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AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele

RECIP.NAME RECIPIENT AFFILIATION
MURLEY,T.E. Document Control Branch (Document Control Desk)

SUBJECT: Forwards rept of LOCA evaluation model changes for 1992 & determination of effect of LOCA model changes on plant LOCA analyses,prepared by Westinghouse,per 10CFR50.46(a)(3)(ii).

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Indiana Michigan
Power Company
P.O. Box 16631
Columbus, OH 43216

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AEP:NRC:1118D
10 CFR 50.46(a)(3)(ii)

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 LOCA EVALUATION MODEL CHANGES
PURSUANT TO 10 CFR 50.46(a)(3)(ii)

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

Attn: T. E. Murley

March 12, 1993

Dear Dr. Murley:

Pursuant to the requirements of 10 CFR 50.46(a)(3)(ii), this letter provides our annual submittal of LOCA model changes.

Attachment 1, which was provided to us by Westinghouse Electric Corp. (Westinghouse), describes LOCA model changes which have been permanently implemented, and provides a discussion in general terms of the impact of these changes on calculated peak clad temperatures. Attachment 2 contains the peak clad temperatures calculated specifically for the Donald C. Cook Nuclear Plant Units 1 and 2. All of the changes for 1992, either alone or in combination with changes from previous submittals, result in a peak fuel clad temperature different by more than 50°F from the temperature calculated using the last acceptable model. As a result, these changes meet the definition of significant as defined in 10 CFR 50.46. This letter, therefore, also serves as our notification of these significant changes, required pursuant to 10 CFR 50.46(a)(3)(ii). With the exception of one Unit 2 SBLOCA run, where the peak clad temperature is very low, the calculated peak clad temperatures have been found to be only slightly increased or reduced. In all cases, the calculated peak clad temperatures remain within the 10 CFR 50.46 acceptance criteria of 2200°F.

One of the issues identified in Attachment 1, resulting in peak clad temperature changes meeting the definition of significant,

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involves fuel hydraulic test results for 15 x 15 optimized fuel assemblies (OFA). Westinghouse informed us of the resolution of this item by letter dated August 17, 1992.

In addition to the changes in peak clad temperatures associated with the licensing basis LOCA analyses, Attachment 2 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 a new analysis of record for SBLOCA for both units. Current plans also 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 work is tentatively planned to be complete and submitted to the staff prior to start of Cycle 16. There are no plans for new LBLOCA analyses for Unit 2 at this time.

Sincerely,

EE Fitzpatrick

E. E. Fitzpatrick
Vice President

eh

Attachments

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

bc: S. J. Brewer
J. B. Kingseed/V. VanderBurg
D. H. Malin/M. S. Ackerman
M. L. Horvath - Bridgman
J. J. Markowsky
J. B. Shinnock
W. G. Smith, Jr.
W. M. Dean, NRC - Washington, D.C.
AEP:NRC:1118D
DC-N-6015.1

ATTACHMENT 1 TO AEP:NRC:1118D

WESTINGHOUSE ELECTRIC CORPORATION

DESCRIPTION OF LOCA MODEL CHANGES

## Structural Metal Heat Modeling

### Background

A discrepancy was discovered during review of the finite element heat conduction model used in the WREFLOOD-INTERIM code to calculate heat transfer from structural metal in the vessel during the reflood phase. It was noted that the material properties available in the code corresponded to those of stainless steel. While this is correct for the internal structures, it is inappropriate for the vessel wall which consists of carbon steel with a thin stainless internal clad. This was defined as a non-discretionary change per Section 4.1.2 of WCAP-13451, since there was thought to be potential for increased PCT with a more sophisticated composite model. The model was revised by replacing it with a more flexible one that allows detailed specification of structures.

### Affected Evaluation Models

1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

### Affected Codes

WREFLOOD-INTERIM

### Estimated Effects

The estimated effect of this correction is a 25°F PCT benefit.

## 15x15 Optimized Fuel Assembly (OFA) Core Pressure Drop Increase

Background

Hydraulic tests performed on the 15x15 Optimized Fuel Assembly (OFA) indicated that the design values for the grid loss coefficients were previously underestimated. This results in the core pressure drop in safety analysis being underestimated by approximately 10%. The effects of this on LOCA analysis has been calculated on a plant specific basis.

This was determined to be a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451. As such, affected plants have been notified previously of the impact of this error in NSAL-92-002.

Affected Evaluation Models

All Large Break and Small Break LOCA Evaluation Models used on the affected plants.

Estimated Effect

Plant specific PCT effects are included on the attached Margin Summary Sheet. (See Attachment 2)

## Steam Generator Secondary Side Modelling Enhancements

Background

A set of related changes which make steam generator secondary side modelling more convenient for the user were implemented into NOTRUMP. This model improvement involved several facets of feedwater flow modelling. First, the common donor boundary node for the standard Evaluation Model nodalization has been separated into two identical boundary nodes. These donor nodes are used to set the feedwater enthalpy. The common donor node configuration did not allow for loop specific enthalpy changeover times in cases where asymmetric AFW flowrates or purge volumes were being modeled for plant specific sensitivities.

The second improvement is the additional capability to initiate main feedwater isolation on either loss of offsite power coincident with reactor trip (low pressurizer pressure) or alternatively on safety injection signal (low-low pressurizer pressure). The previous model allowed this function only on loss of offsite power coincident with reactor trip. The auxiliary feedwater pumps are still assumed to start after a loss of offsite power with an appropriate delay time to model diesel generator start-up and buss loading times.

The final improvement is in the area of modelling the purging of high enthalpy main feedwater after auxiliary feedwater is calculated to start. This was previously modelled through an approximate time delay necessary to purge the lines of the high enthalpy main feedwater before credit could be taken for the much lower enthalpy auxiliary feedwater reaching the steam generator secondary. This time delay was a function of the plant specific purge volume and the auxiliary feedwater flowrate. The new modelling allows the user to input the purge volume directly. This then is used together with the code calculated integrated feedwater flow to determine the appropriate time at which the feedwater enthalpy can be assumed to change.

These improvements are considered to be a Discretionary Change as described in Section 4.1.1 of WCAP-13451. Since they involve only enhancements to the capabilities and useability of the Evaluation Model, and not changes to results calculated consistently with the previous model, these changes were implemented without prior review as discussed in Section 4.1.1 of WCAP-13451.

Affected Evaluation Models

1985 Small Break LOCA Evaluation Model

Estimated Effects

Because these enhancements only allow greater ease in modelling plant specific steam generator secondary side behavior over the previous model, it is estimated that no effect will be seen in Evaluation Model calculations.

ATTACHMENT 2 TO AEP:NRC:1118D

WESTINGHOUSE ELECTRIC CORPORATION

DETERMINATION OF EFFECT OF LOCA MODEL CHANGES ON

COOK NUCLEAR PLANT LOCA ANALYSES

## LARGE BREAK LOCA

PLANT NAME: Donald C. Cook Unit 1

| | | |
|---|---|---|
| A. | ANALYSIS OF RECORD | PCT= 2162°F |
| | (Comments: Evaluation Model: BASH, FQT=2.15, FdH=1.55, SGTP=15%, Other: RHR Cross Tie Valve Closed, 3250MWt Reactor Power) | |
| B. | PRIOR LOCA MODEL ASSESSMENTS - 1989 | ΔPCT= + 0°F |
| C. | PRIOR LOCA MODEL ASSESSMENT - 1990 | ΔPCT= + 0°F |
| D. | PRIOR LOCA MODEL ASSESSMENTS - 1991 | ΔPCT= + 20°F[1] |
| E. | PRIOR LOCA MODEL ASSESSMENTS - July 1992 | ΔPCT= + 0°F |
| F. | 1992 10CFR50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin) | |
| | 1. 15X15 OFA Fuel Hydraulic Test Results | ΔPCT= -92°F[2] |
| | 2. Structural Metal Heat Modeling | ΔPCT= -25°F |
| G. | LICENSING BASIS PCT + PERMANENT ASSESSMENTS | ΔPCT= 2065°F |

1. The 1991 report, AEP:NRC:1118B, dated July 18, 1991 shows 30°F. However, the assessment for fuel rod initial condition inconsistency has been removed from this rack up because the model changes are included in calculations to calculate the assessment for 15X15 OFA fuel hydraulic test results.

2. The large break LOCA licensing basis was reanalyzed incorporating the following LOCA model changes with the year of the change.

   - Fuel Rod Initial Condition Inconsistency (1991)
   - 15X15 OFA Fuel Hydraulic Test Results (1992)
   - IMP Data Base Correction (1992)
   - ECCS Flow Inconsistencies (1989)

LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK UNIT 1

| | | | |
|---|---|---|---|
| A. | ANALYSIS OF RECORD | | PCT= 2181°F |

(Comments: Evaluation Model: BASH, FQT=2.15, FdH=1.55, SGTP=15%,
Other: RHR Cross Tie Valve Open, 3413 MWt Reactor Power,

| | | | |
|---|---|---|---|
| B. | PRIOR LOCA MODEL ASSESSMENTS - 1989 | | ΔPCT=+ 0°F |
| C. | PRIOR LOCA MODEL ASSESSMENTS - 1990 | | ΔPCT=+ 0°F |
| D. | PRIOR LOCA MODEL ASSESSMENTS - 1991 | | ΔPCT=+ 30°F |
| E. | PRIOR LOCA MODEL ASSESSMENTS - July 1992 | | ΔPCT=+ 0°F |
| F. | 1992 10 CFR 50.46 MODEL ASSESSMENTS (Permanent Assessment of PCT Margin) | | |
| | 1. | 15X15 OFA Fuel Hydraulic Test Results | ΔPCT= 0°F[1] |
| | 2. | Structural Metal Heat Modeling | ΔPCT= - 25°F |
| G. | OTHER MARGIN ALLOCATIONS (Use of PCT Margin): | | |
| | 1. | ANALYSIS MARGINS USED: Power Margin | ΔPCT= - 94°F |
| H. | LICENSING BASIS PCT + PERMANENT ASSESSMENTS & POWER MARGIN | | ΔPCT= 2092°F |

1. This issue has only been addressed for the cross ties closed, LBLOCA analysis at 3250 $MW_t$. The result was a PCT benefit. Cross tie valve open operation is also a PCT benefit.

# JUSTIFICATION FOR USE OF POWER MARGIN IN DONALD C. COOK NUCLEAR PLANT UNIT 1 LARGE BREAK PCT RACK UP

The analysis peak clad temperature (PCT) for Donald C. Cook Unit 1 at 3413 $MW_t$ with the RHR cross tie valve open is 2181°F. When the 1991 LOCA model assessment of 30°F was added, the resulting PCT exceeded 2200°F. The following calculation shows that power margin exists for Cook Nuclear Plant Unit 1 since the core is currently licensed at 3250 $MW_t$ versus the analysis power level of 3413 $MW_t$.

A sensitivity to power was previously determined for the Donald C. Cook Nuclear Plant Unit 2 large break analysis. It was conservatively demonstrated that a reduction of $20°F_{PCT}$/% Power could be applied for reduced power. This sensitivity is conservative since it only accounts for the reduction in the LOCBART run. A reduction in power in the blowdown portion of the transient (i.e., SATAN) would be an added benefit which was not accounted for in this sensitivity. Since both Cook Nuclear Plant Unit 1 and Unit 2 are 4 loop ice condenser plants, this sensitivity will be applied to the reduction in power from the Unit 1 analysis power of 3413 $MW_t$ to the licensed operating condition of 3250 $MW_t$ (a 4.7% reduction in power):

$$(20°F_{PCT}/\%\ \text{Power})\ (4.7\%\ \text{Power}) = 94°F$$

When this 94°F margin is applied to the Unit 1, 3413 $MW_t$ analysis with RHR cross tie valves open, the 10 CFR 50.46 PCT limit is not exceeded.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 1

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

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

C. 1992 10 CFR 50.46 MODEL ASSESSMENTS
(Permanent Assessment of PCT Margin)

1. 15X15 OFA Fuel Hydraulic Test Results ΔPCT= 3°F

D. LICENSING BASIS PCT + PERMANENT ASSESSMENTS ΔPCT= 1857°F

1. As discussed in the body of this submission, prior and current LOCA model assessments have been absorbed in new analyses performed to support a request for elevation 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. LOCA model changes with the year of change, which are included in the new analyses, are:

- Fuel Rod Initial Condition Inconsistency (1991)
- NOTRUMP Solution Convergence Reliability (1991)
- ECCS Flow Inconsistencies (1989)
- Clad Creep Model Modifications (1991)
- SG Secondary Side Modeling Concerns
(Including the 1991 AFW Enthalpy Switchover) (1992)

## 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, SCTP=15%,
Other: HHSI Cross Tie Valve Open, 3588 MWt Reactor Power )

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

C. 1992 10 CFR 50.46 MODEL ASSESSMENTS
(Permanent Assessment of PCT Margin)

1. 15X15 OFA Fuel Hydraulic Test Results — ΔPCT= 0°F[1]

D. LICENSING BASIS PCT + PERMANENT ASSESSMENTS — PCT= 1570°F

1. As discussed in the body of this submission, prior and current 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 new analyses. LOCA model changes with the year of change, which are included in the new analyses, are:

- Fuel Rod Initial Condition Inconsistency (1991)
- NOTRUMP Solution Convergence Reliability (1991)
- ECCS Flow Inconsistencies (1989)
- Clad Creep Model Modifications (1991)
- SG Secondary Side Modeling Concerns (Including the 1991 AFW Enthalpy Switchover) (1992)
- 15X15 OFA Fuel Hydraulic Test Results (1992)

## LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

| | | |
|---|---|---|
| A. | ANALYSIS OF RECORD<br>(Comments: Evaluation Model: BASH, FQT=2.335, FdH=1.644, SGTP=15%, Other: RHR Cross Tie Valve Closed, 3413 MWt Reactor Power) | PCT= 2090°F |
| | TRANSITION CORE PENALTY | ΔPCT=+ 50°F |
| B. | PRIOR LOCAL MODEL ASSESSMENTS - 1989<br>(Analysis of record was completed in January 1990. No prior LOCA Model assessments were made.) | ΔPCT=+ NA°F |
| C. | PRIOR LOCA MODEL ASSESSMENTS - 1990 | ΔPCT=+ 0°F |
| D. | PRIOR LOCA MODEL ASSESSMENTS - 1991 | ΔPCT=+ 30°F |
| E. | PRIOR LOCA MODEL ASSESSMENTS - July 1992 | ΔPCT=+ 0°F |
| F. | 1992 10 CFR 50.46 MODEL ASSESSMENTS<br>(Permanent Assessment of PCT Margin) | |
| | 1. Structural Metal Heat Modeling | ΔPCT= -25°F |
| G. | LICENSING BASIS PCT + PERMANENT ASSESSMENTS | ΔPCT= 2145°F |

# LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

| | | |
|---|---|---|
| A. | ANALYSIS OF RECORD<br>(Comments: Evaluation Model: BASH, FQT = 2.22, FdH = 1.62, SGTP = 15%, Other: RHR Cross Tie Valve Open, 3588 MWt Reactor Power) | ΔPCT = 2140°F |
| | TRANSITION CORE PENALTY | ΔPCT = + 50°F |
| B. | PRIOR LOCA MODEL ASSESSMENTS - 1989<br>(Analysis of record was completed in January 1990. No prior LOCA model assessments were made.) | ΔPCT = + NA°F |
| C. | PRIOR LOCA MODEL ASSESSMENTS - 1990 | ΔPCT = + 0°F |
| D. | PRIOR LOCA MODEL ASSESSMENTS - 1991 | ΔPCT = + 30°F |
| E. | PRIOR LOCA MODEL ASSESSMENTS - July 1992 | ΔPCT = + 0°F |
| F. | 1992 10 CFR 50.46 MODEL ASSESSMENTS<br>(Permanent Assessment of PCT Margin) | |
| | 1. Structural Metal Heat Modeling | ΔPCT = - 25°F |
| G. | OTHER MARGIN ALLOCATIONS | |
| | 1. Power Margin | ΔPCT = - 98°F[1] |
| H. | LICENSING BASIS PCT + PERMANENT ASSESSMENTS | ΔPCT = 2097°F |

1. This value was obtained by temporarily allocating 4.9% of power margin using a sensitivity of 20°F/% power. See the Unit 1 justification for the use of power margin in the Donald C.Cook Nuclear Plant Unit 1 Large Break PCT Rack up on page 3 of this attachment.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A. ANALYSIS OF RECORD ΔPCT= 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. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 1956°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. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 1947°F

1. As discussed in the body of this submission, prior and current 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 new analyses. LOCA model changes with the year of change, which are included in the new analyses, are:

- Fuel Rod Initial Condition Inconsistency (1991)
- NOTRUMP Solution Convergence Reliability (1991)
- ECCS Flow Inconsistencies (1989)
- Clad Creep Model Modifications (1991)
- SG Secondary Side Modeling Concerns (1992)

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A. ANALYSIS OF RECORD PCT= 1357°F
(Comments: Evaluation Model:NOTRUMP, FQT=2.32, FdH=1.62, SGTP=15%,
Other: HHSI Cross Tie Valve Open, 3588 MWt Reactor Power )

B. 10 CFR 50.92 SAFETY EVALUATIONS ΔPCT=+ 174°F[1]

C. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 1531°F

1. As discussed in the body of this submission, prior and current 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 new analyses. LOCA model changes with the year of change, which are included in the new analyses, are:

- Fuel Rod Initial Condition Inconsistency (1991)
- NOTRUMP Solution Convergence Reliability (1991)
- ECCS Flow Inconsistencies (1989)
- Clad Creep Model Modifications (1991)
- SG Secondary Side Modeling Concerns (1992)