Vessel Phenomena—CORA" depicting cladding temperature plots from different cladding elevations of "heated rod 5.3" in CORA-16 show that the temperature differences between the lower predicted (with no enhancement) and higher

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[27] *Id.*
[28] *Id.*
[29] *Id.*, p. 27.
[30] *Id.*
[31] *Id.*
[32] F. J. Erbacher, S. Leistikow, "A Review of Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident," Kernforschungszentrum Karlsruhe, KfK 3973, September 1985, p. 6.
[33] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 30.

actual cladding temperatures have a general trend of increasing above approximately 1832°F—up to above 2552°F.[34] Hence, the CORA-16 analysts' use of "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation" for CORA-16,[35] primarily enhances predicted zirconium-steam reaction rates at cladding temperatures of approximately 1832°F and greater—up to above 2552°F. As stated above, the graph in NUREG/CR-4412 of the relatively sparse database on the phenomenon of cladding strain enhancing zirconium-steam reaction rates indicates that the general trend is for cladding strain enhancements of zirconium-steam reaction rates to *decrease as cladding temperatures increase*.[36]

One phenomenon NRC did not consider in its 2011 analysis of CORA-16 is that "[t]he swelling of the [fuel] cladding...alters [the] pellet-to-cladding gap in a manner that provides less efficient energy transport from the fuel to the cladding,"[37] which would cause the local cladding temperature heatup rate to decrease as the cladding ballooned, moving away from the internal heat source of the fuel. The CORA experiments were internally electrically heated (with annular uranium dioxide pellets to replicate uranium dioxide fuel pellets), so in CORA-16, the ballooning of the cladding would have had a mitigating factor on the local cladding temperature heatup rate, which, in turn, would have had a mitigating factor on zirconium-steam reaction rates.

(In its comments on the CORA-16 experiment, NRC explains that "[t]he mechanisms causing [oxidation] enhancement are highly unlikely to occur for typical pre-pressurized [fuel] rods, which will deform and rupture before the oxidation rate is significant."[38] NRC is correct; for example, in a PWR LB LOCA, the ballooning of the

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[34] See Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations.
[35] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[36] R. E. Williford, "An Assessment of Safety Margins in Zircaloy Oxidation and Embrittlement Criteria for ECCS Acceptance," p. 29.
[37] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey and Bert M. Dunn on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041950059, p. 43.
[38] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.

cladding could commence at a local cladding temperature of approximately 1225°F[39] and rupture at local cladding temperatures between 1290°F and 1470°F,[40] temperatures below those at which zirconium-steam reaction rates become rapid. In CORA-16, the test rods' internal pressures were in a range from 4.6 to 6.1 bar (66.7 to 88.5 psi) (far lower than the internal pressures of fuel rods in a reactor core) and the external system pressure, outside of the test rods was 2.2 bar (31.9 psi); hence, there was not much of a difference between the internal and external pressures, which explains why cladding strain and rupture occurred at higher temperatures in CORA-16.)

A plausible explanation for why zirconium-steam reaction rates for CORA-16 were under-predicted in the cladding temperature range from 1652°F to 2192°F by computer safety models would be that the currently used zirconium-steam reaction correlations are inadequate for use in computer safety models.

ORNL papers on the BWR CORA experiments do not report that any experiments were conducted in order to confirm if in fact cladding strain actually increased zirconium-steam reaction rates and accounted for why reaction rates were under-predicted in the 1652°F to 2192°F cladding temperature range for CORA-16. The analysts seem to have merely assumed that the available zirconium-steam reaction correlations could not possibly be inadequate for use in computer safety models; hence, they did not seem to think it was necessary to support and confirm their *estimates* of cladding strain enhanced zirconium-steam reaction rates with solid experimental data.

In NRC's 2011 evaluation of CORA-16, NRC concluded that the fact zirconium-steam reaction rates were under-predicted by computer safety models—using the available zirconium-steam reaction correlations—"is inadequate as a basis to revise regulations or invalidate the use of [the] Baker-Just and Cathcart-Pawel [correlations] for

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[39] Winston & Strawn LLP, "Duke Energy Corporation, Catawba Nuclear Station Units 1 and 2," Enclosure, Testimony of Robert C. Harvey on Behalf of Duke Energy Corporation, "MOX Fuel Lead Assembly Program, MOX Fuel Characteristics and Behavior, and Design Basis Accident (LOCA) Analysis," July 1, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML041950059, p. 43.

[40] NRC, "Acceptance Review of Proposed Generic Issue on Dispersal of Fuel Particles During a Loss-of-Coolant Accident," October 21, 2011, Enclosure, "Fuel Dispersal During a LOCA: Generic Issue Proposal," located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML112910156, p. 2.

design basis calculations of oxidation."[41] (The Baker-Just and Cathcart-Pawel correlations are among the available zirconium-steam reaction correlations.)

NRC's conclusion is unsubstantiated, as the information presented in this section indicates. When NRC chooses to invalidate experimental data, which is important for simulating accidents, with unsubstantiated postulations, NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative computer safety models.[42]

B. Concurrent Cladding Strain and Oxidation has Been Reported to have Occurred in the BWR CORA Experiments as a Whole

In NRC's 2011 evaluation of the CORA-16 experiment, NRC notes that "In-Vessel Phenomena—CORA," states that computer safety models using the available zirconium-steam reaction correlations accurately predicted the zirconium-steam reaction (oxidation) rates that occurred in the CORA-17 experiment.[43] NRC is correct; "In-Vessel Phenomena—CORA" also states that "cladding strain *was not* a factor in the CORA-17 experiment" [emphasis not added].[44] However, a 1997 ORNL paper states that *concurrent cladding strain and oxidation* occurred in the BWR CORA experiments as a whole. The 1997 ORNL paper discusses *all* of the BWR CORA experiments—CORA-16, -17, -18, -28, -31, and -33—and states that "concurrent cladding strain and oxidation in the β Zircaloy phase regime [which commences above approximately 1922°F] must be considered in the experimental analysis" of the BWR CORA experiments.[45] In other words, the 1997 ORNL paper claims that concurrent cladding strain and oxidation caused oxidation rates to be *enhanced* in the BWR CORA experiments as a whole.

[41] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[42] 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 Dai-ichi Accident," SECY-11-0093, July 12, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML111861807, p. 3.
[43] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," p. 3.
[44] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[45] L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," ORNL, 1997, p. 4.

(Perhaps CORA-17 was an exception (not having enhanced oxidation) or perhaps CORA-17 was reanalyzed and found to have had enhanced oxidation, after all. It would also seem that the CORA-33 experiment might be an exception, because CORA-33 was conducted under relatively steam-starved conditions; that is, with "minimal steaming;" in CORA-33, there was "[n]o temperature escalation as a result of [the] limited steam input."[46] Nonetheless, in CORA-33, "[c]oncurrent cladding strain and oxidation in the zircaloy β phase [which commences above approximately 1922°F] occurred"[47] and "the computed cladding strain was significant over 400 mm [15.75 inches] of the rod length."[48])

If concurrent cladding strain and oxidation caused oxidation rates to be *enhanced* in some of the BWR CORA experiments (in addition to CORA-16), that indicates computer safety models using the available zirconium-steam reaction correlations did *not* accurately predict the oxidation rates that occurred in BWR CORA experiments (in addition to CORA-16), at temperatures above approximately 1922°F and greater. Furthermore, if "concurrent cladding strain and oxidation in the β Zircaloy phase regime [which commences above approximately 1922°F] must be considered in the experimental analysis" of the BWR CORA experiments, it follows that the BWR CORA experiment analyses (in addition to the analysis of CORA-16) employed "a simple multiplicative factor (function of strain) to enhance the [predicted] Zircaloy oxidation."[49]

C. For the PWR CORA-2 Experiment, the Thickness of Oxide Layers Was Under-Predicted at Locations that Did Not have Cladding Ballooning

A computer safety model (a CORA experiment-specific, modified version of SCDAP/MOD1[50]) using available zirconium-steam reaction correlations, under-predicted the thickness of oxide layers that occurred at different locations of the multi-rod bundle

[46] L. J. Ott, Siegfried Hagen, "Interpretation of the Results of the CORA-33 Dry Core Boiling Water Reactor Test," Nuclear Engineering and Design, 167, 1997, p. 291.
[47] *Id.*, p. 297.
[48] *Id.*, p. 298.
[49] L. J. Ott, W. I. van Rij, "In-Vessel Phenomena—CORA: BWR Core Melt Progression Phenomena Program, Oak Ridge National Laboratory."
[50] K. Minato *et al.*, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," KfK 4827, July 1991, p. 10.

that was used in the PWR CORA-2 experiment.[51] A 1991 Kernforschungszentrum Karlsruhe ("KfK") report, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," discussing CORA-2, states that "[a] comparison...shows that the measured oxide layers were thicker than those of the calculation."[52]

(It is important to clarify that the computer simulation of CORA-2 predicted the growth of the oxide layer thicknesses up to 5010 seconds, the point at which, in the simulation, the growth of the oxide layers ceased,[53] because "[d]ue to the preceding melt relocation, a complete consumption of the [Zircaloy] stopped the oxidation."[54]

It is possible that the predicted oxide layer thicknesses would have been thicker in the computer simulation if the melt relocation had not consumed the Zircaloy and stopped the oxidation at 5010 seconds. However, in a scenario without a complete consumption of the Zircaloy, if the computer simulation accurately simulated the quantity of steam that would have been available in CORA-2 after 5010 seconds, it is likely that the oxidation would have either been insignificant or would have stopped, anyway, because there would not have been much (if any) available steam. In the actual CORA-2 experiment at 5010 seconds, the oxidation of the Zircaloy would have either been insignificant or would have stopped, because there would not have been much (if any) available steam. In CORA-2, the steam flow rate of 6 grams per second was terminated at 4600 seconds.[55]

The progression of steam availability in CORA-2 is as follows: in the beginning phase of CORA-2 there was an argon flow rate of 10 grams per second through the test bundle; at 3300 seconds into the experiment, there was an argon flow rate of 4 grams per second and steam flow rate of 6 grams per second; at 4600 seconds into the experiment, the steam flow was turned off and the argon flow rate increased to 10 grams per second;

[51] *Id.*, Appendix E, Figures 10, 11, and 12.
[52] *Id.*, p. 10.
[53] *Id.*, Appendix E, Figure 12.
[54] *Id.*, p. 10.
[55] 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. 54.

at 4900 seconds the electrical power of the bundle was shutoff and the cool-down phase commenced.[56])

For CORA-2, the *calculated* maximum thickness of oxide layers was approximately 0.40 mm at the 350 mm elevation of the test bundle.[57] And for CORA-2, on non-ballooned locations of the test bundle, the thicknesses of oxide layers were *measured* at 0.52 and 0.54 mm (at the 268 mm elevation), 0.54 mm (at the 298 mm elevation), and 0.52 mm (at the 480 mm elevation).[58] The oxide layers which were measured at 0.54 mm thick were 35 percent thicker than the predicted the maximum thickness of 0.40 mm. Furthermore, the *calculated* maximum thicknesses of oxide layers were less than 0.40 mm at the locations/elevations at which oxide layers were *measured* at 0.52 and 0.54 mm thick.

Therefore, a computer safety model using available zirconium-steam reaction correlations, significantly under-predicted oxide layer thicknesses at a number of non-ballooned locations of the CORA-2 bundle. However, there are potential problems with making a comparison of the measured and predicted oxide layer thicknesses of CORA-2, because, as mentioned above, in the computer simulation, at 5010 seconds, the growth of the oxide layers ceased, because a melt relocation consumed the Zircaloy and stopped the oxidation. Yet, in the actual CORA-2 experiment, after 5010 seconds, it is likely that the oxidation would have either been insignificant or would have stopped, because there would not have been much (if any) available steam.

Furthermore, because a computer safety model using available zirconium-steam reaction correlations, significantly under-predicted oxide layer thicknesses at a number of non-ballooned locations of the CORA-2 bundle, it means that the zirconium-steam reaction rates were also significantly under-predicted at a number of non-ballooned locations of the CORA-2 bundle. The CORA-2 data indicates that it cannot be legitimately claimed that cladding stain increased zirconium-steam reaction (oxidation) rates at a number of locations of the CORA-2 bundle.

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[56] *Id.*
[57] K. Minato *et al.*, "Zircaloy Oxidation and Cladding Deformation in PWR-Specific CORA Experiments," p. 10.
[58] *Id.*, Appendix E, Figure 11.

The CORA-2 data calls into question the validity of the postulation that cladding strain was a factor in increasing the zirconium-steam reaction rates that occurred in CORA-16 and the BWR CORA experiments as a whole. The CORA-2 data also calls into question the validity of the claim that currently used zirconium-steam reaction rate correlations are adequate for use in computer safety models. Furthermore, the CORA-2 data calls into question the validity of the claim that data from single rod tests—in which a tiny specimen is held at a constant temperature—is adequate for deriving the zirconium-steam reaction correlations used in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA.

1. Additional Information on the PWR CORA-2 Experiment

The 1991 KfK report states that in CORA-2, there was a slight circumferential elongation that occurred at the non-ballooned locations of the test bundle. The intact (non-ballooned) cladding at the 268 mm, 298 mm, and 480 mm elevations had "a circumferential elongation widening the gap between the pellet and the cladding."[59] It is postulated that the "elongation [was] caused by the high inner rod pressure and/or the volume growth due to the oxidation of the [zirconium alloy] cladding."[60] Therefore, it is possible that the slight circumferential elongation that occurred at the non-ballooned locations of the CORA-2 bundle was *only* due to "the volume growth due to the oxidation of the...cladding."[61] In CORA-2, it is also possible that the cladding deformation was caused either solely by the inner rod pressure or caused by both the volume growth due to oxidation and the inner rod pressure.

The 1991 KfK report also states that the "process of [ballooning] starts with a slight circumferential elongation over nearly the whole length of the rod, widening the gap between pellet and cladding. Then the cladding balloons in the hot region of the rod until it ruptures due to mechanical stress at the hottest azimutal position."[62] (The 1991 KfK report neither states the temperature at which the pellet-cladding gap widening would commence nor states the temperature at which the pellet-cladding gap widening

[59] *Id.*, p. 12.
[60] *Id.*
[61] *Id.*
[62] *Id.*, p. 29.

would mostly cease, at non-ballooned locations of the of the rod. The 1991 KfK report also does not state what the time duration would be between the time the pellet-cladding gap widening commenced and mostly ceased, at non-ballooned locations of the of the rod.)

In CORA-2, the initial value of the outer diameter of the cladding was 10.75 mm. After the CORA-2 experiment was conducted, for four rods that were intact at certain elevations, the outer diameter of the cladding was *measured* at 11.3-12.5 mm and 11.5-12.0 mm on two separate rods (at the 268 mm elevation); at 11.5-12.0 mm and 11.3-12.3 mm on two separate rods (at the 298 mm elevation); and at 11.5-12.3 mm on one rod (at the 480 mm elevation). This means that the measured percentage increase of the circumferential elongation for intact cladding was limited between values of 5 and 16 percent.[63]

D. A 2011 IAEA Report States that the Zirconium-Steam Reaction Correlations Used in Computer Safety Models have Limitations

A 2011 IAEA report states that the zirconium-steam reaction correlations used in computer safety models have limitations; one being that the correlations were derived from experiments that tested zirconium in isothermal conditions—in conditions in which the zirconium specimens were kept at a constant temperature.[64]

Such experiments use tiny zirconium specimens heated in a steam environment to investigate the reaction rates of zirconium in steam. The experiments are termed "single rod tests," which is a misnomer, because the specimens used in the experiments are tiny segments of fuel cladding, usually about one or two inches long.

To perform a test at a constant temperature (at higher temperatures—above 1800°F—at which the zirconium-steam reaction generates a great deal of heat), it is necessary that the specimen be a single rod, because a single rod will have radiative heat losses to its surrounding cooler environment, which removes the heat generated by the zirconium-steam reaction. This allows the specimen to be held at a constant temperature when the specimen temperature exceeds 1800°F, because in such experiments "any

[63] *Id.*, p. 48.
[64] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.

failure to remove the heat of the Zircaloy-steam reaction from the fuel cladding can result in an increase in the temperature of the cladding [specimen]."[65]

It would not be possible to conduct an experiment under isothermal conditions—at a constant temperature—with a multi-rod zirconium alloy bundle at temperatures above 1800°F, because the heat generated from the exothermic reaction of the zirconium in steam would be overpowering and cause the local bundle temperatures to increase rapidly. Although it is not explicitly stated, one could argue that the 2011 IAEA report essentially points out that one of the limitations of the experiments used to derive zirconium-steam reaction correlations is that they were single rod experiments, not more realistic multi-rod bundle experiments.

(It is noteworthy that in 1971, Daniel Ford of Union of Concerned Scientists pointed out that computer safety models use a zirconium-steam correlation[66] "derived from experimental data...completely outside of the context of nuclear systems"[67]—from small-scale experiments conducted with tiny zirconium alloy specimens that were held at a constant temperature.)

The 2011 IAEA report also states that a consequence of the zirconium-steam reaction correlations—used in computer safety models—being derived from experiments that tested zirconium in isothermal conditions is that "[t]he temperature gradient is less than [9°F per second] to use the correlation[s] for transient conditions."[68] If this is true, it means that computer safety models using such correlations could perhaps only accurately predict zirconium-steam reaction rates for LOCA conditions in which the local fuel cladding temperature would increase at a rate of *less* than 9°F per second. Unfortunately, the 2011 IAEA report does not discuss how inaccurately such correlations predict

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[65] J. V. Cathcart, R. E. Pawel, *et al.*, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," Oak Ridge National Laboratory, ORNL/NUREG-17, August 1977, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML052230079, pp. 118-119.

[66] The Baker-Just correlation.

[67] Atomic Energy Commission, "In the Matter of: Consolidated Edison Company of New York, Inc.: Indian Point Station Unit No. 2," Docket No. 50-247, November 3, 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML100350611, p. 2551.

[68] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.

reaction rates for LOCA conditions in which the local fuel cladding temperature would increase at a rate of *greater* than 9°F per second.

The 2011 IAEA report states that "post test calculations of temperature-transient experiments...have confirmed the use of [zirconium-steam reaction] correlations under...conditions [in which the local fuel cladding temperature would increase at a rate of less than 9°F per second];"[69] however, it does not provide information on either how accurate or inaccurate calculations were for experiments in which the local fuel cladding temperature increased at a rate of greater than 9°F per second. (The report merely states that "the reaction rate...can differ with time/temperature during the transient."[70])

In a PWR LB LOCA, the maximum local cladding temperature could possibly increase as rapidly as 30°F per second[71] (caused by the residual heat (stored energy) in the fuel, decay heating, and at higher temperatures, heat generated by the zirconium-steam reaction); for example, a computer simulation of a LB LOCA occurring at Indian Point Unit 2, *predicted* that cladding temperatures would increase from about 600°F to above 2100°F in about 50 seconds, indicating an *average* cladding temperature increase of about 30°F per second. The information in the 2011 IAEA report seems to imply that if cladding temperatures were to increase at a rate of 30°F per second that computer safety model predictions of reaction rates would be inaccurate.

An example of "the reaction rate...differ[ing] with time/temperature during [a] transient,"[72] is the progression of the zirconium-steam reaction rates that occurred in the LOFT LP-FP-2 experiment, the only severe fuel damage experiment conducted with actual decay heat.[73] The initial heat up rate of the fuel cladding in LOFT LP-FP-2 was

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[69] *Id.*
[70] *Id.*
[71] A plot of maximum cladding temperatures derived from a computer simulation of a LB LOCA occurring at Indian Point Unit 2, depicts cladding temperatures increasing from about 600°F to above 2100°F in about 50 seconds, indicating an *average* cladding temperature increase of about 30°F per second; see Entergy, "Reply to Supplemental Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate (TAC MC1865)," August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, Attachment 1, p. 2.
[72] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 11.
[73] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development "Degraded Core Quench: A Status Report," August 1996, p. 13.

approximately 1.8°F per second.[74] At high cladding temperatures at which the zirconium-steam reaction became rapid, the local heat up rate of the fuel cladding began increasing. For example, at one location on the central fuel bundle (at the 42-inch elevation) when cladding temperatures had reached just below 2200°F, the fuel cladding heat up rate had increased to approximately 21.4°F per second;[75] at the same location, between approximately 2200°F and 2780°F, the *average* heat up rate was approximately 36.3°F per second.[76] Fuel cladding temperatures were also rapidly increasing at other locations, indicating that zirconium-steam reaction rates were increasing at a number of locations in the fuel bundle.[77]

It should be clarified that the 2011 IAEA report does opine that the two zirconium-steam reaction correlations—the Baker-Just and Cathcart-Pawel correlations—that computer safety models use for NRC's legally-binding simulations of LOCAs are reliable for intact fuel cladding.[78] However, there is experimental data that indicates that currently used zirconium-steam reaction correlations are inadequate for use in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA. (Such data is discussed above in sections II.A. and II.C. of these comments.)

For example, the fact that computer safety models using the available zirconium-steam reaction correlations under-predicted the reaction rates that occurred in CORA-16, in the cladding temperature range from 1652°F to 2192°F, indicates that the currently used correlations are inadequate for use in computer safety models that simulate LOCAs. Furthermore, the results of CORA-16 indicate that data from single rod tests—in which a tiny specimen is held at a constant temperature—is inadequate for deriving the zirconium-steam reaction correlations used in computer safety models intended to accurately predict the reaction rates that would occur in a LOCA.

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[74] *Id.*
[75] NRC, "Draft Interim Review of PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," 2011, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML112650009, p. 4.
[76] *Id.*, p. 5.
[77] *Id.*, pp. 3-5.
[78] IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," p. 8.

III. CONCLUSION

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

Respectfully submitted,

/s/

Mark Edward Leyse
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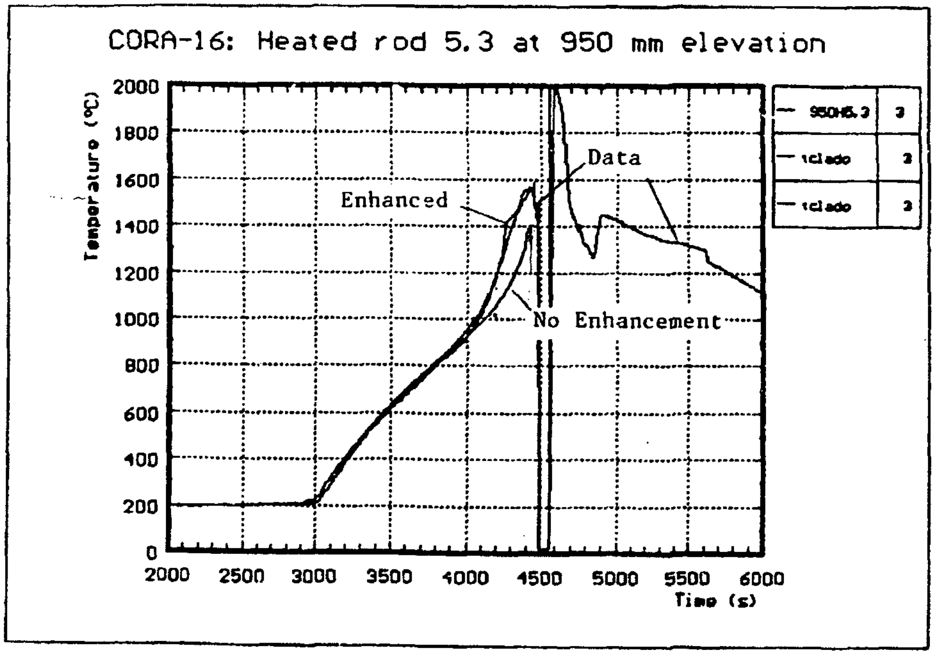
markleyse@gmail.com

Dated: April 16, 2012

Appendix A CORA-16, Heated Rod 5.3 at 950 mm, 750 mm, and 550 mm Elevations[1]

[1] 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.

Use Of A Simple Multiplicative Factor (Function Of Strain) To Enhance The Zircaloy Oxidation Yields Reasonable Predictions For CORA-16



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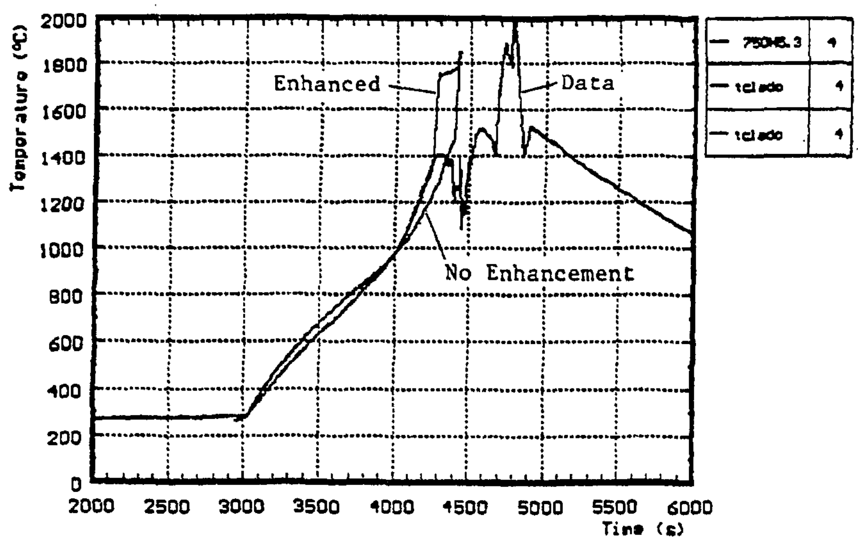
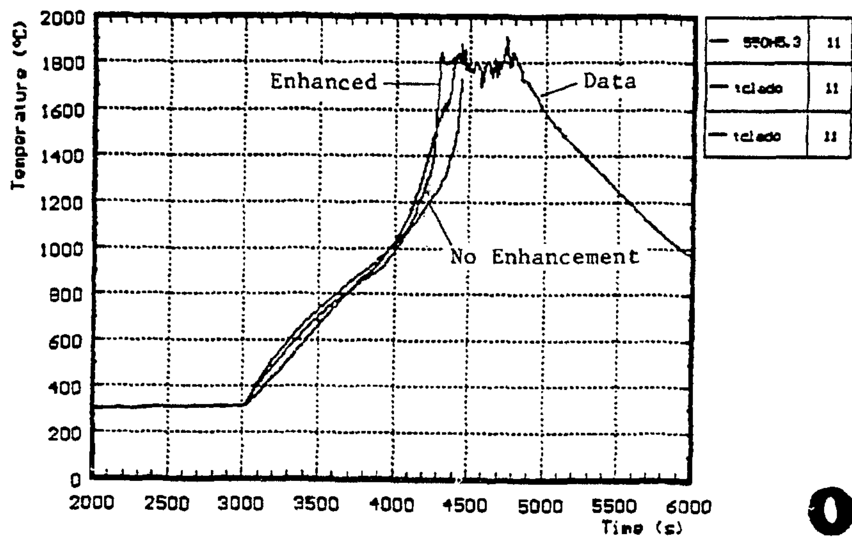
Strain Enhanced Zircaloy Oxidation
(Continued)

CORA-16: Heated rod 5.3 at 750 mm elevation



CORA-16: Heated rod 5.3 at 550 mm elevation



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Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Monday, April 16, 2012 11:45 PM
To: Rulemaking Comments; PDR Resource; Inverso, Tara; Dudley, Richard; Clifford, Paul; CHAIRMAN Resource
Cc: Robert H. Leyse; Christopher Paine; Thomas B. Cochran; Weaver, Jordan; Matthew G. McKinzie; Nuclear; Dave Lochbaum; Ed Lyman
Subject: NRC-2009-0□554 (Sixth)
Attachments: Final April 2012 COMMENTS ON PRM-50-93 and PRM-50-95.pdf

Dear Ms. Vietti-Cook:
Attached to this e-mail is Mark Leyse's, Petitioner's, sixth response, dated April 16, 2012, 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.

In these comments, among other things, Petitioner responds to NRC's "Draft Interim Review of PRM-50-93/95 Issues Related to the CORA Tests," regarding the fact that it has been postulated that cladding strain was a factor in significantly increasing the zirconium-steam reaction rates that occurred in the BWR CORA-16 experiment. As discussed in these comments, there appears to be no data to support such a postulation.
Sincerely,
Mark Leyse