

NORTHEAST UTILITIES

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July 19, 1991

Docket No. 50-423
B13878

Re: 10CFR50.46(a)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Unit No. 3
Reporting of Changes to, and Errors in
ECCS Models or Applications

In accordance with 10CFR50.46(a), transmitted herewith are the changes to, and errors in, the ECCS evaluation model or application of the model for the Millstone Unit No. 3.

The Millstone Unit No. 3 ECCS evaluation models are based on the Westinghouse methodology which has been undergoing an evaluation of safety issues identified in the models. The resolution of these issues has required the vendor to implement changes in the ECCS methodology. The changes reported in this submittal encompass the time period from August 1990 to May 1991 and include both the Small Break and Large Break LOCA evaluation models.

The changes have resulted in revised peak clad temperatures mostly as a result of permanent PCT margin allocations. These allocations have resulted in a significant change to the PCT in the small break LOCA model and, as such, are being reported within 30 days based on the criteria established in 10CFR50.46.

The change in the PCT for the large break LOCA model is not significant. However, it is also being reported to keep the Staff informed.

In neither case have the licensing limits of 10CFR50.46 been exceeded. An adequate margin in the PCT remains, and no reanalysis is required.

Attachment 1 to this letter provides a report describing changes to the Westinghouse ECCS evaluation models, as applicable to Millstone Unit No. 3.

Attachment 2 provides information regarding recently identified safety issues that are currently under investigation by Westinghouse.

Attachment 3 provides an ECCS Evaluation Model Margin Assessment which accounts for the permanent changes from the resolution of the issues described in Attachment 1. The assessment does not include temporary margin

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allocation from the issues identified in Attachment 2 since these issues are still under evaluation and need not be reported.

We trust that this information satisfies the reporting requirements of 10CFR50.46.

Very truly yours

NORTHEAST NUCLEAR ENERGY COMPANY


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Senior Vice President

cc: T. T. Martin, Region I Administrator
D. H. Jaffe, NRC Project Manager, Millstone Unit Nos. 1 and 3
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ATTACHMENT 1

CHANGES TO THE
WESTINGHOUSE ECCS EVALUATION MODELS

AUGUST 1990 - MAY 1991

CHANGES TO THE WESTINGHOUSE ECCS EVALUATION MODELS

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.

Westinghouse has completed the evaluation of several items related to the Westinghouse ECCS Evaluation Models. 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 Millstone Unit No. 3. These are discussed in Section 3.0. 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.

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.

Change Description:

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.

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)(3)(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.

1. 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:

a. 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.

b. 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.

c. 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 10°F for the BASH Evaluation Model.

2. 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 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.

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.

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 uncover 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 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 NOTRUMP Code Solution Convergence

Change Description:

In the development of the NOTRUMP small break LOCA ECCS Evaluation Model, a number of nodding 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.

Effect of Change:

Generally, the modifications result in small shifts in timing or core uncover 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 life-time.

The current basis for all Westinghouse large break 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 (FAH). 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, FAH, 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 FAH. Further studies revealed that, in addition to FQT, FAH, 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.

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 Large 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 so 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 occurred 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.

3.3 Steam Generator Flow Area

Background:

Licenses 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 1970s and early 1980s. 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 10CFR50.46 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 percent, 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 undergraded, 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 undergraded 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 1980s when the tube collapse item was evaluated in depth. There is ample data available which demonstrates that undergraded 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.

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 credit 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 sensitivity studies shall be performed for each evaluation model to evaluate the effect on the calculated results of variations in nodding 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.

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), Volumes 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

SUMMARY OF LOCA ISSUES
CURRENTLY BEING EVALUATED
BY WESTINGHOUSE

INFORMATION REGARDING CURRENT POTENTIAL ITEMS

BACKGROUND

Westinghouse is evaluating potential items in the Westinghouse ECCS Evaluation Models for the following areas. Information is being developed to allow a temporary allocation of ECCS analysis margin to reasonably assure that compliance with the requirements of 10CFR50.46 would be maintained upon resolution of the Potential Items.

Core Average Zircaloy-Water Reaction

10CFR50.46(b)(3) requires that the total amount of hydrogen generated from the chemical reaction of the cladding with water or steam be less than 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. Historically, a value of less than 0.3 percent was provided with the ECCS analysis results for reporting in the plant final safety analysis report (FSAR). This value was based upon Westinghouse calculations performed in 1974 which indicated that the total Zircaloy-water reaction would be less than 0.3 percent if the hot rod calculated peak cladding temperature was less than 2200°F and was based upon an assumed limiting fuel rod power distribution census.

A correction to the Westinghouse ECCS Evaluation Models which doubled the heat generation from the Zircaloy-water reaction was performed in 1978, and core design methods have continued to evolve which could affect the fuel rod census. Consequently, Westinghouse is evaluating the appropriate value for reporting in the FSAR. Even if the total core Zircaloy-water reaction was twice as large as currently reported in the FSAR, the amount of cladding and water reacted would still be less than the 1 percent regulatory limit. Westinghouse is recommending a temporary ECCS analysis margin allocation of 0.7 percent of the Zircaloy-water reaction pending resolution of this item. Final resolution may involve a revision to the FSAR which indicates that the Zircaloy-water reaction was less than 1 percent.

Small Break LOCA Rod Drop Time

For small break LOCAs, Westinghouse assumes that all but the most reactive rod control cluster assembly (RCCA) is inserted. The value for the RCCA drop time is typically the same value which is used in non-LOCA analyses. However, during a LOCA there are three principal loads on the control rod guide tubes that must be considered in calculating the capability of the control rods to insert: (1) the loads from the dynamic decompression wave due to the postulated break, (2) the loads from fluid cross flow due to the depressurization from the postulated LOCA, and (3) the seismic loads due to a postulated safe shutdown earthquake (SSE). While the probability of a concurrent seismic event and small break LOCA is extremely low, the NRC has indicated that the combined loads need to be considered.

It was determined that the potential effects of the loads from the dynamic decompression wave due to the postulated break and the loads from fluid cross flow due to the depressurization from the postulated LOCA may not

have been appropriately considered in the RCCA drop time used in the LOCA analyses. Consideration of these loads could result in an increased time for RCCA insertion and the failure of additional RCCAs to insert. Delays in the insertion of the RCCAs could have a small effect on the peak cladding temperature due to assumptions regarding the transition from full power to shutdown decay heat conditions.

While this item requires further evaluation, preliminary examination of the effect of the loads for a typical three-loop plant with 17 X 17 guide tubes, a typical four-loop plant with 17 X 17 guide tubes, and a typical four loop plant with 15 X 15 guide tubes indicates that the control rod insertion and rod drop time is not significantly affected by the deflections of the guide tubes due to the loads.

Small Break Secondary Side Modeling

For small break LOCAs, Westinghouse assumes that the main feedwater flow is isolated early in the event. The early isolation was assumed to be conservative since it increased the potential for uncovering of the steam generator tube bundle on the secondary side which would reduce the primary to secondary heat removal capability. Typically, Westinghouse analyses assumed that main feedwater flow during a small break LOCA was isolated 5 seconds after the reactor trip signal as a result of the closure of the main feedwater isolation valve or main feedwater flow regulation valve when off-site power is lost.

However, a longer time for isolation of the main feedwater flow could also reduce the total secondary side decay heat removal capability due to the modeling of the effect of throttling of the lower enthalpy auxiliary feedwater flow. If increased amounts of main feedwater flow were delivered, a reduced amount of lower enthalpy auxiliary feedwater flow would be delivered prior to throttling, thereby reducing the secondary side heat removal capability. The effect of this item is complicated by the discovery of a discrepancy in the modeling of the steam generator secondary side mixture level tracking inputs. The input discrepancy would delay the time at which throttling of the auxiliary feedwater flow would occur in the model.

While this item requires further evaluation, there is reasonable assurance that analysis results obtained with the conservative representation of the steam generator secondary side in the NOTRUMP small break LOCA ECCS Evaluation Model will bound analysis results obtained with a more realistic representation of the reactor coolant system.

Small Break LOCA Burst and Blockage Considerations

For small break LOCAs, Westinghouse assumes that, consistent with the large break LOCA assumptions, the fuel rod initial conditions that result in the highest calculated peak cladding temperature (PCT) result from beginning of life conditions. Typically, rupture of the fuel cladding is not calculated for beginning of life fuel rod conditions in a small break LOCA. As the fuel rod internal pressure increases with burnup, rupture of the cladding would be calculated during a small break LOCA potentially resulting in an increase in the PCT due to flow blockage effects and the effect of the metal water reaction on the inside of the cladding.

Based upon a limited number of sensitivity studies, Westinghouse observed lower calculated PCTs for the middle and end of life fuel rod conditions due to the benefits available from reductions in the fuel operating temperature and the cladding high temperature creep performance, both of which offset the penalties resulting from flow blockage due to fuel cladding rupture and the metal-water reaction on the inside of the cladding. However, in the fuel rod studies performed, a rupture of the cladding resulted early in the cladding heatup transient. The inside metal-water reaction resulted in a small PCT penalty relative to a case which did not burst, but the overall burnup benefits resulted in a lower PCT. However, the metal-water reaction, in these studies, was not as vigorous as could be calculated should the rupture of the cladding occur later in the transient when the cladding temperatures were higher. A rupture of the cladding later in the heatup transient could lead to an increased metal-water reaction rate and a higher PCT. Consequently, a more limiting time in life could be calculated if the rupture of the cladding were calculated to occur just prior to reversal of the cladding heatup transient.

It should be noted that rupture of the fuel rod will occur at some time in fuel rod life for certain postulated small break LOCAs. However, rupture of the fuel rod at low cladding temperatures does not significantly affect the PCT results. For the Westinghouse small break LOCA ECCS Evaluation Model, fuel rod rupture at cladding temperatures lower than approximately 1700°F would result in a benefit to the calculated PCT since the cladding strains away from the fuel pellets.

Higher fuel rod internal pressures representative of the middle-of-life and end-of-life conditions tend to result in a rupture of the cladding early in the cladding heatup transient. This allows inside metal-water reaction to build up an oxide layer at lower temperatures and results in a small PCT penalty relative to a case which did not burst. However, the metal-water reaction is not as vigorous as could be calculated if the rupture of the cladding were to occur later in the transient when the cladding temperatures were higher. A rupture of the cladding later in the heatup transient would lead to an increased metal-water reaction rate and a higher PCT.

The limiting time in life will not result at end-of-life conditions and probably not middle-of-life conditions even though fuel rod rupture, metal-water reaction on the inside of the cladding, and flow blockage would be calculated to occur in these instances. The higher rod internal pressures will cause the fuel rods to rupture early in the cladding heatup transient. The buildup of an oxide layer at lower cladding temperatures will inhibit the metal water reaction and prevent the heat addition which would result from a rupture of the fuel rod at higher cladding temperatures.

The limiting time in life consists of the fuel rod pressure and temperature conditions which would result in burst occurring at the PCT time if rupture were inhibited. For lower calculated peak cladding temperatures, the effect is insignificant. However, at higher cladding temperatures, the effect could be significant. While potentially significant, margin in the small break LOCA power distribution assumption, and other small break LOCA analysis assumptions could offset the deleterious effects. Westinghouse

will continue to evaluate this item and provide additional information as it becomes available.

ATTACHMENT 3

LOCA PCT SUMMARIES

PCT SUMMARY INFORMATION

Attachment 3 provides a summary of the current licensing basis ECCS Evaluation Model margin for the large break and small break loss-of-coolant accident (LOCA) analysis. The ECCS analysis results are modified to accommodate permanent margin allocations. A margin allocation of zero indicates that no change to the Westinghouse ECCS Evaluation Model was necessary, or indicates that the analysis result incorporate the necessary change. In addition, current issues not related to the ECCS Evaluation Model, but which may affect input to the model, are listed.

REPORTING OF 10CFR50.46 MARGIN UTILIZATION
LARGE BREAK LOCA

PLANT NAME: MILLSTONE UNIT 3

A.	ANALYSIS OF RECORD	PCT = <u>2133.2°F</u>
	(Comments: Evaluation Model: BASH	
	FQT = 2.6	
	FdH = 1.7	
	SGTP = 10%	
	Other: VANTAGE 5H FUEL)	
B.	PRIOR LOCA MODEL ASSESSMENTS - 1989	Δ PCT = <u>+ 0.0°F</u>
	(Permanent Assessment of PCT Margin)	
C.	PRIOR LOCA MODEL ASSESSMENTS - 1990	Δ PCT = <u>+ 0.0°F</u>
	(Permanent Assessment of PCT Margin)	
D.	CURRENT LOCA MODEL ASSESSMENTS - 06/1991	
	(Permanent Assessment of PCT Margin)	
	1. FUEL ROD INITIAL CONDITION INCONSISTENCY	Δ PCT = <u>+ 10.0°F</u>
	2. LB-LOCA BURST & BLOCKAGE ASSUMPTION	Δ PCT = <u>+ 0.0°F</u>
	3. SG TUBE SEISMIC/LOCA ASSUMPTION	Δ PCT = <u>+ 6.4°F</u>
E.	10CFR50.59 SAFETY EVALUATIONS - (Permanent Assessment of PCT Margin)	
		Δ PCT = <u>+ 0.2°F</u>
	LICENSING BASIS PCT + MARGIN ALLOCATION	Δ PCT = <u>2149.8°F</u>

REPORTING OF 10CFR50.46 MARGIN UTILIZATION

SMALL BREAK LOCA

PLANT NAME: MILLSTONE UNIT 3

A. ANALYSIS OF RECORD

PCT = 1890.5°F

(Comments: Evaluation Model: NOTRUMP
FQT = 2.6
FdH = 1.7
SGTP = 10%
Other: VANTAGE 5H FUEL)

B. PRIOR LOCA MODEL ASSESSMENTS - 1989
(Permanent Assessment of PCT Margin)

Δ PCT = + 0.0°F

C. PRIOR LOCA MODEL ASSESSMENTS - 1990
(Permanent Assessment of PCT Margin)

Δ PCT = + 0.0°F

D. CURRENT LOCA MODEL ASSESSMENTS - 06/1991
(Permanent Assessment of PCT Margin)

1. FUEL ROD INITIAL CONDITION INCONSISTENCY
2. NOTRUMP SOLUTION CONVERGENCE RELIABILITY
3. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION

Δ PCT = + 37.0°F

Δ PCT = + 0°F

Δ PCT = + 10.0°F

E. 10CFR50.59 SAFETY EVALUATIONS - (Permanent Assessment of PCT Margin)

Δ PCT = + 13.4°F

LICENSING BASIS PCT + MARGIN ALLOCATION

Δ PCT = 1950.9°F