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AEP
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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

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

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), 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. I&M previously submitted the annual 10 CFR 50.46 report for Unit 1 and Unit 2 in the referenced letter. The reported analysis of record PCT values remain the same as those of the referenced letter. New PCT assessments against the CNP Unit 1 and Unit 2 large break LOCA analyses of record have been performed. Attachment 1 to this letter describes these assessments. 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 analyses of record 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). Because the previously reported changes to the analyses of record were classified as significant, I&M, in the referenced letter, submitted a schedule for performing new analyses. That schedule remains unchanged.

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This letter contains no new commitments.

Should you have any questions, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

Joel P. Gebbie
For

S. A. Greenlee
Director of Nuclear Technical Services

RV/bjb

Attachments

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ATTACHMENT 1 TO AEP:NRC:2046-01

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

Indiana and Michigan Power Company (I&M) previously submitted the annual 10 CFR 50.46 report for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2 in the referenced letter. The reported analysis of record peak cladding temperature (PCT) values in Attachment 2 of this letter remain the same as in the referenced letter, as no new LOCA analyses have been performed. New PCT assessments against the CNP Units 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 since the last annual report was submitted via the referenced letter.

Recent changes in CNP procedures allow the circulation of essential service water (ESW) through the containment spray (CTS) heat exchanger (HX) during normal operations. These changes will potentially result in a containment spray temperature lower than that used in the analyses of record. The new PCT assessments against the CNP Units 1 and 2 LBLOCA analyses of record are described below.

During the Unit 2 Cycle 13 refueling outage, fuel rods fabricated with zirconium alloy material ZIRLO were introduced into the Unit 2 reactor. The new PCT assessment against the CNP Unit 2 LBLOCA analyses of record are included below.

Assessments Against the LBLOCA Analysis of Record

Lower Containment Spray Temperature

Background:

Recent changes in CNP procedures allow the circulation of ESW through the CTS HX during normal operations. The purpose of circulating ESW through the normally isolated CTS HX during normal operations is to enable ESW system flow to achieve the 2000 gpm minimum flow prescribed for ESW pumps. These changes will potentially result in a containment spray temperature lower than that used in the analyses of record.

Estimated Effect:

The impact of lower containment spray temperature on the LBLOCA has been evaluated for Unit 1 and Unit 2. For Unit 1, the impact of the lower containment spray temperature is determined to be +23°F, and for Unit 2, the impact is determined to be +47°F. The resulting new cumulative PCTs have been determined to be 2081°F for Unit 1, and 2114°F for Unit 2. These resulting new PCTs remain below the 10 CFR 50.46 limit of 2200°F.

Unit 2 Safety Fuel Assemblies with ZIRLO Cladding

Background:

Westinghouse's new zirconium based alloy, known a ZIRLO, enhances fuel reliability and achieves extended burnup. I&M inserted the fuel assemblies containing fuel rods fabricated with the advanced zirconium alloy material ZIRLO into CNP Unit 2, Cycle 13.

Estimated Effect:

A new Unit 2 LBLOCA limiting case evaluation was performed to support the change to ZIRLO fuel initiated in Cycle 13. The LBLOCA evaluation incorporated ZIRLO cladding in the rod heat-up portion of the LBLOCA transient. This evaluation used a version of LOCBART that corrected for the Vapor Film Flow Regime Heat Transfer and Cladding Emissivity Errors. As reported in the referenced letter, these errors were −15°F and −10°F respectively. The impact of the ZIRLO cladding along with using a later version of LOCBART is determined to be -50°F. The evaluation demonstrates that the 10 CFR 50.46 Acceptance Criteria is satisfied for Unit 2 with ZIRLO clad fuel.

Conclusion

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

Reference

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

ATTACHMENT 2 TO AEP:NRC:2046-01

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]	
	1. LOCBART Cladding Emissivity Errors[2]	-11°F
B.	NEW 10 CFR 50.46 ASSESSMENTS	
	1. Reduced Containment Spray Temperature	+23°F
C.	OTHER	
	1. Transition Core Penalty[3]	+31°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT= 2081°F

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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.
2. This is a revised assessment. The prior generic assessment of +6°F has been changed to –11°F based on plant specific information.
3. 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 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]	+58°F
B.	NEW 10 CFR 50.46 ASSESSMENTS		
	1.	Cycle 13 ZIRLO Fuel Evaluation[3]	-50°F
	2.	Reduced Containment Spray Temperature	+47°F
C.	OTHER		0°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS		PCT= 2114°F

1. 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 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. 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 the previous report, 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 [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 SBLOCA 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 smallbreak 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 (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 24, 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 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.