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SUBJECT: Provides notification of LOCA model changes or errors reported by Westinghouse.per 10CFR50.46.

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AEP:NRC:1118F

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. 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

October 25, 1993

Dear Dr. Murley:

Pursuant to the requirements of 10CFR50.46(a)(3)(ii), this letter provides notification of LOCA model changes or errors reported to us by Westinghouse Electric Corporation (Westinghouse) that meet the definition of significant as defined in 10CFR50.46.

Attachment 1, which was provided to us by Westinghouse, describes errors which Westinghouse has discovered in their NOTRUMP computer code used for small break LOCA analysis for Units 1 and 2 of Donald C. Cook Nuclear Plant. The evaluation of the impact performed by Westinghouse of these errors on calculated peak clad temperature, is in the range of -13°F to -55°F. Since the absolute value of the change in calculated peak clad temperature could exceed 50°F, the change meets the definition of significant provided in 10CFR50.46.

Attachment 2, which was also provided to us by Westinghouse, describes an error in their Emergency Core Cooling System evaluation methodology used by Westinghouse for Units 1 and 2 of the Donald C. Cook Nuclear Plant. Westinghouse has evaluated the impact of this error on peak clad temperature. An increase of approximately 150°F in peak clad temperature has been evaluated. However, Westinghouse has informed us (Attachment 2) that there may not be any impact on calculated peak clad temperature caused by the error, due to competing effects. The Westinghouse Owners Group is reviewing this issue and considering the possibility of

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a generic program for resolution. Since the change in peak clad temperature due to the error indicated in Attachment 2 is more than 50°F, the change meets the definition of significant provided in 10CFR50.46.

Attachment 3 contains the peak clad temperatures calculated specifically for the small break LOCA (SBLOCA) analyses for Donald C. Cook Nuclear Plant Units 1 and 2. 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 Donald C. Cook Nuclear Plant includes the 1993 10CFR50.46 model assessments (Attachment 3). This assessment consists of PCT change of: 150°F for the effect of SI in broken loop, -150°F for the effect of improved condensation model as described in Attachment 2, and -13°F for drift flux flow regime errors. Attachment 3 also contains changes to LOCA analyses that were submitted via our letter AEP:NRC:1169, dated November 11, 1992, in support of a proposed technical specification change to increase main steam safety valve (MSSV) setpoint tolerances. This proposed technical specification change is currently under NRC review and therefore the referenced SBLOCA analyses which are affected are not yet part of our licensing basis. We have elected to prepare this report using the analyses performed for the MSSV setpoint tolerance relaxation because 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.

Regarding plans for future analysis, the MSSV analyses will provide the new analysis of record for SBLOCA for both units. As indicated in Attachment 3, however, the MSSV analyses are affected by the errors reported in this letter. Current plans do include both LBLOCA and SBLOCA reanalyses in conjunction with evaluations and analyses to support an increase in allowable steam generator tube plugging for Unit 1. This reanalysis will address the errors noted above in the NOTRUMP model. This work is tentatively planned to be complete in 1994 and will be submitted to the staff for review of any technical specification changes needed prior to start of Cycle 16. There are no plans for new SBLOCA or LBLOCA analyses for Unit 2 at this time.

Sincerely,

E E Fitzpatrick

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:1118F

WESTINGHOUSE ELECTRIC CORPORATION

DESCRIPTION OF NOTRUMP COMPUTER CODE ERRORS

NOTRUMP DRIFT FLUX FLOW REGIME MAP ERRORS

Background

Errors were discovered in both WCAP-10079-P-A and related coding in NOTRUMP SUBROUTINE DFCORRS where the improved TRAC-P1 vertical flow regime map is evaluated. In Evaluation Model applications, this model is only used during counter-current flow conditions in vertical flow links. The affected equation in WCAP-10079-P-A is Equation G-65 which previously allowed for unbounded values of the parameter C_∞ contrary to the intent of the original source of this equation. This allowed a discontinuity to exist in the flow regime map under some circumstances. This was corrected by placing an upper limit of 1.3926 on the parameter C_∞ as reasoned from the discussion in the original source. As stated, this correction returned NOTRUMP to consistency with the original source for the affected equation.

Further investigation of the DFCORRS uncovered an additional closely related logic error which led to discontinuities under certain other circumstances. This error was also corrected and returned the coding to consistency with WCAP-10079-P-A.

This was determined to be a Non-discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant calculations indicated PCT effects ranging from -13 degrees to -55 degrees. For the purposes of tracking PCT, the minimum benefit of -13 degrees has been assigned to these changes. When considering reportability under 10 CFR 50.46(a)(3)(i), however, it has been demonstrated that the effect of these changes may exceed 50 degrees F. Westinghouse, therefore, recommends that these changes be considered significant with respect to 10 CFR 50.46(a)(3)(i) requirements.

ATTACHMENT 2 TO AEP:NRC:1118F

WESTINGHOUSE ELECTRIC CORPORATION

DESCRIPTION OF ECCS EVALUATION METHODOLOGY ERROR

TECHNICAL DESCRIPTION

ISSUE DESCRIPTION

Westinghouse emergency core cooling system (ECCS) evaluation models are considered to be composed of several features which include underlying assumptions. Westinghouse recently completed an evaluation of a potential issue concerning the modeling of Safety Injection (SI) flow into the broken RCS loop for small break loss of coolant accident (SBLOCA). Westinghouse previously assumed that SI to the broken RCS loop would result in a lower calculated PCT and, therefore, modeled the ECCS broken loop branch line to spill the SI to the containment sump. The basis for this assumption included consideration for the effect of back pressure on the spilling ECCS line for cold leg breaks, which would see a higher back pressure for SI connected to the broken RCS loop when compared to spilling against containment back pressure. Spilling to the higher RCS pressure would increase SI to the intact loops, which is a benefit for PCT. The effect on intact loop SI flow rates as well as the assumption that some of the SI to the broken loop would aid in RCS/Core recovery resulted in the Westinghouse ECCS model assumption that SI to the broken loop was a benefit. However, when SI is modeled to enter into the broken loop, a significant PCT penalty is calculated by the NOTRUMP small break evaluation model (approximately 150 degrees F for a typical Westinghouse 3-loop design).

TECHNICAL EVALUATION

An analysis by Westinghouse indicates that the penalty (as described above) occurs as a result of competition between the steam venting out the break and the SI to the broken loop, which also exits through the break. The competition between the steam and the SI results in higher RCS pressures for the identical core steaming rates. Since the ECCS uses centrifugal pumps, higher RCS pressure results in lower delivered SI flow rates to the intact RCS loops, leading to the calculated PCT penalty. This penalty is somewhat aggravated by the use of the Moody two-phase break flow model, which is a thermal equilibrium model being used to model a clearly nonequilibrium process. However, the penalty is large enough such that a change to a nonequilibrium break flow model would not be expected to offset the break flow RCS pressure interaction seen when SI is assumed to enter into the broken loop.

However, when a newer conservative model based on prototypic test is used which modeled the configuration of the SI piping to the RCS cold leg in a Westinghouse designed PWR, a net PCT benefit is calculated. Improved condensation of the loop steam in the intact loops results in lower RCS pressure and larger SI flow rates. The increase in SI flow rates, due to lower RCS pressure, leads to the lower calculated PCT. Thus, the negative effects of SI into the broken loop can be offset by an improved SI condensation model in the intact RCS loops.

The improved condensation model is based on data obtained from the COSI test facility. The COSI test facility is a 1/100 scale representation of the cold leg and SI injection ports in a Westinghouse designed PWR. The COSI tests demonstrated that the current NOTRUMP condensation model under-predicted condensation in the intact loops during SI and thus is a conservative model. Use of the improved condensation model has demonstrated that the current NOTRUMP small break LOCA analyses without the improved condensation model and no SI into the broken loop is more conservative (higher calculated PCT) than a case which includes SI into the broken loop and the improved condensation model.

Additionally, the effects of SI in the broken loop have been determined to not change RCP trip symptoms developed in response to US-NRC Generic Letters 83-10C and 85-12 or SI termination criteria found in the Westinghouse Owners Group Emergency Response Guidelines.

ASSESSMENT OF SAFETY SIGNIFICANCE

The COSI tests demonstrated that the current NOTRUMP condensation model under-predicted condensation in the intact loops during SI and thus is a conservative model. Furthermore, recent evaluations have shown that the current NOTRUMP small break LOCA analyses without the improved condensation model and no SI into the broken loop is more conservative (higher calculated PCT) than a case which includes SI into the broken loop and the improved condensation model. Based on these evaluations, Westinghouse determined that this issue does not involve a Substantial Safety Hazard as defined in 10 CFR Part 21. Reanalyses are not necessary since current NOTRUMP based small break LOCA analyses have a conservatively calculated PCT and, therefore, remain valid.

Therefore, Westinghouse is electing at this time not to incorporate these changes into the current NOTRUMP based small break LOCA evaluation models. Westinghouse has notified the NRC in accordance with 10CFR50.46(a)(3)(ii). This information was also provided to the NRC since information in Westinghouse Topical Reports (References 2, 3 & 4) is affected. A copy of the NRC notification letter is attached to this letter.

ATTACHMENT 3 TO AEP:NRC:1118F

WESTINGHOUSE ELECTRIC CORPORATION

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 (Comments: Evaluation Model: NOTRUMP, FQT= 2.32, FdH= 1.55, Other: HHSI Cross Tie Valve Closed, 3250 Mwt Reactor Power)	PCT= 2122°F
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	PCT= 1844°F

1. 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 Closed, 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	PCT= 1577°F

1. 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 2

A. ANALYSIS OF RECORD PCT= 2124°F
(Comments: Evaluation Model: NOTRUMP, FQT= 2.34, FdH= 1.64, SGTP=15%
Other: HHSI Cross Tie Valve Closed, 3413 Mwt Reactor Power)

B. 10 CFR 50.92 SAFETY EVALUATIONS ΔPCT= -168°F[1]
(Comments: Evaluation Model: NOTRUMP, FQT= 2.357, FdH= 1.666, SGTP=15%,
Other: HHSI Cross Tie Valve Closed, 3250 Mwt Reactor Power)

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= 1943°F

1. This rack up is provided for information and completeness only. It is part of the main steam safety valve (MSSV) setpoint tolerance relaxation submittal, AEP:NRC:1169 dated November 11, 1992.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A. ANALYSIS OF RECORD PCT= 2124°F
(Comments: Evaluation Model: NOTRUMP, FQT= 2.34, FdH= 1.64, SGTP=15%,
Other: HHSI Cross Tie Valve Closed, 3413 Mwt Reactor Power)

B. 10 CFR 50.92 SAFETY EVALUATIONS ΔPCT= -177°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 ΔPCT= 1934°F

1. 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 177°F change indicated above is based on an analysis to develop sensitivities. The MSSV tolerance is ± 1%. The analyses were submitted for NRC review with our letter AEP:NRC:1169 dated November 11, 1992. 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 Closed, 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

1. 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.