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INDIANA
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POWER

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C0801-19
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Donald C. Cook Nuclear Plant Units 1 and 2
ANNUAL REPORT OF LOSS-OF-COOLANT ACCIDENT
EVALUATION MODEL CHANGES

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company, the licensee for the Donald C. Cook Nuclear Plant (CNP), is submitting 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. Attachment 2 demonstrates that all PCT values remain within the 2200 degree Fahrenheit PCT limit specified in 10 CFR 50.46(b)(1).

The overall changes to the Unit 1 limiting small break, Unit 2 limiting large break, and Unit 2 limiting small break LOCA analysis are classified as significant in accordance with 10 CFR 50.46(a)(3)(i). These significant changes were previously reported as required by 10 CFR 50.46(a)(3)(ii). A revised schedule for reanalysis of each of these events is provided in Attachment 3, which lists new commitments made in this letter.

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Should you have any questions, please contact Mr. Ronald W. Gaston, Manager of Regulatory Affairs, at (616) 697-5020.

Sincerely,

SK [signature] for

S. A. Greenlee
Director of Design Engineering and Regulatory Affairs

/bjb

Attachments

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ATTACHMENT 1 TO C0801-19

ASSESSMENTS AGAINST THE LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSES OF RECORD

Indiana and Michigan Power Company (I&M) previously submitted annual 10 CFR 50.46 reports for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2 in References 1 and 2, respectively. The reported analysis of record peak cladding temperature (PCT) values in Attachment 2 remain the same as in References 1 and 2, as no new LOCA analyses have been performed since the previously submitted annual reports. New PCT assessments against the CNP Unit 1 and 2 large break LOCA (LBLOCA) analyses of record are described below. These new assessments are reflected in the PCT accounting in Attachment 2. There are no new PCT assessments against the CNP Unit 1 and 2 small break LOCA analyses of record.

The new error in the LBLOCA emergency core cooling system evaluation model described in this letter was identified and evaluated by Westinghouse Electric Company (Westinghouse), the nuclear steam supply system vendor for CNP, as well as the current supplier of the CNP fuel assemblies and licensing basis LOCA analyses. I&M was notified of this error by a routine Westinghouse 10 CFR 50.46 mid-year notification, dated December 2000. This error is not unique to CNP. It applies to plants using the affected computer codes and methodologies in their LOCA evaluation models.

Assessment Against the LBLOCA Analysis of Record

LOCBART Cladding Emissivity Error

Background:

The LOCBART computer code is used for fuel rod heat-up calculations in the LBLOCA analysis. LOCBART models radiation heat exchange between the rod, grid, and fluid during the reflood phase of the LOCA transient. An error was discovered in LOCBART whereby the cladding surface emissivity values used were substantially lower than the values that would be expected to exist during a LBLOCA reflood transient.

Estimated Effect:

As indicated in the PCT accounting in Attachment 2, the effect of the LOCBART cladding emissivity error is a six degree Fahrenheit (°F) penalty for Unit 1 and a 10°F benefit for Unit 2.

Conclusion

This submittal satisfies the annual reporting requirement of 10 CFR 50.46(a)(3)(ii).

References

1. Letter from M. W. Rencheck (I&M) to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Report of Loss-of-Coolant Accident Evaluation Model Changes," submittal C1200-03, dated December 20, 2000.

2. Letter from M. W. Rencheck (I&M) to U. S. 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 24, 2000.

ATTACHMENT 2 TO C0801-19

DONALD C. COOK NUCLEAR PLANT (CNP) UNITS 1 AND 2

LARGE AND SMALL BREAK LOSS-OF-COOLANT ACCIDENT (LOCA)
PEAK CLAD TEMPERATURE (PCT) 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 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 [1]	0°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	
	1. LOCBART cladding emissivity error	+ 6°F
C.	OTHER	
	1. Transition Core Penalty	+ 31°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT= 2075°F

1. 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 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 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 [1]	
	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

1. 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

$F_Q = 2.335$ $F_{\Delta H} = 1.644$ SGTP = 15% Break Size: $C_d = 0.6$

Operational Parameters: RHR System Cross-Tie Valves Closed, 3413 MWt Reactor Power [1]

LICENSING BASIS			
	Analysis-of-Record, December 1995 [1]		PCT = 2051°F
MARGIN ALLOCATIONS (Δ PCT)			
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS [2]		
	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 [1,3]	+58°F
	3.	LOCBART vapor film flow regime heat transfer error	-15°F
B.	NEW 10 CFR 50.46 ASSESSMENTS		
	1. LOCBART cladding emissivity error		-10°F
C.	OTHER		0°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS		PCT= 2092°F

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1. Power level used as basis for PCT acceptance is 3413 MWt due to the reanalysis (see item A2) 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 performed 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 3411 MWt steady-state power limit in the operating license.

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. On Table 1 of the previous PCT report, a 2°F penalty is listed for 1996 and a 56°F penalty is listed for 1999. Since the reanalysis of the limiting case in 1999 incorporated the effect of the error that was estimated to be a 2°F, the error is being reported as part of the reanalysis penalty since it is accounted for in the reanalysis.

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 [1]

LICENSING BASIS		
	Analysis-of-Record, March 1992	PCT = 1956°F
MARGIN ALLOCATIONS (Δ PCT)		
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS [2]	
	1. Limiting NOTRUMP and SBLOCTA analysis [3]	-214°F
	2. Burst and blockage / time in life	+60°F
	3. Asymmetric HHSI delivery	+50°F
	4. NOTRUMP mixture level tracking/region depletion errors	+13°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	0°F
C.	OTHER	0°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 1865°F

1. Unit 2 is licensed to a 3411 MWt steady-state power level. However, 3250 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 3250 MWt with a safety injection cross-tie valve closed.

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 reanalysis is considered an evaluation because a full spectrum of break sizes was not analyzed. This reanalysis incorporated the errors previously reported (Attachment 1, Reference 2) 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 1865°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 [1]	
	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 error	+13°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	0°F
C.	OTHER	0°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 1542°F

1. 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 C0801-19

PROPOSED REANALYSIS SCHEDULES

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.

A new Unit 1 SBLOCA analysis of the safety injection cross-tie valves closed case will be submitted.	Six months prior to expected Unit 1 Cycle 20 start-up date.
A new Unit 2 SBLOCA analysis of both the safety injection cross-tie valves closed case and the safety injection cross-tie valves open case will be submitted.	One year prior to expected Unit 2 Cycle 15 start-up date.
A new Unit 2 LBLOCA analysis will be submitted.	One year prior to expected Unit 2 Cycle 15 start-up date.