Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1395

INDIANA
MICHIGAN
POWER

August 26, 2004

AEP:NRC:4046
10 CFR 50.46

Docket Nos: 50-315
50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D. C. 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
ANNUAL REPORT OF LOSS-OF-COOLANT ACCIDENT
EVALUATION MODEL CHANGES

References:

1. Letter from S. A. Greenlee, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2 Annual Report of Loss-Of-Coolant Accident Evaluation Model Changes," C0801-19, dated August 31, 2001.

2. Letter from John A. Zwolinski, I&M, to U. S. NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2, Thirty-Day Report of Loss-of-Coolant Accident Evaluation Model Changes," AEP:NRC:3046-02, dated November 24, 2003.

3. Letter from John A. Zwolinski, I&M, to U. S. NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Annual Report of Loss-of-Coolant Accident Evaluation Model Changes," AEP:NRC:3046, dated August 27, 2003.

4. Letter from John A. Zwolinski, I&M, to U. S. NRC Document Control Desk, "Withdrawal of Proposed Technical Specification Changes and Exemption Requests to Support Use of Framatome ANP, Inc. Fuel," AEP:NRC:4565-02, dated June 14, 2004.

A001

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), the licensee for the Donald C. Cook Nuclear Plant (CNP), is transmitting an annual report of loss-of-coolant accident (LOCA) model changes affecting the peak cladding temperature (PCT) for CNP Units 1 and 2. Attachment 1 to this letter describes the current assessments against the large break and small break LOCA analyses of record. Attachment 2 provides the large break and small break LOCA analyses of record PCT values and error assessments.

There are no new PCT assessments against the Unit 1 limiting large break, Unit 1 limiting small break, and Unit 2 limiting large break LOCA analyses. The latest Westinghouse Electric Company evaluation, utilizing the NOTRUMP model, demonstrated a +35°F PCT increase for the Unit 2 small break LOCA analysis with the safety injection system cross-tie valves open because of changes to the NOTRUMP bubble rise/drift flux model. The PCT increase of 70°F for the Unit 2 small break LOCA analysis with the safety injection system cross-tie valves closed, was previously reported in Reference 2.

Because previously reported changes to the Unit 1 limiting small break, the Unit 2 limiting large break, and Unit 2 limiting small break analyses of record were classified as significant, I&M submitted a schedule for performing new analyses in Reference 1. I&M submitted a revised Unit 2 schedule in Reference 3. This revised schedule was submitted to support the transition to a new fuel vendor for CNP, Unit 1 and Unit 2. I&M had contracted with Framatome, ANP, Inc. as the new CNP fuel vendor and initiated a fuel transition license amendment request. I&M has subsequently withdrawn the license amendment request (Reference 4) and contracted with a different fuel vendor to perform these services. For this reason, I&M cannot meet the currently proposed schedule for performing the Unit 1 limiting small break, the Unit 2 limiting large break, and the Unit 2 limiting small break analyses. I&M has evaluated the technical basis for meeting the criteria defined in 10 CFR 50.46 and has determined that the acceptance criteria continue to be met. Therefore, I&M plans to submit a new schedule by December 31, 2004, for providing reanalysis or taking other action as may be needed to show continued compliance with 10 CFR 50.46 requirements. This commitment is provided in Attachment 3.

Should you have any questions, please contact Mr. Michael K. Scarpello, Supervisor of Nuclear Licensing, at (269) 697-5020.

Sincerely,

John A. Zwolinski
Safety Assurance Director

DB/rdw

Attachments

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o attachments
J. T. King, MPSC, w/o attachments
MDEQ – WHMD/HWRPS
NRC Resident Inspector
J. G. Lamb, NRC Washington, DC

ATTACHMENT 1 TO AEP:NRC:4046

ASSESSMENT AGAINST THE LOSS-OF-COOLANT ACCIDENT ANALYSES OF RECORD

Small Break Loss-of-Coolant Accident (LOCA) Analysis of Record

NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections

Background

NOTRUMP was updated to resolve some inconsistencies in several drift flux models as well as the nodal bubble rise/droplet fall models.

Estimated Effect

The NOTRUMP update does not impact the Unit 1 plant specific calculation. The Unit 2 representative plant calculation impact is discussed below.

As shown in the peak fuel cladding temperature (PCT) accounting provided in Attachment 2, Table 4, Item A.5 and Table 5, Item B.1, implementation of the NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections leads to a bounding 35 degree Fahrenheit (°F) increase of the calculated PCT for 10 CFR 50.46 purposes.

The 35°F PCT increase from the NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections results in a further 35°F PCT increase due to the SPIKE Correlation Revision penalty shown on Table 4. The Spike computer program and associated methodology computes PCT increases that would result from fuel rod burst PCT penalties for SBLOCA analyses and is applicable only for cases where PCT exceeds 1700°F. The 35°F PCT penalty associated with the Spike Correlation is included in Attachment 2, Table 4, Item A.2, Burst and Blockage/Time in Life. The 95°F value listed includes this 35°F penalty.

The referenced letter previously reported Table 4, Items A.2 and A.5 pursuant to 10 CFR 50.46(a)(3)(ii).

Conclusion

This transmittal satisfies the annual reporting requirement of 10 CFR 50.46(a)(3)(ii). Attachment 2 demonstrates that the PCT value remains within the 2200°F PCT limit specified in 10 CFR 50.46(b)(1).

Reference

Letter from John A. Zwolinski, Indiana Michigan Power Company, to Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2, Thirty-Day Report of Loss-of-Coolant Accident Evaluation Model Changes, dated November 24, 2003.

ATTACHMENT 2 TO AEP:NRC:4046

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
LARGE AND SMALL BREAK LOSS-OF-COOLANT ACCIDENT
PEAK CLAD TEMPERATURE SUMMARY

TABLE 1

CNP UNIT 1

LARGE BREAK LOCA

Evaluation Model: BASH
F_Q=2.15 $F_{\Delta H}$=1.55 SGTP=15% Break Size: C_d=0.4
Operational Parameters: RHR System Cross-Tie Valves Closed, 3250[1] MWt Reactor Power
Notes: ZIRLO clad, IFM grids

LICENSING BASIS		
	Analysis-of-Record, December 2000	PCT= 2038°F
MARGIN ALLOCATIONS (Δ PCT)		
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS[2]	
	1. LOCBART Cladding Emissivity Errors[3]	-11°F
	2. Reduced Containment Spray Temperature	+23°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	0°F
C.	OTHER	
	1. Transition Core Penalty[4]	+31°F
D.	LICENSING BASIS PCT+ MARGIN ALLOCATIONS	PCT= 2081°F

[1] The 3250 MWt power level used in the reanalysis is acceptable because it bounds the Unit 1 3304 MWt steady state power limit in the operating license after adjusting for recapture of feedwater flow measurement and power calorimetric uncertainty.

[2] ECCS model assessments are no longer being listed by year of occurrence. Instead the errors are being identified by error type. This is consistent with Westinghouse reporting methods and does not change the overall PCT.

[3] This is a revised assessment. The prior generic assessment of +6°F has been changed to -11°F based on plant specific information.

[4] This penalty will be dropped once all fuel assemblies include the Intermediate Flow Mixing (IFM) Grids.

TABLE 2

CNP UNIT 1

SMALL BREAK LOCA

Evaluation Model: NOTRUMP
F_Q=2.32 $F_{\Delta H}$=1.55 SGTP=30% 3" cold leg break
Operational Parameters: SI System Cross-Tie Valves Closed, 3250[5] MWt Reactor Power
Notes: ZIRLO clad, IFM grids

LICENSING BASIS		
	Analysis-of-Record, December 2000	PCT= 1720°F
MARGIN ALLOCATIONS (Δ PCT)		
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS[6]	
	1. Asymmetric HHSI Delivery	+50°F
	2. Reduction in Turbine Driven Auxiliary Feedwater Flow	+109°F
	3. Burst and Blockage / Time in Life	+111°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	0°F
C.	OTHER	0°F
D.	LICENSING BASIS PCT+ MARGIN ALLOCATIONS	PCT= 1990°F

[5] The 3250 MWt power level used in the reanalysis is acceptable because it bounds the Unit 1 3304 MWt steady state power limit in the operating license after adjusting for recapture of feedwater flow measurement and power calorimetric uncertainty.

[6] ECCS model assessments are no longer being listed by year of occurrence. Instead the errors are being identified by error type. This is consistent with Westinghouse reporting methods and does not change the overall PCT.

TABLE 3

CNP UNIT 2

LARGE BREAK LOCA

Evaluation Model: BASH
F_Q=2.335 $F_{\Delta H}$=1.64 SGTP=15% Break Size: C_d=0.6
Operational Parameters: RHR System Cross-Tie Valves Closed, 3413 MWt Reactor Power[7]

LICENSING BASIS			
	Analysis-of-Record, December 1995		PCT= 2051°F
MARGIN ALLOCATIONS (Δ PCT)			
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS[8]		
	1.	ECCS double disk valve leakage	+8°F
	2.	BASH current limiting break size reanalysis to incorporate LOCBART spacer grid single phase heat transfer and LOCBART zirc-water oxidation error	+58°F
	3.	Cycle 13 ZIRLO Fuel Evaluation[9]	-50°F
	4.	Reduced Containment Spray Temperature	+47°F
B.	NEW 10 CFR 50.46 ASSESSMENTS		0°F
C.	OTHER		0°F
D.	LICENSING BASIS PCT+ MARGIN ALLOCATIONS		PCT= 2114°F

[7] Power level used as basis for PCT acceptance is 3413 MWt due to the reanalysis (see item A.2) to provide an integrated error effect on the limiting case. This reanalysis (item A.2) is not considered the analysis-of-record due to the spectrum of break sizes not being reanalyzed to ensure that the limiting break size at 3413 MWt with the errors incorporated would not change. Thus, the analysis-of-record remains as the 1995 analysis at a power level of 3588 MWt. The difference between the limiting case PCT (2051°F) and the PCT from the reanalysis of that limiting break size at 3413 MWt is the 58°F being reported. The 3413 MWt power level used in the reanalysis is acceptable because it bounds the Unit 2 3468 MWt steady state power limit in the operating license after adjusting for recapture of feedwater flow measurement and power calorimetric uncertainty.

[8] ECCS model assessments are no longer being listed by year of occurrence. Instead the errors are being identified by error type. This is consistent with Westinghouse reporting methods and does not change the overall PCT.

[9] The ZIRLO fuel evaluation used a version of LOCBART that corrected for the Vapor Film Flow Regime Heat Transfer and Cladding Emissivity Errors. As reported in previous reports, these errors were -15°F and -10°F respectively. Thus, since this reanalysis incorporates the errors previously reported, the errors are no longer being reported individually.

TABLE 4

CNP UNIT 2

SMALL BREAK LOCA

Evaluation Model: NOTRUMP
F_Q=2.45 $F_{\Delta H}$=1.666 SGTP=15% 3″ cold leg break
Operational Parameters: SI System Cross-Tie Valves Closed, 3250 MWt Reactor Power[10]

LICENSING BASIS			
	Analysis-of-Record, March 1992		PCT= 1956°F
MARGIN ALLOCATIONS (Δ PCT)			
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS[11]		
	1.	Limiting NOTRUMP and SBLOCA analysis[12]	-214°F
	2.	Burst and blockage / time in life	+95°F
	3.	Asymmetric HHSI Delivery	+50°F
	4.	NOTRUMP mixture level tracking/region depletion errors	+13°F
	5.	NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections	+35°F
B.	NEW 10 CFR 50.46 ASSESSMENTS		0°F
C.	OTHER		0°F
D.	LICENSING BASIS PCT+ MARGIN ALLOCATIONS		PCT= 1935°F

[10] Unit 2 is licensed to a 3468 MWt steady-state power level. However, 3304 MWt is assumed for the small break LOCA analysis with the SI system cross-tie valves closed. This is because Unit 2 Technical Specification 3.5.2 limits thermal power to 3304 MWt with a safety injection cross-tie valve closed. The 3250 MWt power level used in the reanalysis is acceptable because it bounds the Unit 2 3304 MWt steady state power limit in the operating license after adjusting for recapture of feedwater flow measurement and power calorimetric uncertainty.

[11] ECCS model assessments are no longer being listed by year of occurrence. Instead the errors are being identified by error type. This is consistent with Westinghouse reporting methods and does not change the overall PCT.

[12] This reanalysis is considered an evaluation because a full spectrum of break sizes was not analyzed. This reanalysis incorporated the errors previously reported (Letter from M. W. Rencheck, Indiana Michigan Power Company to Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2 Annual Report of Loss-of-Coolant Accident Evaluation Model Changes," submittal C1000-07, dated October 27, 2000) in the individual years in which they occurred. The difference between the analysis-of-record limiting break size PCT and the reanalysis PCT is -214°F. Thus, since this reanalysis incorporates the errors previously reported, the errors are no longer being reported individually. Note that this does not impact the resulting PCT as it remains at 1935°F. It is only an accounting change.

TABLE 5

CNP UNIT 2

SMALL BREAK LOCA

Evaluation Model: NOTRUMP
F_Q=2.32 $F_{\Delta H}$=1.62 SGTP=15% 4″ cold leg break
Operational Parameters: SI System Cross-Tie Valves Open, 3588 MWt Reactor Power

LICENSING BASIS			
		Analysis-of-Record, August 1992	PCT= 1531°F
MARGIN ALLOCATIONS (Δ PCT)			
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS[13]		
	1.	Effect of SI in Broken Loop	+150°F
	2.	Effect of Improved Condensation Model	-150°F
	3.	Drift Flux Flow Regime Errors	-13°F
	4.	LUCIFER Error Corrections	-16°F
	5.	Containment Spray During Small Break LOCA	+20°F
	6.	Boiling Heat Transfer Correlation Error	-6°F
	7.	Steam Line Isolation Logic Error	+18°F
	8.	Axial Nodalization, and SBLOCTA correction	+3°F
	9.	NOTRUMP Specific Enthalpy Error	+20°F
	10.	SBLOCTA Fuel Rod Initialization Error	+10°F
	11.	Loop Seal Elevation Error	-38°F
	12.	NOTRUMP Mixture Level Tracking / Region Depletion Errors	+13°F
B.	NEW 10 CFR 50.46 ASSESSMENTS		
	1.	NOTRUMP Bubble Rise/Drift Flux Model Inconsistency Corrections	+35°F
C.	OTHER		0°F
D.	LICENSING BASIS PCT+ MARGIN ALLOCATIONS		PCT= 1577°F

[13] ECCS model assessments are no longer being listed by year of occurrence. Instead the errors are being identified by error type. This is consistent with Westinghouse reporting methods and does not change the overall PCT.

ATTACHMENT 3 TO AEP:NRC:4046

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
Provide a new schedule for performing the Unit 1 limiting small break, the Unit 2 limiting large break, and the Unit 2 limiting small break reanalyses or taking other action as may be needed to show continued compliance with 10 CFR 50.46 requirements.	December 31, 2004