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AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards rept of significant LOCA evaluation model changes per 10CFR50.46(a)(3)(ii) reported to util by Westinghouse.

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AEP:NRC:1118B
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 SIGNIFICANT LOCA EVALUATION
MODEL CHANGES PURSUANT TO 10 CFR 50.46(a)(3)(ii)

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

Attn: T. E. Murley

July 18, 1991

Dear Dr. Murley:

Pursuant to the requirements of 10 CFR 50.46(a)(3)(ii), this letter provides notification of LOCA model changes reported to us by Westinghouse Electric Corp. (Westinghouse) that meet the definition of significant as defined in 10 CFR 50.46.

Attachment 1, which was provided to us by Westinghouse, describes changes which have been or will soon be permanently implemented, and discusses, in general, the impact of these issues on calculated peak clad temperatures. Attachment 2 contains the change in calculated peak clad temperature determined specifically for the Donald C. Cook Nuclear Plant Units 1 and 2.

Attachment 2 and 3 discuss several model changes, impacting both large and small break LOCA analyses. Only the small break model changes actually meet the definition of significant provided in 10 CFR 50.46 (i.e., 50°F.) The other changes are being provided for your information. In all cases, peak clad temperature remains below the 2200°F limit.

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This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

E. E. Fitzpatrick

E. E. Fitzpatrick
Vice President

ldp

Attachments

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

ATTACHMENT 1 TO AEP:NRC:1118B

WESTINGHOUSE ELECTRIC CORPORATION

DESCRIPTION OF LOCA MODEL CHANGES

1.0 INTRODUCTION

Provisions in 10CFR50.46 require the reporting of corrections to or changes in the ECCS Evaluation Model (EM) approved for use in performing safety analyses for the loss of coolant accident (LOCA). This report describes corrections and revisions to the Westinghouse ECCS EM in the period from August 1990 through May 1991. The current Westinghouse ECCS EM are named as listed in Table 1, and consist of several computer codes with specific functions.

Westinghouse has completed the evaluation of several items related to the Westinghouse ECCS Evaluation Models listed in Table 1. Each of these items is discussed in the following sections, which include a description of the item, the assessment which was performed, the resulting change to the Evaluation Model, and the effect of the change on the PCT.

Some of the subjects discussed represent changes to program coding or to inputs directly related to the physical models or solution technique. These are described in Section 2.0.

Some items represent changes to the assumptions made when the Evaluation Model is applied to a specific plant. These are discussed in Section 3. Also included, for information, are items for which a technical assessment is continuing, and items for which it was concluded that no change was necessary.

TABLE 1
SUMMARY OF WESTINGHOUSE ECCS EVALUATION MODELS

NAME: 1978 MODEL

APPLICATION: Analysis of Large Break LOCA

CODES USED:	PURPOSE:	REFERENCE:
SATAN-VI	Blowdown hydraulic transient	1.
WREFLOOD	Reflood hydraulic transient	2.
LOCTA	Fuel rod thermal transient	3.
COCO or LOTIC	Containment pressure transient	4.,5.

NOTE: The NRC has determined that this EM is no longer acceptable for use in new analyses. However, it serves as the licensing basis for some plants.

NAME: 1981 MODEL

APPLICATION: Analysis of Large Break LOCA

CODES USED:	PURPOSE:	REFERENCE:
SATAN-VI	Blowdown hydraulic transient	1.,6.
WREFLOOD	Reflood hydraulic transient	2.
LOCTA	Fuel rod thermal transient	3.
COCO or LOTIC	Containment pressure transient	4.,5.

NOTE: This model superseded the 1978 EM and included changes to the flow blockage model, consistent with requirements in NUREG 0630.

NAME: 1981 MODEL WITH BART

APPLICATION: Analysis of Large Break LOCA

CODES USED:	PURPOSE	REFERENCE:
SATAN-VI	Blowdown hydraulic transient	1.,6.
INTERIM-WREFLOOD	Reflood hydraulic transient	2.,7.
BART	Hot assembly thermohydraulics	7.
INTERIM-LOCTA	Fuel rod thermal transient	3.
COCO or LOTIC	Containment pressure transient	4.,5.

NOTE: This model was developed to provide a more realistic calculation of heat transfer during the reflood portion of the transient.

TABLE 1 (CONTINUED)

NAME: 1981 MODEL WITH BASH

APPLICATION: Analysis of Large Break LOCA

CODES USED:	PURPOSE	REFERENCE:
SATAN-VI	Blowdown hydraulic transient	1.,6.
BASH	Reflood hydraulic transient	8.
LOCBART	Hot assembly thermohydraulics and fuel rod thermal transient	3.,7.,8.
WREFLOOD/COCO/LOTIC	Containment pressure transient	2.,4.,5.,8.

NOTE: this model was developed to further improve the reflood portion of the Evaluation Model.

NAME: UPI WCOBRA/TRAC

APPLICATION: Analysis of Large Break LOCA for plants with upper plenum safety injection.

CODES USED:	PURPOSE	REFERENCE
COBRA/TRAC	Combined thermal and hydraulic transient	9.

NOTE: This model uses a best estimate computer code, but includes required features of Appendix K.

NAME: 1975 SBLOCA MODEL

APPLICATION: Analysis of Small Break LOCA

CODES USED:	PURPOSE	REFERENCE
WFLASH	System hydraulic transient	10., 11.
SBLOCTA	Fuelrod Thermal transient	3.

NOTE: This model is no longer used, but some plants are licensed under this methodology.

NAME: 1985 SBLOCA MODEL

APPLICATION: Analysis of Small Break LOCA

CODES USED:	PURPOSE	REFERENCE
NOTRUMP	System Hydraulic transient	12., 13
SBLOCTA	Fuel rod thermal transient	3.

NOTE: This model was developed to provide more realistic SBLOCA simulations, as required by NRC, following TMI.

2.0 EVALUATION MODEL CODE CHANGES

This section describes changes and revisions to the Westinghouse ECCS Evaluation Model computer codes. Except where noted, these corrections will be implemented in all future applications of the Evaluation Model.

2.1 FUEL ROD MODEL REVISIONS

During the review of the original Westinghouse ECCS Evaluation Model following the promulgation of 10CFR50.46 in 1974, Westinghouse committed to maintain consistency between future loss-of-coolant accident (LOCA) fuel rod computer models and the fuel rod design computer models used to predict fuel rod normal operation performance. These fuel rod design codes are also used to establish initial conditions for the LOCA analysis.

<u>Change Description</u>:

It was found that the large break and small break LOCA code versions were not consistent with fuel design codes in the following areas:

1. The LOCA codes were not consistent with the fuel rod design code relative to the flux depression factors at higher fuel enrichment.

2. The LOCA codes were not consistent with the fuel rod design code relative to the fuel rod gap gas conductivities and pellet surface roughness models.

3. The coding of the pellet/clad contact resistance model required revision.

Modifications were made to the fuel rod models used in the LOCA Evaluation Models to maintain consistency with the latest approved version of the fuel rod design code.

In addition, it was determined that integration of the cladding strain rate equation used in the large break LOCA Evaluation Model, as described in Reference 3, was being calculated twice each time step instead of once. The coding was corrected to properly integrate the strain rate equation.

<u>Affected Evaluation Models:</u>

1981 Large Break LOCA Evaluation Model
1981 Large Break LOCA Evaluation Model, With BART
1981 Large Break LOCA Evaluation Model, With BASH
1975 Small Break LOCA Evaluation Model
1985 Small Break LOCA Evaluation Model

Effect of Changes:

The changes made to make the LOCA fuel rod models consistent with the fuel design codes were judged to be insignificant, as defined by 10CFR50.46(a)(i). To quantify the effect on the calculated peak cladding temperature (PCT), calculations were performed which incorporated the changes, including the cladding strain model correction for the large break LOCA. For the large break LOCA Evaluation Model, additional calculations, incorporating only the cladding strain corrections were performed and the results supported the conclusion that compensating effects were not present. The PCT effects reported below will bound the effects taken separately for the large break LOCA.

a) Large Break LOCA

The effect of the changes on the large break LOCA peak cladding temperature was determined using the BASH large break LOCA Evaluation Model. The effects were judged applicable to older Evaluation Models. Several calculations were performed to assess the effect of the changes on the calculated results as follows:

1. Blowdown Analysis -

It was determined that the changes will have a small effect on the core average rod and hot assembly average rod performance during the blowdown analysis. The effect of the changes on the blowdown analysis was determined by performing a blowdown depressurization computer calculation for a typical three-loop plant and a typical four-loop plant using the SATAN-VI computer code.

2. Hot Assembly Rod Heatup Analysis -

The hot rod heatup calculations would typically show the largest effect of the changes. Hot rod heatup computer analysis calculations were performed using the LOCBART computer code to assess the effect of the changes on the hot assembly average rod, hot rod and adjacent rod.

3. Determination of the Effect on the Peak Cladding Temperature

The effect of the changes on the calculated peak cladding temperature was determined by performing a calculation for typical three-loop and four-loop plants using the BASH Evaluation Model. The analysis calculations confirmed that the effect of the ECCS Evaluation Model changes were insignificant as defined by 10CFR50.46(a)(3)(i). The calculations showed that the peak cladding temperatures increased by less than by 10°F for the BASH Evaluation Model. It was judged that 25°F would bound the effect on the peak cladding temperature for the BART Evaluation Model, while calculations performed for the Westinghouse 1981 Evaluation Model showed that the peak cladding temperature could increase by approximately 41°F.

b) Small Break LOCA

The effect of the changes on the small break LOCA analysis peak cladding temperature calculations was determined using the 1985 small break LOCA Evaluation Model by performing a computer analysis calculations for a typical three-loop plant and a typical four-loop plant. The analysis calculations confirmed that the effect of the changes on the small break LOCA ECCS Evaluation Model were insignificant as defined by 10CFR50.46(a)(3)(i). The calculations showed that 37°F would bound the effect on the calculated peak cladding temperatures for the four-loop plants and the three-loop plants. It was judged that an increase of 37°F would bound the effect of the changes for the 2-loop plants.

Status:

Changes completed and implemented.

2.2 SMALL BREAK LOCA ROD INTERNAL PRESSURE INITIAL CONDITION ASSUMPTION

Change Description:

The Westinghouse small break loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) Evaluation Model analyses assume that higher fuel rod initial fill pressure leads to a higher calculated peak cladding temperature (PCT), as found in studies with the Westinghouse large break LOCA ECCS Evaluation Model. However, lower fuel rod internal pressure could result in decreased cladding creep (rod swelling) away from the fuel pellets when the fuel rod internal pressure was higher than the reactor coolant system (RCS) pressure. A lower fuel rod initial fill pressure could then result in a higher calculated peak cladding temperature.

The Westinghouse small break LOCA cladding strain model is based upon a correlation of Hardy's data, as described in Section 3.5.1 of Reference 3. Evaluation of the limiting fuel rod initial fill pressure assumption revealed that this model was used outside of the applicable range in the small break LOCA Evaluation Model calculations, allowing the cladding to expand and contract more rapidly than it should. The model was corrected to fit applicable data over the range of small break LOCA conditions. Correction of the cladding strain model affects the small break LOCA Evaluation Model calculations through the fuel rod internal pressure initial condition assumption.

Affected Evaluation Models:

1975 Small Break LOCA Evaluation Model
1985 Small Break LOCA Evaluation Model

Effect of Changes:

Implementation of the corrected cladding creep equation results in a small reduction in the pellet to cladding gap when the RCS pressure exceeds the rod

internal pressure and increases the gap after RCS pressure falls below the rod internal pressure. Since the cladding typically demonstrates very little creep toward the fuel pellet prior to core uncovery when the RCS pressure exceeds the rod internal pressure, implementation of the correlation for the appropriate range has a negligible benefit on the peak cladding temperature calculation during this portion of the transient. However, after the RCS pressure falls below the rod internal pressure, implementation of an accurate correlation for cladding creep in small break LOCA analyses would reduce the expansion of the cladding away from the fuel compared to what was previously calculated and results in a PCT penalty because the cladding is closer to the fuel.

Calculations were performed to assess the effect of the cladding strain modifications for the limiting three-inch equivalent diameter cold leg break in typical three-loop and four-loop plants. The results indicated that the change to the calculated peak cladding temperature resulting from the cladding strain model change would be less than 20°F. The effect on the calculated peak cladding temperature depended upon when the peak cladding temperature occurs and whether the rod internal pressure was above or below the system pressure when the peak cladding temperature occurs. For the range of fuel rod internal pressure initial conditions, the combined effect of the fuel rod internal pressure and the cladding strain model revision is typically bounded by 40°F. However, in an extreme case the combined effect could be as large as 60°F.

Status:

Modifications to the small break LOCA cladding strain model for application to the appropriate range of conditions have been implemented and the effect of the rod internal pressure initial condition assumption assessed. Since changes to the strain model may also affect assumptions concerning the limiting time in the core cycle due to the propensity for cladding burst, the small break LOCA limiting time in the core cycle assumptions are being reviewed and a conclusion regarding their continued validity will be determined by the end of 1991.

2.3 UPI MODEL REVISIONS

Change Description:

Revisions were made to the WCOBRA/TRAC large break LOCA Evaluation Model used for plants equipped with upper plenum injection (UPI). These changes, and their effects, were previously reported to the NRC (Reference 14).

Affected Evaluation Model

UPI WCOBRA/TRAC

Status:

Complete.

2.4 NOTRUMP CODE SOLUTION CONVERGENCE

Change Description:

In the development of the NOTRUMP small break LOCA ECCS Evaluation Model, a number of noding sensitivity studies were performed to demonstrate acceptable solution convergence as required by Appendix K to 10CFR50. Temporal solution convergence sensitivity studies were performed by varying input parameters which govern the rate of change of key process variables, such as changes in the pressure, mass, and internal energy. Standard input values were specified for the input parameters which govern the time step size selection. However, since the initial studies, modifications were made to the NOTRUMP computer program to enhance code performance and implement necessary modifications (Reference 15). Subsequent to the modifications, solution convergence was not re-confirmed.

To analyze changes in plant operating conditions, sensitivity studies were performed with the NOTRUMP computer code for variations in initial RCS pressure, auxiliary feedwater flow rates, power distribution, etc., which resulted in peak cladding temperature (PCT) variations which were greater than anticipated based upon engineering judgement. In addition, the direction of the PCT variation conflicted with engineering judgement expectations in some cases. The unexpected variability of the sensitivity study results indicated that the numerical solution may not be properly converged.

Sensitivity studies were performed for the time step size selection criteria which culminated in a revision to the recommended time step size selection criteria inputs. Fixed input values originally recommended for the steady state and all break transient calculations were modified to assure converged results. The NOTRUMP code was re-verified against the SUT-08 Semiscale experiment and it was confirmed that the code adequately predicts key small break phenomena.

Affected Models:

1985 Small Break LOCA Evaluation Model

Effect of Changes:

Generally, the modifications result in small shifts in timing of core uncovery and recovery. However, these changes may result in a change in the calculated peak cladding temperature which exceeds 50°F for some plants. Based on representative calculations, however, this change will most likely result in a reduction in the calculated peak cladding temperature. Since the potential beneficial effect of a non-converged solution is plant specific, a generic PCT effect cannot be provided. However, it has been concluded that current licensing basis results remain valid since the results are conservative relative to the change.

Status:

This change has been implemented and will be used in all future analyses.

3.0 EVALUATION MODEL APPLICATION CHANGES

The following section describes changes in the way the LOCA evaluation model is applied, or provides additional information on the method of application.

3.1 LARGE BREAK LOCA POWER DISTRIBUTION ASSUMPTION

Background:

Appendix K to 10CFR50 requires that the power distribution which results in the most severe calculated consequences be used in the ECCS Evaluation Model calculations. The power distributions to be studied are those expected to occur during the core lifetime.

The current basis for all Westinghouse large LOCA Evaluation Model is the chopped cosine power distribution. This distribution is symmetrical and is defined by two quantities: the ratio of peak linear power relative to the average (FQT), and the ratio of hot rod integral power relative to the average (FΔH). This power distribution was found to produce the highest peak cladding temperature (PCT) when compared to power distributions skewed to the top or bottom of the core in studies performed by Westinghouse and submitted to the NRC. Typically the power distributions were assumed to peak at discrete elevations in the core (4, 6, 8, and 10 feet). It was also assumed that the key parameters affecting PCT were the FQT, FΔH, the peak power location, and integral of power to the peak power elevation.

Calculations performed with the advanced LOCA Evaluation Models, BART and BASH, which examined peak power locations and power distributions which were not considered in the original analyses, under some circumstances lead to PCTs greater than those calculated with the cosine distribution. This behavior was revealed when performing power distribution studies for core designs with relatively low FQT and relatively high FΔH. Further studies revealed that, in addition to FQT, FΔH, and the peak power location, the nature of the axial distribution of power affected the results. That is, two power distributions with the same FQT, FΔH, and peak power location, but whose power was distributed differently along the rod could result in significantly different PCTs.

Westinghouse has completed an analysis effort to understand and properly account for the effect of skewed power distributions on the calculated large break LOCA PCT. This effort included the identification of the worst power distributions that could occur during core life with full consideration of the current generation of reload core designs.

Change Description:

As a result of these studies, revisions have been made to the current reload and safety analysis methodology which accounts for the variability in power distributions from cycle to cycle and plant to plant. This revision provides a means of determining that the current licensing basis (i.e., the chopped cosine) is expected to remain limiting, but also provides for identifying and analyzing the most severe expected power distribution, if different from the chopped cosine.

Affected Evaluation Models:

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

Status:

In order to verify that a plant was not affected by this item, a large break LOCA power distribution surveillance factor was applied to confirm that the power shapes identified as potentially being more limiting are not present. The owners of the affected plants were advised to temporarily apply this surveillance factor to their normal flux map measurements. In some cases, a temporary 100°F PCT margin allocation was applied, rather than the surveillance factor. This margin assured that, if limiting power shapes did occur, 10CFR50.46 limits would still be met.

The process described in Reference 16 will be used to assess specific core designs. In this process, each power distribution calculated in the core design will be evaluated to determine whether it is more limiting than the cosine power distribution. Adjustments will be made to the core design operating bands to eliminate these limiting distributions and surveillance factors will be defined to assure that plant safety limits are met. This will assure that a change to the ECCS Evaluation Model is not required, since the chopped cosine power distribution will remain limiting.

3.2 LA .E BREAK LOCA BURST AND BLOCKAGE ASSUMPTION

Background:

The cladding swelling and flow blockage models were reviewed in detail during the NRC's evaluation of the Westinghouse Evaluation Model. However, the use of the average rod in the hot assembly may not have been documented in a manner detailed enough to allow the staff to adequately assess this aspect of the model.

Appendix K to 10CFR50 requires consideration of the effects of flow blockage resulting from the swelling and rupture of the fuel rods during a loss-of-coolant accident (LOCA). 10CFR50 Appendix K Paragraph I.B states:

> "...To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated."

In Westinghouse ECCS Evaluation Model calculations, the average rod in the hot assembly is used as the basis for calculating the effects of flow blockage. If a significant number of fuel rods in the hot assembly are operating at power levels greater than that of the average rod, the time at which cladding swelling and rupture is calculated to occur may be predicted later in the LOCA transient, since the lower power rod will take longer to heat up to levels where swelling and rupture will occur.

A review of the Westinghouse model used to predict assembly blockage was performed. This model was developed from the Westinghouse Multi-Rod Burst Tests (MRBT) and was the model used to determine assembly wide blockage until replaced by the NUREG-0630 model starting in 1980. These models provide the means for determining assembly wide blockage once the mean burst strain has been established. Implementation of these burst models has relied upon the average rod to provide the mean burst strain. The average rod is a low power rod producing the power of the average of rods in the hot assembly and is primarily used to calculate the enthalpy rise in the hot assembly. Use of the average rod in the model assumes that the time at which blockage is calculated to occur is represented by the burst of the average rod. A review of current hot assembly power distributions indicates that in general the average rod in the hot assembly is also representative of the largest number of rods in the assembly, so that burst of this rod adequately represents when most of the rods will burst. With this representation, however, the true onset of blockage would likely begin earlier, as the highest power rods reach their burst temperature. This time is estimated to be a few seconds prior to the time when the average rod bursts.

Large break LOCA Evaluation Models which use BART or BASH simulate the hot assembly rod with the actual average power, while older Evaluation Models use an average rod power which is adjusted downward to account for thimbles (this is described in detail in Addendum 3 to reference (7)). If burst occurs after the flooding rate has fallen below one inch per second, the time at which the blockage penalty is calculated will be delayed for these older Evaluation Models.

Change Description:

Ample experimental evidence currently exists which shows that flow blockage does not result in a heat transfer penalty during a LOCA. In addition, newer Evaluation Models have been developed and licensed which demonstrate that the older Evaluation Models contain a substantial amount of conservatism. Westinghouse concluded that further artificial changes to the ECCS Evaluation Models to force the calculation of an earlier burst time were not necessary. In rare instances where burst has not occured prior to the flooding rate falling below 1.0-inch/second, the results of the ECCS analysis calculation are supplemented by a permanent assessment of margin. Typically this will only occur in cases where the calculated PCT is low. Westinghouse concludes that no model change is required to calculate an earlier burst time.

Affected Evaluation Models:

1978 ECCS Evaluation Model
1981 ECCS Evaluation Model
1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Status:

Complete.

3.3 STEAM GENERATOR FLOW AREA

Background:

Licensees are normally required to provide assurance that there exists only an extremely low probability of abnormal leakage or gross rupture of any part of the reactor coolant pressure boundary (General design criteria 14 and 31). The NRC issued a regulatory guide (RG 1.121) which addressed this requirement specifically for steam generator tubes in pressurized water reactors. In that guide, the staff required analytical and experimental evidence that steam generator tube integrity will be maintained for the combinations of the loads resulting from a LOCA with the loads from a safe shutdown earthquake (SSE). These loads are combined for added conservatism in the calculation of structural integrity. This analysis provides the basis for establishing criteria for removing from service tubes which had experienced significant degradation.

Analyses performed by Westinghouse in support of the above requirement for various utilities, combined the most severe LOCA loads with the plant specific SSE, as delineated in the design criteria and the Regulatory Guide. Generally, these analyses showed that while tube integrity was maintained, the combined loads led to some tube deformation. This deformation reduces the flow area through the steam generator. The reduced flow area increases the resistance through the steam generator to the flow of steam from the core during a LOCA, which potentially could increase the calculated PCT.

The effect of tube deformation and flow area reduction in the steam generator was analyzed and evaluated for some plants by Westinghouse in the late 1970's and early 1980's. The combination of LOCA and SSE loads led to the following calculated phenomena:

1. LOCA and SSE loads cause the steam generator tube bundle to vibrate.

2. The tube support plates may be deformed as a result of lateral loads at the wedge supports at the periphery of the plate. The tube support plate deformation may cause tube deformation.

3. During a postulated large LOCA, the primary side depressurizes to containment pressure. Applying the resulting pressure differential to the deformed tubes causes some of these tubes to collapse, and reduces the effective flow area through the steam generator.

4. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the calculated peak cladding temperature (PCT).

The ability of the steam generator to continue to perform its safety function was established by evaluating the effect of the resulting flow area reduction on the LOCA PCT. The postulated break examined was the steam generator outlet break, because this break was judged to result in the greatest loads on the steam generator, and thus the greatest flow area reduction. It was concluded that the steam generator would continue to meet its safety function because the degree of flow area reduction was small, and the postulated break at the steam generator outlet resulted in a low PCT.

In April of 1990, in considering the effect of the combination of LOCA + SSE loadings on the steam generator component, it was determined that the potential for flow area reduction due to the contribution of SSE loadings should be included in other LOCA analyses. With SSE loadings, flow area reduction may occur in all steam generators (not just the faulted loop). Therefore, it was concluded that the effects of flow area reduction during the most limiting primary pipe break affecting LOCA PCT, i.e., the reactor vessel inlet break (cold leg break LOCA), had to be evaluated to confirm that 10CFR 50.46 limits continue to be met and that the affected steam generators will continue to perform their intended safety function.

Consequently, the action was taken to address the safety significance of steam generator tube collapse during a cold leg break LOCA. The effect of flow area reduction from combined LOCA and SSE loads was estimated. The magnitude of the flow area reduction was considered equivalent to an increased level of steam generator tube plugging. Typically, the area reduction was estimated to range from 0 to 7.5%, depending on the magnitude of the seismic loads. Since detailed non-linear seismic analyses are not available for Series 51 and earlier design steam generators, some area reductions had to be estimated based on available information. For most of these plants, a 5 percent flow area reduction was assumed to occur in each steam generator as a result of the SSE. For these evaluations, the contribution of loadings at the tube support plates from the LOCA cold leg break was assumed negligible, since the additional area reduction, if it occurred, would occur only in the broken loop steam generator.

Westinghouse recognizes that, for most plants, as required by GDC 2, "Design Basis for Protection against Natural Phenomena", that steam generators must be able to withstand the effects of combined LOCA + SSE loadings and continue to perform their intended safety function. It is judged that this requirement applies to undegraded as well as locally degraded steam generator tubes. Compliance with GDC 2 is addressed below for both conditions.

For tubes which have not experienced cracking at the tube support plate elevations, it is Westinghouse's engineering judgment that the calculation of steam generator tube deformation or collapse as a result of the combination of LOCA loads with SSE loads does not conflict with the requirements of GDC 2. During a large break LOCA, the intended safety functions of the steam generator tubes are to provide a flow path for the venting of steam generated in the core through the RCS pipe break and to provide a flow path such that the other plant systems can perform their intended safety functions in mitigating the LOCA event.

Tube deformation has the same effect on the LOCA event as the plugging of steam generator tubes. The effect of tube deformation and/or collapse can be taken into account by assigning an appropriate PCT penalty, or accounting for the area reduction directly in the analysis. Evaluations completed to date show that tube deformation results in acceptable LOCA PCT. From a steam generator structural integrity perspective, Section III of the ASME Code recognizes that inelastic deformation can occur for faulted condition loadings. There are no requirements that equate steam generator tube deformation, per se, with loss of safety function. Cross-sectional bending stresses in the tubes at the tube support plate elevations are considered secondary stresses within the definitions of the ASME Code and need not be

considered in establishing the limits for allowable steam generator tube wall degradation. Therefore, for undegraded tubes, for the expected degree of flow area reduction, and despite the calculation showing potential tube collapse for a limited number of tubes, the steam generators continue to perform their required safety functions after the combination of LOCA + SSE loads, meeting the requirements of GDC 2.

During a November 7, 1990 meeting with a utility and the NRC staff on this subject, a concern was raised that tubes with partial wall cracks at the tube support plate elevations could progress to through-wall cracks during tube deformation. This may result in the potential for significant secondary to primary inleakage during a LOCA event; it was noted that inleakage is not addressed in the existing ECCS analysis. Westinghouse did not consider the potential for secondary to primary inleakage during resolution of the steam generator tube collapse item. This is a relatively new item, not previously addressed, since cracking at the tube support plate elevations had been insignificant in the early 1980's when the tube collapse item was evaluated in depth. There is ample data available which demonstrates that undegraded tubes maintain their integrity under collapse loads. There is also some data which shows that cracked tubes do not behave significantly differently from uncracked tubes when collapse loads are applied. However, cracked tube data is available only for round or slightly ovalized tubes.

It is important to recognize that the core melt frequency resulting from a combined LOCA + SSE event, subsequent tube collapse, and significant steam generator tube inleakage is very low, on the order of 10^{-8} /RY or less. This estimate takes into account such factors as the possibility of a seismically induced LOCA, the expected occurrence of cracking in a tube as a function of height in the steam generator tube bundle, the localized effect of the tube support plate deformation, and the possibility that a tube which is identified to deform during LOCA + SSE loadings would also contain a partial through-wall crack which would result in significant inleakage. To further reduce the likelihood that cracked tubes would be subjected to collapse loads, eddy current inspection requirements can be established. The inspection plan would reduce the potential for the presence of cracking in the regions of the tube support plate elevations near wedges that are most susceptible to collapse which may then lead to penetration of the primary pressure boundary and significant inleakage during a LOCA + SSE event.

Change Description

As noted above, detailed analyses which provide an estimate of the degree of flow area reduction due to both seismic and LOCA forces are not available for all steam generators. The information that does exist indicates that the flow area reduction may range from 0 to 7.5 percent, depending on the magnitude of the postulated forces, and accounting for uncertainties. It is difficult to estimate the flow area reduction for a particular steam generator design, based on the results of a different design, due to the differences in the design and materials used for the tube support plates.

While a specific flow area reduction has not been determined for some earlier design steam generators, the risk associated with flow area reduction and tube leakage from a combined seismic and LOCA event has been shown to be exceedingly low. Based on this low risk, it is considered adequate to assume, for those plants which do not have a detailed analysis, that 5 percent of the tubes are susceptible to deformation.

The effect of potential steam generator area reduction on the cold leg break LOCA peak cladding temperature has been either analyzed or estimated for each Westinghouse plant. A value of 5 percent area reduction has been applied, unless a detailed non-linear analysis is available. The effect of tube deformation and/or collapse will be taken into account by allocating the appropriate PCT margin, or by representing the area reduction by assuming additional tube plugging in the analysis.

Affected Evaluation Models:

1978 Large Break ECCS Evaluation Model
1981 Large Break ECCS Evaluation Model
1981 Large Break ECCS Evaluation Model with BART
1981 Large Break ECCS Evaluation Model with BASH

Status:

Complete.

3.4 BROKEN LOOP SAFETY INJECTION FLOW IN SMALL BREAKS

Background:

In the Westinghouse NOTRUMP small break Evaluation Model, it is assumed that the safety injection water which flows to the loop in which the break is postulated to occur is entirely discharged to the containment. The practice of not taking cr. it for safety injection into the broken loop preceded the development of calculational models used to satisfy the requirements of 10CFR50.46 or the older Interim Acceptance Criteria (IAC).

It was assumed that neglecting safety injection flow to the broken loop would reduce the capability for core cooling because the flow would not contribute to the reactor coolant system inventory. It was also assumed that the interactive effects on the break flow and the condensation of steam would overall result in better core cooling. The basis for these assumptions was questioned.

The spatial representation of the reactor coolant system, the representation of safety injection flow into the intact loops, and the model for the calculation of the amount of steam condensation as a result of interaction with the safety injection water were selected for conservatism in the Westinghouse 1985 small break LOCA Evaluation Model. This model was reviewed in detail by the NRC and approved. This model, however, is not appropriate for evaluating the effects of safety injection

flow into the broken loop due to the interactive effects of the safety injection fluid with the break flow and steam condensation. To evaluate the effect of safety injection flow into the broken loop, a change was made to the ECCS Evaluation Model to provide a more appropriate representation of the interaction of the safety injection fluid with steam in the RCS. The revised model for condensation of steam due to interaction with the safety injection (SI) fluid was developed based upon test 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 W PWR). The revised steam-SI condensation model was incorporated into a modified version of the Evaluation Model and analysis calculations were performed for a typical three-loop plant.

Analysis calculations which included safety injection flow into the broken loop with the more appropriate revised steam-SI condensation model showed a 54°F benefit over the current model analysis calculation, in which SI into the broken loop is not modeled. However, an increase in PCT was noted when SI was modeled in the broken loop with the revised steam-SI condensation model, when compared to the revised steam-SI condensation model case without SI injection (see summary of results below). Although incorporation of safety injection flow into the broken loop shows a penalty on the peak cladding temperature calculation, it is Westinghouse judgement that the penalty results from the required models of Appendix K to 10CFR50 regarding break flow for the existing spatial representation of the RCS. It is Westinghouse judgement that the actual system response to a small break LOCA event would demonstrate that inclusion of safety injection flow into the loop containing the break would mitigate the consequences of the event to a greater extent than if safety injection flow to the loop containing the break was not delivered to the reactor coolant system.

Westinghouse concluded that the practice of neglecting safety injection flow into the broken loop in combination with a conservative condensation model as in the current version of the Westinghouse 1985 small break LOCA Evaluation Model is conservative and in compliance with the regulatory requirements. Therefore a model change is unnecessary. In order to reach this conclusion, however, the Evaluation Model was changed for application to this analysis scenario.

SUMMARY OF RESULTS

	PCT°F
Current model without safety injection into the broken loop	2037
Revised model with safety injection into the broken loop	1983
Revised model without safety injection into the broken loop	1806

While no change to the Evaluation Model is contemplated as a result of this evaluation, it is possible to view the effect of safety injection flow into the broken loop as significant, since the revised steam-SI condensation model significantly reduces the calculated PCT overall. In accordance with 10CFR50 Appendix K, II.3:

> "Appropriate sensistivity studies shall be performed for each evaluation model to evaluate the effect on the calculated results of variations in noding phenomena assumed to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which the results are shown to be sensitive, the choices made shall be justified."

The existing model is justified as adequately conservative under the requirements of Appendix K to 10CFR50 and will therefore not be revised. This was discussed informally with representatives of the NRC staff at a meeting on January 22, 1991.

Change Description:

Upon evaluation, it was determined that no change to the ECCS Evaluation Model was necessary.

Affected Evaluation Models:

1985 Small Break LOCA Evaluation Model

Status:

Complete

4.0 REFERENCES

1. "SATAN-VI Program: Comprehensive Space Time Analysis of Loss-of-Coolant", WCAP-8306 (Non-Proprietary), June 1974.

2. "Calculational Model for Core Reflooding after a Loss of Coolant Accident (WREFLOOD Code)", WCAP-8171 (Non-Proprietary), June 1974.

3. "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8305, (Non-Proprietary), June 1974.

4. "Containment Pressure Analysis Code (COCO)", WCAP-8326 (Non-Proprietary), June 1974.

5. "Long Term Ice Condenser Containment Code - Lotic Code", WCAP-8355 (Non-Proprietary), July 1974.

6. "Westinghouse ECCS Evaluation Model: 1981 Version," WCAP-9221-A, Revision 1, (Non-Proprietary).

7. "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients", WCAP-9695-A (Non-Proprietary), March 1984.

8. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", WCAP-11524-A (Non-Proprietary), March 1987.

9. "Westinghouse Large Break LOCA Best Estimate Methodology", WCAP-12130-A, (Non-Proprietary), Vols. 1, 2, December 1988.

10. "WFLASH - A Fortran IV Computer Program for Simulation of Transients in a Multi-Loop PWR", WCAP-8261-A (Non-Proprietary).

11. "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", WCAP-8971-A (Non-Proprietary).

12. "NOTRUMP: A Nodal Transient Small Break and General Network Code", WCAP-10080-A (Non-Proprietary).

13. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10081-A (Non-Proprietary).

14. "Westinghouse Large Break LOCA Best Estimate Methodology, Volume 1: Model Description and Validation, Addendum 4: Model Revisions," WCAP-12130, Revision 2, Volume 1, Addendum 4, (Non-Proprietary), Nissley. M. E., et al, August 1990.

15. "10CFR50.46 Annual Notification for 1989 of Modifications in Westinghouse ECCS Evaluation Models," NS-NRC-89-3633, Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), Dated October 5, 1988.

16. "Large Break LOCA Power Distribution Methodology", WCAP-12935 (Non-Proprietary), May 1991.

ATTACHMENT 2 TO AEP:NRC:1118B

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, 3250 MWt Reactor Power)	
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989	ΔPCT= + 0°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1990	ΔPCT= + 0°F
D.	CURRENT LOCA MODEL ASSESSMENTS - 06/1991 (Permanent Assessment of PCT Margin)	
	1. FUEL ROD INITIAL CONDITION INCONSISTENCY	ΔPCT= + 10°F
	2. LB-LOCA BURST & BLOCKAGE ASSUMPTION	ΔPCT= + 0°F
	3. SG TUBE SEISMIC/LOCA ASSUMPTION*	ΔPCT= + 20°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	PCT= 2192°F

*Flow area reduction estimates are not available for Model 51 steam generators. Using a more likely than not argument, a 5% area reduction was used.

LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK UNIT 1

A.	ANALYSIS OF RECORD (Comments: Evaluation Model: BASH, FQT=2.15, FdH=1.55, SGTP= 15%, Other: RHR Cross Tie Valve Open, 3413 MWt Reactor Power)	PCT= 2181°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989	ΔPCT= + 0°F
C.	PRIOR LOCA MODEL ASSESSMENTS - 1990	ΔPCT= + 0°F
D.	CURRENT LOCA MODEL ASSESSMENTS - 06/1991 (Permanent Assessment of PCT Margin)	
	1. FUEL ROD INITIAL CONDITION INCONSISTENCY	ΔPCT= + 10°F
	2. LB-LOCA BURST & BLOCKAGE ASSUMPTION	ΔPCT= + 0°F
	3. SG TUBE SEISMIC/LOCA ASSUMPTION*	ΔPCT= + 20°F
E.	OTHER MARGIN ALLOCATIONS (Use of PCT Margin):	
	1. ANALYSIS MARGINS USED: Power Margin**	ΔPCT= - 94°F
F.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS & POWER MARGIN	PCT= 2117°F

Note: See next page for footnotes.

* Flow area reduction estimates are not available for Model 51 steam generators. Using a more likely than not argument, a 5% area reduction was used.

** JUSTIFICATION FOR USE OF POWER MARGIN IN DONALD C. COOK NUCLEAR PLANT UNIT 1 LARGE BREAK PCT RACKUP

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

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 MWt to the licensed operating condition of 3250 MWt (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 Mwt 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 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. PRIOR LOCA MODEL ASSESSMENTS - 1989 ΔPCT= + 0°F

C. PRIOR LOCA MODEL ASSESSMENTS - 1990 ΔPCT= + 0°F

D. CURRENT LOCA MODEL ASSESSMENTS - 06/1991
(Permanent Assessment of PCT Margin)
1. FUEL ROD INITIAL CONDITION INCONSISTENCY ΔPCT= + 37°F
2. NOTRUMP SOLUTION CONVERGENCE RELIABILITY ΔPCT= - 214°F
3. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION ΔPCT= + 20°F
4. AUXILIARY FEEDWATER ENTHALPY SWITCHOVER ΔPCT= + 26°F

E. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 1991°F

LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A.	ANALYSIS OF RECORD (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
B.	PRIOR LOCAL MODEL ASSESSMENTS - 1989 (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.	CURRENT LOCA MODEL ASSESSMENTS - 06/1991 (Permanent Assessment of PCT Margin)	
	1. FUEL ROD INITIAL CONDITION INCONSISTENCY	ΔPCT= + 10°F
	2. LB-LOCA BURST & BLOCKAGE ASSUMPTION	ΔPCT= + 0°F
	3. SG TUBE SEISMIC/LOCA ASSUMPTION*	ΔPCT= + 20°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	ΔPCT= 2120°F

*Flow area reduction estimates are not available for Model 51 steam generators. Using a more likely than not argument, a 5% area reduction was used.

LARGE BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A. ANALYSIS OF RECORD PCT- 2140°F
(Comments: Evaluation Model: BASH, FQT-2.22, FdH-1.62, SGTP- 15%,
Other: RHR Cross Tie Valve Open, 3588 Mwt Reactor Power)

B. PRIOR LOCA MODEL ASSESSMENTS - 1989 ΔPCT- + NA°F
(Analysis of record was completed in January 1990. No prior LOCA model assessments were made.)

C. PRIOR LOCA MODEL ASSESSMENTS - 1990 ΔPCT- + 0°F

D. CURRENT LOCA MODEL ASSESSMENTS - 06/1991
(Permanent Assessment of PCT Margin)
1. FUEL ROD INITIAL CONDITION INCONSISTENCY ΔPCT- + 10°F
2. LB-LOCA BURST & BLOCKAGE ASSUMPTION ΔPCT- + 0°F
3. SG TUBE SEISMIC/LOCA ASSUMPTION* ΔPCT- + 20°F

E. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT- 2170°F

*Flow area reduction estimates are not available for Model 51 steam generators. Using a more likely than not argument, a 5% area reduction was used.

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A. ANALYSIS OF RECORD PCT= 2124°F
(Comments: Evaluation Model:NOTRUMP, FQT=2.44, FdH=1.64, SGTP= 15%,
Other:HHSI Cross Tie Valve Closed, 3413 Mwt Reactor Power)

B. PRIOR LOCA MODEL ASSESSMENTS - 1989 ΔPCT= NA°F
(Analysis of record was completed in January 1990. No prior LOCA Model assessments were made.)

C. PRIOR LOCA MODEL ASSESSMENTS - 1990 ΔPCT= + 0°F

D. CURRENT LOCA MODEL ASSESSMENTS - 06/1991
(Permanent Assessment of PCT Margin)
1. FUEL ROD INITIAL CONDITION INCONSISTENCY ΔPCT= + 37°F
2. NOTRUMP SOLUTION CONVERGENCE RELIABILITY ΔPCT= + 0°F
3. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION ΔPCT= + 20°F

E. LICENSING BASIS PCT + PERMANENT ASSESSMENTS PCT= 2181°F

SMALL BREAK LOCA

PLANT NAME: DONALD C. COOK NUCLEAR PLANT UNIT 2

A.	ANALYSIS OF RECORD (Comments: Evaluation Model:NOTRUMP, FQT=2.44, FdH=1.64, SGTP= 15%, Other:HHSI Cross Tie Valve Closed, 3413 Mwt Reactor Power)	PCT= 2124°F
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989 (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.	CURRENT LOCA MODEL ASSESSMENTS - 06/1991 (Permanent Assessment of PCT Margin)	
	1. FUEL ROD INITIAL CONDITION INCONSISTENCY	ΔPCT= + 37°F
	2. NOTRUMP SOLUTION CONVERGENCE RELIABILITY	ΔPCT= + 0°F
	3. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION	ΔPCT= + 20°F
E.	LICENSING BASIS PCT + PERMANENT ASSESSMENTS	~~Δ~~PCT= 2181°F

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. PRIOR LOCAL MODEL ASSESSMENTS - 1989 ΔPCT= NA°F
(Analysis of record was completed in January 1990. No prior LOCA Model assessments were made.)

C. PRIOR LOCA MODEL ASSESSMENTS - 1990 ΔPCT= 0°F

D. CURRENT LOCA MODEL ASSESSMENTS - 06/1991
(Permanent Assessment of PCT Margin)
1. FUEL ROD INITIAL CONDITION INCONSISTENCY ΔPCT= + 37°F
2. NOTRUMP SOLUTION CONVERGENCE RELIABILITY ΔPCT= + 0°F
3. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION ΔPCT= + 20°F

E. LICENSING BASIS PCT & PERMANENT ASSESSMENTS PCT= 1414°F