7/31/08

73 FR 44778

(8)

September 5, 2008

Michael Lesar
Chief, Rulemaking, Directives and Editing Branch
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemaking and Adjudications Staff

Subject: Response to the Nuclear Regulatory Commission's (NRC's) notice of solicitation of public comments on documents under consideration to establish the technical basis for new performance-based emergency core cooling system requirements

Dear Mr. Lesar:

Enclosed is a letter I sent to the NRC, as a public comment on PRM-84; it has information that pertains to establishing the technical basis for new performance-based emergency core cooling system ("ECCS") requirements.

My name is Mark Edward Leyse; I submitted a petition for rulemaking (ADAMS Accession No. ML070871368, Docket PRM-50-84) to the NRC, dated March 15, 2007, requesting new regulations, regarding limiting the thickness of crud (corrosion products) and/or oxide layers on fuel cladding surfaces, and stipulating a maximum allowable percentage of hydrogen content in fuel cladding; I also requested that the NRC amend Appendix K to Part 50—ECCS Evaluation Models. PRM-50-84 was summarized briefly in the American Nuclear Society's *Nuclear News*'s June 2007 issue[1] and commented on and deemed "a well-documented justification for…recommended changes to the [NRC's] regulations"[2] by the Union of Concerned Scientists. I am a private citizen concerned about nuclear safety issues; my father, Robert H. Leyse, worked for several decades in the nuclear industry, and worked on two of the PWR Full Length Emergency Cooling Heat Transfer tests mentioned in Appendix K to Part 50.

In this response to the NRC's notice of solicitation, I will respond to the query, regarding the affects of crud deposits and how crud deposits should be addressed from a

[1] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[2] Union of Concerned Scientists, Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84), July 31, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072130342, p. 3.

SUNSI Review Complete
Template = ADM-013

E-RIDS = ADM-03
Add = P.M. Clifford (PMC3)

regulatory point of view, mentioned in the third section of the notice, "Implementation," on page 44779. The notice states:

> Crud deposits on the fuel cladding surface may affect fuel stored energy, fuel rod heat transfer, and cladding corrosion.
>
> a. What role does plant chemistry and crud deposits play on these items?
>
> b. How should normal and abnormal levels of crud deposits be addressed from a regulatory point of view?

In this response, I cannot "include references to the section and page numbers of the document to which the comments applies," as the notice requested, because the documents under review—"Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46" and "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents"—do not mention any of the affects of crud deposits.

Respectfully submitted,

Mark Edward Leyse
P.O. Box 1314
New York, NY 10025
mel2005@columbia.edu

September 5, 2008

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

Attention: Rulemaking and Adjudications Staff

Public Comment on PRM-50-84

Dear Ms. Vietti-Cook:

This public comment on PRM-50-84 (ADAMS Accession No. ML070871368), is a response to comment-letter number five, regarding PRM-50-84, submitted by Nuclear Energy Institute ("NEI") on August 3, 2007 (ADAMS Accession No. ML072150609) and comment-letter number eleven, submitted by Strategic Teaming and Resource Sharing (STARS) on August 15, 2007 (ADAMS Accession No. ML072360363).

Petitioner will first address NEI comments no. 2 and no. 3, regarding guidelines in NUREG-0800 and approved fuel performance models, respectively.

NEI comment no. 2 states:

> It is well recognized that the effects of corrosion on the cladding and grid spacer surfaces and other fuel system structural components need to be considered to ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analyses. Guidelines in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design" do not specify an explicit limit on the maximum allowable corrosion thickness. The guidance contained in SRP Section 4.2 does require that the impact of corrosion on the thermal and mechanical performance be considered in the fuel design analysis, when comparing to the design stress and strain limits. For the fuel rod cladding, the effects include:
>
> (I) The heat transfer resistance provided by the cladding oxide and crud layers, thereby increasing cladding and fuel pellet temperatures,
>
> (II) The metal loss as a result of the corrosion reaction, thereby reducing the cladding load carrying ability.

> These effects are already considered in the design analyses to ensure that the cladding does not exceed the mechanical design limits e.g. design stress and design strain.

First, the guidelines in NUREG-0800 are not legally binding rules or regulations; *i.e.*, they are "guidelines" for the NRC staff, not legally binding requirements for licensees. (NUREG-0800, SRP, Section 4.2, states: "The SRP is not a substitute for the NRC's regulations, and compliance with it is not required.") But the aspects of NUREG-0800, SRP, Section 4.2, cited by NEI, should be made into legally binding regulations for licensees; this would help licensees operate nuclear power plants more safely.

NUREG-0800, SRP, Section 4.2(II)(3)(C)(i) Fuel Temperatures (Stored Energy) states:

> Fuel temperatures and stored energy during normal operation serve as input to ECCS performance calculations. Temperature calculations require complex computer codes that model many different phenomena. RG [(Regulatory Guide)] 1.157 describes models, correlations, data, and methods to realistically calculate ECCS performance during a LOCA and to estimate the uncertainty in that calculation. Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of Appendix K to 10 CFR Part 50. Phenomenological models that should be reviewed include the following:

SRP § 4.2(II)(3)(C)(i) lists 21 of the "phenomenological models that should be reviewed;" the 4th being: "Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers," the 19th being: "Cladding oxide and crud layer thickness[es]."

Currently, Appendix K to Part 50—ECCS Evaluation Models I(A)(1), *The Initial Stored Energy in the Fuel* (and any NRC approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations) does not require that the thermal conductivity of the cladding, cladding crud and cladding oxidation layers be modeled in ECCS evaluation calculations.

These aspects of SRP § 4.2(II)(3)(C)(i); *i.e.*, modeling the thermal conductivity of the cladding, and crud and/or oxide layers on cladding in ECCS evaluation calculations, must be made legally binding for all holders of operating licenses for nuclear power plants. (PRM-50-84 does not request that the thermal conductivity of the cladding be factored into ECCS evaluation calculations; however, the argument of PRM-50-84,

suggests that the thermal conductivity of cladding must also be modeled, because it also affects the stored energy in the fuel.)

Furthermore, if industry already conducts ECCS evaluation calculations by accurately modeling the thermal conductivity of crud and/or oxide layers on cladding as readily as NEI claims, then NEI and the nuclear industry should not be opposed to PRM-50-84, or making the above mentioned aspects of SRP § 4.2(II)(3)(C)(i) into a legally binding regulation. It is perhaps because the nuclear industry does not conduct ECCS evaluation calculations by accurately modeling the thermal conductivity of crud and/or oxide layers on cladding, as readily as NEI claims, that NEI is opposed to PRM-50-84.

PRM-50-84 provides amble evidence that the nuclear industry does not conduct ECCS evaluation calculations by accurately modeling the thermal conductivity of crud and/or oxide layers on cladding. Unfortunately, the impact of crud and oxidation is often not properly modeled addressed; PRM-50-84 (page 16), states that there is little or no evidence that crud has ever been properly factored into peak cladding temperature ("PCT") calculations for postulated LOCAs, for nuclear power plants and cites an attachment to a letter dated June 17, 2003 from Gary W. Johnsen, RELAP5-3D Program Manager, Idaho National Engineering and Environmental Laboratory ("INEEL"), to Robert H. Leyse. The attachment states:

> [W]e are not aware of any user who has modeled crud on fuel elements with SCDAP/RELAP5-3D. ... We suspect that none of the other [severe accident analysis] codes have been applied to consider [fuel crud buildup] (because it has not been demonstrated conclusively that this effect should be considered). ... SCDAP/RELAP5-3D *can* be used to consider this effect, it is simply that users have not chosen to consider this phenomen[on] [emphasis not added].[1]

Furthermore, PRM-50-84 (pages 16-17) provides an example of a utility not properly factoring the thermal conductivity of crud into a PCT calculation for a postulated LOCA. "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," dating from 2002, states, "+4.0°F Cycle 6 crud deposition penalty has been deleted. A PCT penalty of 0°F has been assessed for 4 mils [(~100 μm)] of crud,

[1] From an attachment of a letter from Gary W. Johnsen, RELAP5-3D Program Manager, INEEL to Robert H. Leyse, June 17, 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML032050508.

provided BOL conditions remain limiting. In the event that the SBLOCA cumulative PCT becomes ≥ 1700°F, this issue must be reassessed."[2] Clearly, little attention was given to the thermal resistance of the heavy crud layer at Callaway Cycle 6 (1993), which affected high-duty, one-cycle cladding, at the upper spans 4, 5, and 6 of the fuel assembly.[3]

A recent paper, "The Chemistry of Fuel Crud Deposits and its Effect on AOA in PWR Plants," describing computer codes that model chemical conditions and heat transfer within crud deposits, helps clarify the magnitude of the error of the Callaway Cycle 6 ECCS evaluation: it states that a crud layer that is 59 μm thick is modeled so that "the rise in temperature [from the water side to the fuel side of the layer] is dramatic, reaching temperatures near 400°C [at the fuel side]," up from around 345°C at the water side of the layer.[4] This means, according to the calculations of these codes, that a 59 μm crud layer increases cladding surface temperatures by approximately 55°C or 100°F during operation. And also, according to the calculations of these codes, that a 100 μm crud layer would increase cladding temperatures by more than 100°F during operation. Therefore, according to these codes, at onset of a postulated LOCA, at Callaway Cycle 6, the temperature of the cladding, at some locations, would be over 100°F higher than it would be if the cladding were clean: this would result in a substantially higher than "+4.0°F…crud deposition penalty"[5] for the Cycle 6 calculated PCT.

It is significant that "The Chemistry of Fuel Crud Deposits and its Effect on AOA in PWR Plants" states that the "rise in temperature [across crud layers] was not accounted for in previous models [of crud layers]."[6] And significant that these computer codes that

---

[2] Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," October 14, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML023010263, Attachment 2, p. 6, note 3.

[3] Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs."

[4] Jim Henshaw, John C. McGuire, Howard E. Sims, Ann Tuson, Shirley Dickinson, Jeff Deshon "The Chemistry of Fuel Crud Deposits and Its Effect on AOA in PWR Plants," 2005/2006, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML063390145, p. 8.

[5] Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," Attachment 2, p. 6, note 3.

[6] Jim Henshaw, John C. McGuire, Howard E. Sims, Ann Tuson, Shirley Dickinson, Jeff Deshon "The Chemistry of Fuel Crud Deposits and Its Effect on AOA in PWR Plants," p. 8.

model chemical conditions and heat transfer within crud deposits do not seem to model morphologies of crud that have been documented to increase local cladding temperatures by over 180 or 270°F or greater during PWR operation. Therefore, it is possible that the actual thermal resistance of the crud at Callaway Cycle 6 was greater than what these computer codes would predict. In reality, the increase in temperature across the 100 μm crud layer might have been significantly greater than what these codes would have calculated in 2005/2006, when the paper was written.

***

NEI comment no. 3 states:

> Approved fuel performance models are used to determine fuel rod conditions at the start of a postulated LOCA. The impact of crud and oxidation on fuel temperatures and pressures may be determined explicitly or implicitly in the system of models used. The impact of crud and oxidation is addressed, since the system of approved models is benchmarked to temperature and fission gas release data which inherently include corrosion up to high burnup levels.

Unfortunately, as illustrated above, the impact of crud and oxidation is not properly modeled in ECCS evaluation calculations, as readily as NEI claims. And again, if these models truly determine "[t]he impact of crud and oxidation on fuel temperatures and pressures...explicitly or implicitly in the system models used" as NEI claims, then NEI and the nuclear industry should not be opposed to PRM-50-84.

Additionally, NEI's statement that "[t]he impact of crud and oxidation on fuel temperatures and pressures may be determined explicitly or implicitly in the system models used" is further evidence that the NRC should amend Appendix K to Part 50, as Petitioner requests. As stated in PRM-50-84 (page 33), the primary purpose of Appendix K to Part 50, regarding the stored energy in the fuel, is to require that the stored energy in the fuel be calculated that "yields the highest calculated cladding temperature" or PCT. Because crud and oxide layers on fuel cladding impact fuel temperatures, Appendix K to Part 50 must be amended to require that the thermal conductivity of layers of crud and/or oxide be modeled for calculations of the stored energy in the fuel, as PRM-50-84 requests.

Furthermore, emergency core cooling system ("ECCS") evaluation calculations must not be conducted in violation of NRC requirement 10 C.F.R. § 50.46(a)(1)(i). 10

C.F.R. § 50.46(a)(1)(i) states: "ECCS cooling performance must be calculated...to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." And because crud and/or oxide layers on fuel would affect the severity of a LOCA, as SRP § 4.2(II)(3)(C)(i) states, then Appendix K to Part 50 must be amended to require that the thermal conductivity of layers of crud and/or oxide be modeled for calculations of the stored energy in the fuel, as PRM-50-84 requests.

***

Petitioner will now address the remaining four NEI comments on PRM-50-84.

NEI comment no. 1 states:

> It should be noted that the petitioner relies heavily on four [sic] abnormal operating experiences at River Bend (1998-99 and 2001-03), Three Mile Island Unit 1 (1995), Palo Verde Unit 2 (1997), and Seabrook Nuclear Operating Unit (1997). These units all experienced localized sections of thick crud formation during normal operation. The Industry has taken corrective actions to mitigate both general and localized crud formation during operation. These actions include developing revisions to existing water chemistry guidelines.

First, NEI's statement that Petitioner "relies heavily on [five] abnormal operating experiences" seems to imply that PRM-50-84 addresses issues pertinent only to abnormal operating conditions. This is not the case, because crud can affect all light water reactors ("LWRs"). In a 2005 press release, titled "Employees at Entergy's River Bend Station Earn Top Industry Practice 'Best of the Best' Award," regarding Entergy's "first-of-a-kind approach for examining nuclear fuel to determine the cause of deposits that form on the fuel [cladding]" (Entergy had examined crud from River Bend Cycle 11 (2001-03)), Skip Bowman, NEI's president and chief executive officer, stated: "All U.S. power reactors can use this technique. Indeed, River Bend's efforts are now serving as a basis for additional incentives that will further the entire industry's knowledge of residue formation and behavior."[7]

Second, PRM-50-84—in addition to the five crud-related "abnormal operating experiences" cited by NEI—states that crud or corrosion related fuel failures had

[7] Nuclear Energy Institute, News Release: "Employees at Entergy's River Bend Station Earn Top Industry Practice 'Best of the Best' Award," 05/18/05, located at: http://www.nei.org/newsandevents/riverbenaward/ (accessed on 08/31/07).

occurred at BWRs in six of the years from 1997 to 2004[8] (page 32), and also cites "An Integrated Approach to Maximizing Fuel Reliability" (page 31), regarding the recent trend of corrosion-related fuel failures at BWRs. "An Integrated Approach to Maximizing Fuel Reliability" states:

> [An] increase in BWR failures is due to a great extent to [four] cases that have affected a large number of fuel assemblies. One of these cases is clearly related to crud-accelerated corrosion failures. The other three are also corrosion-related failures and are currently under investigation. The root cause of the failures or the reason for the high crud levels has not been established yet. The analysis is complicated because of coolant chemistry changes introduced for IGSCC and dose control, and the *lack of understanding* of the interplay among materials, fuel duty and the water chemistry variables [emphasis added]."[9]

One of the BWRs that had fuel failures in recent years, mentioned in PRM-50-84 (p. 31), was Browns Ferry-2. Browns Ferry-2 Cycle 12 (April, 2001 to March, 2003) had heavy corrosion at the upper elevations of 300 fuel-rod assemblies; a total of 63 of these assemblies had fuel rods that failed (most likely, at the upper elevations).[10]

Furthermore, PRM-50-84, mentions that crud deposits on cladding in pressurized-water reactors (PWRs) have been measured at up to 125 μm thick[11] (pp. 7, 20). And mentions that AOA helps provide an indication of how often fuel rods have crud deposits that are at least 35 μm thick, which is approximately the minimum thickness of crud that enables AOA to occur (p. 19). Additionally, PRM-50-84, mentions that as of 2003 more than 30 fuel cycles in 16 U.S. PWRs had exhibited AOA[12] (p. 20).

---

[8] Rosa Yang, Odelli Ozer, Kurt Edsinger, Bo Cheng, Jeff Deshon, "An Integrated Approach to Maximizing Fuel Reliability," American Nuclear Society, Proceedings of the *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida, September 19-22, 2004, p. 11.
[9] Id.
[10] TA Keys, James F. Lemons, Conrad Ottenfeld, "Fuel Corrosion Failures in the Browns Ferry Nuclear Plant," American Nuclear Society, Proceedings of the *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida, September 19-22, 2004, pp. 229-231.
[11] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2003/rf093003.pdf (accessed on 01/21/07), p. 133.
[12] U. S. Department of Energy, Nuclear Energy Plant Optimization ("NEPO"), "Current NEPO Projects," located at: http://nepo.ne.doe.gov/NEPO2002projects.asp (accessed on 01/21/07).

It is also noteworthy that PRM-50-84 comment-letter number eleven, submitted by STARS, on August 15, 2007, states that PWRs are "susceptible to heavy crud deposition."[13]

Third, the statement that "industry has taken corrective actions to mitigate both general and localized crud formation during operation [that include] developing revisions to existing water chemistry guidelines," does not mean that the problem is solved. It is significant that the paper "An Integrated Approach to Maximizing Fuel Reliability," states that: "the lack of understanding of the interplay among materials, fuel duty and the water chemistry variables"[14] makes the analysis of BWR fuel failures complicated.

And significant that, discussing current trends in the nuclear industry, EPRI document "2006 Portfolio, 41.002 Fuel Reliability" states:

> [T]he overall industry fuel failure rate has risen in the last couple of years as increased fuel duty and new water chemistry environments have presented increasing challenges to cladding integrity in today's extended fuel cycle operation. [Additionally], front-end economics and reliability are not always harmonious. Fuel vendor research and development, for example, has been significantly scaled back to keep the business competitive, while utilities are operating the fuel more aggressively than ever before.[15]

Additionally, regarding increased fuel duty, "2006 Portfolio, 41.002 Fuel Reliability" states:

> Extended fuel cycle operation and power up-rates have increased fuel duty appreciably since the 1980s. Accompanying this transition to higher duty cores have been many crud-related incidents causing anomalous and unanticipated core behavior in pressurized water reactors, fuel integrity problems, and adverse radiological events. These included axial offset anomaly as well as fuel failure cases in which crud played a significant role. ... [AOA] is a phenomenon where anomalous neutron flux behavior has been observed at many plants operating with high-energy cores. Excessive crud deposition creates operational difficulties for plant operators and has safety implications. [AOA] bears an immediate threat to

---

[13] STARS, Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84), August 15, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072360363. p. 2.

[14] Rosa Yang, Odelli Ozer, Kurt Edsinger, Bo Cheng, Jeff Deshon, "An Integrated Approach to Maximizing Fuel Reliability," p. 11.

[15] EPRI, "2006 Portfolio, 41.002 Fuel Reliability," located at: http://www.epriweb.com/public/2006_P041-002.pdf (accessed on 01/21/07), p. 1.

nuclear power's competitiveness; utilities would like to solve this problem as soon as possible.[16]

***

NEI comment no. 4 states:

For those Pressurized Water Reactor (PWR) cases cited in the petition in which unusual crud patterns and deposits where observed, post-cycle fuel inspection has shown that there was no significant increase in over-all cladding corrosion compared to existing approved corrosion models. Thus, the cladding temperature was not significantly affected by the presence of the crud with the exception of a very limited number of localized damage sites. These localized damage sites are limited both axially and azimuthally such that their thermal resistance effect on the overall fuel temperature and stored energy is small. Furthermore, any damage on the limited surface area of the cladding affected by unusual crud patterns is no different than other types of cladding damage such as fretting wear or secondary hydriding of leaking rods. Thus, assuming that cladding with localized crud damage has failed using existing fuel acceptance criteria (e.g. SRP 4.2), consequences associated with unusual crud patterns and deposits are no different than the other types of fuel rod failure modes already accounted for in the plant Technical Specification limits. The Reactor Coolant System (RCS) iodine levels in plant Technical Specification limits inherently restrict the number of damaged rods in a core. Crud-affected fuel rods are not expected to have any significant effects on initial core conditions that could affect LOCA consequences. Any impact on LOCA analyses would be negligible.

NEI mischaracterizes the extent of the cladding damage that occurred at TMI-1 Cycle 10; NEI claims "the cladding temperature was not significantly affected by the presence of the crud with the exception of a *very limited number* of localized damage sites [emphasis added]." The paper, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," states that after cycle 10, a total of 253 fuel rods in 38 fuel assemblies were observed with a Distinctive Crud Pattern ("DCP"); i.e., "a mottled appearance of a dark, nearly black surface with jagged patches of white showing through."[17] Out of the 253 rods observed with a DCP, nine had failed and, based on eddy current measurements, 101

[16] Id., pp. 2-3.

[17] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," American Nuclear Society, Proceedings of the *2004 International Meeting on LWR Fuel Performance*, Orlando, Florida, September 19-22, 2004, p. 340.

of the rods had thinned, without failure—"a substantial number of rods with the DCP and clad wall thinning."[18]

At TMI-1 Cycle 10, the crud layer was observed to be heaviest in span six of the fuel assemblies, which was "the hottest span" of the assemblies during cycle 10.[19] Additionally, the transcript of proceedings from NRC ACRS, Reactor Fuels Subcommittee, September 30, 2003, states that nine fuel rods failed at the span-six elevation.[20] "Localized cladding penetration due to crud induced localized corrosion was identified as the failure mechanism."[21] Crud was also observed in spans five and seven.[22]

"Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10" states:

> Several samples from both the sound and failed rods showed recrystallization in the DCP areas of the cladding. The recrystallization occurred in an arc of cladding that ranged from 50° to 75° of the outside face of the fuel rod and extended through the entire cladding wall. ...
>
> Recrystallization of cold-worked stress-relief-annealed Zircaloy-4 cladding indicates that the cladding was subjected to temperatures in the range 450 to 500°C or greater, for an indeterminate time, but within the range of ~1000 to 10 hours for the respective temperature limits.[23]

NEI is incorrect, regarding cladding with thick oxide layers or cladding perforated by oxidation, when it claims "[c]rud-affected fuel rods are not expected to have any significant effects on initial core conditions that could affect LOCA consequences. Any impact on LOCA analyses would be negligible."

NEI does not consider the guidelines for calculating ECR (equivalent cladding reacted; the percentage of the cladding of a fuel rod that has oxidized) that are stated in "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," which states that the oxidation considered for ECR during a postulated LOCA "includes

---

[18] Id.
[19] See R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 340; see also NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 236.
[20] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 236.
[21] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.
[22] Id., p. 340.
[23] Id., p. 342.

both pre-accident oxidation and oxidation occurring during a LOCA."[24] Nor does NEI consider that C.F.R. § 10 50.46(b)(2) *Maximum cladding oxidation* states: "[t]he calculated total oxidation of the cladding shall *nowhere* exceed 0.17 times the total cladding thickness before oxidation [emphasis added]." Concerning this 17% limit NRC Information Notice 98-29 warns: "[i]f this...oxidation limit [of 17%] were to be exceeded during an accident, the cladding could become embrittled. The cladding could then fracture and fragment during the reflood period and lose structural integrity. This in turn could compromise the structural soundness and coolable geometry of the core and ultimately the ability to keep the core cooled."[25]

NEI states that the "localized damage sites are limited both axially and azimuthally such that their thermal resistance effect on the *overall* fuel temperature and stored energy is small [emphasis added];" it does not consider *localized* temperatures and *localized* areas where the thermal resistance of crud would increase the stored energy in the fuel. In the event of a LOCA, the initial stored energy in the fuel plays a role in determining the peak cladding temperature. During a LOCA, the peak cladding temperature is a localized temperature, not an overall temperature. 10 C.F.R. § 50.46(b)(1) *Peak cladding temperature* states: "[t]he calculated maximum fuel element cladding temperature shall not exceed 2200°F."

This also illustrates that NEI is incorrect, regarding the PCT, when it claims that "[c]rud-affected fuel rods are not expected to have any significant effects on initial core conditions that could affect LOCA consequences. Any impact on LOCA analyses would be negligible." At TMI-1 Cycle 10, during operation, local cladding temperatures were estimated to have possibly increased by more than 270°F,[26] because of the effects of the thermal resistance of the crud. That means that the initial stored energy in the fuel—which plays a role in determining the PCT—would be substantially higher at some locations than if the cladding were clean.

---

[24] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation."
[25] Id.
[26] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 342.

NEI is also incorrect when it claims "any damage on the limited surface area of the cladding affected by unusual crud patterns is *no different* than other types of cladding damage such as fretting wear or secondary hydriding of leaking rods [emphasis added]."

First, crud causes rapid hydriding; it also causes rapid oxidation. "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10" states that in rod 011 there was massive absorption of hydrogen, to the extent that "hydrided material seems to have broken away from the outer portions of the cladding."[27] Cladding hydrogen content was measured on a non-failed rod at 700 ppm.[28] Therefore, it is highly probable that rod 011 absorbed at least 700 ppm of hydrogen at locations of its upper elevation. Incidentally, this value for hydrogen content in one-cycle cladding is similar to values that have been measured in high-burnup cladding: at (PWR) H. B. Robinson-2, high-burnup cladding hydrogen content was measured at 800 ppm.[29]

Second, "damage on the limited surface area of the cladding affected by unusual crud patterns"—including thick oxidation layers and excessive oxidation that perforated the cladding—is very different than fretting wear. Again, the oxidation considered for ECR during a postulated LOCA "includes both pre-accident oxidation and oxidation occurring during a LOCA (NRC Information Notice 98-29);" and "[t]he calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation (C.F.R. § 10 50.46(b)(2))."

Finally, NEI is incorrect, regarding many aspects of LOCAs, when it claims "[c]rud-affected fuel rods are not expected to have any significant effects on initial core conditions that could affect LOCA consequences. Any impact on LOCA analyses would be negligible."

For conditions where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures (conditions that occurred at PWRs: TMI-1 Cycle 10 (1995), Palo Verde Unit 2 (1997), and Seabrook Nuclear Operating Unit (1997)), an ECCS design basis that did not model crudded and oxidized cladding would be substantially non-conservative in at least the following aspects: 1)

---

[27] Id.
[28] Id., p. 347.
[29] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, July 27, 2005, p. 99.

heavily crudded and oxidized cladding surface temperatures (at some locations) would be higher at the onset of a LOCA than the licensing basis for temperatures based on clean cladding; 2) the stored energy in the fuel sheathed within cladding with heavy crud and oxide layers would be substantially greater than that of fuel sheathed within clean cladding at the onset of a LOCA; 3) the amount of coolant in the vicinity of cladding with heavy crud and oxide layers at the onset of a LOCA would be substantially less than if the cladding were clean; 4) during blowdown and also during reflood the amount of coolant flow past cladding with heavy crud and oxide layers would be substantially less than the flow past clean cladding; 5) the increased quantity of the stored energy in the fuel and the delay in the transfer of that stored energy to the coolant caused by a heavy crud layer would cause the cladding to be subjected to extremely high temperatures for a substantially longer time duration than the time duration used in the licensing basis, providing more time for heatup and degradation of the fuel and cladding; 6) the severity of the fuel and cladding degradation occurring in the event of a LOCA and its effect on obstructing coolant flow would be substantially greater than those calculated by an ECCS design based on clean cladding; 7) the increased quantity of the stored energy in the fuel and the delay in the transfer of that stored energy to the coolant would increase the time until quench; 8) at the onset of a LOCA, there would already be severe cladding degradation, massive oxidation and absorption of hydrogen at some locations, which would contribute to a loss of cladding ductility.

***

NEI comment no. 5 states:

> For the one Boiling Water Reactor (BWR) case cited, in River Bend Cycle 8 significant increases in cladding corrosion were observed only in conjunction with unusually heavy tenacious crud formation. Such crud formation occurred only at lower elevations and thus would have had an impact on the initial stored energy in the fuel only for these locations. Whereas it is true that flow through the affected bundles would be reduced leading to higher initial voiding in the upper part of these bundles, this effect is of secondary importance for a postulated LOCA and is within the envelope determined for core operations with reduced core flow. The calculated Peak Clad Temperature (PCT) in a BWR LOCA event is relatively insensitive to the initial stored energy because PCT values that can challenge the licensing limit occur later in the event and are dominated by the balance between the decay heat and the amount of steam cooling

> after the initial stored energy difference has been mitigated. It is true that a very early peak in the calculated PCT is sensitive to stored energy but this value is seldom the most limiting value and when it is, this peak is far from the licensing limit of 2200°F. The similar crud anomaly that occurred for River Bend Cycle 11 was generally considered to be less severe than the Cycle 8 occurrence in that the heavier crud deposition was even more localized. Both events were operational experiences and would not have been prevented or mitigated by the imposition of specific licensing limits on crud thickness. After the second event, River Bend implemented specific hardware changes to prevent further high-crud events based on the root cause determination for these anomalies. These changes have been effective to date.

NEI has not provided a thorough enough analysis to back its claims, regarding what would have occurred at River Bend Cycles 8 and 11, in the event of LOCAs.

General Electric estimated that cladding temperatures, in local areas, approached 1200°F during cycles 8 and 11, because of heavy layers of crud and oxide.[30] During cycle 11, high temperatures caused significant fuel rod bowing in addition to fuel rod failures.[31]

The lower elevation of the heavy crud layer would not be a compensating factor for the following deficiencies in the LOCA analyses for heavily crudded cladding at River Bend in at least the following aspects: 1) the cladding surface temperature (at some locations) at River Bend Cycles 8 and 11 was reported to have reached temperatures approaching 1200°F; therefore, the starting temperature in the event of a LOCA would be almost 1200°F, not the licensing basis for temperatures around 578°F; 2) the stored energy in the fuel with cladding that had surface temperatures approaching 1200°F (at some locations) would be substantially greater than that of fuel with cladding surface temperatures in the range of 578°F at the onset of a LOCA; 3) the amount of coolant in the vicinity of cladding with heavy crud and oxide layers at the onset of a LOCA would be substantially less than if the cladding were clean; 4) during blowdown and also during reflood the amount of coolant flow past cladding with a heavy crud layer would be substantially less than the flow past clean cladding; 5) the increased quantity of the stored

---

[30] NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," Report Details, p.12, states that the maximum cladding temperatures were similar at River Bend during cycles 8 and 11.
[31] Id., p. 13.

energy in the fuel and the delay in the transfer of the stored energy to the coolant caused by a heavy crud layer would cause the cladding to be subjected to extremely high temperatures for a substantially longer time duration than the time duration used in the licensing basis, providing more time for heatup and degradation of the fuel and cladding; 6) the increased degradation of the fuel and cladding occurring during the extended duration of the extremely high temperatures would further obstruct reflood coolant flow; 7) the increased quantity of the stored energy in the fuel and the delay in the transfer of that stored energy to the coolant would increase the time until quench; 8) at the onset of a LOCA, there would already be severe cladding degradation, massive oxidation and absorption of hydrogen at some locations, which would contribute to a loss of cladding ductility.

When analyzing what might have occurred at River Bend Cycles 8 and 11, in the event of a LOCA, it is also important to consider the F factor, explained at the NRC's ACRS, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting, in January 2007.

At the January 2007 meeting, there was concern that high-burnup fuel with cladding degradation, high levels of oxidation and hydriding, would exceed the 17% oxidation limit in the event of LOCAs at nuclear power plants. The guideline of "NRC Information Notice 98-29," stipulating that the "[t]otal oxidation [of cladding] includes both pre-accident oxidation and oxidation occurring during a LOCA"[32] is being considered for regulation status for a new revised version of 10 C.F.R. § 50.46, due in 2009.[33]

At the January 2007 meeting, NRC staff member Ralph Meyer stated that the purpose of the 17% limit (and the 2200°F limit) was to ensure that cladding ductility was retained, by remaining below those limits, in the event of a LOCA.[34] He also provided examples regarding cladding ductility where the value 1.2 (the F factor[35]) was multiplied

[32] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation."
[33] See NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, p. 245; see also NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, p. 10.
[34] NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, p. 13.
[35] Id., pp. 179-182.

by the pre-accident ECR in order to calculate the remaining percentage of oxidation allowed to occur during a LOCA.[36] He explained that the F factor "depends most strongly on the temperature transient, on heat-up rates and cool-down rates," and that there could be "several different...transients that [would] have different heat-up rates and cool-down rates, and [that 1.2] is sort of a middle of the road value."[37] (A NRC regulatory guide states that the F factor can vary from 1 to 1.6.)[38]

It is significant that at River Bend Cycle 8, ECR was measured at approximately 12%. And significant that at River Bend Cycles 8 and 11, there were also cladding perforations due to oxidation; therefore, those cycles' maximum ECR was actually 100% on one-cycle, high-powered fuel.

It is highly probable that River Bend Cycles 8 and 11 operated in violation of 10 C.F.R. § 50.46(b).

In its entirety, 10 C.F.R. § 50.46(b)(2) states:

> (2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

It is also significant that another BWR, Browns Ferry-2 Cycle 12 (April, 2001 to March, 2003), had heavy corrosion at the upper elevations of 300 fuel-rod assemblies; a

[36] Id., pp. 31-33.
[37] Id., p. 31.
[38] Id., pp. 181-182.

total of 63 of these assemblies had fuel rods that failed (most likely, at the upper elevations).[39]

***

NEI comment no. 6 states:

> It is true that cladding hydrogen content can have an adverse effect on ductile/brittle behavior of zirconium alloy material heated into the beta phase and quenched (as would occur during a typical LOCA scenario). The hydrogen impact on post-quench cladding ductility is a complex function of the oxidation temperature and pre-quench cooling path. The potential impact of hydrogen on the fuel acceptance criteria specified in 10CFR50.46(b) has been recognized for several years and experimental programs are currently underway to assess this impact on current cladding alloys as well as on newer alloys developed to minimize hydrogen build-up during irradiation. Based on the data being generated from several experimental programs, NRC-RES is in the process of preparing the technical basis for performance-based fuel acceptance criteria in 10CFR50.46 that include the effects of hydrogen.

It is laudable that the NRC had an experimental program at Argon National Laboratory to assess the impact of hydrogen content in cladding on cladding embrittlement, and that the NRC may revise 10 C.F.R. § 50.46(b) to include the effects of hydrogen on cladding during postulated LOCAs. However, the NRC has not considered scenarios where crud layers on one-cycle fuel cladding would cause rapid hydriding of cladding (as well as rapid oxidation).

It is significant that at TMI Cycle 10, rod 011 had massive absorption of hydrogen, to the extent that "hydrided material seems to have broken away from the outer portions of the cladding."[40] Cladding hydrogen content was measured on a non-failed rod at 700 ppm.[41] Therefore, it is highly probable that rod 011 absorbed at least 700 ppm of hydrogen at locations of its upper elevation. Incidentally, this value for hydrogen content in one-cycle cladding is similar to values that have been measured in high-burnup cladding: at (PWR) H. B. Robinson-2, high-burnup cladding hydrogen content was

---

[39] TA Keys, James F. Lemons, Conrad Ottenfeld, "Fuel Corrosion Failures in the Browns Ferry Nuclear Plant," pp. 229-231.
[40] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 342.
[41] Id., p. 347.

measured at 800 ppm.[42] (The NRC's ISG-11, Rev. 1 states that hydrogen concentrations of 400-500 ppm are associated with oxide thicknesses of 70-80 μm. And such values for hydrogen concentration and oxide thickness are typically associated with high burnup fuel.)

***

Finally, NEI concludes its comment letter, stating:

> In summary, the Industry opinion is that the requirement to consider the impact of crud and/or corrosion layers resident on the fuel rod cladding surface is adequately specified within the current regulations and Staff guidance documents used to prepare and review fuel design and plant safety analyses. The specific incidents referenced by Mr. Leyse in his petition were isolated operational events and would not have been prevented by imposition of specific limits on crud thickness. The Industry is actively pursuing root cause evaluations and has developed corrective actions, including specific hardware changes, to mitigate further cases of excessive crud formation. Any effects of cladding hydrogen content will be addressed in upcoming revisions to criteria under preparation by the NRC Staff.
>
> The Industry position is that the petition for rulemaking submitted by Mr. Leyse is not needed and should not be considered for action by the Nuclear Regulatory Commission.

Again, as stated above, if the nuclear industry, performs ECCS evaluation calculations as proficiently as NEI claims, then NEI and the nuclear industry should not be opposed to PRM-50-84. Perhaps the reason NEI and the nuclear industry oppose PRM-50-84 is because ECCS evaluation calculations are not performed as proficiently as NEI claims. And PRM-50-84 provides evidence, cited above, that the nuclear industry does not often properly model the thermal conductivity of layers of crud and/or oxide on fuel cladding in ECCS evaluation calculations. Petitioner also suggests that the NRC make aspects of NUREG-0800, SRP § 4.2(II)(3)(C)(i); *i.e.*, modeling the thermal conductivity of the cladding, and crud and/or oxide layers on cladding in ECCS evaluation calculations, into a legally binding regulation for all holders of operating licenses for nuclear power plants.

***

[42] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, July 27, 2005, p. 99.

Petitioner will now respond to comment-letter number eleven, regarding PRM-50-84, submitted by Strategic Teaming and Resource Sharing (STARS) on 08/15/07.

The STARS[43] comments on PRM-50-84, stated that STARS supports the industry comments that were provided on PRM-50-84, in the NEI letter from James H. Riley to the NRC, dated August 3, 2007, titled "Leyse Petition for Rulemaking: PRM-50-84." STARS also stated that it had its own additional comments on PRM-50-84.

The additional STARS comments were:

> The STARS alliance does not support [PRM-50-84]. The industry has funded extensive research that has resulted in chemistry controls, core design constraints, and operational guidance that reduce susceptibility to heavy crud deposition. Chemistry indicators and core power distribution measurements can be evaluated to look for evidence of heavy crud deposition or crud movement. Visual inspections of fuel assemblies during refueling outages may also provide evidence of heavy crud deposition. Many PWRs, especially those most susceptible to heavy crud deposition, make extensive use of this industry guidance.
>
> The requested rulemaking would not make a significant contribution to maintaining safety because current regulations and regulatory guidance already address consideration of crud-related parameters for core cooling.
>
> The proposed revisions would decrease efficiency and effectiveness because licensees would be required to generate additional information as part of the development of their ECCS evaluation models and the NRC staff would need to evaluate the licensee's data and analysis. The resources expended to promulgate the rule and supporting regulatory guidance would be significant with little return of value.
>
> The existing regulatory requirements and guidance already require a nuclear power plant Applicant/licensee to address the impacts of the core geometry change on cooling in ECCS analyses and transient analyses. Therefore, nuclear safety will not be enhanced by adopting the subject petition for rulemaking.

[43] STARS consists of six plants operated by Luminant Power, AmerenUE, Wolf Creek Nuclear Operating Corporation, Pacific Gas and Electric Company, STP Nuclear Operating Company and Arizona Public Service Company.

Discussing increased fuel duty and new water chemistry environments, EPRI document "2006 Portfolio, 41.002 Fuel Reliability" states:

> [T]he overall industry fuel failure rate has risen in the last couple of years as increased fuel duty and new water chemistry environments have presented increasing challenges to cladding integrity in today's extended fuel cycle operation. [Additionally], front-end economics and reliability are not always harmonious. Fuel vendor research and development, for example, has been significantly scaled back to keep the business competitive, while utilities are operating the fuel more aggressively than ever before.[44]

Additionally, regarding crud-related incidents in PWRs, "2006 Portfolio, 41.002 Fuel Reliability" states:

> Extended fuel cycle operation and power up-rates have increased fuel duty appreciably since the 1980s. Accompanying this transition to higher duty cores have been many crud-related incidents causing anomalous and unanticipated core behavior in pressurized water reactors, fuel integrity problems, and adverse radiological events. These included axial offset anomaly as well as fuel failure cases in which crud played a significant role.[45]

Furthermore, the guidelines in NUREG-0800 are not legally binding rules or regulations; *i.e.*, they are "guidelines" for the NRC staff, not legally binding requirements for licensees. (NUREG-0800, SRP, Section 4.2, states: "The SRP is not a substitute for the NRC's regulations, and compliance with it is not required.")

And finally, the STARS sentence, "Many PWRs, especially those most susceptible to heavy crud deposition, make extensive use of this industry guidance," refers to PWRs that are "susceptible to heavy crud deposition." If PWRs are indeed susceptible to heavy crud deposition, then it follows that new regulations are needed to ensure that cladding does not have unsafe thicknesses of crud, and that Appendix K to Part 50 needs to be amended to require that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of crud and/or oxide layers on cladding plays in increasing the stored energy in the fuel.

---

[44] EPRI, "2006 Portfolio, 41.002 Fuel Reliability," p. 1.
[45] Id., pp. 2-3.

Respectfully submitted,

Mark Edward Leyse
P.O. Box 1314
New York, NY 10025
mel2005@columbia.edu