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NL-12-083

John A. Ventosa
Site Vice President

June 14, 2012

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: 30-Day 10 CFR 50.46 Report of Change in Emergency Core Cooling System Model

Indian Point Unit Nos. 2 and 3
Docket Nos. 50-247 & 50-286
License Nos. DPR-26 & DPR-64

Reference: 1. LTR-NRC-12-27, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (Entergy) is submitting a 30-day report of changes in the emergency core cooling system (ECCS) models for Indian Point Units 2 (IP2) and 3 (IP3). This report is submitted in accordance with 10 CFR 50.46(a)(3)(ii) as a significant change in peak cladding temperature (PCT) due to accumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F.

On December 13, 2011, the Nuclear Regulatory Commission (NRC) issued an Information Notice (IN 2011-21) regarding the effect of a potentially significant error associated with fuel pellet thermal conductivity degradation (TCD) on peak cladding temperature (PCT) in the Westinghouse Electric Company LLC (Westinghouse) furnished realistic emergency core cooling system evaluation models. Several utilities subsequently received 10 CFR 50.54(f) letters and in response to those letters, Westinghouse submitted directly to the NRC (Reference 1) a description of the methodology and assumptions used to determine the estimated PCT impact due to TCD. The reports in Attachments 1 and 2 are based on this methodology.

As required by 10 CFR 50.46(a)(3)(ii), Attachments 1 and 2 provide the reports for the evaluation of fuel pellet TCD in the large break loss-of-coolant accident (LBLOCA) analysis. COLR changes have been implemented to assure the inputs of the evaluation remain valid.

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NRR

The proposed schedule for providing a reanalysis and the regulatory commitment associated with this correspondence is identified in Attachment 3.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-254-6710.

Sincerely,

Patrick W. Conroy for John A Ventosa

JAV/rw/jc

Attachments:

1. Indian Point Unit 2 10 CFR 50.46 Report for the Evaluation of Fuel Pellet Thermal Conductivity Degradation.
2. Indian Point Unit 3 10 CFR 50.46 Report for the Evaluation of Fuel Pellet Thermal Conductivity Degradation.
3. Indian Point Units 2 and 3 Reanalysis Schedule and Regulatory Commitment List.

cc: Mr. William Dean, Regional Administrator, NRC Region 1
Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
IPEC NRC Resident Inspector's Office
Mr. Francis J. Murray, President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Department of Public Service

Attachment 1

Indian Point Unit 2 10 CFR 50.46 Report for the Evaluation of Fuel Pellet Thermal Conductivity Degradation

Entergy Nuclear Operations, Inc.
Indian Point Unit 2
Docket No. 50-247

Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown
(Non-Discretionary Change)

Background

Fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown were not explicitly considered in the Indian Point Unit 2 Large Break Loss-of-Coolant Accident (LBLOCA) Analysis of Record (AOR). NRC Information Notice 2011-21 (Reference 1) notified addressees of recent information obtained concerning the impact of irradiation on fuel thermal conductivity and its potential to cause significantly higher predicted peak cladding temperature (PCT) results in realistic emergency core cooling system (ECCS) evaluation models (EMs). This evaluation provides an estimated effect of TCD on the PCT calculation for the ECCS at Indian Point Unit 2.

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A quantitative evaluation as discussed in Reference 2 was performed to assess the PCT effect of TCD and peaking factor burndown with other considerations of burnup on the Indian Point Unit 2 LBLOCA analysis and concluded that the estimated PCT impact is 209 °F for 10 CFR 50.46 reporting purposes. The peaking factor burndown included in the evaluation is provided in Table 1. Entergy and its vendor, Westinghouse Electric Company LLC, utilize processes, which ensure that the LOCA analysis input values conservatively bound the as-operated plant values for those parameters.

Table 1. Peaking Factors Assumed in the Evaluation of TCD

Rod Burnup [MWd/MTU]	$F_{\Delta H}$ (1), (2)	F_Q Transient (1)	F_Q Steady-State
0	1.65	2.3	1.8
30,000	1.65	2.3	1.8
60,000	1.30	1.8	1.4
62,000	1.30	1.8	1.4

(1) Includes uncertainties.
(2) Hot assembly average power follows the same burndown, since it is a function of $F_{\Delta H}$.

References

1. NRC Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011. [NRC ADAMS Accession Number ML113430785]

2. LTR-NRC-12-27, Letter from J.A. Gresham (Westinghouse) to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

Evaluation of Design Input Changes with Respect to Plant Operation

Background

To demonstrate compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning peak cladding temperature (PCT) when explicitly considering fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown in the Indian Point Unit 2 Large Break Loss-of-Coolant Accident (LBLOCA) analysis, design input values were revised to more closely represent current plant operation. These input changes are not changes to the approved 2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM. The updated inputs for Indian Point Unit 2 include:

- Reduction in $F_{\Delta H}$
- Reduction in hot assembly average power
- Reduction in Transient F_Q
- Reduction in Steady-State F_Q
- Reduction in upper bound steam generator tube plugging

Entergy and its vendor, Westinghouse Electric Company LLC, utilize processes, which ensure that the LOCA analysis input values conservatively bound the as-operated plant values for those parameters.

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A quantitative evaluation as discussed in Reference 1 was performed to estimate an overall PCT change due to changes in design input parameters. The evaluation concluded that the estimated PCT impact of these design input changes is -63 °F for 10 CFR 50.46 reporting purposes.

References

1. LTR-NRC-12-27, Letter from J.A. Gresham (Westinghouse) to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Plant Name: Indian Point Unit 2
Utility Name: Entergy Nuclear Northeast
Revision Date: 5/7/2012

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	2/15/2005	**Limiting Break Size:** Guillotine
F_Q:	2.5	**$F_{\Delta H}$:**	1.7	
Fuel:	15x15 Upgraded	**SGTP (%):**	10	
Notes:	See note (c) for current peaking factor and SGTP limitations.			

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1962	1	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1. HOTSPOT Fuel Relocation Error	0	3	(a)
B. PLANNED PLANT MODIFICATION EVALUATONS			
1. Bent Fuel Assembly Alignment Pins	5	1	
2. Changes to Containment Sump Strainer Evaluation	0	2	
3. Evaluation of Design Input Changes with Respect to Plant Operation	-63	4,5	(b, c)
C. 2012 ECCS MODEL ASSESSMENTS			
1. Evaluation of Pellet Thermal Conductivity Degradation and Peaking Factor Burndown	209	4,5	(b)
D. OTHER			
1. None.	0		
LICENSING BASIS PCT + PCT ASSESSMENTS PCT =	2113		

References:

1. WCAP-16405-P, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for Indian Point Unit 2 Nuclear Plant Using the ASTRUM Methodology," May 2005.

2. LTR-LIS-06-299, "Evaluation of Sump Strainer Modification on Indian Point Unit 2 (IPP) Best Estimate Large Break LOCA Analyses and Transmittal of Revised PCT Sheets," May 2006.

3. LTR-LIS-07-379, "10 CFR 50.46 Reporting Text for HOTSPOT Fuel Relocation Error for Indian Point 2," June 2007.

4. LTR-NRC-12-27, Letter from J.A. Gresham (Westinghouse) to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 2012.

5. NF-ECH-12-23, "Information Regarding the Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown Including Design Input Changes for Indian Point Unit 2," May 2012.

Notes:

(a) The magnitude of this assessment was changed from prior versions of the PCT rackup sheet due to a recalculation in Reference 5.

(b) These assessments are coupled via an evaluation of burnup effects, which include thermal conductivity degradation, peaking factor burndown and design input changes.

(c) Design input changes were a reduction in $F_Q(tr)$ from 2.5 to 2.3, $F_Q(ss)$ from 2.0 to 1.8, $F_{\Delta H}$ from 1.7 to 1.65 and a corresponding reduction in Pbar-HA; and maximum steam generator tube plugging from 10% to 5%. These peaking factor limits and steam generator tube plugging limit supersede the values cited for the analysis-of-record.

Attachment 2

Indian Point Unit 3 10 CFR 50.46 Report for the Evaluation of Fuel Pellet Thermal Conductivity Degradation

Entergy Nuclear Operations, Inc.
Indian Point Unit 3
Docket No. 50-286

Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown
(Non-Discretionary Change)

Background

Fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown were not explicitly considered in the Indian Point Unit 3 Large Break Loss-of-Coolant Accident (LBLOCA) Analysis of Record (AOR). NRC Information Notice 2011-21 (Reference 1) notified addressees of recent information obtained concerning the impact of irradiation on fuel thermal conductivity and its potential to cause significantly higher predicted peak cladding temperature (PCT) results in realistic emergency core cooling system (ECCS) evaluation models (EMs). This evaluation provides an estimated effect of TCD on the PCT calculation for the ECCS at Indian Point Unit 3.

Affected Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

Estimated Effect

A quantitative evaluation as discussed in Reference 2 was performed to assess the PCT effect of TCD and peaking factor burndown with other considerations of burnup on the Indian Point Unit 3 LBLOCA analysis and concluded that the estimated PCT impact is 185 °F for 10 CFR 50.46 reporting purposes. The peaking factor burndown included in the evaluation is provided in Table 1. Entergy and its vendor, Westinghouse Electric Company LLC, utilize processes, which ensure that the LOCA analysis input values conservatively bound the as-operated plant values for those parameters.

Table 1. Peaking Factors Assumed in the Evaluation of TCD

Rod Burnup [MWd/MTU]	$F_{\Delta H}$ (1), (2)	F_Q Transient (1)	F_Q Steady-State
0	1.65	2.3	1.8
30,000	1.65	2.3	1.8
60,000	1.30	1.8	1.4
62,000	1.30	1.8	1.4

(1) Includes uncertainties.
(2) Hot assembly average power follows the same burndown, since it is a function of $F_{\Delta H}$.

References

1. NRC Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011. [NRC ADAMS Accession Number ML113430785]

2. LTR-NRC-12-27, Letter from J.A. Gresham (Westinghouse) to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

Evaluation of Design Input Changes with Respect to Plant Operation

Background

To demonstrate compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning peak cladding temperature (PCT) when explicitly considering fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown in the Indian Point Unit 3 Large Break Loss-of-Coolant Accident (LBLOCA) analysis, design input values were revised to more closely represent current plant operation. These input changes are not changes to the approved 1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model. The updated inputs for Indian Point Unit 3 include:

- Reduction in $F_{\Delta H}$
- Reduction in hot assembly average power
- Reduction in Transient F_Q
- Reduction in Steady-State F_Q

Entergy and its vendor, Westinghouse Electric Company LLC, utilize processes, which ensure that the LOCA analysis input values conservatively bound the as-operated plant values for those parameters.

Affected Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

Estimated Effect

A quantitative evaluation as discussed in Reference 1 was performed to estimate an overall PCT change due to changes in design input parameters. The evaluation concluded that the estimated PCT impact of these design input changes is -95 °F for 10 CFR 50.46 reporting purposes.

References

1. LTR-NRC-12-27, Letter from J.A. Gresham (Westinghouse) to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Plant Name: Indian Point Unit 3
Utility Name: Entergy Nuclear Northeast
Revision Date: 5/10/2012

Composite

Analysis Information

EM:	CQD (1996)	**Analysis Date:**	1/23/2004	**Limiting Break Size:** Guillotine
F_Q:	2.5	**$F_{\Delta H}$:**	1.7	
Fuel:	OFA w/IFMs	**SGTP (%):**	10	

Notes: Analysis also supports 15×15 Upgraded Fuel (Reference 1). See note (c) for current peaking factor limitations.

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1944**	**1**	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1. Revised Blowdown Heatup Uncertainty Distribution	5	3	
2. HOTSPOT Fuel Relocation Error	0	6	(a)
B. PLANNED PLANT MODIFICATION EVALUATONS			
1. Bent Fuel Assembly Alignment Pins	5	2	
2. Changes Due to Containment Sump Strainer Metal Evaluation	2	4	
3. 80 °F Initial Containment/Accumulator Temperature	12	4	
4. Thimble Plug Removal	0	5	
5. Evaluation of Design Input Changes with Respect to Plant Operation	-95	6	(b, c)
C. 2012 ECCS MODEL ASSESSMENTS			
1. Evaluation of Pellet Thermal Conductivity Degradation and Peaking Factor Burndown	185	6	(b)
D. OTHER			
1. None.	0		
LICENSING BASIS PCT + PCT ASSESSMENTS	PCT = 2058		

References:

1. WCAP-16178-P, "Best-Estimate Analysis of the Large Break Loss of Coolant Accident for Indian Point Unit 3 Nuclear Plant Stretch Power Uprate," March 2004.

2. INT-99-211, "10 CFR 50.46 Annual Notification and Reporting for 1998," S.M. Ira, 03/05/99.

3. INT-05-15, "10 CFR 50.46 Annual Notification and Reporting for 2004," April 2005.

4. INT-08-14, Revision 1, "Evaluation of Containment Sump Strainer Modifications, Reduced Containment Initial Temperature and Accumulator Water Temperature on Indian Point Unit 3 (INT) Best Estimate Large Break LOCA Analysis," February 2010.

5. INT-10-15, "10 CFR 50.46 Report for Thimble Plug Removal," December 2010.

6. NF-ECH-12-24, "Information Regarding the Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown Including Design Input Changes for Indian Point Unit 3," May 2012.

Notes:

(a) The magnitude of this assessment was changed from prior versions of the PCT rackup sheet due to a recalculation in Reference 6.

(b) These assessments are coupled via an evaluation of burnup effects, which include thermal conductivity degradation, peaking factor burndown and design input changes.

(c) Design input changes were a reduction in F_Q(tr) from 2.5 to 2.3, F_Q(ss) from 2.0 to 1.8, $F_{\Delta H}$ from 1.7 to 1.65 and a corresponding reduction in Pbar-HA. These peaking factor limits supersede the values cited for the analysis-of-record.

Attachment 3

Indian Point Units 2 and 3 Reanalysis Schedule and Regulatory Commitment List

Entergy Nuclear Operations, Inc.
Indian Point Units 2 and 3
Docket Nos. 50-247 and 50-286

INDIAN POINT UNITS 2 AND 3 LARGE BREAK LOSS-OF-COOLANT ACCIDENT REANALYSIS SCHEDULE

The estimated impact on Indian Point Units 2 and 3 Large Break LOCA (LBLOCA) Evaluation Model (EM) from fuel pellet thermal conductivity degradation (TCD) results in a significant change in peak cladding temperature (PCT), as defined in 10 CFR 50.46(a)(3)(i). 10 CFR 50.46(a)(3)(ii) requires the licensee to provide a report within 30 days, including a proposed schedule for a reanalysis or taking other action as may be needed to show compliance with 10 CFR 50.46. Entergy Nuclear Operations, Inc. (Entergy) has reviewed the information provided by Westinghouse Electric Company LLC (Westinghouse) and determined that the adjusted LBLOCA PCT values and the manner in which they were derived continue to comply with the requirements of 10 CFR 50.46. Entergy has evaluated the requirement for reanalysis specified in 10 CFR 50.46(a)(3)(ii) and hereby proposes a schedule for reanalysis.

On or before December 15, 2016, Entergy will submit to the Nuclear Regulatory Commission (NRC) LBLOCA analyses that apply NRC approved methods that include the effects of fuel pellet TCD. The date for the analyses submittal is contingent on the following milestones needed to perform a revised licensing basis LBLOCA analysis with an NRC approved emergency core cooling system (ECCS) EM that explicitly accounts for TCD:

1) Submittal by Westinghouse, to the NRC for review and approval, of revised fuel performance and LBLOCA EM methodologies that include the effects of TCD.

2) Prior NRC approval of a fuel performance analysis methodology that includes the effects of TCD. The new NRC-approved methodology would replace the current licensing basis methodology for Indian Point Unit Nos. 2 and 3 that is described in WCAP-15063-P-A, Revision 1 with Errata "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

3) Prior NRC approval of a LBLOCA EM that includes the effects of TCD and accommodates the ongoing 10 CFR 50.46(c) rulemaking process. The new methodology would replace the current licensing basis methodology for Indian Point Unit No. 2 that is described in WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005 and for the Indian Point Unit No. 3 that is described in WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

This information satisfies the reporting requirements of 10 CFR 50.46(a)(3)(ii).

Regulatory Commitment List

The following list identifies those actions committed to by Entergy Nuclear Operations, Inc. (Entergy) for Indian Point Unit Nos. 2 and 3 in this document. Any other actions discussed in the submittal represent intended or planned actions by Entergy. They are described only as information and are not Regulatory Commitments. Please notify Mr. Robert Walpole, Manager, Licensing, at (914) 254-6710 of any questions regarding this document or associated Regulatory Commitments.

Regulatory Commitment	**Due Date**
1. Entergy will submit to the NRC LBLOCA analyses that apply NRC approved methods that include the effects of fuel pellet thermal conductivity degradation (TCD) for Indian Point Unit Nos. 2 and 3.	On or before December 15, 2016.