EDO Principal Correspondence Control

FROM: DUE: 04/09/08 EDO CONTROL: G20080162
DOC DT: 03/07/08
FINAL REPLY:

Mark Edward Leyser
New York, NY

TO:

Reyes, EDO

FOR SIGNATURE OF : ** GRN ** CRC NO:

Wiggins, NRR

DESC: ROUTING:

2.206 - Indian Point (EDATS: OEDO-2008-0178)

Reyes
Virgilio
Mallett
Ash
Ordaz
Burns
Collins, RI
Carpenter, OE
Caputo, OI
Cyr, OGC
West, OEDO

DATE: 03/10/08

ASSIGNED TO: CONTACT:

NRR Wiggins

SPECIAL INSTRUCTIONS OR REMARKS:

Template: EDO-001

E-RIDS: EDO-01

EDATS
Electronic Document and Action Tracking System

EDATS Number: OEDO-2008-0178 — Source: OEDO

General Information

Assigned To: NRR — **OEDO Due Date:** 4/9/2008 5:00 PM
Other Assignees: — **SECY Due Date:** NONE
Subject: 2.206 - Indian Point Units 2 and 3
Description:

CC Routing: Region I; OE; OI; OGC
ADAMS Accession Numbers - Incoming: NONE — **Response/Package:** NONE

Other Information

Cross Reference Number: G20080162 — **Staff Initiated:** NO
Related Task: — **Recurring Item:** NO
File Routing: EDATS — **Agency Lesson Learned:** NO
Roadmap Item: NO

Process Information

Action Type: 2.206 Review — **Priority:** Medium
Sensitivity: None
Signature Level: NRR — **Urgency:** NO
OEDO Concurrence: NO
OCM Concurrence: NO
OCA Concurrence: NO
Special Instructions:

Document Information

Originator Name: Mark Edward Leyse, New York, NY — **Date of Incoming:** 3/7/2008
Originating Organization: Citizens — **Document Received by OEDO Date:** 3/10/2008
Addressee: L. Reyes, EDO — **Date Response Requested by Originator:** NONE
Incoming Task Received: Letter

March 7, 2008

Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: 10 C.F.R. § 2.206 Request for Emergency Shutdown of Indian Point Units 2 and 3: the Current Power Levels of Both Plants were Qualified by Emergency-Core-Cooling-System Evaluation Calculations that Violated 10 C.F.R. § 50.46(a)(1)(i)

Dear Mr. Reyes:

Enclosed is a petition for an enforcement action, regarding Indian Point Units 2 and 3 ("IP-2 and -3"), dated March 7, 2008, submitted pursuant to 10 C.F.R. § 2.206. 10 C.F.R. § 2.206(a) states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper." A copy of this petition has also been mailed to the Nuclear Regulatory Commission ("NRC") Public Document Room.

This petition is similar to a petition for an enforcement action, regarding IP-2 and -3, that I submitted, dated April 24, 2007 (ADAMS Accession No. ML071150299); my current petition for an enforcement action discusses significant information not covered in my previous petition.

I am submitting this petition because the emergency-core-cooling-system ("ECCS") evaluation calculations that qualified the recent IP-2 and -3 stretch power uprates of 3.26%[1] and 4.85%,[2] respectively (authorized by the NRC in 2004 and 2005, respectively), were conducted in violation of 10 C.F.R. § 50.46(a)(1)(i). 10 C.F.R. § 50.46(a)(1)(i) states that "ECCS cooling performance must be calculated...to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." The ECCS evaluation calculations done to qualify the recent IP-2 and -3 stretch power uprates did not model scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures, operating conditions that have occurred at U.S. pressurized water reactors ("PWRs") in recent years. In the event of a loss-of-coolant accident ("LOCA") such fuel would yield significantly higher peak cladding temperatures ("PCTs") than the fresh, beginning-of-life ("BOL") fuel modeled

[1] 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, p. 1.
[2] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," March 24, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050600380, p. 1.

by the licensee of IP-2 and -3.[3] Furthermore, the cladding, in scenarios where crud-induced corrosion failures would occur, would be substantially more oxidized than the maximum oxidation values claimed by the licensee in the ECCS evaluation calculations. Additionally, it is significant that the recent ECCS evaluation calculations for IP-2 and -3 did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period[, possibly] sufficiently imped[ing] the desired heat transfer [enough] to lead to fuel cladding failure due to thermal stresses."[4] For these reasons, it is highly probable that IP-2 and -3 are currently operating at unsafe power levels, 3216 megawatts thermal ("MWt") and 3216 MWt, respectively, and highly probable that if a large break (LB) LOCA were to occur at either IP-2 or -3 under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated.

I will now discuss my pervious petition for an enforcement action, dated April 24, 2007, regarding IP-2 and -3 (ADAMS Accession No. ML071150299). In a letter to me, dated May 31, 2007, Jennifer Golder, Deputy Director (Acting), Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, stated that "[t]he PRB's final decision is that [my] petition [for an enforcement action] does not meet the criteria for acceptance under 10 CFR 2.206 because [I]...identified no facts to indicate that IP-2 or IP-3 is in violation of any NRC requirement, or that operation of IP-2 or IP-3 presents a safety hazard." And in your letter to me, dated August 21, 2007, you stated that you supported the NRC's Petition Review Board's ("PRB's") decision not to review my pervious petition; specifically, you stated:

> I have reviewed the information you presented and I concur with the PRB's decisions. The NRC previously reviewed the emergency core cooling system evaluation calculations for IP2 and IP3 in accordance with our review criteria and found them acceptable. Therefore, you did not provide facts sufficient to constitute a basis for the requested action and did not meet the criteria for acceptance of a petition as stated in NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions." The PRB appropriately did not accept your petition for review.
>
> As the PRB noted in its letter to you, your concerns are generic which are more appropriately addressed by a petition for rulemaking, and you have submitted such a petition (ADAMS Accession No. ML070871368, Docket

[3] The PCTs were calculated with Westinghouse's WCOBRA/TRAC computer code. Westinghouse maintains that BOL fuel is the most limiting condition of fuel during LOCAs. See Westinghouse, "Code Qualification Document for Best Estimate Small Break LOCA Analysis; Volume 3: PWR Uncertainties and Sensitivities for Small Break LOCA," 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML031570508, Section 29, p. 25.
[4] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," 2006, p. C-24.

> PRM-50-84). The NRC is now reviewing your petition for rulemaking which was published for public comment in the *Federal Register* on May 23, 2007 (72 FR 28902). We will communicate with you periodically to inform you of the status of our review.[5]

I am aware that "[t]he NRC previously reviewed the emergency core cooling system evaluation calculations for IP2 and IP3...and found them acceptable." However, I see no reason why that is proof that I "did not provide facts sufficient to constitute a basis for the requested action and did not meet the criteria for acceptance of a petition as stated in NRC Management Directive 8.11, 'Review Process for 10 CFR 2.206 Petitions.' " In fact, the enforcement petition I submitted, dated April 24, 2007, provided facts that illustrate that the NRC did not review the ECCS evaluation calculations in question, "in accordance with [the NRC's] review criteria," because those ECCS evaluation calculations violated 10 C.F.R. § 50.46(a)(1)(i).

Regarding rejecting petitions under 10 C.F.R. 2.206; *i.e.*, specifically, the criterion that petitions do not provide facts sufficient to constitute a basis for a requested action, "Review Process for 10 CFR 2.206 Petitions," handbook 8.11 states that a petition will be rejected if:

> The incoming correspondence...fails to provide sufficient facts to support the petition...simply alleg[ing] wrongdoing, violations of NRC regulations, or existence of safety concerns. The request cannot be simply a general statement of opposition to nuclear power or a general assertion without supporting facts (*e.g.*, the quality assurance at the facility is inadequate).

As any good faith reading will reveal, the enforcement petition I submitted, dated April 24, 2007, provides sufficient facts supporting my contention that the licensee of IP-2 and -3 qualified the recent IP-2 and -3 stretch power uprates of 3.26%[6] and 4.85%,[7] respectively (authorized by the NRC in 2004 and 2005, respectively), with non-conservative ECCS evaluation calculations. I provided sufficient facts to support my contention that those ECCS evaluation calculations did not calculate the most severe postulated LOCAs that could occur at both plants.

In my April 24, 2007 petition, I illustrated that the recent ECCS evaluation calculations qualifying the stretch power uprates were non-conservative because those calculations did not model scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures (100% oxidation); *i.e.*, cladding

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[5] NRC, Reyes's letter to Leyse, concerning Leyse's petition for an enforcement action (dated April 24, 2007), August 21, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072140819.

[6] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," p. 1.

[7] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," p. 1.

that would be significantly more degraded than the most degraded cladding conditions modeled in the ECCS evaluation calculations for both plants (see pages 2-6 and 39-40 of my April 24, 2007 petition). Furthermore, I discussed the fact that one-cycle fuel sheathed in heavily crudded and oxidized cladding has higher quantities of stored energy than the BOL fuel modeled by the licensee of IP-2 and -3 in the ECCS evaluation calculations done to qualify the IP-2 and -3 stretch power uprates (see pages 2-3, 7-9, 14-15, 25-27, 31-32, and 38-39 of my April 24, 2007 petition). It is significant that the guidelines in the NRC's NUREG-0800, Standard Review Plan, Section 4.2(II)(3)(C)(i), regarding calculating the stored energy in the fuel at the onset of a postulated LOCA, state that the "[t]hermal conductivity of the fuel, cladding, cladding crud, and [cladding] oxidation layers" are phenomena that should be modeled in ECCS evaluation calculations.

Additionally, as I state in my April 24, 2007 petition, the licensee of IP-2 and -3 claimed that the pre-accident oxidation and transient oxidation would "always be below 15%"[8] at both IP-2 and -3 in the event of LOCAs (see page 6 of my April 24, 2007 petition). I illustrated that this claim is non-conservative because it failed to consider operating conditions that have occurred at U.S. PWRs in recent years: three PWRs in the United States (Three Mile Island Unit 1 (1995), Seabrook (1997), and Palo Verde Unit 2 (2000)) operated with cladding that had 100% local oxidation—oxidation had locally perforated cladding at those plants (see page 6 of my April 24, 2007 petition).

The PRB's claim that my April 24, 2007 petition "identified no facts to indicate that...operation of IP-2 or IP-3 presents a safety hazard" is also incorrect. My April 24, 2007 petition clearly documents facts that illustrate that IP-2 and -3 currently have power levels that were qualified by non-conservative ECCS evaluation calculations; *i.e.*, calculations that did not "provide assurance that the most severe postulated [LOCAs were] calculated" (see pages 2-12, 14-17, and 39-40 of my April 24, 2007 petition). For this reason it is highly probable that the current operations of IP-2 and -3 present a safety hazard, because it is highly probable that if a LB LOCA were to occur at either plant under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated.

Discussing my April 24, 2007 petition, you state that my "concerns are generic...more appropriately addressed by a petition for rulemaking, and [that I] have submitted such a petition (ADAMS Accession No. ML070871368, Docket PRM-50-84)."[9] It is laudable

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[8] See 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; see also NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," March 24, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050600380, Enclosure 2, p. 16.

[9] NRC, Reyes's letter to Leyse, concerning Leyse's petition for an enforcement action (dated April 24, 2007).

that the NRC is presently considering my petition for rulemaking; however, that should not have precluded the PRB from considering the fact that the licensee of IP-2 and -3 recently conducted non-conservative ECCS evaluation calculations.

Indeed, my petition for rulemaking is related to my April 24, 2007 petition for an enforcement action; however, my petition for rulemaking does not propose any regulations that would require nuclear power plants to reduce their power levels. My petition for rulemaking proposes that the NRC ensure that nuclear power plants do not operate with unsafe thicknesses of crud and oxide layers on cladding, set a maximum allowable hydrogen content in cladding, and amend Appendix K to Part 50—ECCS Evaluation Models 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. My April 24, 2007 petition for an enforcement action, asserted the fact that the recent ECCS evaluation calculations that qualified the recent IP-2 and -3 stretch power uprates were non-conservative and asserted that it is highly probable that both plants are currently operating at unsafe power levels. I claimed (among other things) that it is highly probable that if a LB LOCA were to occur at either IP-2 or -3 under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated.

My current petition for an enforcement action, regarding IP-2 and -3, dated March 7, 2008, explicitly states that the current power levels of IP-2 and -3 were qualified by non-conservative ECCS evaluation calculations, conducted in violation of 10 C.F.R. § 50.46(a)(1)(i), not "provid[ing] assurance that the most severe postulated [LOCAs were] calculated." It is for this reason that I request that conservative ECCS evaluation calculations be conducted in compliance with 10 C.F.R. § 50.46(a)(1)(i) for IP-2 and -3, and request that the NRC order the licensee of IP-2 and -3 to reduce the power levels of IP-2 and -3 to legally acceptable levels if the results of the conservative ECCS evaluation calculations show that the plants are operating in violation of 10 C.F.R. § 50.46(b). A violation of 10 C.F.R. § 50.46(a)(1)(i) should not be construed as an issue best resolved in a petition for rulemaking as you and the PRB suggest: it does not make sense to request new regulations to resolve violations of existing regulations.

To uphold its congressional mandate to protect the lives, property, and environment of the people of New York, the NRC must not allow the power levels of IP-2 and -3 to be based on ECCS evaluation calculations that violate 10 C.F.R. § 50.46(a)(1)(i). In the case of IP-2 and -3, located less than 40 miles north of New York City, this lack of ECCS evaluation model conservatism puts millions of people at risk. I respectfully request that the PRB consider the current petition for an enforcement action that I am submitting, dated March 7, 2008, concerning IP-2 and -3.

Sincerely,

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

March 7, 2008

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of: ENTERGY CORPORATION (Indian Point Nuclear Generating Units No. 2 and 3; Facility Operating Licenses DPR-26 and DPR-64)	: : : : : :	TO: LUIS A. REYES Executive Director for Operations U.S. Nuclear Regulatory Commission Washington D.C. 20555-0001 Docket No. ______

MARK EDWARD LEYSE,
Petitioner

10 C.F.R. § 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3: THE CURRENT POWER LEVELS OF BOTH PLANTS WERE QUALIFIED BY EMERGENCY-CORE-COOLING-SYSTEM EVALUATION CALCULATIONS THAT VIOLATED 10 C.F.R. § 50.46(a)(1)(i)

I. REQUEST FOR ACTION

This petition for an enforcement action is submitted pursuant to 10 C.F.R. § 2.206 by Mark Edward Leyse. Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") either 1) revoke the operating license of Indian Point Units 2 and 3 ("IP-2 and -3"), 2) order the licensee of IP-2 and -3 to immediately suspend the operations of IP-2 and -3, or 3) temporarily shutdown IP-2 and -3, per 10 C.F.R. § 2.202, because recent emergency-core-cooling-system ("ECCS") evaluation calculations done to qualify the current power levels of IP-2 and -3 did not calculate the most severe postulated loss-of-coolant accidents ("LOCAs") that could occur at both plants, in violation of 10 C.F.R. § 50.46(a)(1)(i). In the event of option 3, Petitioner requests that the NRC order the licensee to conduct conservative ECCS evaluation calculations for IP-2 and -3 that are compliant with 10 C.F.R. § 50.46(a)(1)(i); *i.e.*, calculations that model scenarios where one-cycle fuel would have heavily crudded[1] and/or oxidized fuel rods ("cladding") or would have crud-induced corrosion failures (operating conditions that have occurred at pressurized water reactors ("PWRs") in recent years) at the onset of postulated LOCAs. These ECCS evaluation calculations must also model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period[, possibly] sufficiently imped[ing] the desired heat transfer [enough] to lead to fuel cladding failure due to thermal stresses."[2] 10 C.F.R. § 50.46(a)(1)(i) states that "ECCS cooling performance must be calculated...to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." After conservative ECCS evaluation calculations are conducted in compliance with 10 C.F.R. § 50.46(a)(1)(i), Petitioner requests that the NRC order the licensee of IP-2 and -3 to reduce the power levels of IP-2 and -3 to legally acceptable levels if the results of the conservative ECCS evaluation calculations show that the plants are operating in violation of 10 C.F.R. § 50.46(b).

The ECCS evaluation calculations done to qualify the recent IP-2 and -3 stretch power uprates of 3.26%[3] and 4.85%,[4] respectively (authorized by the NRC in 2004 and 2005, respectively), were non-conservative, because those calculations did not model scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures. In the event of a LOCA such fuel would yield significantly higher peak cladding temperatures ("PCTs") than the fresh, beginning of life ("BOL") fuel modeled by the licensee of IP-2 and -3 in the ECCS evaluation calculations done to qualify the IP-2 and -3 stretch power uprates.[5] Additionally, the

[1] "Crud" is a term meaning corrosion products.
[2] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," 2006, p. C-24.
[3] 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, p. 1.
[4] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," March 24, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050600380, p. 1.
[5] The PCTs were calculated with Westinghouse's <u>W</u>COBRA/TRAC computer code. Westinghouse maintains that BOL fuel is the most limiting condition of fuel during LOCAs. See

cladding, in scenarios where crud-induced corrosion failures would occur, would be substantially more oxidized than the maximum oxidation values claimed by the licensee in the ECCS evaluation calculations. Therefore, it is highly probable that IP-2 and -3 are currently operating at unsafe power levels, because the current power levels of 3216 megawatts thermal ("MWt") and 3216 MWt, respectively, were qualified by the results of non-conservative ECCS evaluation calculations.

Furthermore, the ECCS evaluation calculations that qualified the recent IP-2 and -3 stretch power uprates did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[6] It is also highly probable that IP-2 and -3 were operating at unsafe power levels prior to the 2004 and 2005 stretch power uprates, at 3114.4 MWt and 3067.4 MWt, respectively, because the ECCS evaluation calculations that were done to qualify power levels prior to 2004 and 2005 also did not model scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, and also did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."

II. STATEMENT OF PETITIONER'S INTEREST

Petitioner, Mark Edward Leyse, is a private citizen who is concerned about nuclear safety issues; his 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 the NRC's Appendix K to Part 50—ECCS Evaluation Models. Petitioner's petition for rulemaking, dated March 15, 2007, PRM-50-84 (ADAMS Accession No. ML070871368), was summarized briefly in the American

Westinghouse, "Code Qualification Document for Best Estimate Small Break LOCA Analysis; Volume 3: PWR Uncertainties and Sensitivities for Small Break LOCA," 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML031570508, Section 29, p. 25.

[6] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," p. C-24.

Nuclear Society's *Nuclear News*'s June 2007 issue[7] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[8] by the Union of Concerned Scientists.

10 C.F.R. § 2.206(a) states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper." Petitioner submits this petition on account of the fact that the recent ECCS evaluation calculations that qualified the recent IP-2 and -3 stretch power uprates of 3.26%[9] and 4.85%,[10] respectively (authorized by the NRC in 2004 and 2005, respectively), were conducted in violation of 10 C.F.R. § 50.46(a)(1)(i). This petition is similar to a petition for an enforcement action, regarding IP-2 and -3, that Petitioner submitted, dated April 24, 2007 (ADAMS Accession No. ML071150299); Petitioner's current petition for an enforcement action discusses significant information not covered in his previous petition.

The NRC's Petition Review Board ("PRB") decided to not consider Petitioner's pervious petition. In a letter to Petitioner, dated May 31, 2007, Jennifer Golder, Deputy Director (Acting), Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, stated that "[t]he PRB's final decision is that [Petitioner's] petition [for an enforcement action] does not meet the criteria for acceptance under 10 CFR 2.206 because [Petitioner]...identified no facts to indicate that IP-2 or IP-3 is in violation of any NRC requirement, or that operation of IP-2 or IP-3 presents a safety hazard." Additionally, Luis A. Reyes wrote a letter to Petitioner, dated August 21, 2007, where he stated that he supported the PRB's decision not to review Petitioner's pervious petition.

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[7] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[8] 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.
[9] 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, p. 1.
[10] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," March 24, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050600380, p. 1.

In his letter to Petitioner Reyes stated:

> I have reviewed the information you presented and I concur with the PRB's decisions. The NRC previously reviewed the [ECCS] evaluation calculations for IP2 and IP3 in accordance with [the NRC's] review criteria and found them acceptable. Therefore, you did not provide facts sufficient to constitute a basis for the requested action and did not meet the criteria for acceptance of a petition as stated in NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions." The PRB appropriately did not accept your petition for review.
>
> As the PRB noted in its letter to you, your concerns are generic which are more appropriately addressed by a petition for rulemaking, and you have submitted such a petition (ADAMS Accession No. ML070871368, Docket PRM-50-84). The NRC is now reviewing your petition for rulemaking which was published for public comment in the *Federal Register* on May 23, 2007 (72 FR 28902). We will communicate with you periodically to inform you of the status of our review.[11]

Petitioner is aware that "[t]he NRC previously reviewed the emergency core cooling system evaluation calculations for IP2 and IP3...and found them acceptable." However, Petitioner sees no reason why that is proof that he "did not provide facts sufficient to constitute a basis for the requested action and did not meet the criteria for acceptance of a petition as stated in NRC Management Directive 8.11, 'Review Process for 10 CFR 2.206 Petitions.'" In fact, the enforcement petition Petitioner submitted, dated April 24, 2007, provided facts that illustrate that the NRC did not review the ECCS evaluation calculations in question, "in accordance with [the NRC's] review criteria," because those ECCS evaluation calculations violated 10 C.F.R. § 50.46(a)(1)(i).

Regarding rejecting petitions under 10 C.F.R. 2.206; *i.e.*, specifically, the criterion that petitions do not provide facts sufficient to constitute a basis for a requested action, "Review Process for 10 CFR 2.206 Petitions," handbook 8.11 states that a petition will be rejected if:

> The incoming correspondence...fails to provide sufficient facts to support the petition...simply alleg[ing] wrongdoing, violations of NRC regulations, or existence of safety concerns. The request cannot be simply a general statement of opposition to nuclear power or a general assertion

[11] NRC, Reyes's letter to Leyse, concerning Leyse's petition for an enforcement action (dated April 24, 2007), August 21, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072140819.

without supporting facts (*e.g.*, the quality assurance at the facility is inadequate).

As any good faith reading will reveal, the enforcement petition Petitioner submitted, dated April 24, 2007, provides sufficient facts supporting Petitioner's contention that the licensee of IP-2 and -3 qualified the recent IP-2 and -3 stretch power uprates with non-conservative ECCS evaluation calculations. Petitioner provided sufficient facts to support Petitioner's contention that those ECCS evaluation calculations did not calculate the most severe postulated LOCAs that could occur at both plants.

The PRB's claim that Petitioner's petition "identified no facts to indicate that...operation of IP-2 or IP-3 presents a safety hazard" is also incorrect. Petitioner's petition clearly documents facts that illustrate that IP-2 and -3 currently have power levels that were qualified by non-conservative ECCS evaluation calculations; *i.e.*, calculations that did not "provide assurance that the most severe postulated [LOCAs were] calculated."

Discussing Petitioner's previous petition for an enforcement action, dated April 24, 2007, Reyes stated that Petitioner's "concerns are generic...more appropriately addressed by a petition for rulemaking, and [that Petitioner has] submitted such a petition (ADAMS Accession No. ML070871368, Docket PRM-50-84)."[12] It is laudable that the NRC is presently considering Petitioner's petition for rulemaking; however, that should not have precluded the PRB from considering the fact that the licensee of IP-2 and -3 recently conducted non-conservative ECCS evaluation calculations.

Indeed, Petitioner's petition for rulemaking, dated March 15, 2007, is related to Petitioner's petition for an enforcement action, dated April 24, 2007; however, Petitioner's petition for rulemaking does not propose any regulations that would require nuclear power plants to reduce their power levels. Petitioner's petition for rulemaking proposes that the NRC ensure that nuclear power plants do not operate with unsafe thicknesses of crud and oxide layers on cladding, set a maximum allowable hydrogen content in cladding, and amend Appendix K to Part 50—ECCS Evaluation Models 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

[12] Id.

resistance of crud and/or oxide layers on cladding plays in increasing the stored energy in the fuel. Petitioner's petition for an enforcement action, dated April 24, 2007, asserted the fact that the recent ECCS evaluation calculations that qualified the recent IP-2 and -3 stretch power uprates were non-conservative and asserted that it is highly probable that both plants are currently operating at unsafe power levels. Petitioner claimed (among other things) that it is highly probable that if a LB LOCA were to occur at either IP-2 or -3 under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated.

Petitioner's current petition for an enforcement action, regarding IP-2 and -3, dated March 7, 2008, explicitly states that the current power levels of IP-2 and -3 were qualified by non-conservative ECCS evaluation calculations, conducted in violation of 10 C.F.R. § 50.46(a)(1)(i), not "provid[ing] assurance that the most severe postulated [LOCAs were] calculated." It is for this reason that Petitioner requests that conservative ECCS evaluation calculations compliant with 10 C.F.R. § 50.46(a)(1)(i) be conducted for IP-2 and -3 and that the NRC order the licensee of IP-2 and -3 to reduce the power levels of IP-2 and -3 to legally acceptable levels if the results of the conservative ECCS evaluation calculations show that the plants are operating in violation of 10 C.F.R. § 50.46(b).

A violation of 10 C.F.R. § 50.46(a)(1)(i) should not be construed as an issue best resolved in a petition for rulemaking as Reyes and the PRB suggest: it does not make sense to request new regulations to resolve violations of existing regulations.

III. FACTS CONSTITUTING THE BASIS FOR PETITIONER'S REQUEST

The low thermal conductivity of crud and/or oxide layers on cladding inhibits heat transfer, causing local cladding surface temperatures to increase during the operation of nuclear power plants (sometimes in excess of 300°F or even 600°F);[13] temperatures also

[13] See 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; see also NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," 02/28/06, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession

increase in the fuel sheathed within the cladding (*i.e.*, the stored energy in the fuel increases). In the event of a LOCA, the thermal resistance of insulating layers of crud and/or oxide on cladding, and increased fuel temperatures, would cause the PCT to be higher than it would be if the cladding were clean. If a large break ("LB") LOCA had occurred at several nuclear power plants that operated with heavy crud and oxide layers in recent years, there is a high probability that their PCTs would have exceeded 2200°F, in violation of the parameter set forth in 10 C.F.R. § 50.46(b)(1), *Peak cladding temperature*.

Furthermore, at the onset of a LOCA, at plants that experienced crud-induced corrosion fuel failures (like Three Mile Island Unit 1 Cycle 10), the percentage of maximum local cladding oxidation would already be 100%, because these plants operated with fuel rods that had been perforated by oxidation. This is significant, because "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation" states that the "[t]otal oxidation [of cladding] includes both pre-accident oxidation and oxidation occurring during a LOCA."[14] (This is a NRC recommendation for guidance; it is not a legally binding regulation. But it is being considered for regulation status for a new revised version of 10 C.F.R. § 50.46, due in 2009.)[15] This NRC information notice applies to C.F.R. § 10 50.46(b)(2), *Maximum cladding oxidation*, which dictates the rule for the maximum allowed value of the equivalent cladding reacted ("ECR") calculated by severe accident analysis programs (codes)—used for ECCS evaluations—when simulating LOCAs. C.F.R. § 10 50.46(b)(2) states: "[t]he calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." 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

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Number: ML060600503, Report Details, pp.10-12. River Bend, a boiling water reactor ("BWR"), operated with local cladding temperatures approaching 1200°F during cycles 8 and 11.

[14] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998, located at: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1998/in98029.html (accessed on 01/21/07).

[15] 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), p. 245; see also NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/fullcommittee/2007/ac020207.pdf (accessed on 02/27/07), p. 10.

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."[16]

It is highly probable that if a LB LOCA were to occur at either IP-2 or -3 under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated, because the current power levels of IP-2 and -3 were qualified by non-conservative ECCS evaluation calculations. In its entirety 10 C.F.R. § 50.46(b) states:

> (1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
>
> (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.
>
> (3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
>
> (4) *Coolable geometry.* Calculated changes in core geometry shall be such that the core remains amenable to cooling.

[16] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation."

> (5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

It is also significant that for LOCAs in general, irrespective of cladding conditions at the onset, the non-conservative ECCS evaluation calculations that qualified the current power levels of IP-2 and -3 did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period[, possibly] sufficiently imped[ing] the desired heat transfer [enough] to lead to fuel cladding failure due to thermal stresses."[17]

NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," explains that "[t]he coolant water, bearing the dissolved and suspended solids, would flash away as it encountered the hot fuel cladding and reactor vessel surfaces and leave its solids load behind;" and that "the deposited solids could undergo higher temperature hydrothermal reactions and likely undergo self-cementation."[18]

A. Entergy's Non-Conservative ECCS Evaluation Calculations Used to Qualify the Stretch Power Uprates for Indian Point Units 2 and 3.

When Entergy, the licensee of IP-2 and -3, did LOCA related calculations to qualify the stretch power uprates for IP-2 and -3 (authorized by the NRC in 2004 and 2005, respectively), the calculated maximum cladding oxidation percentages were calculated for fresh, BOL fuel, with Westinghouse's WCOBRA/TRAC code.[19] These ECCS evaluation calculations were done for the safety evaluations that helped qualify the recent plant stretch uprates of 3.26 % for IP-2 and of 4.85% for IP-3. The percentage of

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[17] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," p. C-24.
[18] Id.
[19] 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, pp. 6-7. These calculations were also done for BOL fuel at IP-3.

maximum local cladding oxidation occurring in the event of a LB LOCA at IP-2 was calculated at 13.2%,[20] and for IP-3 it was calculated at 7.6%.[21] However, scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, operating conditions that have occurred at PWRs in recent years, were not modeled in these calculations. Because of the omission of such scenarios, the ECCS evaluation calculations done to qualify the recent plant stretch uprates of IP-2 and -3 are non-conservative.

Additionally, Entergy claimed that the pre-accident oxidation and transient oxidation would "always be below 15%,"[22] at both IP-2 and -3, in the event of LOCAs. This claim is also non-conservative: in recent years at three PWRs in the United States (Three Mile Island Unit 1 (1995), Seabrook (1997), and Palo Verde Unit 2 (2000)) local oxidation was 100% during operation, where oxidation perforated cladding.

It is also significant that these same ECCS evaluation calculations did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[23]

1. Entergy's Non-Conservative ECCS Evaluation Calculations of the Maximum Cladding Oxidation that could Occur during a LB LOCA at Indian Point Unit 2.

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 of 3.26%), 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:

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[20] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.

[21] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," Enclosure 2, p. 16.

[22] See Id.; see also NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.

[23] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," p. C-24.

> 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.[24]

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 approximately 20 or 25 GWd/MTU yield the highest PCTs (and have the maximum stored energy). He also stated that at burnups of around 30 GWd/MTU there is an approximate 10% reduction in achievable power, which yields PCTs that are approximately 100°C (180°F) lower than those of fresher fuel.[25]

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

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[24] 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. 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.

[25] NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, pp. 251-252.

of operation"[26] 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,"[27] 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. Also, at higher levels of oxidation, spalling of the oxide layer can lead to the formation of hot spots forming on the bare cladding surface. Accelerated oxidation at the hot spots can produce through-wall holes. 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.[28]

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 1 Cycle 10, such cladding was perforated by oxidation only 121 days into the cycle.[29] Therefore, it is highly probable that quickly formed layers of crud and oxide would either slow down or stop the cladding from creeping down towards the fuel pellets, not reducing

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[26] 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.
[27] Id.
[28] 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.
[29] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.

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."[30]

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.[31]

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 increase of the stored energy in the fuel caused by a heavy crud layer is substantial (in some cases, enough to increase local cladding temperatures in excess of 300°F or even 600°F during operation).[32] This increase raises the stored energy in the fuel to levels higher than that of fresh, BOL fuel, or fuel with burnups between 30 to 35 GWd/MTU, which are considered the times of life or burnups that represent the maximum stored energy that fuel has during operation. (Fresh, BOL fuel is generally considered to have the maximum stored energy in fuel; however, COPERNIC and

[30] 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.
[31] 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.
[32] See R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 342; see also NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," Report Details pp.10-12. River Bend, a BWR, operated with local cladding temperatures approaching 1200°F during cycles 8 and 11.

FRAPCON-3 (computer codes, programs that simulate LOCAs) calculate that mid-life fuel with burnups of about 30 to 35 GWd/MTU have the maximum stored energy.)[33]

a. How the Stored Energy in the Fuel Affects the Oxidation.

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. When the cladding reacts with steam, an exothermic reaction occurs which generates heat, additionally heating up the cladding. Regarding the significance of time and temperature during a LOCA, NRC staff member, Ralph Meyer, states:

> [I]n 10 CFR 50.46, part [b]...[t]here is an oxidation limit of 17[%]. This is really a *time* limit because it was understood at the beginning and we know it now that the embrittling process does not take place on the surface where the oxide is accumulating [during a LOCA]. It is related to the diffusion of oxygen in the metal. The diffusion process and the oxidation process run at about the same speed. And so an oxidation limit was used. It [is] very convenient. ... It gives you a nearly constant number that you can use as a limit. ... [A] basic LOCA transient calculation is just *time* and *temperature*. And then you run along with that some equation for oxidation and get a calculated oxidation amount during the transient [emphasis added].[34]

Regarding oxidation-induced cladding embrittlement, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> Embrittled cladding can fragment upon introduction of the emergency cooling water in a severe accident. During a high-temperature transient accident, the cladding becomes embrittled by steam oxidation of the zircaloy cladding and the formation of thick reaction layers of brittle oxide and oxygen-stabilized alpha zircaloy. The extent of cladding oxidation, and hence embrittlement, is a function of *temperature*, *time*, and the

[33] "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," p. 10.

[34] NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, pp. 15-16.

> supply of steam and zircaloy. Embrittlement of the cladding may lead to loss of coolable geometry and is thus relevant to the safety analysis of fuel rods [emphasis added].[35]

The increase of the stored energy in the fuel (caused by a heavy crud layer) and the delay in the transfer of that stored energy to the coolant would also increase the time until quench. "Compendium of ECCS Research for Realistic LOCA Analysis" states, "[t]he amount of residual thermal energy [in the fuel rod] influences the *time* required to quench the reactor core with emergency cooling water [emphasis added]."[36]

It is significant that the ECCS evaluation calculations that helped qualify the 2004 power uprate of IP-2 did not model scenarios where cladding would be heavily crudded and oxidized, even though suchlike cladding conditions have occurred at PWRs in recent years. 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]...[37]

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

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[35] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.14-6.

[36] Id., p. 6-14-2.

[37] 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.

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%)[38] are substantially non-conservative.

It is also significant that for LOCAs in general, irrespective of cladding conditions at the onset, the ECCS evaluation calculations that helped qualify the 2004 power uprate of IP-2 did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[39]

b. Entergy's Non-Conservative Conclusion that the Sum of the Pre-Accident and Transient Oxidation would Remain Below 15%.

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-transient oxidation increases from zero at BOL to a very conservative maximum at EOL of <15%. Additional W̲COBRA/TRAC and HOTSPOT [(with oxidation calculations using "corresponding W̲COBRA/TRAC transient boundary conditions")[40]] 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

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[38] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.
[39] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," p. C-24.
[40] 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.

conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.[41]

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 that have occurred 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. Furthermore, at non-perforated cladding locations there were thick oxide layers on cladding at the affected plants.

2. Recent Peak Cladding Temperatures Calculated for Indian Point Unit 2.

a. The PCT Calculated in 2001 with Westinghouse's 1996 WCOBRA/TRAC Computer Code.

In 2001, IP-2 had a PCT of 2188°F in a computer simulated LB LOCA—only 12°F shy the requirements of 10 C.F.R. § 50.46(b)(1).[42]

b. PCTs Calculated with WCOBRA/TRAC to Help Qualify the 2004 Stretch Power Uprate.

The ECCS evaluation calculations that helped qualify IP-2's 2004 stretch power uprate, calculated IP-2's PCT at 2137°F for ZIRLO cladding in Vantage assemblies and at 2115°F for fuel in 15x15 assemblies during a postulated LB LOCA.[43] (The ECCS evaluation calculations that helped qualify IP-3's 2005 stretch power uprate, calculated IP-3's PCT at 1944°F during a postulated LB LOCA.)[44]

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[41] Id., p. 7.
[42] Consolidated Edison Company of New York, Inc., "Indian Point Unit 2 – 30 Day and Annual 10 CFR 50.46 Report," April 10, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011150434, p. 1.
[43] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: 3.26 Percent Power Uprate," Enclosure 2, p. 18.
[44] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters," Enclosure 2, p. 16.

c. PCTs Calculated in 2005 with Westinghouse's ASTRUM Methodology, Bounded in the 1996 WCOBRA/TRAC Code.

In 2006, IP-2 was issued an amendment to its operating license for its LB LOCA analysis methodology; it converted to the "Realistic Large Break LOCA Evaluation Methodology using the Automated Statistical Treatment of Uncertainty Method ("ASTRUM")."[45] With the ASTRUM methodology, bounded in the 1996 WCOBRA/TRAC code,[46] IP-2's PCT was calculated at 1962°F for ZIRLO cladding in 422 Vantage assemblies and at 1814°F for ZIRLO cladding in 15x15 assemblies.[47]

d. Why Recent PCTs Calculated for IP-2, Using the 1996 WCOBRA/TRAC Computer Code and the ASTRUM methodology, are Non-Conservative.

It is significant that, in 1995, Three Mile Island Unit 1 ("TMI-1") Cycle 10 (a PWR) operated with crud deposits on the surface of fuel rods that caused regions of the cladding to be "subjected to temperatures in the range 450 to 500°C or greater."[48] Under typical operating conditions at TMI-1 the maximum cladding temperature is 346°C,[49] meaning that crud deposits raised local cladding temperatures by over 100 or 150°C (180 or 270°F) or greater, during cycle 10. Hence, if IP-2 had operated with heavy crud and oxide layers on its cladding in recent years, the calculated PCT in a computer simulation of a LB LOCA would have with high probability increased, from recently calculated values of 1962°F to 2188°F, by hundreds of degrees Fahrenheit, and violated 10 C.F.R. §

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[45] NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Large Break Loss-of-Coolant Accident (LBLOCA) Analysis Methodology," July 24, 2006, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML061710291.

[46] Cesare Frepoli, Katsuhiro Ohkawa, Robert M. Kemper, "Realistic Large Break LOCA Analysis of AP1000 with ASTRUM," The 6th International Conference on Nuclear Thermal Hydraulics, Operations and Safety, Nara, Japan, October 4-8, 2004, pp. 7-8.

[47] See NRC, letter to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Large Break Loss-of-Coolant Accident (LBLOCA) Analysis Methodology," Enclosure 2, p. 3; see also Entergy, Attachment 1 to NL-05-058, "Reanalysis of Large Break Loss of Coolant Accident Using ASTRUM," April 22, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML051230311, pp. 1-2.

[48] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 342.

[49] *World Nuclear Industry Handbook, 1995*, Nuclear Engineering International (England), p. 80.

50.46(b)(1) by exceeding 2200°F (if the thermal resistance of such layers were taken into account in the calculation).

3. Why Recent ECCS Evaluation Calculations for IP-2, Using the 1996 WCOBRA/TRAC Computer Code and the ASTRUM methodology, are Non-Conservative for All Requirements of 10 C.F.R. § 50.46(b).

Recent ECCS evaluation calculations for IP-2, using the WCOBRA/TRAC code, dating from 1996, and the ASTRUM methodology,[50] bounded in the 1996 WCOBRA/TRAC code, are non-conservative, because they did not model scenarios where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures (operating conditions that have occurred at PWRs in recent years).

For this reason the design basis (modeled with the WCOBRA/TRAC code) for the ECCS of IP-2 is substantially non-conservative in regard to 10 C.F.R. § 50.46(b), because it is based on clean cladding: the most limiting ECCS evaluation calculations were done for fresh, BOL fuel. This is standard for modeling fuel with the WCOBRA/TRAC code, despite the fact that heavy crud and oxide layers on cladding increase the stored energy in the fuel to quantities greater than that of fresh, BOL fuel.

Discussing the WCOBRA/TRAC code, Westinghouse's "Code Qualification Document for Best Estimate Small Break LOCA Analysis" states:

> Hot assembly burnup affects fuel average temperature during normal operation. Fuel temperatures and steady-state peaking factors are typically highest early in the [first] fuel cycle. Because this results in higher calculated PCT during large break LOCAs, the hot rod and the hot assembly rod are assumed at BOL conditions in the scoping analysis.[51]

[50] "The ASTRUM method is based on a non-parametric (distribution free) statistics. For the same conditions this technique is expected to reduce the final PCT estimate (95/95 PCT) by about 150°F. This is achieved because of the elimination of the superposition penalty." See Cesare Frepoli, Katsuhiro Ohkawa, Robert M. Kemper, "Realistic Large Break LOCA Analysis of AP1000 with ASTRUM," p. 17.

[51] Westinghouse, "Code Qualification Document for Best Estimate Small Break LOCA Analysis; Volume 3: PWR Uncertainties and Sensitivities for Small Break LOCA," 2003, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML031570508, Section 29, p. 25.

This is significant because heavy crud and oxide layers on cladding have been documented to increase local cladding temperatures in excess of 300°F or even 600°F during operation;[52] temperatures also increase in the fuel sheathed within the cladding (*i.e.*, the stored energy in the fuel increases). And even though such layers on cladding would substantially increase a plant's PCT in the event of LOCA, there is little or no evidence that crud has ever been properly factored into PCT calculations for postulated LOCAs. 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 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].[53]

As in the cases of other nuclear power plants, it is a serious oversight that recent IP-2 ECCS evaluation calculations have not modeled conditions of heavy crud and oxide layers on cladding. For situations where one-cycle fuel would have heavily crudded and oxidized cladding or would have crud-induced corrosion failures, the ECCS design basis for IP-2 is substantially non-conservative in at least the following aspects: 1) 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

[52] See R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 342; see also NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," Report Details pp.10-12. River Bend, a BWR, operated with local cladding temperatures approaching 1200°F during cycles 8 and 11.

[53] 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.

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.

It is also significant that for LOCAs in general, irrespective of cladding conditions at the onset, the non-conservative ECCS evaluation calculations that qualified the current power level of IP-2 did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a LOCA that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[54]

B. An Example of Crud-Induced Cladding Corrosion Failures: Three Mile Island Unit 1 Cycle 10.

In 1995, Three Mile Island Unit 1 ("TMI-1"), a PWR, operated with crud deposits on the surface of fuel rods that caused regions of the cladding to be "subjected to temperatures in the range 450 to 500°C or greater."[55] Under typical operating conditions at TMI-1 the maximum cladding temperature is 346°C,[56] meaning that crud deposits raised local cladding temperatures by over 100 or 150°C (180 or 270°F) or greater.

[54] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program;" p. C-24.
[55] 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.
[56] *World Nuclear Industry Handbook, 1995*, Nuclear Engineering International (England), p. 80.

Therefore, it is highly probable that if a LB LOCA had occurred at TMI-1 during a significant period of cycle 10, the heavy crud and oxide layers on the cladding would have caused the PCT to exceed 2200°F.

Discussing crud and its effect on increasing cladding temperature, the paper "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10" states:

> The cause of the higher temperature on the outer face of the peripheral rods is believed to result from local deposition of a crud layer, which impeded heat transfer. Steam blanketing within a layer of dense crud could significantly increase local temperatures, and it has been implicated in past fuel failures in low duty PWRs, and more recently in failures in higher duty plants. The effect of steam blanketing would be similar to a dryout, both would preclude water to effectively remove heat from the fuel rod surface, causing the fuel rod to over-heat.[57]

At TMI-1 Cycle 10, the first leaking rod (a symptom of a cladding perforation) was detected 121 days into the cycle.[58] When cladding is perforated by corrosion, increases in offgas activity are detected in the coolant. Different steps can be taken: the power can be suppressed at the assemblies where leaking rods are detected or the fuel cycle can be terminated in order to remove the failed fuel rods. But because corrosion is not detected during plant operation, there is often a significant length of time before corrosion progresses and perforates cladding and causes an increase in offgas activity, meaning that heavily corroded fuel rods are often operated at full power for significant periods of time. It is hypothesized that at TMI-1 Cycle 10 cladding temperatures of a range of 450 to 500°C or greater lasted "for an indeterminate time, but within the range of ~1000 to 10 hours for the respective temperature limits."[59]

In 1995, TMI-1 had PWR Zr-4 fuel-rod cladding with a thickness of .67 mm or 670 μm (microns).[60] After cycle 10, 38 fuel assemblies were observed with a Distinctive Crud Pattern ("a mottled appearance of a dark, nearly black surface with jagged patches

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[57] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 343.
[58] Id., p. 339.
[59] Id., p. 342.
[60] *World Nuclear Industry Handbook 1995*, p. 80.

of white showing through").[61] Additionally, after cycle 10, the maximum oxide thickness measured on a fuel rod was 111.1 μm, at an axial elevation of 118.5 inches.[62] Therefore, the equivalent cladding reacted ("ECR"); that is, the percentage of the cladding of that rod that had oxidized, was 10.6% (this percentage is calculated by dividing the oxide thickness (111.1 μm) by the oxide to metal ratio of 1.56[63] (the value 1.56 is derived from the atomic weights of the elements involved in the chemical reaction of oxygen and Zircaloy cladding) and then dividing that value (71.2 μm) by the cladding thickness (670 μm)).

It is pertinent that, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," from 2000, states, "[r]ecent out-of-reactor measured elastic and plastic cladding strain values from high burnup cladding from two PWR fuel vendors have shown a decrease in Zr-4 cladding ductilities when oxide thicknesses begin to exceed 100 μm. As a result, the NRC staff has encouraged fuel vendors to establish a maximum oxide thickness limit of 100 μm."[64] (This is a NRC recommendation for guidance; it is not a legally binding regulation.) (It is also significant, that the TMI-1 Cycle 10 cladding—because of the low thermal conductivity of the crud layer—had an oxide thickness measured at over 100 μm (on one-cycle cladding), and that one-cycle cladding was initially perforated by oxidation only 121 days into the cycle.)[65]

If there had been a LOCA at TMI-1 Cycle 10, it is highly probable that the ECR, at the location where oxide thickness was measured at 111.1 μm, would have increased from a pre-accident value of 10.6% to a during-accident value exceeding 17%. Petitioner's point, however, is not to make an issue out of this supposition about the ECR; after all, during cycle 10, fuel rods had failed due to local corrosion perforations,[66]

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[61] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 340.
[62] Id., p. 344.
[63] NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, p. 243.
[64] 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.
[65] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 339.
[66] Id., p. 343.

and at the cladding perforations the ECR was already 100%. The point is rather to focus on the role that the thermal resistance of heavy layers of oxide and crud on cladding would play during a LOCA.

The maximum observed crud thickness from TMI-1 Cycle 10 was measured at 33 μm.[67] However, the analysis of the crud deposits on the cladding conducted after cycle 10 could not be thorough because most of the crud samples that had been collected disappeared into a storage pool, with a pH of about 4.5, before they were examined.[68] Typically, a great deal of PWR crud comes off the cladding during reactor shutdown: as much as four kilograms of crud can depart from cladding surfaces during reactor shutdown. Hence, the thickness of the crud that deposits on the cladding during plant operation is often unknown.[69] Thus, in the case of TMI-1 Cycle 10, the crud thicknesses were almost certainly much thicker than the values measured; perhaps they were 100 μm or greater. In fact, crud deposits on cladding in PWRs have been measured at up to 125 μm thick.[70]

1. The Thermal Conductivities of Crud and Zirconium Dioxide.

As already mentioned, crud layers increased local cladding surface temperatures by over 180 or 270°F or greater during cycle 10 because the thermal conductivity of the crud was very low. Pertaining to the thermal conductivity of crud, Bo Cheng of Electric Power Research Institute ("EPRI"), at the NRC's Advisory Committee on Reactor Safeguards ("ACRS"), Reactor Fuels Subcommittee, September 30, 2003, stated:

> [T]he thermal conductivity of the crud all depends on the morphology more than from the type, the chemical composition because the crud, say, it comes as a solid, the solid iron oxide conductivity is better than zirconium by maybe a factor of two to five. ... If the morphology is such that it would cause a steam blanketing, then your steam has extremely poor conductivity, maybe two orders of magnitude lower than the... The

[67] Id., p. 340.
[68] 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. 241.
[69] Id., pp. 241-242.
[70] Id., p. 133.

crud is so difficult to characterize. And the conductivities all so much depend on the morphology.[71]

The thermal conductivity of crud is reported to be 0.8648 W/mK in volume two of the code manual, "Frapcon-3: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup."[72] This same value for the thermal conductivity of crud is given in NUREG-1230, dating back to 1988.[73] So it is evident that—although 0.8648 W/mK is a very low thermal conductivity—Cheng thought a crud layer with steam blanketing would have an even lower thermal conductivity than 0.8648 W/mK. He stated that steam trapped within a crud layer (with steam blanketing) would have "extremely poor conductivity." (This is because the thermal conductivity of steam is extremely low: it has been measured between values of 0.0154 and 0.0678 Btu/hrftF (0.0267 and 0.1173 W/mK) between temperatures of 250 and 1500°F (394.26 and 1088.7°K) and pressures of 20 and 2000 psia.)[74] And he also stated that "crud is...difficult to characterize" and that its thermal "conductivities...depend on [its] morpholog[ies]."[75] (For example, a ~100 μm crud flake, from a boiling water reactor ("BWR") that experienced crud-induced fuel failures, has been described as having a 50% porosity with voids and plugged up steam chimneys.)[76] Therefore, it is clear that certain morphologies of crud have thermal conductivities that are less than 0.8648 W/mK and of unknown values.

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[71] Id., p. 240.
[72] Pacific Northwest National Laboratory, NUREG/CR-6534, Volume 2, "Frapcon-3: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," 1997, p. 2.8.
[73] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.14-4.
[74] C. A. Meyer, R. B. McClintock, G. J. Silvestri, R. C. Spencer, Jr., ASME Steam Tables, The American Society of Mechanical Engineers, 1983, p. 281.
[75] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 240.
[76] 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. 14.

In fact, EPRI currently (to be completed in 2008) has a goal to "[p]erform crud simulation tests to determine the effect of tenacious crud on fuel surface heat transfer."[77] This study is for BWR crud but its results could also be applied to PWRs. As the article "Fuel Crud Formation and Behavior," describing a project for sampling BWR crud flakes, claims: "methods developed to determine the number and distribution of chimneys and capillaries on fuel crud surface, essential in understanding the adequacy of heat transfer within...crud deposit[s] have large applications for both PWR and BWR fuel depositions."[78] Whether or not the findings of this research will be applied to modeling crud for calculations of PCTs during postulated LOCAs is open to conjecture.

Zirconium dioxide (ZrO2) or zirconia also has a low thermal conductivity, and is used industrially as an insulating material.[79] The thermal conductivity of zircaloy-cladding oxide has been measured between 1.354 and 1.586 W/mK at temperatures between 297 and 1450°K, dipping as low as 0.955 W/mK at 668°K.[80] Additionally, volume one of the code manual, "Frapcon-3" (published in 1997) states that the current MATPRO function for ZrO2, uses values of approximately 2.0 W/mK for the thermal conductivity of ZrO2 at typical LWR operating cladding temperatures. But it also states that in 1995 an EPRI-sponsored Halden Reactor experiment gave indications that the value for the thermal conductivity of ZrO2 at the same temperatures may be much lower, at values close to 1.0 W/mK.[81] Like crud, oxide also impedes heat transfer:

> Crud inhibits heat transfer, increasing clad temperature and oxide layer growth rate. ... Oxide can form, with or without the benefit of crud, in the presence of sustained elevated cladding temperatures. Like crud,

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[77] EPRI, "2007 Portfolio, AP41.02 Fuel Reliability," located at: http://mydocs.epri.com/docs/Portfolio/PDF/2007_P041-002.pdf (accessed on 01/21/07), p. 5.

[78] Charles Turk, "Fuel Crud Formation and Behavior," *Nuclear Plant Journal*, January-February 2006, located at: http://npj.goinfo.com/NPJMain.nsf/504ca249c786e20f85256284006da7ab/89609e291af0b7b286257194007576c1?OpenDocument (accessed on 01/21/07).

[79] The following is from a description of the "Hot Spot 110: 1700°C Lab Furnace": "The zirconia insulation incorporated in the Hot Spot 110 has the lowest thermal conductivity of any commercially available high temperature insulation," located at: http://www.zircarzirconia.com/doc/F-HS.pdf (accessed on 01/21/07).

[80] K. E. Gilchrist, "Thermal Property Measurements on Zircaloy-2 and Associated Oxide Layers," *Journal of Nuclear Materials*, 62, 1976, pp. 257-264.

[81] Pacific Northwest National Laboratory, NUREG/CR-6534, Volume 1, "Frapcon-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application," 1997, p. 8.3.

formation of an oxide layer inhibits heat transfer causing accelerated corrosion which can potentially lead to fuel failure.[82]

2. A Discussion of an Individual Fuel Rod at TMI-1 Cycle 10.

Fuel rod (rod 011) was one of the fuel rods that failed at TMI-1 Cycle 10. As already mentioned, the maximum oxide thickness measured on rod 011 was 111.1 μm, and elsewhere on the same rod oxidation had perforated the cladding. There is a high probability that during cycle 10, on rod 011 there had been a crud layer that was approximately 100 μm thick on top of the 111.1 μm oxide layer. Such a crud layer would have been the primary cause of the 111.1 μm oxide layer, as well as the perforations on rod 011. Therefore, it is highly probable that rod 011 had an approximately 200 μm layer of oxide and crud combined; that is, a heavy layer with a very low thermal conductivity (with plausible values of approximately 1.4 W/mK or less for the oxide portion of the layer and a value less than 0.8648 W/mK—most likely, substantially less—for the crud portion).

If a LB LOCA had occurred at TMI-1 Cycle 10, the very low thermal conductivity of the 200 μm layer of oxide and crud combined would have inhibited effective heat transfer and with high probability caused the PCT to exceed 2200°F (~1204°C), in violation of the parameter set forth in 10 C.F.R. § 50.46(b)(1).

The 111.1 μm oxide layer and the crud layer of a possible thickness of approximately 100 μm were on rod 011 at an elevation 118.5 inches above the bottom of the end plug, or about 80% above the base of the active core.[83] 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.[84] Additionally, the transcript of proceedings from NRC ACRS, Reactor Fuels Subcommittee, September 30, 2003, states

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[82] Yovan D. Lukic and Jeffery S. Schmidt, "Taming the Crud Problem: The Evolution," *Advances in Nuclear Fuel Management III Conference*, Hilton Head Island, South Carolina, October 2003.
[83] *World Nuclear Industry Handbook 1995*, p. 80. At Three Mile Island during cycle 10 the active core height was 3.6 meters or 143.9 inches.
[84] 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.

that nine fuel rods failed at the span-six elevation.[85] Crud was also observed in spans five and seven[86] or at elevations from around 80 to 120 inches above the bottom of the end plug[87] (around 55 to 80% above the base of the active core). Typically, during a postulated LOCA the PCT occurs approximately 60% above the base of the active core. (At TMI-1 Cycle 10, the PCT would have most likely occurred in span six: "the hottest span.") Therefore, for clean cladding at TMI-1, during a postulated LOCA, it seems highly probable that at an elevation of 118.5 inches, the temperature would have been calculated within 100°F of the PCT. (Of course, this is a simple assessment: the phenomena occurring during a LOCA are very complex; the actual elevation of the PCT for clean cladding at TMI-1, around 1995, can be researched, as well as what the temperature would have been at the 118.5 inch elevation for clean cladding.) Therefore, during cycle 10, it is highly probable that the cladding temperature would have exceeded 2200°F during a postulated LB LOCA at the 118.5 inch elevation on rod 011, as well as at the span-six elevation where rod 011 failed.

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."[88] Cladding hydrogen content was measured on a non-failed rod at 700 ppm.[89] 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.[90]

An increase in cladding hydrogen content contributes to cladding embrittlement. The transcript of proceedings of NRC, ACRS, Reactor Fuels Subcommittee Meeting,

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[85] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 236.
[86] R. Tropasso, J. Willse, B. Cheng, "Crud-Induced Cladding Corrosion Failures in TMI-1 Cycle 10," p. 340.
[87] Id., p. 344.
[88] Id., p. 342.
[89] Id., p. 347.
[90] 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.

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 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].[91]

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."[92]

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.

C. The Stored Energy in Fuel Sheathed within Crudded and Oxidized Cladding.

When cladding temperatures are increased by layers of crud and oxide, there is also an increase in the stored energy in the fuel, because the thermal resistance of insulating layers of crud and oxide increase fuel temperatures. Describing how the quantity of stored energy in the fuel is partly related to heat transfer through cladding NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," states:

> The amount of stored energy [in the fuel] is directly related to the temperature of the fuel center and the temperature gradient from the fuel center to the fuel surface. The temperature of the fuel center and the temperature gradient are a function of thermal conduction within the

[91] 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).
[92] David B. Mitchell and Bert M. Dunn, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," p. xviii.

> pellet, fuel pellet cracking, heat transfer through the fuel cladding gap, *and conduction through the cladding* [emphasis added].[93]

It is significant that the quantity of stored energy in the fuel is partly related to heat transfer through cladding, because crud and oxide layers impede heat transfer through cladding. For this reason, the stored energy in the fuel increases when the cladding sheathing it is heavily crudded and oxidized. And the stored energy in the fuel at the onset of a LOCA is significant for determining the PCT during a LOCA; "Compendium of ECCS Research for Realistic LOCA Analysis," states, "[d]uring the blowdown period, fuel and cladding temperatures are in part determined by the initial stored thermal energy in the fuel rods."[94]

Concerning the effect that fuel temperatures (or stored energy), at the onset of a LOCA, have on the PCT (during a postulated LOCA), the NRC, discussing Westinghouse's PAD 4.0 code, states:

> The PAD 4.0 code is used to provide initial thermal conditions (fuel centerline and volume average temperatures) and rod pressures for the start of the LOCA analysis. The fuel volume average temperature is the *primary* PAD input that impacts the calculation of maximum peak cladding temperatures (PCTs) to verify that Westinghouse meets the 10 CFR 50.46 requirement of PCT not exceeding 2200°F. Traditionally, the NRC has required that a best estimate code such as PAD 4.0 maintain a 95 percent bounding estimate of centerline and volume average temperatures at a 95 percent confidence level for input to LOCA analysis. ... From the example LOCA calculation provided by Westinghouse, the *maximum fuel temperatures* (*generally corresponds to maximum PCTs*) calculated by PAD 4.0 are consistent with the FRAPCON-3 code results [emphasis added].[95]

Furthermore, concerning stored energy in the fuel at the onset of a LOCA, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> The amount of stored energy in the fuel at the start of a reactor transient plays an important role in the response of the fuel rod during the transient.

[93] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 6.14-2.
[94] Id., p. 6.14-1.
[95] NRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report WCAP-15063-P, Revision 1, 'Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)'," April 24, 2000, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML003706392, pp. 7-8.

> A portion of the stored energy (typically more than 50%) is removed during the blowdown period of LOCA. The residual thermal energy is in the fuel rod at the beginning of the adiabatic heatup phase of the LOCA. The amount of residual thermal energy influences the time required to quench the reactor core with emergency cooling water.[96]

And 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.[97]

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 increase of the stored energy in the fuel caused by a heavy crud layer is substantial (in some cases, enough to increase local cladding temperatures in excess of 300°F or even 600°F during operation).[98] This increase raises the stored energy in the fuel to levels higher than that of fresh, BOL fuel, or fuel with burnups between 30 to 35 GWd/MTU, which are considered the times of life or burnups that represent the maximum stored energy that fuel has during operation. (Fresh, BOL fuel is generally considered to have the maximum stored energy in fuel; however, COPERNIC and FRAPCON-3 (computer codes, programs that simulate LOCAs) calculate that mid-life

[96] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 6-14-2.

[97] 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.

[98] NRC, "River Bend Station – NRC Problem Identification and Resolution Inspection Report 0500458/2005008," Report Details, pp.10-12. River Bend, a BWR, operated with local cladding temperatures approaching 1200°F during cycles 8 and 11.

fuel with burnups of about 30 to 35 GWd/MTU have the maximum stored energy.)[99] The quantities of the stored energy in BOL fuel or fuel with burnups between 30 to 35 GWd/MTU are what are used to calculate PCTs during postulated LOCAs by computer codes because the maximum stored energy in the fuel corresponds to the maximum PCT.[100]

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. When the cladding reacts with steam, an exothermic reaction occurs which generates heat, additionally heating up the cladding. Regarding the significance of time and temperature during a LOCA, NRC staff member, Ralph Meyer, states:

> [I]n 10 CFR 50.46, part [b]...[t]here is an oxidation limit of 17[%]. This is really a *time* limit because it was understood at the beginning and we know it now that the embrittling process does not take place on the surface where the oxide is accumulating [during a LOCA]. It is related to the diffusion of oxygen in the metal. The diffusion process and the oxidation process run at about the same speed. And so an oxidation limit was used. It [is] very convenient. ... It gives you a nearly constant number that you can use as a limit. ... [A] basic LOCA transient calculation is just *time* and *temperature*. And then you run along with that some equation for oxidation and get a calculated oxidation amount during the transient [emphasis added].[101]

[99] "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. 10.
[100] NRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report WCAP-15063-P, Revision 1, 'Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)'," pp. 7-8.
[101] NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/fullcommittee/2007/ac020207.pdf (accessed on 02/27/07), pp. 15-16.

Regarding oxidation-induced cladding embrittlement, "Compendium of ECCS Research for Realistic LOCA Analysis" states:

> Embrittled cladding can fragment upon introduction of the emergency cooling water in a severe accident. During a high-temperature transient accident, the cladding becomes embrittled by steam oxidation of the zircaloy cladding and the formation of thick reaction layers of brittle oxide and oxygen-stabilized alpha zircaloy. The extent of cladding oxidation, and hence embrittlement, is a function of *temperature*, *time*, and the supply of steam and zircaloy. Embrittlement of the cladding may lead to loss of coolable geometry and is thus relevant to the safety analysis of fuel rods [emphasis added].[102]

The increase of the stored energy (caused by a heavy crud layer) and the delay in the transfer of that stored energy to the coolant would also increase the time until quench. As cited before, "Compendium of ECCS Research for Realistic LOCA Analysis" states, "[t]he amount of residual thermal energy [in the fuel rod] influences the *time* required to quench the reactor core with emergency cooling water [emphasis added]."[103]

D. There is Little or No Evidence that Crud has Ever been Properly Factored into PCT Calculations for Postulated LOCAs.

As already discussed, the increased stored energy in the fuel and its effect on increasing cladding temperatures during a LOCA, and its effect on delaying the transfer of stored energy to the coolant during the blowdown phase, is very significant for how cladding would be affected during a LOCA. However, there is little or no evidence that crud has ever been properly factored into PCT calculations for postulated LOCAs for nuclear power plants. 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 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

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[102] NRC, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," p. 6.14-6.
[103] Id., p. 6-14-2.

effect, it is simply that users have not chosen to consider this phenomen[on] [emphasis not added].[104]

An example of not properly factoring the thermal conductivity of crud into a PCT calculation for a postulated LOCA is in "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," dating from 2002. It 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, provided BOL conditions remain limiting. In the event that the SBLOCA cumulative PCT becomes ≥ 1700°F, this issue must be reassessed."[105] 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.[106]

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.[107] 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

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104 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.
105 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.
106 Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs."
107 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.

would be if the cladding were clean: this would result in a substantially higher than "+4.0°F…crud deposition penalty"[108] 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]."[109] 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.

E. The Non-Conservatism of Not Factoring Crud into PCT Calculations.

The fact that a heavy crud layer would increase the quantity of stored energy in the fuel at the onset of a LOCA is significant; it means that the value of the PCT would also increase, above that of fuel with the same burnup, sheathed within clean cladding. (Of course, this does not hold for fresh, BOL fuel, because such fuel has clean cladding at the beginning of its use.) And heavily crudded one-cycle fuel has a higher quantity of stored energy in the fuel than BOL fuel; crud has caused local cladding temperatures to increase by over 300°F during the operation of PWRs. Furthermore, the effects of crud can be quick; *e.g.*, at TMI-1 Cycle 10, one-cycle fuel had a cladding perforation, caused by corrosion, detected only 121 days into the cycle. It is also significant that most of the cladding that experienced crud-induced corrosion failures recently at PWRs was high-

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[108] Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," Attachment 2, p. 6, note 3.
[109] 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.

power, one-cycle cladding,[110] and that crud layers approximately 100 μm thick at Callaway Cycle 6 were on high-power, one-cycle cladding.[111]

BOL fuel or fuel with burnups between 30 to 35 GWd/MTU (sheathed within clean cladding) are the times of life or burnups considered to have the maximum stored energy that fuel has during operation. For this reason, the quantities of stored energy in BOL fuel or fuel with burnups between 30 to 35 GWd/MTU are what are used to calculate PCTs during postulated LOCAs by computer codes, because the maximum stored energy in the fuel corresponds to the maximum PCT.[112]

(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 to yield the highest PCTs for postulated LOCAs. At the January 2007, NRC, 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 approximately 20 or 25 GWd/MTU yield the highest PCTs. He also stated that at burnups of around 30 GWd/MTU there is an approximate 10% reduction in achievable power, which yields PCTs that are approximately 100°C (180°F) lower than those of fresher fuel.)[113]

It is significant that the stored energy of fuel (high-power, one-cycle fuel) sheathed within heavily crudded and oxidized cladding is substantially greater than the BOL quantities of stored energy that are factored into calculating PCTs during postulated LOCAs. PCT calculations that helped qualify power uprates at a number of PWRs (including IP-2 and -3) were not calculated with the maximum stored energy that fuel can

[110] NRC, Advisory Committee on Reactor Safeguards, Reactor Fuels Subcommittee Meeting Transcript, September 30, 2003, p. 235.
[111] See Bo Cheng, David Smith, Ed Armstrong, Ken Turnage, Gordon Bond, "Water Chemistry and Fuel Performance in LWRs," see also Union Electric Company, "Callaway Plant, 10 CFR 50.46 Annual Report, ECCS Evaluation Model Revisions," 2002, Attachment 2, p. 6, note 3.
[112] "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," p. 10. WCOBRA/TRAC calculates that fresh, BOL fuel has the maximum stored energy in fuel; COPERNIC and FRAPCON-3 calculate that mid-life fuel with burnups of about 30 to 35 GWd/MTU have the maximum stored energy.
[113] 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.

attain during operation: recent experiences with fuel at TMI-1, Palo Verde Unit 2, Seabrook, and Callaway were not modeled. Hence, the values of the PCTs generated by these ECCS evaluation calculations are non-conservative. Furthermore, the power uprates that these non-conservative PCTs helped qualify make it highly probable that nuclear power plants will operate in violation of 10 C.F.R. § 50.46(b)(1).

F. Crud and Axial Offset Anomaly.

Axial offset anomaly ("AOA") or CIPS (crud induced power shift) is a phenomenon caused by crud deposition on cladding; it helps provide an indication of how frequently crud affects the operation of PWRs. AOA occurs in PWRs when crud deposits on cladding have a level of boron sufficient to reduce the rate of fission in the vicinity of the crud. "NRC Information Notice 97-85: Effects of Crud Buildup and Boron Deposition on Power Distribution and Shutdown Margin" provides a brief description of AOA and how it occurs:

> High core power results in increased subcooled nucleate boiling in the upper core, which, in turn, causes greater crud accumulation on the fuel assemblies. Lithium borate is absorbed and concentrated in the crud layer, reducing the fission rate in the upper portion of the core. ... As a result of the reduced fissioning in the upper core, the power distribution shifts toward the bottom of the core.[114]

AOA is caused by crud deposits on fuel rods; therefore, the number of occurrences of AOA helps provide an indication of how often PWR 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.[115] However, there can also be crud deposits on fuel rods thicker than 35 μm that do not cause AOA, because not all crud deposits have the quantity of boron that causes AOA. As mentioned before, the thickest layer of crud to be

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[114] NRC, "NRC Information Notice 97-85: Effects of Crud Buildup and Boron Deposition on Power Distribution and Shutdown Margin," December 11, 1997, located at: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1997/in97085.html (accessed on 01/21/07).

[115] 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. 7.

measured in a PWR was 125 μm thick (it caused AOA but not cladding perforations). As of 2003 more than 30 fuel cycles in 16 U.S. PWRs had exhibited AOA.[116]

Current problems caused by crud at PWRs—AOA among them—are discussed in EPRI document "2006 Portfolio, 41.002 Fuel Reliability" as follows:

> 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 nuclear power's competitiveness; utilities would like to solve this problem as soon as possible.[117]

AOA is detectable during the operation of PWRs; if necessary, after it is detected, a plant can be operated at a lower power level, as H. A. Sepp of Westinghouse points out:

> Several PWRs have experienced [AOAs] due to buildup of boron within crud deposits, in portions of the reactor core which experience subcooled boiling. AOA is characterized by axial power distributions that are more skewed to the bottom of the core than would be expected. These AOA are detectable, and are closely monitored to ensure that adequate shutdown margins can be maintained. In extreme cases, reductions in operating power level have been required to maintain adequate shutdown margin.[118]

What Sepp describes is a case of reducing operating power according to the severity of AOAs, not according to the thickness of crud deposits. In PWRs there can be heavy crud deposits with low levels of boron; in such cases there would only be slight AOAs or no AOAs at all. For example, TMI-1 Cycle 10 had only a slight AOA even though it had enough crud to induce corrosion fuel failures. In common practice, if a

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[116] 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).
[117] EPRI, "2006 Portfolio, 41.002 Fuel Reliability," located at: http://www.epriweb.com/public/2006_P041-002.pdf (accessed on 01/21/07), pp. 2-3.
[118] Attachment of a letter from H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse Electric Company to Annette L. Vietti-Cook, Secretary of the Commission, NRC, December 17, 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML020530290.

heavy crud layer was detected during plant operation that did not cause an AOA, it is unlikely that the operating power level would be reduced, because the thermal resistance of the crud and how it would raise the PCT in the event of a LOCA would most likely not be considered problematic.

IV. CONCLUSION

After decades of operating experience, heavy crud and/or oxide layers on cladding or crud-induced corrosion failures remain within the realm of anticipated operational occurrences at nuclear power plants. Moreover, power uprates and longer fuel cycles increase the likelihood of heavy crud and/or oxide layers on cladding. Discussing current trends in the nuclear industry for both PWRs and BWRs (crud or corrosion related fuel failures occurred at BWRs in six of the years from 1997 to 2004)[119] an 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.[120]

This EPRI document also refers to the "many operational surprises utilities have experienced recently"[121] at nuclear power plants, stating that among the operational surprises were "higher than expected [levels of] cladding corrosion and hydriding."[122] (Petitioner would add higher than expected levels of crud.) Meanwhile, in recent years, numerous power uprates and license renewals, largely based on non-conservative ECCS evaluation calculations (like those that helped qualify the recent power uprates of IP-2 and -3), have been granted for nuclear power plants.

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[119] Rosa Yang, Odelli Ozer, Kurt Edsinger, Bo Cheng, Jeff Deshon, "An Integrated Approach to Maximizing Fuel Reliability," p. 11.
[120] EPRI, "2006 Portfolio, 41.002 Fuel Reliability," p. 1.
[121] Id., p. 2.
[122] Id.

At the NRC's 539th ACRS Meeting, in February 2007, Jennifer Uhle, Deputy Division Director of Materials Engineering in the Office of Nuclear Regulatory Research, stated that the current criteria of 10 C.F.R. § 50.46 are non-conservative.[123] When discussing possible revisions to 10 C.F.R. § 50.46 at the same meeting, and at the NRC's ACRS, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting, in January 2007, 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"[124] is being considered for regulation status for a new revised version of 10 C.F.R. § 50.46, due in 2009.[125]

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.[126] He also provided examples regarding cladding ductility where the value 1.2 (the F factor)[127] was multiplied by the pre-accident ECR in order to calculate the remaining percentage of oxidation allowed to occur during a LOCA.[128] 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."[129] (A NRC regulatory guide states that the F factor can vary from 1 to 1.6.[130] The F factor's use in LOCA calculations is also being considered for regulation status.)[131]

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[123] NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, located at: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/fullcommittee/2007/ac020207.pdf (accessed on 02/27/07), pp. 8, 10.
[124] NRC, "NRC Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation."
[125] 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.
[126] NRC, Advisory Committee on Reactor Safeguards, Materials, Metallurgy, and Reactor Fuels Subcommittee Meeting Transcript, January 19, 2007, p. 13.
[127] Id., pp. 179-182.
[128] Id., pp. 31-33.
[129] Id., p. 31.
[130] Id., pp. 181-182.
[131] Id., p. 246.

At the January 2007 meeting, Meyer cited the following "worst case zircaloy," postulated-LOCA example:

> [W]e have a de facto corrosion limit [that is] used in safety analyses of 100 microns, and zircaloy can get that much corrosion on it if you push it hard enough. And so [I have] taken this example right at the limit. So this would be what I call a worst case zircaloy example, and the 100 microns is about [10%] ECR, and you multiply that by 1.2, subtract the 12 from 17, and you get five percent, a fairly small number."[132]

At the same meeting, in response to Meyer's "worst-case zircaloy" example, Mitch Nissley of Westinghouse Electric Company, stated:

> [W]e anticipated an F factor on the order of 1.5 or 1.6, and I went through and did a shorthand calculation just to show this was similar to Dr. Meyer's use of the 100 micron Zr-4 design limit. One hundred microns...is effectively a design limit at least for Westinghouse fuel, for all of our cladding types. ... If you use a large F factor, [you have] got no room to work with with curb design limits on fuel.[133]

Then to argue that high-burnup fuel would not be subjected to extremely high temperatures in the event of a LOCA, Nissley added:

> Once [the fuel] starts to burn down in terms of its achievable power levels, achievable peak cladding temperatures and the corresponding transient oxides drop off dramatically, and that comment is valid for all break sizes, both large and small beaks. The important conclusion from this [is that] high burnup fuel [used in the U.S.] cannot [have PCTs that] approach 1200[°C].[134]

Then, after citing data from sample LOCA calculations that demonstrated that one-cycle fuel from burnups of zero to approximately 20 or 25 GWd/MTU yield the highest PCTs,[135] Nissley concluded:

> I showed you in that one example [LB LOCA] calculation that even using more or less an upper bound for the high burnup fuel in terms of relative power, it was more than 1000[°F], less limiting than the fresh fuel. I think the real message here is [we have] done a lot of testing at 1200[°C] with high burnup fuel. The double-sided [oxidation] reaction is also a limit that I know of to [occur at] very high temperature[s, above approximately

[132] Id., p. 33.
[133] Id., p. 243.
[134] Id., pp. 250-251.
[135] Id., p. 251.

1100°C[136]]. [A]nd [with high burnup fuel] you just [cannot reach temperatures that high].[137]

The conclusion to be drawn from Nissley's argument is that the F factor would only apply to cladding sheathing high-burnup fuel that would not have enough power (or stored energy) to reach PCTs above temperatures where rapid oxidation occurs. Hence, pre-accident oxidation (and the phenomena the F factor accounts for) would not cause a loss of cladding ductility for properly managed high-burnup fuel in the event of a LOCA.[138]

However, Nissley did not mention scenarios involving one-cycle fuel of burnups between zero and 25 GWd/MTU, with heavily crudded cladding. Such fuel would yield substantially higher PCTs than the examples he cited. Furthermore, the cladding, in such scenarios, where there are crud-induced corrosion failures, would be substantially more degraded than that of Meyer's "worst-case zircaloy" example, where cladding had an ECR value of 10%. At TMI-1 Cycle 10, cladding was measured with approximately 10% ECR; however, there were also cladding perforations due to corrosion at TMI-1 Cycle 10, so its maximum ECR was actually 100% on one-cycle, high-powered fuel. The fuel at TMI-1 Cycle 10 (and any other nuclear power plant with crud-induced corrosion failures on one-cycle, high power fuel rods) would yield higher PCTs than fresh, BOL fuel; and this fuel was sheathed within cladding that was more degraded than that of Meyer's "worst case zircaloy" example. Hence, such fuel is similar to BOL fuel but it yields even higher PCTs, and such cladding is similar to high-burnup cladding but it is even more degraded.

Uhle is certainly correct that the current criteria of 10 C.F.R. § 50.46 are non-conservative, though the NRC still has not addressed the extent of this non-conservatism. For example, the NRC has not addressed the role that the thermal resistance of crud and oxide layers on cladding play in determining the quantity of stored energy in the fuel at the onset of a postulated LOCA. (Petitioner recently submitted a petition for rulemaking (PRM-50-84), requesting that the NRC amend Appendix K to Part 50—ECCS Evaluation

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136 Id.

137 Id., p. 261.

138 This is discussed in more detail in NRC, Advisory Committee on Reactor Safeguards 539th Meeting Transcript, February 2, 2007, pp. 60-64.

Models I(A)(1), *The Initial Stored Energy in the Fuel*, 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.)[139]

It is significant that Westinghouse as well as other vendors have not modeled fuel and cladding conditions that have occurred at nuclear power plants in recent years for their NRC approved best-estimate ECCS evaluation models (like the WCOBRA/TRAC code), used in lieu of Appendix K to Part 50 calculations.[140] In the case of IP-2 and -3, less than 40 miles north of New York City, this lack of ECCS evaluation model conservatism puts millions of people at risk. It is highly probable that if a LB LOCA were to occur at either IP-2 or -3 under circumstances where one-cycle fuel would have heavily crudded and oxidized cladding or crud-induced corrosion failures, the parameters set forth in 10 C.F.R. § 50.46(b) would be violated.

It is also significant that for LOCAs in general, irrespective of cladding conditions at the onset, the NRC-approved ECCS evaluation calculations of Westinghouse as well as other vendors do not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a postulated accident that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[141]

For conditions where one-cycle fuel would have heavily crudded and/or oxidized cladding or would have crud-induced corrosion failures the current ECCS design basis for either IP-2 or -3 is substantially non-conservative in at least the following aspects: 1) 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

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[139] Mark Edward Leyse, Petition for Rulemaking 50-84, March 15, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070871368.

[140] NRC, "10 CFR Part 50: Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," 2005, located at: http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2005/secy2005-0052/2005-0052scy.pdf (accessed on 01/21/07), p. 11. Best-estimate ECCS evaluation models used in lieu of Appendix K calculations are described in NRC Regulatory Guide 1.157.

[141] NRC, NUREG-1861, "Peer Review of GSI-191 Chemical Effects Research Program," p. C-24.

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.

It is also significant that for LOCAs in general, irrespective of cladding conditions at the onset, recent ECCS evaluation calculations for IP-2 and -3 did not model "[t]he [dissolved and suspended] solids [in the reactor coolant system water following a postulated accident that] might...cling tenaciously to the fuel cladding and compromise the heat transfer meant to occur during the post-LOCA period."[142]

To uphold its congressional mandate to protect the lives, property, and environment of the people of New York, the NRC must not allow the power levels of IP-2 and -3 to be based on ECCS evaluation calculations that violated 10 C.F.R. § 50.46(a)(1)(i). Petitioner requests that the NRC either 1) revoke the operating license of IP-2 and -3, 2) order the licensee of IP-2 and -3 to immediately suspend the operations of IP-2 and -3, or 3) temporarily shutdown IP-2 and -3, per 10 C.F.R. § 2.202. In the event of option 3, Petitioner requests that the NRC order the licensee to conduct conservative

[142] Id.

ECCS evaluation calculations for IP-2 and -3 that are compliant with 10 C.F.R. § 50.46(a)(1)(i). After conservative ECCS evaluation calculations are conducted in compliance with 10 C.F.R. § 50.46(a)(1)(i), Petitioner requests that the NRC order the licensee of IP-2 and -3 to reduce the power levels of IP-2 and -3 to legally acceptable levels if the results of the conservative ECCS evaluation calculations show that the plants are operating in violation of 10 C.F.R. § 50.46(b). If implemented, the enforcement actions proposed in this petition would improve public and plant worker safety.

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