PRM-50-84
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RULEMAKINGS AND
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August 5, 2007

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:

Petitioner would like to add supplementary information to PRM-50-84.

First, Petitioner would like to add information pertinent to the section of PRM-50-84 titled, "A Discussion of an Individual Fuel Rod at TMI-1 Cycle 10" (pp. 10-12):

It is significant 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."[1] Cladding hydrogen content was measured on a non-failed rod at 700 ppm.[2] 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 pressurized water reactor ("PWR"), H. B. Robinson-2, high-burnup cladding hydrogen content was measured at 800 ppm.[3]

An increase in cladding hydrogen content contributes to cladding embrittlement. The transcript of proceedings of NRC, ACRS, Reactor Fuels Subcommittee Meeting, April 4, 2001, relates the opinions of two experts regarding hydrogen content's role in reducing cladding ductility:

> Hee Chung [of Argonne National Laboratory] now points out that for Zircaloy, that there seems to be a threshold around 600 or 700 ppm

[1] 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. 342.
[2] Id., p. 347.
[3] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, July 27, 2005, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2005/rf072705.pdf (accessed on 01/21/07), p. 99.

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SECY-02

> hydrogen. When you get that much hydrogen in the specimen, then it also contributes to the reduction of ductility. .... Griger [of KFKI Atomic Energy Research Institute] believes that he sees a threshold [for a reduction of ductility for Zircaloy] at a much lower level, down around 150 to 200 [ppm].[4]

It is also significant that rod 011 was perforated by oxidation and that it had a 111.1 μm oxide layer at the 118.5 inch elevation, because there is "a decrease in Zr-4 cladding ductilities when oxide thicknesses begin to exceed 100 μm."[5]

Because rod 011 was degraded from substantial oxidation and massive absorption of hydrogen it would have been somewhat embrittled during cycle 10. Therefore, if a real-life LB LOCA had occurred during cycle 10, rod 011 would have with high probability been subjected to temperatures exceeding 2200°F and also with high probability fractured and fragmented during the reflood period (of the LOCA) and lost structural integrity.

Second, Petitioner would also like to add information pertinent to the section of PRM-50-84 titled, "There is Little or No Evidence that Crud has Ever been Properly Factored into PCT Calculations for Postulated LOCAs" (pp. 16-17):

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.[6] 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

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[4] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, April 4, 2001, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2001/rf010404.html (accessed on 01/21/07).

[5] David B. Mitchell and Bert M. Dunn, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML003686365, p. xviii.

[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," 2005/2006, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML063390145, p. 8.

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"[7] for the Cycle 6 calculated peak cladding temperature ("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]."[8] And significant that these computer codes that 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.

Third, in PRM-50-84, Petitioner claims that ECCS evaluation calculations that helped qualify recent power uprates at a number of nuclear power plants were non-conservative (p. 19). Petitioner would like to provide information regarding a recent ECCS evaluation calculation for Indian Point Unit 2 (PWR) that illustrates Petitioner's claim:

When Entergy, the licensee of Indian Point Unit 2 ("IP-2"), did LOCA related calculations to qualify the stretch power uprate for IP-2 (authorized by the NRC in 2004), the calculated maximum cladding oxidation percentages were calculated for fresh, BOL fuel, with Westinghouse's WCOBRA/TRAC code.[9] These ECCS evaluation calculations

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[7] 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.

[8] 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.

[9] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate,"

were done for the safety evaluations that helped qualify the recent stretch power uprate of 3.26 % for IP-2 (to a power level of 3216 megawatts thermal ("MWt")).

Discussing ECCS evaluation calculations of the maximum local oxidation that could occur during a LB LOCA at IP-2 (to qualify the 2004 stretch power uprate), Entergy, states:

> The maximum local oxidation was calculated for fresh fuel, at the beginning of the cycle. This represents the maximum amount of transient oxidation that could occur at any time in life. As burnup increases, the transient oxidation decreases for the following reasons:
>
> 1) The cladding creeps down towards the fuel pellets, due to the system pressure exceeding the rod internal pressure. This will reduce the average internal stored energy at the hot spot by several hundred degrees [Fahrenheit] relatively early in the first cycle of operation. Accounting only for this change, which occurs early in the first cycle, reduces the transient oxidation significantly.
>
> 2) Later in life, the clad creep-down benefit still remains in effect. In addition, with increasing irradiation, the power production from the fuel will naturally decrease as a result of depletion of the fissionable isotopes. Reductions in achievable peaking factors in the burned fuel relative to the fresh fuel are realized before the middle of the second cycle of operation. The achievable linear heat rates decrease steadily from this point until the fuel is discharged, at which point the transient oxidation will be negligible.[10]

As Entergy states, fresh, BOL or one-cycle fuel with low burnups are usually the conditions of the fuel that are considered to have the maximum stored energy, and during postulated LOCAs to yield the maximum amount of transient oxidation (and the highest PCTs) that could occur at any time in the fuel's life. At the January 2007, NRC, Advisory Committee on Reactor Safeguards ("ACRS"), Subcommittee Meeting on Materials, Metallurgy, and Reactor Fuels, Mitch Nissley of Westinghouse cited data from sample LOCA calculations that showed that one-cycle fuel from burnups of zero to

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August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, pp. 6-7.

[10] Id., p. 6. Fuel-cladding gap closure typically takes place within 500 days of operation for M5 cladding; see "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report BAW-10231P, 'COPERNIC Fuel Rod Design Computer Code,' Framatome Cogema Fuels, Project No. 693," 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML020070158, p. 7.

approximately 20 or 25 GWd/MTU yield the highest PCTs (and have the maximum stored energy).[11]

However, Entergy's claim that "the average internal stored energy [will decrease] at the hot spot by several hundred degrees [Fahrenheit] relatively early in the first cycle of operation"[12] is misleading: burnups of 25 GWd/MTU occur in fuel well past the early part of its first cycle of operation. Furthermore, for conditions where cladding would be crudded and oxidized it is highly probable that the cladding would not "[creep] down towards the fuel pellets, due to the system pressure exceeding the [fuel] rod internal pressure...relatively early in the first cycle of operation,"[13] because crud and oxide layers on cladding increase fuel rod internal pressure.

Regarding this phenomenon, NRC document, "Safety Evaluation by the Office of Nuclear Regulation, Topical Report WCAP-15604-NP. REV. 1, 'Limited Scope High Burnup Lead Test Assemblies' Westinghouse Owners Group, Project No. 694," states:

> Clad[ding] oxidation can lead to significantly increased fuel rod internal pressures. Above certain oxidation levels, the impacts on rod internal pressure and the significant impacts on the cladding pressure limit characteristics could result in the rod internal pressure criterion being exceeded. Therefore, if oxidation is kept to a minimum, the fuel rod internal pressure criterion is less limiting than simply the oxidation criterion by itself. In addition to oxidation causing increases in rod internal pressures, crud deposition has a similar effect since crud is a poor conductor of heat. Keeping crud deposition to a minimum also reduces the impact on rod internal pressures.[14]

It is significant that, in some cases, thick crud and oxide layers have quickly accumulated on one-cycle cladding sheathing high-duty fuel. At Three Mile Island Unit

[11] See NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/2007/mm011907.pdf (accessed on 02/27/07), pp. 251-252.

[12] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," August 12, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042380253, p. 6.

[13] Id.

[14] NRC, "Safety Evaluation by the Office of Nuclear Regulation, Topical Report WCAP-15604-NP. REV. 1, 'Limited Scope High Burnup Lead Test Assemblies' Westinghouse Owners Group, Project No. 694," 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070740225 (See Section A), p. 4.

1 Cycle 10, such cladding was perforated by oxidation only 121 days into the cycle.[15] Therefore, it is highly probable that quickly accumulated layers of crud and oxide would either slow down or stop the cladding from creeping down towards the fuel pellets, not reducing the average stored energy in the fuel or the average temperature "at the hot spot by several hundred degrees [Fahrenheit] relatively early in the first cycle of operation."[16]

And even more significantly, Entergy does not consider that the stored energy in one-cycle fuel sheathed within heavily crudded and oxidized cladding would increase to levels greater than that of BOL fuel sheathed within clean cladding.

To clarify how a heavy crud layer would affect the stored energy in the fuel during a LOCA is a citation from a letter from James F. Klapproth, Manager, Engineering and Technology at GE Nuclear Energy, to the NRC:

> The primary effects of [a] heavy crud layer during a postulated LOCA would be an increase in the fuel stored energy at the onset of the event, and a delay in the transfer of that stored energy to the coolant during the blowdown phase of the event.[17]

The fact that a heavy crud layer would: 1) increase the stored energy in the fuel at the onset of a LOCA; and 2) delay the transfer of that stored energy to the coolant during the blowdown phase of a LOCA, is very significant for how cladding would be affected during a LOCA.

The increased stored energy (caused by a heavy crud layer) and the delay in the transfer of that stored energy to the coolant during the blowdown phase would increase the PCT and cause the cladding to be subjected to extremely high temperatures for a substantially longer time duration than if the cladding where clean at the onset of the LOCA. This would provide more time for heatup and degradation of the fuel and cladding, including rapid oxidation and embrittlement of the cladding.

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[15] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.

[16] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," p. 6.

[17] Letter from James F. Klapproth, Manager, Engineering and Technology, GE Nuclear Energy to Annette L. Vietti-Cook, Secretary of the Commission, NRC, April 8, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML021020383.

Regarding the time until quench, Entergy's "Reply to Request for Additional Information Regarding Indian Point 2 Stretch Power Uprate," states:

> In order to demonstrate stable and sustained quench, the WCOBRA/TRAC calculation for the maximum local oxidation analysis was extended. Figure 1 shows the peak cladding temperatures for the five rods modeled in WCOBRA/TRAC. This figure indicates that quench occurs at approximately 275 seconds for the low power rod (rod 5), 400 seconds for the core average rods (rods 3 and 4), and 500 seconds for the hot rod (rod 1) and hot assembly average rod (rod 2). Once quench is predicted to occur, the rod temperatures remain slightly above the fluid saturation temperature for the remainder of the simulation. ... This is consistent with the expected result based on *the removal of the initial core stored energy* [emphasis added]...[18]

The time period until quench for each of the five rods modeled in Entergy's ECCS evaluation calculations would have been significantly increased if scenarios where cladding would be heavily crudded and oxidized had been modeled, because the removal of the initial core stored energy would have taken more time. Because such scenarios were not modeled, Entergy's results for the time period until quench are non-conservative. And because heavy crud and oxide layers on cladding would cause the cladding to be subjected to extremely high temperatures for a substantially longer time duration than the rods modeled in the ECCS evaluation calculations, there would be substantially more degradation of the fuel and cladding, including rapid oxidation and embrittlement of the cladding. Therefore, the results of Entergy's ECCS evaluation calculations for the maximum cladding oxidation that could occur during a LB LOCA at IP-2 (13.2%)[19] are substantially non-conservative.

Discussing calculations of the maximum local oxidation that could occur during a LB LOCA and the maximum sum of the pre-accident and transient oxidation that could occur at IP-2, Entergy, states:

> [T]he transient oxidation decreases from a very conservative maximum of 13.2% at BOL to a negligible value at EOL [(end of life)], while the pre-

[18] Entergy, Attachment 1 to NL-04-121, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," September 24, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042720432, Attachment 1, p. 8.

[19] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," October 27, 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML042960007, Enclosure 2, p. 18.

> transient oxidation increases from zero at BOL to a very conservative maximum at EOL of <15%. Additional WCOBRA/TRAC and HOTSPOT [(with oxidation calculations using "corresponding WCOBRA/TRAC transient boundary conditions")[20]] calculations were performed at intermediate burnups, accounting for burnup effects on fuel performance data (primarily initial stored energy and rod internal pressure). These calculations support the conclusion that the sum of the transient and pre-transient oxidation remains below 15% at all times in life. This conclusion is applicable to each of the fuel designs that will be included in the SPU [(stretch power uprate)] cores, and confirms IP-2 conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.[21]

Entergy's statement that its "calculations support the conclusion that the sum of the transient and pre-transient oxidation remains below 15% at all times in life," is non-conservative. Entergy's analysis omits cladding conditions experienced at PWRs in recent years in the United States where there were crud-induced corrosion fuel failures. In such cases the pre-transient oxidation would have been 100%, because local oxidation perforated cladding at the affected plants.

Clearly, the 2004 stretch power uprate of 3.26 % for IP-2 was qualified by the results of ECCS evaluation calculations that were noncompliant with 10 C.F.R. § 50.46(a)(1)(i), which requires that "ECCS cooling performance must be calculated...to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." Therefore, it is highly probable that IP-2 is currently operating in noncompliance with the parameters set forth in 10 C.F.R. § 50.46(b), at an unsafe power level (3216 MWt).

Sincerely,

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mel2005@columbia.edu

[20] Entergy, Attachment 1 to NL-04-100, "Reply to NRC Request for Additional Information Regarding Proposed License Amendment Request for Indian Point 2 Stretch Power Uprate," p. 6.
[21] Id., p. 7.

From: <mel2005@columbia.edu>
To: <SECY@nrc.gov>
Date: Mon, Aug 6, 2007 12:21 PM
Subject: Attn: Rulemaking and Adjudications Staff; Public Comments on PRM-50-84

Dear Ms. Vietti-Cook:

Attached in two PDF files are two separate letters with comments on PRM-50-84.

Thank you,

Mark Edward Leyse

CC: Dave Lochbaum <dlochbaum@ucsusa.org>

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