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MURLEY,T.E. Document Control Branch (Document Control Desk)

SUBJECT: Notifies of LOCA model changes or errors that meet definition of significant as defined in 10CFR50.46. Determination of effect of LOCA model changes on plnt SBLOCA analyses encl.

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AEP:NRC:1118G

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License No. DPR-58 and DPR-74
REPORT OF SIGNIFICANT LOCA EVALUATION
MODEL CHANGES PURSUANT TO 10CFR50.46(a)(3)(ii)

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Attn: T. E. Murley

January 12, 1994

Dear Dr. Murley:

Pursuant to the requirements of 10CFR50.46(a)(3)(ii), this letter provides notification of LOCA model changes or errors that meet the definition of significant as defined in 10CFR50.46. Specifically, this letter reports on the effects of an input error in auxiliary feedwater flow rate used to analyze a small break LOCA analysis for Unit 1. This letter also serves to provide additional information regarding an error in the emergency core cooling system evaluation methodology used by Westinghouse for Units 1 and 2 that was reported previously in our letter AEP:NRC:1118F, dated October 19, 1993.

The original small break LOCA (SBLOCA) analysis for Units 1 and 2 of Donald C. Cook Nuclear Plant was performed by Westinghouse Electric Corporation (Westinghouse) using their NOTRUMP computer code. Subsequently, Cook Nuclear Plant Units 1 and 2 were reanalyzed to determine the impact of an increase in the main steam safety valve (MSSV) setpoint tolerance from ± 1% to ± 3%. This reanalysis was submitted to the NRC in support of our request to modify the MSSV Technical Specifications via our letter AEP:NRC:1169, dated November 11, 1992. We have discovered that the Unit 1 SBLOCA reanalysis submitted via letter AEP:NRC:1169 used nominal auxiliary feedwater flow rates (1258 gpm total delivery) instead of minimum auxiliary feedwater flow rates (750 gpm total delivery). Since minimum auxiliary

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feedwater flow rates are more limiting, the small break LOCA for Cook Nuclear Plant Unit 1 for ± 3% MSSV setpoint tolerance has been reanalyzed using minimum auxiliary feedwater flow rates (750 gpm total delivery).

The reanalysis is being reported under the provisions of 10CFR50.46 because the MSSV setpoint tolerance relaxation SBLOCA analyses have been included in our 10CFR50.46 reports since AEP:NRC:1118D, dated March 12, 1993, was submitted. Therefore, we believe it is appropriate to report errors found in these analyses under 10CFR50.46 even though they have not yet been approved and are not yet part of our licensing basis. These analyses include in their modeling non-discretionary changes to the SBLOCA model as defined in WCAP-13251. The MSSV SBLOCA analyses bound currently approved operational limits and are therefore conservative.

The revised analysis shows that the total peak clad temperature (PCT) with minimum auxiliary feedwater flow rate with high head safety injection cross ties closed is 2068°F, while the PCT reported in AEP:NRC:1169 with nominal auxiliary feedwater was 1878.7°F. Since the absolute value of the change in calculated PCT exceeds 50°F, the change meets the definition of significant provided in 10CFR50.46.

Attachment 1 contains the peak clad temperatures calculated specifically for the small break LOCA analyses for Cook Nuclear Plant Unit 1. The peak clad temperatures for the large break LOCA (LBLOCA) remain the same as reported to the NRC via our letter AEP:NRC:1118D, dated March 12, 1993. The licensing basis PCT plus permanent assessment for SBLOCA for Cook Nuclear Plant includes all previous permanent 10CFR50.46 model assessments. The assessment of this report consists of a PCT change of 189.3°F to the Unit 1 SBLOCA cross ties closed analysis because of the NOTRUMP code input error in auxiliary feedwater flow rate. The SBLOCA analysis with high head safety injection cross-tie valve open has not yet been performed by Westinghouse. For this case, the change in PCT for model assessments has been estimated to be 97°F.

Regarding the second issue of error in Emergency Core Cooling System (ECCS) methodology, our letter AEP:NRC:1118F, dated October 19, 1993, reported errors which Westinghouse discovered in their NOTRUMP computer code used for SBLOCA analysis, and an error in ECCS evaluation methodology used by Westinghouse for

both Units 1 and 2 of Cook Nuclear Plant. The following additional information on the Westinghouse ECCS evaluation methodology has been provided to us by the Westinghouse Owners Group (WOG).

Westinghouse plans to meet with the NRC in January 1994 to present the results of the investigations performed for safety injection (SI) in the broken loop, and discuss the condensation on safety injection test data and resulting correlation. This meeting will serve as the focal point for determining the level of detail and schedule required by the NRC in order to complete their review of the SI in the broken loop issue. Should final resolution of the SI in the broken loop issue not change the conclusion of Westinghouse letter ET-NRC-93-3971, dated September 21, 1993, and our letter AEP:NRC:1118F, we will not file an additional report. However, should final resolution of this issue change the conclusion of our current 30 day report, Indiana Michigan Power Company will provide further information to the NRC under 10CFR50.46 requirements.

Attachment 2 provides replacement pages for pages 1, 2, and 5 of Attachment 3 to our submittal AEP:NRC:1118F, which contained inadvertent errors.

Regarding plans for future analysis, the MSSV analysis will provide the new analysis of record for SBLOCA for Unit 1. The cross-tie valve open case is scheduled to be completed in June 1994.

Sincerely,

SJBrewer for

E. E. Fitzpatrick
Vice President

cad

Attachments

cc: A. A. Blind - Bridgman
G. Charnoff
J. B. Martin - Region III
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

ATTACHMENT 1 TO AEP:NRC:1118G

DETERMINATION OF EFFECT OF LOCA MODEL

CHANGES ON COOK NUCLEAR PLANT

SMALL BREAK LOCA ANALYSES

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 1

A. ANALYSIS OF RECORD PCT- 2122°F
(Comments: Evaluation Model: NOTRUMP, FQT- 2.32, FdH- 1.55, SGTP-15%,
Other: HHSI Cross Tie Valve Closed, 3250 Mwt Reactor Power)

B. 10 CFR 50.92 SAFETY EVALUATIONS ΔPCT- -268°F[1]

C. PRIOR PERMANENT LOCA MODEL ASSESSMENTS ΔPCT- -10°F

D. 1993 10 CFR 50.46 MODEL ASSESSMENTS
(Permanent Assessment of PCT Margin)
1. Auxiliary Feedwater Flow Rate Input Error in NOTRUMP Computer Code ΔPCT- 189.3°F

E. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT- 2033.3°F

Note: As discussed in the body of this submission, prior LOCA model assessments have been absorbed in new analyses performed to support a request for relaxation of the main steam safety valve (MSSV) setpoint tolerance. The analyses were submitted for NRC review with our letter AEP:NRC:1169, dated November 11, 1992. Since these analyses bound currently licensed operating conditions, the resulting changes are being reported in lieu of developing a rack up of evaluations for each issue that has been absorbed in the analyses.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 1

A. ANALYSIS OF RECORD PCT= 2122°F
(Comments: Evaluation Model: NOTRUMP, FQT= 2.32, FdH= 1.55, SGTP=15%,
Other: HHSI Cross Tie Valve Open, 3588 Mwt Reactor Power)

B. 10 CFR 50.92 SAFETY EVALUATIONS ΔPCT= -552°F[1]

C. PRIOR PERMANENT LOCA MODEL ASSESSMENTS ΔPCT= -13°F

D. 1993 10 CFR 50.46 MODEL ASSESSMENTS
(Permanent Assessment of PCT Margin)
1. Auxiliary Feedwater Flow Rate Input Error in NOTRUMP Computer Code ΔPCT= 97°F

E. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 1654°F

Note: As discussed in the body of this submission, prior LOCA model assessments have been absorbed in new analyses performed to support a request for relaxation of the main steam safety valve (MSSV) setpoint tolerance. The analyses were submitted for NRC review with our letter AEP:NRC:1169, dated November 11, 1992. Since these analyses bound currently licensed operating conditions, the resulting changes are being reported in lieu of developing a rack up of evaluations for each issue that has been absorbed in the analyses.

ATTACHMENT 2 TO AEP:NRC:1118G

REPLACEMENT PAGES FOR

ATTACHMENT 3 TO AEP:NRC:1118F LETTER

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 1

A. ANALYSIS OF RECORD PCT= 2122°F
(Comments: Evaluation Model: NOTRUMP, FQT= 2.32, FdH= 1.55, SGTP=15%,
Other: HHSI Cross Tie Valve Closed, 3250 Mwt Reactor Power)

B.	10 CFR 50.92 SAFETY EVALUATIONS	ΔPCT= -268°F[1]
C.	PRIOR PERMANENT LOCA MODEL ASSESSMENTS	ΔPCT= 3°F
D.	1993 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)	
	1. Effect of SI in Broken Loop	ΔPCT= 150°F
	2. Effect of Improved Condensation Model	ΔPCT= -150°F
	3. Drift Flux Flow Regime Errors	ΔPCT= -13°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	ΔPCT= 1844°F

Note: As discussed in the body of this submission, prior LOCA model assessments have been absorbed in new analyses performed to support a request for relaxation of the main steam safety valve (MSSV) setpoint tolerance. The analyses were submitted for NRC review with our letter AEP:NRC:1169, dated November 11, 1992. Since these analyses bound currently licensed operating conditions, the resulting changes are being reported in lieu of developing a rack up of evaluations for each issue that has been absorbed in the analyses.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 1

A.	ANALYSIS OF RECORD (Comments: Evaluation Model: NOTRUMP, FQT- 2.32, FdH- 1.55, SGTP-15%, Other: HHSI Cross Tie Valve Open, 3588 Mwt Reactor Power)	PCT- 2122°F
B.	10 CFR 50.92 SAFETY EVALUATIONS	ΔPCT- -552°F[1]
C.	PRIOR PERMANENT LOCA MODEL ASSESSMENTS	ΔPCT- 0°F
D.	1993 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)	
	1. Effect of SI in Broken Loop	ΔPCT- 150°F
	2. Effect of Improved Condensation Model	ΔPCT- -150°F
	3. Drift Flux Flow Regime Errors	ΔPCT- -13°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	ΔPCT- 1557°F

Note: As discussed in the body of this submission, prior LOCA model assessments have been absorbed in new analyses performed to support a request for relaxation of the main steam safety valve (MSSV) setpoint tolerance. The analyses were submitted for NRC review with our letter AEP:NRC:1169, dated November 11, 1992. Since these analyses bound currently licensed operation conditions, the resulting changes are being reported in lieu of developing a rack up of evaluations for each issue that has been absorbed in the analyses.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A.	ANALYSIS OF RECORD (Comments: Evaluation Model: NOTRUMP, FQT= 2.32, FdH= 1.62, SGTP=15%, Other: HHSI Cross Tie Valve Open, 3588 Mwt Reactor Power)	PCT= 1357°F
B.	10 CFR 50.92 SAFETY EVALUATIONS	ΔPCT= +174°F[1]
C.	1993 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin)	
	1. Effect of SI in Broken Loop	ΔPCT= 150°F
	2. Effect of Improved Condensation Model	ΔPCT= -150°F
	3. Drift Flux Flow Regime Errors	ΔPCT= -13°F
D.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT= 1518°F

Note: As discussed in the body of this submission, prior LOCA model assessments have been absorbed in new analyses performed to support a request for relaxation of the main steam safety valve (MSSV) setpoint tolerance. The analyses were submitted for NRC review with our letter AEP:NRC:1169, dated November 11, 1992. Since these analyses bound currently licensed operating conditions, the resulting changes are being reported in lieu of developing a rack up of evaluations for each issue that has been absorbed in the analyses.