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Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS

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1.0 INTRODUCTION

This report provides a description of the Westinghouse Evaluation Models and analysis methodology used to perform Large Break and Small Break LOCA licensing analyses for Combustion Engineering (CE) Nuclear Steam Supply System (NSSS) designs. The models described herein have been developed in accordance with the requirements of 10 CFR 50 Appendix K and Item II.K.3.30 of NUREG-0737 (for Small Break LOCA). The intended application of these models is as a licensed analysis tool to demonstrate compliance with 10 CFR 50.46 for the CE NSSS design.

Previous applications of Westinghouse LOCA methodology to a Combustion Engineering NSSS have relied heavily on the approved Evaluation Models and methods employed for the analysis of Westinghouse plants. Since most of the modeling and methodology is directly applicable to the analysis of a CE NSSS, the licensing precedent has been to directly reference the NRC-approved models and methods used for the analysis of Westinghouse plants. The additional documentation required for licensing the application to a CE NSSS focused primarily on the modifications required to appropriately model the differences between the Westinghouse NSSS and CE NSSS designs, and the differences between the fuel designs for the two systems. This report will employ a similar approach, referencing existing NRC-approved documentation, where possible, and focusing primarily on addressing plant and fuel differences relative to the LOCA analysis methodology.

The specific CE NSSS design examined in the preparation of this report was Fort Calhoun Unit 1. However, the ECCS Evaluation Models described in this report are intended to have generic applicability. Most of the

modifications to the existing Westinghouse ECCS Evaluation Models have been previously employed in the analysis of another CE NSSS, Millstone Unit 2. Where differences were noted between Fort Calhoun Unit 1 and Millstone Unit 2, not readily accommodated by plant-specific code input, the necessary code or methodology modifications were made in a manner permitting the modeling of either configuration. These models will also be applicable for the Large Break and Small Break LOCA analyses for plants, other than those identified here, which have a Combustion Engineering designed NSSS. Section 2.0 of this report contains a description of the CE (specifically Fort Calhoun) plant and fuel design, and the differences from a typical Westinghouse plant and fuel design.

The Large Break LOCA analysis will be performed using a version of the 1981 Evaluation Model with BART, modified for application to a CE NSSS. The BART EM has been used for the analysis of Large Break LOCA in numerous licensing applications for Westinghouse fuel in Westinghouse plants. Modifications to the codes which constitute the BART EM and special modeling methods will be developed and implemented for the analysis of the CE NSSS. The model, as modified for the intended application, will hereafter be referred to as the 1981 + BART for CE NSSS Large Break LOCA ECCS Evaluation Model (BART for CE EM). Modification of the various codes and modeling features of this proposed model are contained in section 3.0 of this report.

The Small Break LOCA EM described in this report is the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP code for the Combustion Engineering NSSS (NOTRUMP EM for CE). The NOTRUMP EM has been used for the analysis of Small Break LOCA in numerous licensing applications for Westinghouse fuel in Westinghouse plants. The version of the NOTRUMP EM

developed for the analysis of a CE NSSS has previously been used to perform the Small Break analysis for Millstone Unit 2, which was subsequently reviewed and approved for licensing applications by the NRC. The documentation contained in this topical, describing the NOTRUMP EM for CE, updates the information previously provided to the NRC. Only minor modifications to the previously approved methodology have been implemented for the modeling of a CE NSSS. However, since the initial approval of the NOTRUMP EM for CE, several modifications have been implemented for the Westinghouse NOTRUMP EM. These modifications have also been included in the version for the analysis of a CE NSSS. The descriptions of the NOTRUMP EM for CE contained in this report do not define a new Evaluation Model, but rather augment the existing definitions and references to describe the model and its application in accordance with 1991 technology and standards. A further discussion of the various codes and modeling features of this model are contained in section 4.0 of this report.

Section 5.0 of this report individually addresses the five Acceptance Criteria of 10 CFR 50.46, and the Westinghouse licensing position on satisfying each of these criteria for a plant-specific application. Section 6.0 contains a summary of the proposed models, methods and licensing approach. References are contained in Section 7.0.

2.0 System Description

The CE NSSS design has been reviewed to ensure the applicability of the models and methodology to be used for the Large and Small Break LOCA analyses. Following is a brief summary of the unique features of the CE

NSSS and proposed "CE-type fuel" design[1], highlighting the differences from a typical Westinghouse plant and fuel design.

Loop Layout: The CE design incorporates a system of two hot legs feeding two Steam Generators. The two steam generators feed four cross-over legs (loop seals) attached to four pumps which, in turn, feed four cold legs. The cold leg diameter is slightly smaller than that in a Westinghouse NSSS design, while the hot leg diameter is larger. In addition, the elevation of the loop seals is near the top of the core elevation, unlike the Westinghouse design, in which the loop seal elevation is near the core midplane.

Control Element Assemb (CEA) Design: The CE design includes a variety of shroud designs for the CEAs. More than one design may be present within a given plant.

Upper Head Bypass/Upper Head Temperature: There is no design cooling flow path for the CE upper head design. Only a very small amount of flow passes from the upper downcomer to the upper head via the upper head alignment keyway, resulting in an upper head temperature very near T_{HOT}.

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1 The "CE-type fuel" design refers to a Westinghouse designed fuel assembly for use in a CE NSSS, specifically Fort Calhoun. This design features a high degree of compatibility with the existing CE fuel design (large thimbles & instrumentation tubes, short active fuel stack, etc.) compared to typical Westinghouse fuel designs, and is thus identified as "CE-type fuel".

Lower Plenum Design: The lower plenum of the CE design differs substantially from the Westinghouse design. The CE design features an extension to the core barrel, the Flow Skirt, which extends to the inner bottom surface of the vessel. A large number of small peripheral holes through this Flow Skirt provide the flow path from the barrel vessel annulus to the lower plenum region directly beneath the reactor core. Compared to a Westinghouse design, the lower plenum region for the CE design is relatively open, with no base plate or absorber rods and comparatively small support columns. Flow restrictions in the lower plenum are present at the Flow Skirt, Lower Flow Plate, and Core Barrel Support Plate.

Core Shroud Bypass: The core shroud region of the CE design differs from the Westinghouse Upflow Barrel/Baffle design in the type of flow paths available and relative flow resistance through the region. For Fort Calhoun, for example, there are no flow holes through the centering plates (horizontal structures) in this region. Slots at the interior edge of the centering plates provide a bypass flow path from lower to upper plenum.

Reactor Coolant Pump: The reactor coolant pump design (single speed vertical shaft centrifugal pumps) for the CE plants are similar to those employed at Westinghouse plants, although the size of the pumps are somewhat smaller.

Steam Generator Inlet Nozzle Angle: The Westinghouse NSSS features a Steam Generator inlet nozzle inclined 40^0 from vertical, while CE designs have a greater angle of inclination.

Safety Injection/Accumulator Nozzle Angle: As in many Westinghouse plants, the CE design provides for a safety injection line which is connected to the cold leg at an angle. For Westinghouse plants, this angle is either 45^{0} or 90^{0}, while this angle is either 60^{0} or 75^{0} from horizontal for the CE designs. The pressure differential due to the interaction of safety injection with steam flow in the cold leg piping is a function of this injection angle.

Safety Injection Tank Pressure: Westinghouse plants feature passive injection tanks (accumulators) with a cover gas pressure on the order of 600 to 700 psig. Some CE designs may include Safety Injection Tanks with a minimum cover gas pressure near 200 psig.

Fuel Assembly Design: The CE fuel assembly design features guide tubes which are several times larger than the Westinghouse 14 x 14 guide thimbles. The one instrumentation tube per assembly is also much larger for the CE design. Grid parameters also vary slightly from the Westinghouse 14 x 14 designs.

Fuel Rod/Pellet Stack Design: The cladding radial dimensions for the Westinghouse designed fuel for a CE NSSS differ somewhat from the Westinghouse designs, including open volumes inside the rod. Also, the active pellet stack height for the CE fuel is often less than the typical 12 foot core length identified with most Westinghouse plants.

3.0 LARGE BREAK LOCA

The Large Break ECCS Evaluation Model for the analysis of a CE NSSS is based upon the Westinghouse 1981 Evaluation Model with BART (BART EM). The BART EM was developed to satisfy the requirements for the Analysis of Large Break Loss-of-Coolant Accidents, as defined in 10 CFR 50.46[1], and employs the required models and assumptions identified in 10 CFR 50, Appendix K[1]. The BART EM incorporates components of the earlier 1981 Evaluation Model[2], plus BART technology[3,4,5]. The codes used in the BART EM are SATAN-VI, WREFLOOD, COCO, BART AND LOCTA-IV.

Using the Westinghouse BART EM as a basis, a number of modifications were introduced to more accurately model the specifics of the CE NSSS design and to provide a more accurate calculation of the transient response. Modifications included to accurately model the CE NSSS design are discussed in greater detail in Sections 3.1 and 3.2.

Aside from the modifications for the CE design, the Westinghouse version of the BART EM was enhanced for application to CE NSSS designs. This enhancement is manifested through a tighter coupling of the BART and LOCTA-IV codes.

The direct communication of these codes provides a more accurate calculation of core heat transfer and rod heat-up. The coordinated interaction of these codes was first described as an intended future application for the BART EM[3,4,5]. The NRC Safety Evaluation Report (SER), contained in the Proprietary, Approved version of the BART topical report[3] identifies that the NRC, "finds the loose coupling acceptable for evaluation model calculations." This SER does not

specifically preclude the use of the "tightly coupled" BART/LOCTA model, presuming sufficient verification is provided and NRC approval is obtained. Subsequent enhancements in Westinghouse Large Break LOCA ECCS modeling capability lead to the development of a "tightly coupled" BART/LOCTA unit, the LOCBART code. The LOCBART code is a synthesis of the existing LOCTA-IV and BART codes and is an integral part of the 1981 Evaluation Model with BART/BASH (BASH EM)(6). The SER contained in the Proprietary, Approved version of the BASH EM topical report(6) identifies the acceptability of using the LOCBART code for Appendix K Large Break LOCA calculations as part of the BASH EM.

The BASH code(6) was developed as a replacement of the WREFLOOD code for the calculation of system hydraulics during the reflood portion of a Large Break LOCA. The Evaluation Model described in this report does not make use of the BASH code. Reference to the BASH topical report is provided because this report provides a detailed description of the LOCBART code. The LOCBART code is used to calculate clad to fluid heat transfer coefficients and fuel rod thermal transient behavior as part of the Evaluation Model described in this report. The LOCBART code is further described in Section 3.1.7.

Because the LOCBART code is a normal part of the BASH EM, the format for reading system hydraulic information (required by LOCBART) is consistent with the output format from the BASH code. This format differs from the output from the WREFLOOD code. Thus to facilitate the transfer of information from the WREFLOOD code to the LOCBART code, the REFBASH code was developed. The REFBASH code serves two purposes - 1) to reformat the WREFLOOD output information to a form readily readable by LOCBART, and 2) to adjust the WREFLOOD flooding rate information to be consistent with the methodology employed as part of the Westinghouse BART EM(4). The REFBASH code is described in further detail in Section 3.1.6.

The Evaluation Model described in this report is referred to as the 1981 + BART for CE NSSS Evaluation Model (BART for CE EM). The codes used in this EM are SATAN-VI, WREFLOOD, COCO, REFBASH and LOCBART.

Among these codes, those which were included in the 1981 Evaluation Model (SATAN-VI, WREFLOOD, COCO and LOCTA-IV) have previously been modified for application to a CE NSSS. Changes were made to the 1978 Evaluation Model[7] to 1) introduce new technology based on analytical techniques that had been approved for the modeling of plants with Upper Head Injection (UHI), and 2) incorporate the necessary features appropriate for the analysis of Large Break LOCA for Westinghouse fuel in a CE NSSS.[8] The 1978 EM with the "UHI technology" eventually became the 1981 EM[2], and the modifications for the application to CE NSSS were approved along with the model improvements for the 1981 EM in the NRC Safety Evaluation Report (SER) which appears in the Proprietary Version of the 1981 EM topical report[2]. The NRC SER for the 1981 Evaluation Model with BART[3] states that the model, "may be applied to PWR's using Westinghouse fuel and which use only cold leg injection," which encompasses the Fort Calhoun Large Break LOCA analysis. However, modifications for the modeling of a CE NSSS have, to date, not been identified and approved for the LOCBART code.

Section 3.1 briefly describes the individual codes of the BART for CE EM including, where applicable, the features which previously have been included in the codes for the modeling of a CE NSSS. Section 3.2 contains a description of the necessary modifications for the BART EM codes and methodology required to ensure the adequacy of the proposed BART EM for CE NSSS. Section 3.3 provides as overview of the application of the model to a CE NSSS, based on the plant geometry and plant operating conditions at Fort Calhoun Unit 1. This overview includes a description of the basic break spectrum analysis plus a description of the sensitivity studies considered in support of this report.

3.1.1 SATAN-VI

The analysis of the blowdown portion of the Large Break LOCA transient will be performed with the SATAN-VI[9] code. SATAN-VI has previously been modified for application to a CE NSSS as part of the 1981 EM[2,8]. The primary modifications to this code were 1) the addition of five new elements (47 - 51) to model the intact cold side piping from the Steam Generator outlet plenum to the vessel (4 elements), plus the Safety Injection Tank (SIT) attached to the broken loop intact cold leg, and 2) the addition of two elements (52 and 53) to model the CEA shroud flow paths from the Upper Head to the Upper Plenum. A more detailed discussion of these modifications is contained in reference 8.

3.1.2 WREFLOOD

The WREFLOOD[10] code is used for calculating the system thermal-hydraulics for the refill and reflood portions of the Large Break transient. In addition, WREFLOOD is interactively linked to the COCO code providing reflood mass and energy release data for the calculation of the containment pressure transient. Two modeling/methodology modifications were incorporated into WREFLOOD for modeling a CE NSSS as part of the 1981 EM[2,8]. As with SATAN-VI, elements were added to model the intact leg in the broken loop[2] (auxiliary loop).

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2 The intact leg of the broken loop is often referred to as the "auxiliary loop" in Westinghouse documentation for CE NSSS analysis.

Because of the difference in the injection angle for Safety Injection flow into the steam-filled cold legs (see Section 2.0), the NRC-specified pressure drops of 0.4 and 1.5 psi for injection angles of 60^{0} and 75^{0}, respectively, are used to model the unrecoverable pressure drop due to steam/water mixing in the Reactor Coolant System (RCS). This feature of the CE NSSS design is accommodated in the WREFLOOD code via input. A more detailed discussion of these modifications is contained in reference 8.

3.1.3 COCO

The COCO code(11) is used to calculate the containment pressure transient during the Large Break LOCA. COCO uses code input, mass and energy information read from the SATAN-VI tape and information on the reflood mass and energy releases provided interactively from WREFLOOD. No modifications (other than plant specific input data) are required to apply COCO to a CE NSSS.

3.1.4 BART

The BART(3) code was developed to provide a more mechanistic calculation of core heat transfer coefficients which had previously been determined from the FLECHT correlation(12) employed in the LOCTA-IV(13) code as part of the 1981 EM. None of the features of the BART code specifically require modification for application to a CE NSSS. With the development of the BART EM, modifications were also added to the WREFLOOD and LOCTA-IV codes for consistency in modeling with BART. The modeling of metal heat release in the vessel downcomer and lower plenum was enhanced from a simple conservative exponential decay to a more mechanistic calculation. The improved version of the WREFLOOD code models the various metal

components of the lower plenum as slabs, spheres and cylinders and heat flux to the fluid in these regions is calculated by solving the conduction equation based on the modeled geometries. In addition, the core mass entrainment calculation in WREFLOOD was modified to account for the effect of boiling below the quench front for compatibility with BART methods. The BART code is discussed further in Sections 3.1.5 and 3.1.6.

Changes in modeling methodology were also introduced as part of the BART EM. The core reflooding rate during the reflood portion of the transient is used as input by the BART code. Due to the limited number of available data points from WREFLOOD, the methodology for inputting flooding rate data into BART was revised(4). The revision permits a more accurate representation of the core reflooding transient and the integral of the flooding rate over the initial oscillatory period at the start of reflood. This methodology is discussed further in section 3.1.6.

Subsequent to the approval of the BART EM, a modification to the method of modeling the control rod guide thimbles (called guide tubes in a CE plant) was introduced(5). This modification changed the manner in which the volume inside the thimbles was represented in the WREFLOOD code, and this revised model is applicable to all BART EM analyses. At the same time, it was identified that due to calculational differences between LOCTA-IV and BART, the identification of the number of rods in the LOCTA-IV input should be limited to only the fuel rods for accurate calculation of enthalpy rise and the fuel rod to fluid heat transfer coefficient(5). To offset the anticipated increase in calculated Peak Clad Temperature resulting from these methodology modifications, a previously unused model in the BART code was activated. The radiation heat transfer model, which had previously been approved by the NRC staff, was implemented as part of the BART EM(5).

The BART Evaluation Model was approved for application to PWR's using Westinghouse fuel and which use only cold leg injection by the NRC SER which is printed in the Proprietary version of reference 3. This SER restricts the use of the BART spacer grid model, pending further studies[14]. Aspects of the spacer grid model later received separate approval[15], and are included as part of the BART EM.

It is noted that the BART code as described here is not directly applied as part of the proposed 1981 + BART for CE NSSS EM. The function of the BART code as described here has been incorporated into the LOCBART code which is the interactive combination of the existing BART and LOCTA-IV codes. The LOCBART code is included as part of the BART for CE EM. The BART EM and BART code have never been applied to a CE NSSS and, consequently, no modifications for this type of modeling have previously been incorporated.

3.1.5 LOCTA-IV

The LOCTA-IV code has long been used to perform the rod heat-up calculations in Westinghouse Evaluation Models, including the 1978 EM[7], the 1981 EM[2] and the BART EM[3]. Several modifications were made to the LOCTA-IV code for modeling a CE NSSS (with "CE-type fuel") based on the 1981 EM[8].

A minor modification was incorporated to accurately model the fuel crack and dish volumes, which are calculated internally in LOCTA-IV. Parameters appropriate to "CE-type fuel" were included in the code, replacing the standard Westinghouse values. The burst and blockage modeling is verified as being applicable without modification. Also, the optional feature for modeling fuel rod to guide tube radiation is described, although this option was not employed in the calculations whose results are documented in reference 8.

As with the BART code, the LOCTA-IV code is not applied directly as part of the BART for CE EM, but LOCTA-IV is incorporated into the LOCBART code. The LOCBART code is discussed further in Section 3.1.7.

3.1.6 REFBASH

The REFBASH code has not previously been used as part of any Westinghouse ECCS Evaluation Model. This code does not perform any system thermal-hydraulic or fuel transient calculations, but is strictly an intermediate input/output data processing tool. The two purposes of the REFBASH code are to rewrite WREFLOOD output data to a format readable by LOCBART and to adjust the WREFLOOD flooding rates to be consistent with the BART EM methodology.

Although a description of the "tight coupling" of BART and LOCTA is provided as part of the BART EM(3), the LOCBART code is normally used as part of the BASH EM(6). Most of the system hydraulic data during the reflood portion of the transient is read from the BASH output tapes in a standard BASH analysis. Similar information may, of course, be obtained from a WREFLOOD output tape, however, the output data from WREFLOOD and BASH are written to tape in different formats. A major portion of the REBASH function is to read the necessary data from the WREFLOOD tape, and rewrite the information to another tape in an output format similar to the BASH output format. The reformatted WREFLOOD data, in the form of the REFBASH output tape, can then be read directly by the LOCBART code.

The BART code which has been incorporated into LOCBART requires as input, flooding rate data for the reflood portion of the transient, which is relatively free of marked oscillatory behavior. The WREFLOOD code

typically calculates several noticeable oscillations during the first few seconds of reflood. To address this modeling mismatch, a methodology was developed for use with the BART EM to limit the oscillations early in reflood while still providing an accurate representation of the flooding rate and flooding rate integral(4). Because the flooding rate tables were "hand input" with the BART code, adjustment of the WREFLOOD flooding rate for BART input was also performed by hand. For LOCBART, flooding rate information is read directly from tape. To accommodate this form of data transfer, the flooding rate adjustment required by the BART portion of LOCBART has been automated within the REFBASH code.

Even though no system, containment or fuel calculations are performed by the REFBASH code, it is an integral part of the BART for CE EM code sequence, and is subject to the same technical and quality assurance requirements as the other safety-related codes.

3.1.7 LOCBART

The LOCBART code(6) is used for performing the rod heat-up calculations. LOCBART is a synthesis of two previously approved Large Break LOCA codes, LOCTA-IV and BART. The combination of these codes permits a direct exchange of transient fuel rod temperatures (from LOCTA) and heat transfer coefficients (from BART) at each time step. This is an improvement over the Westinghouse BART EM methodology which required iteration and "hand transfer" of input between the BART and LOCTA-IV codes as discussed in Sections 3.1.4 and 3.1.5.

The LOCTA-IV portion of LOCBART is the same as that used in the BART EM. The BART portion of the code was modified to model and numerically solve equations for reverse core flow during reflood along with a modified calculation of droplet number density(6).

There is currently no licensing documentation supporting the use of the LOCBART code for analysis of a CE NSSS. It is however noted that the components of the LOCBART code have previously been modified for CE NSSS analyses[8], and had been approved for this application[2,3]. Additional modifications required for LOCBART for analysis of a CE NSSS are described in Section 3.2.

3.2 MODIFICATIONS TO THE LARGE BREAK LOCA ECCS EVALUATION MODELS

Section 3.1 provided a description of the BART Evaluation Model and the codes which are used as part of that EM plus a description of the REFBASH and LOCBART codes used in this proposed EM. Also included was a brief summary of the modifications previously incorporated into some of the existing Large Break codes for the modeling of a CE NSSS. This section will provide a description of the code and methodology changes beyond those described in Section 3.1. These modifications fall into three categories:

1) modifications made to the standard Westinghouse codes and methodology implemented since the publication of the October 1988 version of 10 CFR 50.46 which are not described in the references cited above (Section 3.2.1),

2) modifications to the codes and methods previously modified for application to a CE NSSS beyond those described above, or additional modifications required for the appropriate modeling of Fort Calhoun Unit 1 (Section 3.2.2).

3) modifications to those codes and methods not previously modified for application to a CE NSSS, specifically the REFBASH and LOCBART codes (Section 3.2.3).

3.2.1 MODIFICATIONS TO EXISTING WESTINGHOUSE MODELS

On October 17, 1988, the NRC implemented revised ECCS Evaluation Model reporting requirements through a rule change to 10 CFR 50.46. Under the new rule, changes to Evaluation Models would be required to be reported when 1) a single change would result in a change in calculated Peak Clad Temperature (PCT) of greater than 50^{o}F or, 2) when the sum of the absolute value of the change in PCT resulting from all (unreported) changes becomes greater than 50^{o}F or, 3) annually, independent of the magnitude of the effects or potential effects on calculated PCT. As a result of this rule change, the historical procedure for documentation of EM changes (revisions or addenda to topical reports) has been somewhat altered.

Since the implementation of the "new" 50.46, two annual reporting periods (1989 & 1990) have occurred. In 1989, Westinghouse issued a document(16) detailing the modifications made to the ECCS Evaluation Models through July 1, 1989. Only one item was identified for the 1981 Large Break LOCA ECCS Evaluation Model with BART for Westinghouse plants. This item involved modifications to the WREFLOOD code modeling of downcomer overfill for maximum safeguards assumptions. This modification has been implemented in the WREFLOOD version used in the BART for CE EM.

An additional modification identified in reference 16 addresses pressurizer modeling in the blowdown transient for two-loop plants. The potential for special modeling requirements for two-loop plants is associated with the unique two-loop blowdown phenomena. Two-loop plants show very little positive core flow in the early portion of the blowdown transient, unlike three- and four-loop plants which demonstrate positive

core flow in the first approximately 5 - 12 seconds with a relatively high flow rate. Due to the difference in the CE and Westinghouse designs, an examination of the blowdown behavior for the CE design examined in this report was undertaken. The Cd = 0.4, 0.6 and 0.8 minimum safeguards, loss of offsite power, 8.75 ft. peak power shape (limiting spectrum cases) were examined. The plots of the core flow during blowdown (Figures 3.2.1-1 through 3.2.1-3) indicate an extended period (the first 4.5 - 10 seconds) of positive core flow, reaching relatively high flow rates (>4000 lb/s). This type of behavior is more representative of a Westinghouse three-loop plant blowdown than the two-loop phenomena which resulted in the modeling modification. Therefore, this item does not affect Fort Calhoun, and modeling is based on the traditional Westinghouse modeling for this item.

For the 1990 reporting period, it was concluded that no reportable evaluation model changes were identified for the period from August 1989 to August 1990. Consequently, no formal transmittal detailing modifications during this time period was issued to the NRC.

In late June 1991, reports were issued to the various utilities which are currently licensed with Westinghouse ECCS Evaluation Models identifying changes to the various evaluation models implemented for the time period from August 1990 to May 1991. A copy of the attachment (to the utility letters) identifying these changes is contained in Appendix A. Four modifications are identified which affect the codes which constitute the BART for CE EM, and the methods associated with these codes. The first modification includes miscellaneous changes to the various large break EM codes to provide consistency with the fuel rod design code and to update the pellet/clad contact resistance model coding. These revisions have been incorporated into the code version used as the basis for the proposed EM.

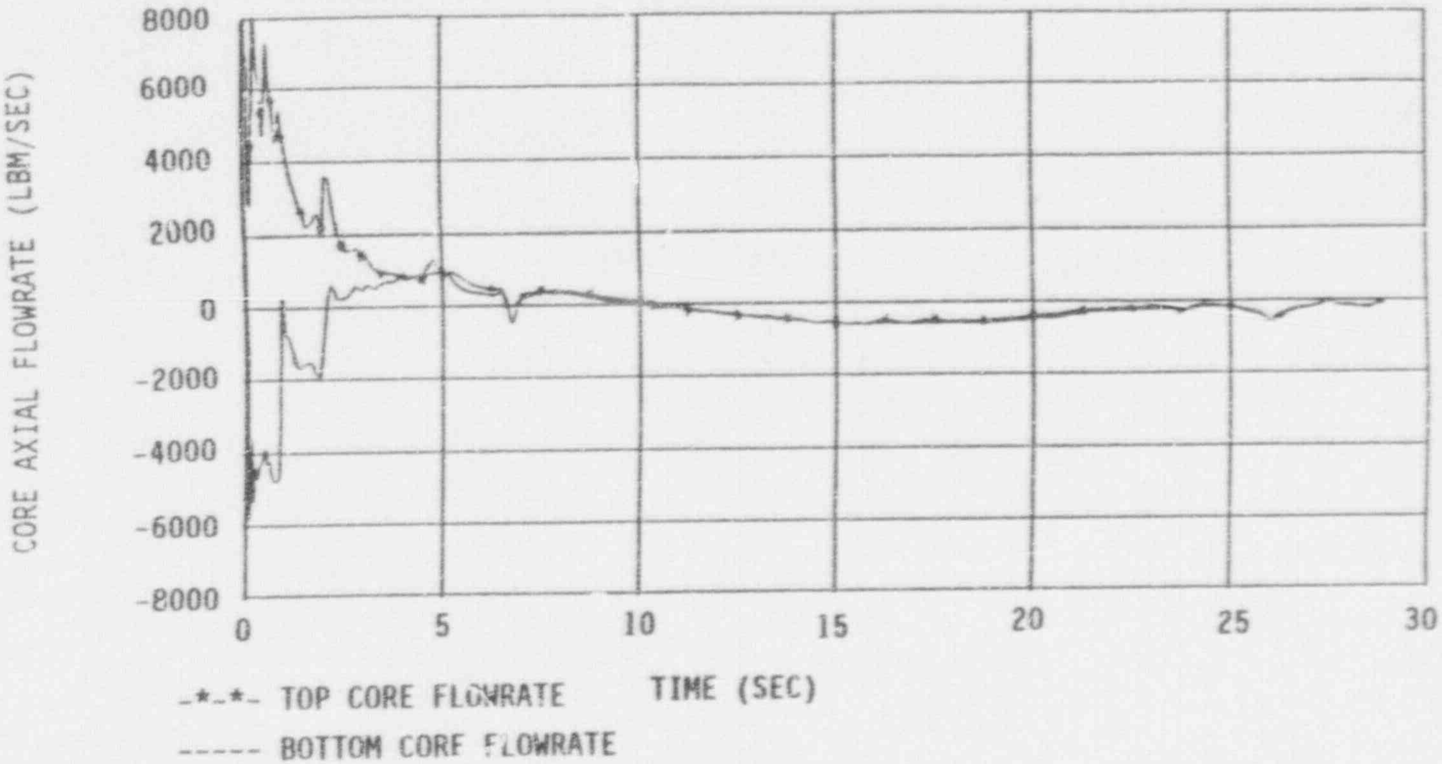

Figure 3.2.1-1

Core Flow during Blowdown, Co=0.4, MINSI, Loop, 8.75 ft peak power shape

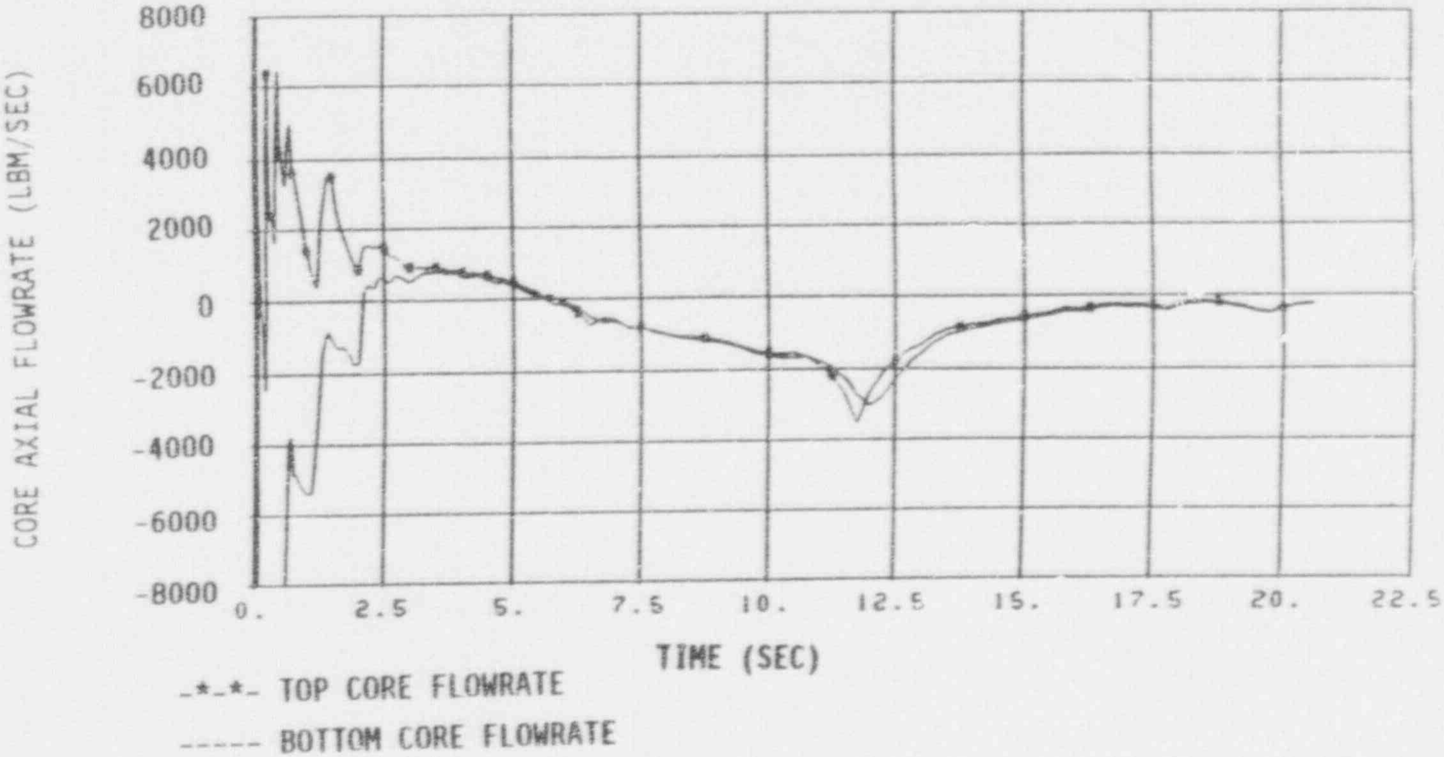

Figure 3.2.1-2

Core Flow during Blowdown, Co=0.6, MINSI, Loop, 8.75 ft peak power shape

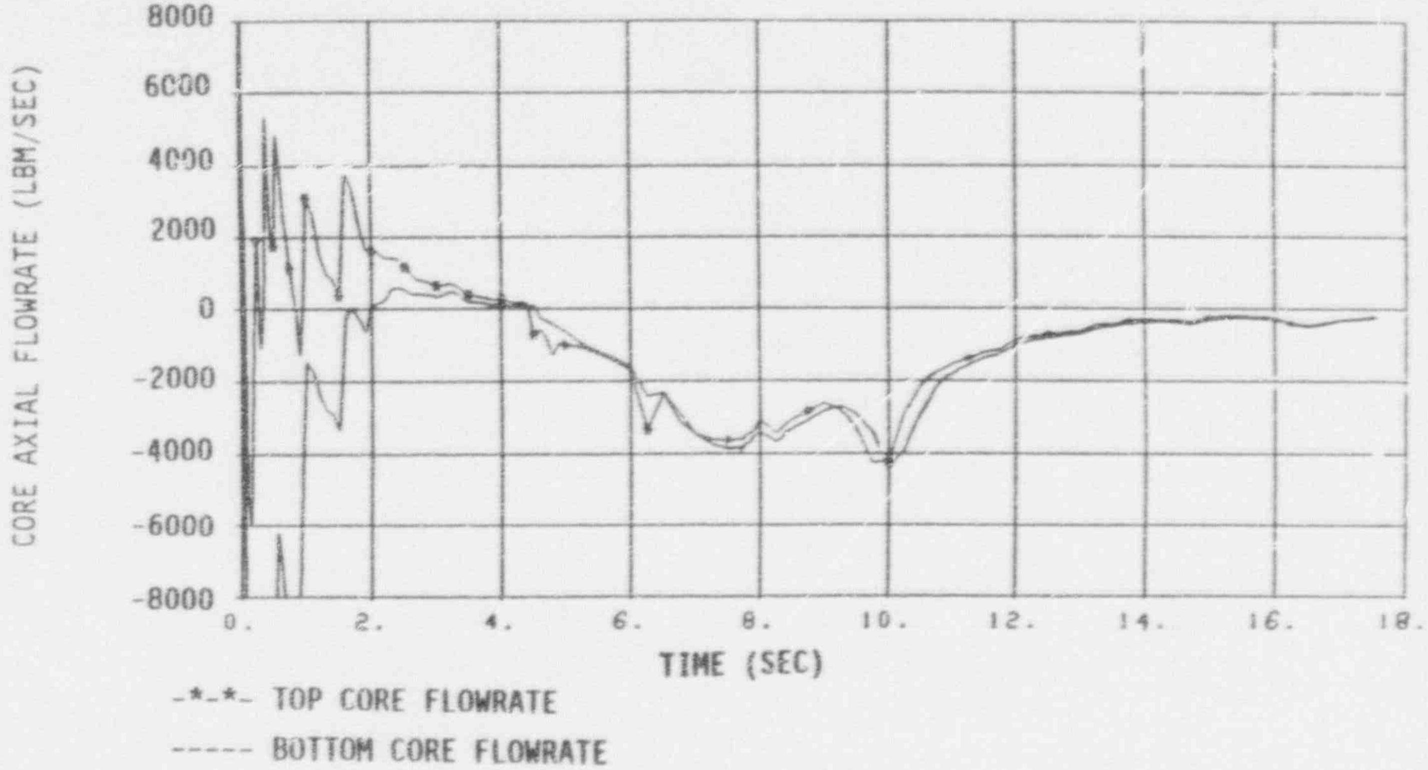

Figure 3.2.1-3

Core Flow during Blowdown, Co=0.8, MINSI, Loop, 8.75 ft peak power shape

The second change involves the methodology for identifying and analyzing the most severe expected axial power distribution for large break LOCA. For CE plant designs, core heights are typically less than the 12 foot active core height typical in a Westinghouse NSSS design. Also, technical specifications for CE plants generally specify an axially independent power peaking envelope in terms of a limiting peak linear heat rate (kw/ft), as opposed to the typical Westinghouse K(z) methodology which imposes peak power limitations at upper core elevations. For these reasons, the methodology described in Appendix A is not employed for the proposed BART for CE EM, in lieu of a sensitivity study examining various axial power distributions. A description of this sensitivity study is contained in Section 3.3.2 with further details provided in Appendix B.

The third modification relates to the burst and blockage assumptions associated with the large break LOCA analysis. The resolution of this issue concluded that no changes would be required to the model. However, in rare instances where bursting of the average rod in the hottest assembly has not occurred prior to the core flooding rate falling below 1 inch per second, the results of the ECCS analysis calculation will be supplemented by a permanent assessment of Peak Clad Temperature (PCT) margin. All of the large break calculations supporting this report have demonstrated burst for the hot assembly average rod at a time substantially before the flooding rate has fallen below 1 inch per second. Further, owing to the relatively high local power levels associated with CE plants, this behavior would generally be expected for the analysis of any CE plant. Still, for instances where the BART for CE EM predicts flooding rates less than 1 inch per second prior to hot assembly rod burst, an assessment of PCT margin will be ma[illegible]

The fourth modification relates to the methodology for modeling reduced Steam Generator (SG) tube flow area due to tube deformation resulting from combined LOCA and Safe Shutdown Earthquake (SSE) loads. NRC Regulatory Guide 1.121[17] describes a method for the evaluation of the deformation of Steam Generator tubes. For Westinghouse plants, this evaluation may require modeling of increased Steam Generator tube plugging levels to account for potential tube deformation resulting from combined LOCA and SSE loads.

3.2.2 MODIFICATIONS TO THE EXISTING WESTINGHOUSE MODELS FOR THE ANALYSIS OF A CE NSSS

A review of the previous modifications to the Large Break LOCA codes[8] for modeling a CE NSSS was undertaken to assess the applicability of these earlier "generic" CE modifications to the analysis of Fort Calhoun Unit 1, or to ensure that where the Westinghouse model was applied directly, such application is still appropriate.

3.2.2.1 SATAN-VI

The modeling of the auxiliary loop and the CEA shroud flow paths in SATAN-VI is appropriate for Fort Calhoun through use of the correct plant-specific input values. Based on evaluation of the fuel assembly and fuel rod designs, [

]a,c

[ ]a,c

] Figure 3.2.1-4 illustrates the SATAN-VI modeling for the BART for CE EM. [ ]a,c

The Fort Calhoun ECCS design, like many Westinghouse plants, features interaction between pumped safety injection and passive safety injection. A standard method of modeling such interaction in SATAN-VI has been developed for Westinghouse plants (see Appendix B of reference 2) by [ ]a,c This modeling was not employed in the earlier version of SATAN-VI modified for modeling of a CE NSSS(8) which used the Millstone Unit 2 plant as the base plant for supporting the model. For a plant design which does not include SIT/SI interaction, [ ]a,c

a,c

Figure 3.2.2.1 1981 + BART for CE NSSS SATAN Diagram

3.2.2.2 WREFLOOD

A diagram of the 1981 + BART for CE NSSS is provided in Figure 3.2.2-2.

For WREFLOOD, the unrecoverable pressure differential of 1.5 psi due to steam/SI interaction for a CE plant with a cold leg injection nozzle angle of 75° will be the basis for the plant specific Fort Calhoun input. Modeling of the auxiliary loop will not require modification to the methods, beyond those previously described[8], only use of the appropriate input for modeling the Fort Calhoun geometry. As with SATAN, the WREFLOOD code models the interaction between active and passive ECCS systems. Modeling of this feature was not included in earlier Westinghouse large break evaluation models used for analyzing a CE NSSS[8]. The same logic employed for Westinghouse plants featuring accumulator/SI interaction was incorporated into the WREFLOOD version used as part of the BART for CE EM to appropriately model SIT/SI interaction. This modeling was implemented via plant specific input to the WREFLOOD code.

The WREFLOOD code version used in conjunction with the BART EM features an enhanced modeling of the heat release from hot metal vessel structures during the refill portion of the transient. The updated metal heat modeling is acceptable by virtue of the NRC SER contained in the Proprietary version of reference 3. The model can be applied as part of the BART for CE EM without code modification, using plant-specific inputs. The code modifications previously included in WREFLOOD([8]) (see Section 3.1.2), have been incorporated into the BART-compatible version of the WREFLOOD code.

a,

a,c

Figure 3.2.2-2 1981 + BART for CE NSSS WREFLOOD Diagram

a,c

3.2.2.3 COCO

There are no changes to the COCO code required for modeling Fort Calhoun.

3.2.2.4 LOCTA-IV

Changes to the LOCTA-IV code are discussed in Section 3.2.3 as part of the LOCBART modifications.

3.2.3 MODIFICATIONS TO BART/REFBASH/LOCBART FOR MODELING A COMBUSTION ENGINEERING NSSS

Changes to the BART code are not explicitly required, however, some features of the BART modeling have not been previously considered for application to a CE NSSS, specifically, the metal heat modeling in the version of the WREFLOOD model used with BART and the application of the spacer grid model. The spacer grid model has been incorporated into the LOCBART code and is discussed in further detail later in this section.

The REFBASH code does not require any coding or methodology changes for application to a CE NSSS.

The LOCBART code uses the more mechanistic calculation of the BART model to determine heat transfer coefficients replacing the FLECHT correlation employed by LOCTA-IV. LOCBART is a synthesis of the existing LOCTA-IV and BART codes. The BART portion of the code has been discussed previously. The LOCBART code uses a mesh of [ ][a,c] nodes to allow appropriate interaction with BART. This noding is different than the standard [ ][a,c] node model used in the LOCTA-IV code, as previously applied to a CE NSSS (Millstone 2). No sensitivities for this change in noding were performed for the specific application of LOCBART to a CE NSSS. Studies of this type have previously been performed and reported to the NRC as part of the LOCBART development effort.

One of the previous modifications incorporated into LOCTA-IV for modeling a CE NSSS was the replacement of the standard Westinghouse crack and dish volumes with values more appropriate for the "CE-type fuel" being analyzed. Current versions of the Westinghouse fuel design code show that predicted crack and dish volumes for a "CE-type fuel" design vary only slightly (<5%) from the values predicted for similar Westinghouse fuel designs. An evaluation has concluded that differences in crack and dish volumes of this magnitude have a negligible effect on calculated Peak Clad Temperature. Therefore, the standard Westinghouse values are used in the LOCBART version which is part of the BART for CE EM.

Another previous modification for LOCTA-IV included specifically for modeling "CE-type fuel" was the fuel rod to guide tube radiation modeling[8]. This modeling provided the option to account for rod to guide tube heat transfer during the adiabatic heat-up of the refill portion of the transient. This term is generally very small for a typical Westinghouse fuel design, but would be somewhat large for "CE-type fuel" owing to the comparatively large guide tube design. While this option was available

for previous analyses for a CE NSSS, the option was not used. This option has not been incorporated into LOCBART for the BART for CE EM in which fuel rod to thimble radiation is conservatively neglected.

In addition, local heat transfer effects in LOCBART are modeled through the use of the spacer grid model[15]. This model permits a more accurate prediction of core heat transfer and allows calculation of such local heat transfer phenomena as grid rewet, resulting in vapor desuperheat and liquid film evaporation yielding higher steam flow and convection heat transfer.

3.2.4 IMPLEMENTATION OF MODELING FEATURES

Loop Layout:

The loop layout modifications are similar to those made in the previous Evaluation Model[8] to accommodate the addition of the auxiliary loop for both SATAN-VI and WREFLOOD. A copy of the noding diagrams for the modeling of a CE NSSS with SATAN-VI and WREFLOOD are provided in Figures 3.2.2-1 and 3.2.2-2, respectively. The additional loop/geometry inputs (relative elevations, reactor coolant piping diameters, etc.) are modeled through the use of appropriate plant-specific input.

Control Element Assembly (CEA) Design:

Elements 52 and 53 in SATAN-VI represent the CEA flow path modeled during the blowdown transient. Flow areas, hydraulic resistance, etc. for the elements are based on plant-specific input. As with the previous EM[8], the CEA flow paths are not specifically modeled in the WREFLOOD code, and are not important in terms of reflooding phenomena.

Upper Head Bypass/Upper Head Temperature:

The small amount of bypass flow passing from the upper downcomer to the Upper Head via the alignment keyway has been modeled in the SATAN-VI calculation to provide an accurate representation of the dynamics of the Upper Head draining. This modeling is implemented through code input. The Upper Head flows and thermal conditions are not specifically modeled in the WREFLOOD code, and are not important in terms of reflooding phenomena.

Lower Plenum Design

[ ] a,c

] Specifics of lower plenum geometry, other than free volume, are not explicitly modeled in WREFLOOD.

Core Shroud Bypass:

[ ] a,c

] Thus the SATAN-VI modeling of core shroud bypass is the same as that previously identified in Reference 8. [

] a,c the modeling of the core shroud (barrel/baffle) region is important in the reflood transient. Because this region refloods at very near the

same rate as the active core region, and because fluid entering this region is far removed from the hot assembly, this quantity of fluid is unavailable for core cooling. Therefore, the core shroud area is carefully modeled in the WREFLOOD code through plant-specific input.

Reactor Coolant Pump:

Specifics of the reactor coolant pump performance and coastdown characteristics are provided to SATAN-VI via plant-specific input. Appendix K[1] requires that the RCPs, "shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free." The locked rotor assumption during reflood for Large Break LOCA has always proven to provide the limiting Large Break LOCA PCT. The greater resistance to steam flow resulting from this assumption suppresses the core reflooding rates in all cases and produces the limiting result. The modeling of the RCPs in WREFLOOD will be based on the locked rotor assumption, and the appropriate plant-specific values for pump geometry and locked rotor flow resistance will be included in the reflood analysis via input to the WREFLOOD code.

Steam Generator Inlet Nozzle Angle:

The pressure drop from the hot leg to the SG inlet is provided to SATAN-VI via code-specific input. Details of the SG inlet nozzle have a negligible influence on blowdown characteristics beyond effects on pressure drop. For the Large Break WREFLOOD modeling, the SG acts as a heat source throughout the reflood transient, and all liquid entrained from the core is assumed to pass through the Steam Generator. Thus, modeling of flooding in the SG inlet nozzle, which is a function of the nozzle inclination, is not considered for analysis of the Large Break core reflooding transient.

Safety Injection/Accumulator Nozzle Angle:

The importante effect of safety injection and/or accumulator (SIT) injection angle is the small unrecoverable pressure drop associated with low velocity loop steam condensation, and this geometry is therefore not modeled in SATAN-VI. In the WREFLOOD code, the unrecoverable pressure differential due to steam/SI interaction for a CE plant with a cold leg injection nozzle angle of 60^0 or 75^0 will be included through the plant-specific input.

Safety Injection Tank Pressure:

The polytropic expansion of the nitrogen cover gas in the SIT during the reflood portion of the transient rapidly reduces injection pressure to below 100 psig. Due to the rapid gas expansion and pressure range during the SIT injection phase, the method of calculating the injection rate will not be dependent on the initial gas pressurization. Thus the SIT modeling in SATAN-VI and in WREFLOOD is applicable to SITs with initial cover gas pressures on the order of 200 psig.

Fuel Assembly Design:

The primary consideration of Fuel Assembly design in the hydraulic codes, SATAN-VI and WREFLOOD, is the core pressure drop. This parameter is accounted for via fuel-specific input for the "CE-type fuel" being analyzed. In the rod heat up calculations, fuel-specific parameters are also input. The large thimbles featured in the "CE-type fuel" design would provide a beneficial radiation sink during the adiabatic heat-up period of the refill transient. However, this effect has been conservatively neglected as part of the BART for CE methodology.

Fuel Rod/Pellet Stack Design:

Fuel compatibility considerations result in designs of Westinghouse fuel for non-Westinghouse plants which are similar in many aspects to those of the NSSS vendor fuel. Accordingly, the dimensions of these designs differ somewhat from those of Westinghouse fuel designed for Westinghouse plants. These dimensional differences have been considered in relation to the rod burst criteria applied in the 1981 + BART for CE NSSS Evaluation Model.

3.2.5 APPLICABILITY OF THE EVALUATION MODEL

Part of the confirmation of the applicability of the evaluation model is the verification that the model and/or application of the model conforms to the restrictions and requirements identified by the NRC as a condition for approval of the model. A review has been conducted of the documentation associated with the BART EM, which serves as the basis for the proposed 1981 + BART for CE EM, and related reports, plus the documentation of the application of the 1981 model components to a CE NSSS. From this review, eight pertinent restrictions/requirements were identified.

Following is a reiteration of the requirements and restrictions associated with the previously approved constituent codes of the proposed BART for CE EM. Each item contains a description of the proposed method for addressing the item. Also, although no formal results are reported herein for the application of this model, responses based on the current Fort Calhoun Unit 1 analysis are provided for each item as an example of the method of applicability confirmation.

1) Application of the BART model is restricted to the range of operation bounded by the data contained in Section 2-7 in the NRC SER for the BART model[2]. These conditions are:

Pressure (psia)	20 - 60
Initial Temperature (°F)	1100 - 1600
Initial Power (kw/ft)	0.45 - 1.2
Inlet Subcooling (°F)	20 - 140
Reflood Rate (in/sec)	0.6 - 1.5

Response: These parameters should be verified against the final licensing basis computer output information and internally documented.

Example: For Ft. Calhoun:

Pressure	~50 psia
Initial Temperature	~1550°F
Initial Power	~0.8 kw/ft
Inlet Subcooling	~80°F
Reflood Rate	$0.95 < V_{in} < 1.4$ (through PCT time)

2) BART nodes must be less than 6 inches in length[2].

Response: Verify BART nodes <6 inches long (0.5 ft.)

Example: For Fort Calhoun, length of all BART nodes

$0.25 \text{ ft.} \leq \Delta x \leq 0.5 \text{ ft.}$

0.5 ft. nodes are all located in lower core elevations, far from burst and/or PCT.

A single 0.6003 ft. node is located in the lower core, far from burst and/or PCT.

3) BART[3] applicable only to PWR using Westinghouse fuel with only cold leg injection.

Response: Confirm conditions.

Example: Fort Calhoun is a PWR with a 4/2 CE NSSS and only cold leg ECCS injection (no upper head, upper plenum or direct vessel injection). The analysis performed assumed this unit to be fueled with Westinghouse designed/manufactured "CE-type fuel".

4) BART model[3] requires that conditions of no single failure being worst case should be considered.

Response: See Section 3.3.2.3

Example: Maximum safeguards cases performed for Fort Calhoun. MAXSI results will be included in the plant-specific results summary report.

5) SER for 1981 EM[2] requires use of the skewed power shape option for less than 12 foot cores.

Response: Skewed power option will be used for these conditions.

Example: For Fort Calhoun, with a 128 inch active core length, the skewed power options were employed in SATAN-VI, WREFLOOD and LOCBART, even for chopped cosine power distribution studies.

6) 1981 EM SER[2] identifies modeling of SI/accumulator interaction.

Response: Employ this modeling to CE plants featuring this ECCS arrangement.

Example: Interaction of SIT/SI (common injection nozzle) has been considered in the SATAN-VI and WREFLOOD modeling for Fort Calhoun.

7) A burnup study was specifically requested for the application of the Westinghouse EM to a CE NSSS[2].

Response: See Section 3.3.2.5

Example: N/A.

8) Spacer grid model applicability restricted to the conditions identified in the SER for this model.

Response: Verify against analysis.

Example: These conditions are roughly equivalent to those highlighted in Section 2.3 of the BART SER. See item 1.

3.3 LARGE BREAK LOCA ANALYSIS

This section describes the application of the 1981 + BART for CE NSSS Evaluation Model. Where reference is made to actual analyses, said analyses have been based on the geometry and plant operating conditions representative of Fort Calhoun Unit 1. Ft. Calhoun is a two hot leg, four cold leg PWR featuring 133 assemblies, a 14 x 14 fuel array and a core thermal power of 1500 MW.

3.3.1 BREAK SPECTRUM ANALYSIS

Considerable experience in the performance of Large Break LOCA analyses for Westinghouse NSSS design has demonstrated that it is difficult to identify a "generically limiting" discharge coefficient. Limiting discharge coefficients for Large Break LOCA Analyses may change from Evaluation Model to Evaluation Model, even with few or no changes in plant- or fuel-specific inputs. Even with no change in Evaluation Model, large changes in tube plugging levels, plant geometry, or ECCS capability could result in a shift in the limiting discharge coefficient from analysis to analysis.

The previous Westinghouse Evaluation Model application to a CE NSSS, documented in reference 8 for Millstone Unit 2, showed a limiting discharge coefficient of $C_D = 0.4$. The Fort Calhoun Unit 1 Updated Safety Analysis Report [18] indicates a limiting discharge coefficient of $C_D = 0.6$ for an analysis performed with the Combustion Engineering Evaluation Model. A break spectrum analysis performed using the 1981 + BART for CE NSSS based on Fort Calhoun Unit 1 found the limiting discharge coefficient to be $C_D = 0.4$.

Due to the high degree of uncertainty associated with parametric variations from application to application (plant geometric differences, ECCS differences, etc.), no attempt will be made to isolate a "generically limiting" discharge coefficient associated with this model. Instead, the application of this model to a CE NSSS shall include the examination of a spectrum of discharge coefficients for the purpose of identifying which discharge coefficient results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of a complete spectrum of no less than three discharge coefficients.

3.3.2 SENSITIVITY STUDIES

This section discusses sensitivity studies which are required either on a plant-specific application basis or to support the proposed EM. Of the many sensitivity studies required, six have been selected for discussion in the body of this report, owing to their relative importance. Other sensitivity studies are discussed in Appendix C of this report.

3.3.2.1 BREAK LOCATION

Because of the large hydraulic resistance between the reactor core and the break location, breaks in the Reactor Coolant System cold legs are expected to provide limiting Large Break LOCA PCT results. Sensitivity studies[19] were performed for double-ended guillotine breaks with various discharge coefficients located in the cold leg, hot leg and pump suction leg as well as split breaks ranging in size from 1.0 $ft.^2$ to the

double-ended area of a cold leg. These studies clearly demonstrate that a double-ended cold leg guillotine (DECLG) break results in the most limiting calculated PCT results for a Westinghouse NSSS. Similar studies, performed by Combustion Engineering[20] for a CE NSSS, also confirm the DECLG break as being limiting for Large Break LOCA. Based on these conclusions, application of the 1981 + BART for CE NSSS Evaluation Model may be limited to the analysis of DECLG breaks.

3.3.2.2 POWER AVAILABILITY

Criterion 35 of 10 CFR 50, Appendix A[21] requires that suitable redundancy be designed into the ECCS to "assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure." The normal assumption for the analysis of a Westinghouse NSSS is that offsite power is lost coincident with the initiation of the Large Break LOCA. Sensitivity studies[19] using the Westinghouse Evaluation Model applied to a Westinghouse NSSS design have confirmed that this scenario results in the highest calculated Peak Clad Temperature.

Availability of Offsite Power allows continued operation of the Reactor Coolant Pumps which enhances initial blowdown cooling and permits an earlier actuation of pumped ECCS compared to the Loss of Offsite Power (LOOP) assumption. This, however, may result in an increase in the amount of SIT (accumulator) water calculated to bypass in accordance with the requirements of 10 CFR 50 Appendix K[1]. While the net result of competing effects resulting from the assumption of available offsite power (No LOOP) consistently result in a reduction in PCT for Westinghouse

plants, these results may not apply to the use of a Westinghouse EM for analysis of a CE NSSS. Previous studies[8] have demonstrated that power availability assumptions may affect calculated PCT results in a different manner for different discharge coefficients. Studies performed using the BART for CE EM modeling Fort Calhoun Unit 1 have also shown that a "generically limiting" power availability assumption cannot be identified for applying the Westinghouse EM to a CE NSSS.

Due to the high degree of uncertainty associated with the effects of the power availability assumption on calculated PCT for variations in plant conditions, blowdown characteristics, etc., no attempt will be made to identify a "generically limiting" power availability assumption for this model. Instead, the application of this model to a CE NSSS shall include the examination of LOOP and No LOOP for purposes of identifying the power availability condition which results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of both the LOOP and No LOOP assumptions.

3.3.2.3 <u>ECCS AVAILABILITY</u>

10 CFR 50, Appendix K[1] requires that, "the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place." However, Westinghouse has determined that for some non-burst node limited, four-loop Westinghouse NSSS plants, the assumption of no single failure may result in a higher calculated Large Break PCT when analyzed with the Westinghouse Evaluation Model. This "no single failure" assumption is

generally referred to as maximum safeguards (MAXSI). Available information on similar studies performed by Combustion Engineering[18] indicates that MAXSI may also be limiting for CE plants analyzed using the CE Evaluation Model.

The NRC SER for the Westinghouse BART EM[3] states, "When applying this model...the condition of no single failure as being the worst case should be considered." Since the BART EM is the basis for the 1981 + BART for CE NSSS EM, this restriction is also applicable to this model. Therefore, no attempt will be made to identify a "generically limiting" ECCS availability assumption (maximum versus minimum safeguards) for this model. Instead, the application of this model to a CE NSSS shall include the examination of MINSI and MAXSI for purposes of identifying the ECCS availability condition which results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of both the MINSI and MAXSI assumptions.

3.3.2.4 AXIAL POWER DISTRIBUTIONS

10 CFR 50, Appendix K[1] requires that, "A range of power distribution shapes...shall be studied, and the one selected should be that which results in the most severe calculated consequences..." Unlike Westinghouse plants which generally limit power peaking in the upper elevations of the core below the midplane value, CE plants are limited, in terms of power peaking, only by a peak linear heat rate and the fraction of integral power above the core midplane (Axial Shape Index). Because this varies from the typical Westinghouse practices, a sensitivity study

for power shapes was performed using the BART for CE EM and input based on Fort Calhoun Unit 1. While the specifics of this study are applicable only to Fort Calhoun Unit 1, the methods employed and the general trend of the study's results are tied to the Evaluation Model. Details of this study are contained in Appendix B.

3.3.2.5 FUEL BURNUP

Substantial experience has been gained in the examination of the effects of fuel burnup on large break LOCA results. Studies performed to support the Westinghouse generic position on fuel burnup[22] have consistently demonstrated a predictable trend in PCT as functions of fuel burnup. These studies, including those performed with the BART EM, demonstrated that the limiting condition in terms of Large Break PCT occurs at the time of maximum densification (near the beginning of life), when fuel stored energy is at a maximum. Although "CE-type fuel" varies in some aspects from the fuel employed in a Westinghouse NSSS, the consistency of burnup behavior across a wide variety of fuel types and plant conditions would indicate a similar trend for the "CE type fuel". The previous Westinghouse Evaluation Model for a CE NSSS[8] was used to perform a study of fuel burnup effects on Large Break LOCA PCT. The results of that study, contained in reference 2, confirm the expected conclusion that beginning of life represents the limiting fuel burnup condition for Large Break LOCA. Based on this information, no burnup study has been performed for the 1981 + BART for CE NSSS EM. Plant specific applications will assume fuel conditions at the time of maximum densification unless special considerations warrant otherwise.

2.3.2.6 INTEGRATED FUEL BURNABLE ABSORBERS

At the time of the most recent previous analysis of a CE plant using the Westinghouse model, Integrated Fuel Burnable Absorbers (IFBA) were not available as a fuel design option. IFBA fuel differs from non-IFBA fuel in that the boron coating covering the fuel pellets in the IFBA rod results in a much higher fission gas release with burnup than the non-IFBA rod. This ultimately yields a higher rod internal pressure late in fuel life for the IFBA rod. [ ][a,c] Burnup studies for IFBA fuel have demonstrated trends very similar to burnup studies for non-IFBA fuel, showing beginning of life conditions to be limiting over higher burnup levels.

In general, IFBA fuel results in lower Large Break Peak Clad Temperatures than non-IFBA fuel when both are examined at the time of maximum densification. At very low burnups, however, the [ ][a,c] Although this burst occurs later than the predicted burst for BOL non-IFBA, which would normally be expected to result in a reduced PCT, the high temperature associated with low pressure burst drives the local zirc/water reaction. Under certain conditions, the temperature increases due to increased local exothermic zirc/water reaction may overshadow the benefit of later burst. In this event, IFBA will result in a higher PCT than non-IFBA at beginning of life conditions.

There are a wide variety of IFBA designs covering a range of coating geometry, absorption capability, initial backfill, etc. A sensitivity study has been performed for IFBA using the 1981 + BART for CE NSSS EM based on Fort Calhoun Unit 1 plant modeling. The IFBA fuel data used in this study was taken at [ ]^{a,c} and was representative of the IFBA design to first be introduced into the Fort Calhoun Unit 1 core. Results of this study show that non-IFBA at time of maximum densification yields a slightly higher PCT (Δ-PCT <20^{o}F) than the IFBA fuel. However, given the variability in IFBA fuel designs, no attempt will be made to identify a "generically limiting" IFBA or non-IFBA fuel design. Instead, the application of this model to a CE NSSS shall include the examination of IFBA and non-IFBA for purposes of identifying which results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of both the IFBA and non-IFBA fuel.

4.0 SMALL BREAK LOCA

The Small Break LOCA ECCS Evaluation Model for the analysis of a CE NSSS is based upon a version of the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code[26,27] as modified for application to a CE NSSS[28] (NOTRUMP EM for CE). The NOTRUMP EM for CE will be used to demonstrate compliance with the requirements for the analysis of Small Break Loss-Of-Coolant Accidents as defined in 10 CFR 50.46[1], and employs the required models and assumptions identified in 10 CFR 50, Appendix K[1]. Specifically, the NOTRUMP EM was developed to address the requirements of NUREG-0737, Item II.K.3.30. The NOTRUMP EM for CE, as modified in reference 28 for the application to the CE NSSS, demonstrated compliance with 10 CFR 50.46 and NUREG-0737, Item II.K.3.31 for Millstone Unit 2. This section describes the minor modifications to the previously approved NOTRUMP EM for CE, which have been implemented to more appropriately model the CE NSSS and bring the EM up to 1991 technology and standards. Additionally, details of the application of the revised NOTRUMP EM for CE to the plant geometry and operating conditions of Fort Calhoun Unit 1 are provided.

The NOTRUMP EM consists of the NOTRUMP and LOCTA-IV codes. The NOTRUMP EM has previously been modified for application to a CE NSSS to perform the licensing basis Small Break LOCA calculation for Millstone Unit 2[28]. Changes were made to the NOTRUMP EM to incorporate the necessary features appropriate for the analysis of a Small Break LOCA for Westinghouse fuel in a CE NSSS, and to address additional Small Break licensing issues not addressed in the Westinghouse version of the NOTRUMP EM. NUREG-0611 addressed NRC concerns regarding Westinghouse NSSS design and the associated Small Break LOCA analysis methods, for which responses were provided in references 26 and 27. The NRC SER on the NOTRUMP code and EM

(contained in the Proprietary versions of references 26 and 27, respectively) evaluated the responses as being adequate. NUREG-0635 addresses concerns similar to those identified in NUREG-0611, but is specific to the CE NSSS design and the associated Small Break LOCA analysis methods. The additional concerns of NUREG-0635, not identified in NUREG-0611 are addressed specifically for the NOTRUMP EM applied to a CE NSSS. The responses to those additional concerns, plus the NRC SER identifying the adequacy of the response are contained in reference 28.

Section 4.1 briefly describes the individual codes of the NOTRUMP EM including, where applicable, the changes to code input methodology which previously have been included in the modeling of a CE NSSS. Section 4.2 contains a description of additional modifications to the NOTRUMP EM since the last application of the NOTRUMP EM and methodology for the CE NSSS with Westinghouse fuel. This section also contains modifications in codes and methods for the application of the model to Fort Calhoun Unit 1. Section 4.3 provides results of the application of the revised NOTRUMP EM for CE to Fort Calhoun Unit 1 and anticipated sensitivity studies which are required to support the Evaluation Model and plant-specific fuel design.

4.1 SMALL BREAK LOCA ECCS EVALUATION MODEL AS APPLIED TO A CE NSSS

4.1.1 NOTRUMP

The NOTRUMP[26,27] code is a nodal transient general network code used to model the system thermal-hydraulic transient throughout the Small Break LOCA. The previous application of the NOTRUMP code to a CE NSSS is documented in reference 28. The design differences between the Westinghouse and CE NSSS which could potentially impact the small break LOCA transient results were evaluated in this report. The necessary modifications to the NOTRUMP code input methodology to properly account for the differences between a typical Westinghouse NSSS and a CE NSSS were also described. These design differences are addressed in section 4.2 to verify continued applicability for Fort Calhoun Unit 1.

4.1.2 LOCTA-IV

The LOCTA-IV[13] code is used to perform the rod heat-up calculation for the Small Break LOCA analysis. The LOCTA-IV code is the same code identified in reference 13, but has been modified slightly to allow consistency with the NOTRUMP output format. The version of LOCTA-IV used in the NOTRUMP EM has also been modified to enhance the modeling of steam cooling and radiation in the rod heat-up calculations. Discussions of these modifications are contained in reference 27. Design differences between the Westinghouse and CE NSSS fuel which may impact the LOCTA-IV calculation methodology were examined in reference 28. The modifications to this code and methodology which were necessary to appropriately model the plant differences were also identified. These differences are also addressed in section 4.2 to verify continued applicability for Fort Calhoun Unit 1.

4.2 MODIFICATIONS TO THE SMALL BREAK LOCA ECCS EVALUATION MODELS

This section provides a description of the code and methodology modifications to the NOTRUMP EM for CE which update the existing model as described in reference 28. These modifications fall into two categories:

1) Modifications made to the standard Westinghouse NOTRUMP codes and methodology implemented since the publication of the October 1988 version of 10 CFR 50.46 which are not described in the references cited above for the NOTRUMP EM (Section 4.2.1).

2) Additional modifications required for the appropriate modeling of Fort Calhoun Unit 1 which are beyond those described in reference 28 (Section 4.2.2).

4.2.1 MODIFICATIONS TO EXISTING WESTINGHOUSE MODELS

As described in section 3.2.1, on October 17, 1988, the NRC implemented revised ECCS Evaluation Model reporting requirements through a rule change to 10 CFR 50.46. Since the implementation of the "new" 50.46 regulation, two annual reporting periods (1989 & 1900) have occurred. In 1989, Westinghouse issued a document(16) detailing the modifications made to the ECCS Evaluation Models through July 1, 1989.

The modifications to the NOTRUMP code identified in this document are equally applicable to the NOTRUMP EM for CE. Therefore, these modifications were contained in the revised NOTRUMP EM for CE used for the Fort Calhoun Unit 1 analysis. The modifications to the NOTRUMP small break LOCA ECCS Evaluation Model and the NOTRUMP EM for CE are summarized below:

A. A modification was made to preclude changing the region designation (upper, lower) for a node in a stack which does not contain the mixture-vapor interface. The purpose of the modification was to enhance tracking of the mixture-vapor interface in a stacked series of fluid nodes and to preclude a node in a stack, which does not contain the mixture-vapor interface, from changing the region designation.

B. Typographical errors in the equations which calculate the heat transfer rate derivatives for subcooled, saturated, and superheated natural convection conditions for the upper region of interior fluid nodes were corrected.

C. Typographical errors in equations which calculate the derivatives of the natural convection mode of heat transfer in the subroutine HEAT were corrected.

D. A typographical error was corrected in an equation which calculates the internal energy for nodes associated with the reactor coolant pump model when the associated reactor coolant pump flow links are calculated to be in critical flow.

E. A modification was made to properly call some doubly dimensioned variables in subroutine INIT and TRANSIT.

F. A modification was made to prevent code aborts due to different treatments of the precision of numbers between FORTRAN compilers resulting from implementation of a new FORTRAN compiler.

G. An error in the implementation of equation 5-33 of reference 26 was corrected. Equation 5-33 describes the calculation of the flow link friction parameter C_k for single phase flow in a non-critical flow link k.

H. A test was added in the rod-to-steam radiation heat transfer coefficient calculation to preclude the use for the correlation when the wall-to-steam temperature differential dropped below the useful range of the correlation.

I. A modification was made to correct an error in implementing equations L-28, L-52, and L-29, L-53 of reference 26. The two pairs of equations respectively describe the partial derivatives of F^k with respect to pressure and specific enthalpy. F^k is an interpolation parameter that is defined by equations L-27 and L-51 of reference 26.

J. An error in the printed value for the break flow link specific volume in the Moody break flow model was identified. Only the printed output value of the specific volume was incorrect.

K. An error in the fuel pellet to cladding contact pressure (used in the fuel rod gap conductance model) was identified. The erroneous coding was called only when no gap between the fuel pellet and the cladding exists.

L. An error in the calculation of the saturated or subcooled boiling critical heat fluxes using the McBeth correlation was identified.

Several modifications to the LOCTA-IV computer code of the Small Break LOCA ECCS Evaluation Model were identified in reference 16. These also apply equally to the revised NOTRUMP EM for CE and were contained in the analysis of Fort Calhoun Unit 1. The modifications identified were:

M. A test was added in the rod-to-steam radiation heat transfer coefficient calculation to preclude the use of the correlation when the wall-to-steam temperature differential dropped below the useful range of the correlation.

N. An update was performed to allow the use of fuel rod performance data from the revised Westinghouse model.

O. Modifications supporting general upgrade of the computer program were implemented. These included removal of unused or redundant coding, better coding organization to increase efficiency of calculations, and user friendliness improvements.

P. Two modifications improving the consistency between the Westinghouse fuel rod performance data (PAD) and the Small Break LOCTA-IV fuel rod models were implemented. The form of the equation for the density of Uranium-Dioxide in the specific heat correlation, which modeled three dimensional expansion was corrected to account for only two-dimensional thermal expansion due to the way the fuel rod is modeled. Also, an error in the equation for the pellet/clad contact pressure was corrected.

Reference 16 concluded that the combined effect of the above modifications to the NOTRUMP and LOCTA-IV computer codes would result in a net reduction in the Peak Cladding Temperature.

For the 1990 reporting period, it was concluded that no reportable evaluation model changes occurred for the period from August 1989 to August 1990. Consequently, no formal transmittal detailing modifications during this time period was issued to the NRC.

In late June, 1991, reports were issued to utilities which are currently licensed with Westinghouse ECCS Evaluation Models identifying changes to the various evaluation models implemented for the time period from August 1990 to May 1991. A copy of the changes is contained in Appendix A. Four modifications were identified which affect the codes which constitute the NOTRUMP EM. The following modifications also apply to the NOTRUMP EM for CE and therefore were included in the code version used for the Fort Calhoun Unit 1 analysis.

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

2) 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 13. However, 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.

3) Unexpected variability of sensitivity study results with the NOTRUMP EM following the 1989 model changes 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 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.

4) 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 into containment. Westinghouse concluded following analyses calculations that the current practice of neglecting safety injection flow into the broken loop in combination with a conservative condensation model is conservative and in compliance with the regulatory requirements. Therefore, no model change was concluded to be necessary.

4.2.2 MODIFICATIONS TO EXISTING WESTINGHOUSE SMALL BREAK MODELS FOR ANALYSIS OF A COMBUSTION ENGINEERING NSSS

The modifications incorporated into the codes and input methodology which were determined to be necessary in reference 28 to appropriately model the CE NSSS were reviewed to assess their applicability to the analysis of Fort Calhoun Unit 1. Also, the additional modifications required for the appropriate modeling of Fort Calhoun Unit 1 which are beyond those described in reference 28 were identified. The results of this review are summarized below along with of a description of code and input methodology modifications which were required to perform the Fort Calhoun Unit 1 analysis.

1) Loop Layout Representation

As described in reference 28, the system modeling was modified to accurately represent the CE two hot leg, four cold leg design by [ ]a,c

thus code modifications were not required to enable modeling of the CE piping geometry. [

]a,c

The limiting single failure for the Small Break LOCA analysis for Fort Calhoun Unit 1 is the loss of a diesel generator which results in Low Pressure Safety Injection into only one of the two intact loop cold legs. In order to more appropriately model this safety injection configuration, [

]a,c

scheme also allowed explicit modeling the Fort Calhoun Unit 1 Trip-2 Leave-2 Reactor Coolant Pump trip Emergency Operating Procedure following an abnormal condition with no loss of offsite power to be explicitly modeled. A noding diagram for this revised CE NSSS noding is presented in Figure 4-1. Also, in order to explicitly model the different combinations of High and Low Pressure Safety Injection into each of the Fort Calhoun Unit 1 cold legs, several changes to the NOTRUMP $USINPUT variables described in section 4-2-1 of reference 27 were necessary. The variables NSIFL, NSIFN, NSIP, PSI, SIDLAY, SIMULT, and WSI were replaced with variables NSIHPFL and NSILPFL,

Figure 4-1

Fort Calhoun NOTRUMP Noding Diagram

NSIHPFN and NSILPFN, NSILP and NSIHP, LPPSI and HPPSI, HSIDLAY and LSIDLAY, HSIMULT and LSIMULT, and HPWSI and LPWSI, respectively, to allow explicit modeling of the High Pressure and Low Pressure Safety Injection pumps.

2) Large Diameter Hot Leg

In reference 28, verification was provided that the correlations and modeling techniques of the NOTRUMP code would be adequate and applicable for the larger diameter CE hot leg piping. Fort Calhoun Unit 1 has a 32" I.D. hot leg which is larger than the standard Westinghouse design of 29.5" I.D. NOTRUMP accounts for the larger diameter hot leg via code input and requires no code or input methodology modification.

3) Hot - Leg SG Inlet Plenum Flooding

In reference 28, the difference in Steam Generator inlet nozzle inclination was addressed and appropriate sensitivity studies were performed. Based on comparisons of the flooding constants in the flooding correlation for the two designs (40^{0} from vertical for a Westinghouse plant versus 50^{0} for a typical CE plant), plus the results of the sensitivity studies performed, it was concluded that appropriate modeling of the CE Steam Generator inlet plenum would be obtained [

] a,c

4) Loop Seal Behavior

The complicated hydraulic behavior of the loop seals during a Small Break LOCA were discussed in reference 28. Based on this discussion, it was verified that the noding scheme chosen for the analysis, in combination with geometric input representative of a CE NSSS (loop seal design) would adequately model the CE loop seal behavior, and no code or input methodology modifications were required. Since the revised noding scheme for the Fort Calhoun Unit 1 analysis contains no changes in the loop seal noding or code input methodology from that developed in reference 28, the loop seal behavior will continue to be adequately modeled.

5) Control Element Assembly Design

The modeling of the flow paths from the upper head to the upper plenum and geometry of the CEA shrouds were examined in reference 28. Sensitivity studies were performed to verify that the noding scheme chosen for the analysis, in combination with geometric input representative of a CE NSSS would conservatively model the upper head venting during a Small Break LOCA transient. [

]a,c was determined to be acceptable for Small Break LOCA modeling of the CE NSSS. The Fort Calhoun Unit 1 CEA design is similar to that assumed in the sensitivities mentioned above, therefore the modeling of the flowpath from the upper head to the upper plenum [ ]a,c

6) Upper Head Bypass Flow and Temperature

The adequacy of the modeling of the upper head bypass flow in the chosen noding scheme was addressed in reference 28 via comparison of the hydraulic resistances of the bypass flow paths in a CE and Westinghouse design. Based on the small amount of upper head cooling flow for a CE NSSS compared to a Westinghouse plant, it was decided [

]a,c Since Fort Calhoun Unit 1 has a small amount of bypass flow, the upper head temperature and bypass flow assumptions were modeled consistent with the CE NSSS methodology described above.

7) Safety Injection Angle

The angle between the cold leg and the safety injection nozzle differ between Westinghouse and CE NSSS designs. The value of the injection angle is important in calculating the [

]a,c

8) Safety Injection Flow Geometry

In reference 28, an evaluation of the behavior of the subcooled Safety Injection (SI) at the location of the SI line/cold leg interface was performed and the SI behavior for a CE NSSS design was determined to be similar to that in a Westinghouse NSSS design. Therefore, no modification to the NOTRUMP code was required to appropriately model the CE SI behavior. The Fort Calhoun Unit 1 Safety Injection flow geometry is similar to that used in the evaluation and therefore the SI behavior will be appropriately modeled.

9) Polytropic Expansion Coefficient

Examination of data from accumulator (SIT) blowdown tests were used to determine a polytropic expansion coefficient for CE SITs with cover gas pressures of approximately 200 or 600 psig in reference 28. Based on this examination, it was determined that the polytropic expansion coefficient which is used for Westinghouse analyses would be applicable to either of the CE SIT conditions. Therefore no code or input methodology changes were required. Since Fort Calhoun Unit 1 has SITs with a cover gas pressure of approximately 255 psia, the use of the Westinghouse assumed value for the polytropic expansion coefficient will remain applicable.

10) Reactor Coolant Pumps

Based on a comparison of the Reactor Coolant Pump (RCP) design, and comparison with applicable test data, it was concluded that the
[ ]a,c

reference 28. Thus, it was concluded by using [

]a,c

11) Fuel Assembly Design

Despite differences in the fuel assembly hydraulic diameter, the steam cooling correlations used in the NOTRUMP and LOCTA-IV code were determined to be applicable over a range encompassing both the CE and Westinghouse fuel rod design in reference 28. The steam cooling heat transfer relations used in the NOTRUMP EM are a function of the fuel assembly hydraulic diameter and transient specific data. Westinghouse fuel hydraulic diameters range from [ ]a,c The Fort Calhoun Unit 1 fuel design has a hydraulic diameter of [ ]a,c which is bounded by the Westinghouse designs. [

]a,c

12) Rod Burst Calculation

Based on rod burst test data, it was determined in reference 28 that clad rupture is directly related to clad stress and therefore the thickness to diameter ratio of the clad design. In determining burst temperature as a function of pressure for fuel being loaded into a CE NSSS, the pressure differential required for burst at a given temperature was [

]a,c

to the CE NSSS in reference 28. [

]a,c Since the Fort Calhoun Fuel design value is larger than Westinghouse fuel values, it will be more resistent to clad burst. Clad burst is a PCT penalty for small break LOCA analyses. Therefore, the use of the Westinghouse pressure differential required for burst at a given temperature in the LOCTA-IV code will result in a conservative early burst time calculation for Fort Calhoun Unit 1 fuel. Thus, the LOCTA-IV code was not modified, as before, for the CE NOTRUMP EM, and the standard Westinghouse LOCTA-IV version was used for the Fort Calhoun Unit 1 analysis.

13) Fuel Crack and Dish Volumes

Fuel rod internal crack and dish volumes, normally calculated internal to the LOCTA-IV code, were replaced with values representative of the Fort Calhoun Unit 1 fuel design, rather than the standard Westinghouse values. This provides more appropriate modeling of the Fort Calhoun Unit 1 fuel design.

4.3 SMALL BREAK LOCA ANALYSIS

This section describes the break spectrum analysis and sensitivity studies which were performed using the revised NOTRUMP EM for CE. It also provides justification for the continued applicability of sensitivities performed previously.

4.3.1 BREAK SPECTRUM ANALYSIS

Considerable experience in the performance of Small Break LOCA analyses for Westinghouse NSSS design has demonstrated that it is difficult to identify a "generically limiting" break size. Limiting break sizes for Small Break LOCA Analyses may change from Evaluation Model to Evaluation Model, even with few or no changes in plant or fuel specific inputs. Even with no change in Evaluation Model, large changes in plant geometry, or ECCS capability could result in a shift in the limiting break size.

The previous application of the NOTRUMP EM to a CE NSSS, documented in reference 28 for Millstone Unit 2, showed a limiting break size of 4 inches in equivalent diameter. The Fort Calhoun Unit 1 Updated Safety Analysis Report[18] indicates a limiting break size of approximately 3.7 inches in equivalent diameter for an analysis performed with the Combustion Engineering Evaluation Model. A break spectrum analysis performed using the NOTRUMP EM for CE based on Fort Calhoun Unit 1 found the limiting break to be 3 inches in equivalent diameter.

Due to the high degree of uncertainty associated with parametric variations from application to application (plant geometric differences, ECCS differences, etc.), the application of the NOTRUMP EM for CE should include the examination of a spectrum of break sizes for the purpose of identifying which break size results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of a complete spectrum of no less than three break sizes.

4.3.2 BREAK LOCATION

Historically, Westinghouse has shown the limiting break location for the NOTRUMP EM as applied to a Westinghouse NSSS to be the cold leg break. Sensitivity studies for the NOTRUMP EM as applied to the CE NSSS were performed for a four inch break located in the cold leg, hot leg, and pump suction leg in reference 28. These studies clearly demonstrate that a cold leg break results in the most limiting calculated PCT results for a CE NSSS. Similar studies, performed by Combustion Engineering[20] for a CE NSSS, also confirm the cold leg break as being limiting for Small Break LOCA. Based on these studies, application of the NOTRUMP EM for CE may be limited to the analysis of cold leg breaks.

4.3.3 POWER AVAILABILITY

For all standard NOTRUMP analyses applied to a Westinghouse NSSS, it is assumed that offsite power is lost coincident with reactor trip during the Small Break LOCA. This assumption is based on sensitivities for a Westinghouse NSSS design which have historically shown that the Loss Of Offsite Power (LOOP) scenario results in the highest calculated Peak Clad Temperature. In order to confirm that the LOOP assumption provides a limiting PCT for the NOTRUMP EM for CE, a sensitivity was performed assuming No Loss Of Offsite Power (No LOOP). The sensitivity included the Reactor Coolant Pump trip strategy developed by CE and currently contained in the Fort Calhoun Unit 1 Emergency Operating Procedures (EOPs). Under this strategy, on an abnormal condition (specifically, reactor trip), the operators are instructed to first manually trip two diametrically opposed pumps. The remaining two pumps are tripped when either 1) the RCS pressure falls below the Safety Injection Setpoint, or 2) use of the EOPs identifies the initiating event as a Small Break LOCA. This strategy is referred to as "Trip 2, Leave 2". Results of the sensitivity study confirmed that the LOOP assumption continues to be limiting for the NOTRUMP EM for CE.

4.3.4 FUEL BURNUP

Studies performed to support the Westinghouse generic position on fuel burnup have consistently demonstrated a predictable trend in PCT as a function of fuel burnup. These studies, have demonstrated that the limiting condition in terms of Small Break PCT occurs at the time of maximum densification (near the beginning of life), when fuel stored energy is at a maximum. However. recent changes to the clad strain and burst models in the LOCTA-IV code may cause later time in life burnups to be limiting, if burst is calculated to occur. In order to confirm the continued applicability of the previous studies to the NOTRUMP EM for CE, a sensitivity was performed at several burnups. This sensitivity verified that beginning-of-life burnup resulted in the limiting PCT for the Fort Calhoun Unit 1 fuel. However, based on the fact that clad burst was not calculated to occur for the limiting PCT case for Fort Calhoun Unit 1, it cannot be concluded that beginning-of-life will be limiting for all analyses performed with the NOTRUMP EM for CE. Therefore, the application of the NOTRUMP EM for CE shall include the examination of fuel burnup for purposes of identifying which time in life results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of the limiting time in life.

4.3.5 INTEGRATED FUEL BURNABLE ABSORBERS

At the time of the development of the NOTRUMP EM for CE in reference 28, Integrated Fuel Burnable Absorbers (IFBA) were not available as a fuel design option. IFBA fuel differs from non-IFBA fuel in that the boron coating covering the fuel pellets in the IFBA rod results in a much

higher fission gas release with burnup than the non-IFBA rod. This ultimately yields a higher rod internal pressure late in fuel life for the IFBA rod. As a structural design consideration, IFBA rods are generally pressurized to a lower critical backfill to partially offset the high EOL pressures. Burnup studies for IFBA fuel have demonstrated trends very similar to burnup studies for non-IFBA fuel, showing beginning of life conditions to be limiting over higher burnup levels.

A sensitivity was performed to determine the limiting fuel type. The IFBA fuel data used in this study was taken at 0 MWD/MTU and was representative of the IFBA design to first be introduced into the Fort Calhoun Unit 1 core. Results of this study showed that non-IFBA and IFBA fuel resulted in the same PCT since rod burst was not calculated to occur. However, given the variability in IFBA fuel designs, no attempt will be made to identify a "generically limiting" IFBA or non-IFBA fuel design. Instead, the application of the NOTRUMP EM for CE shall include the examination of IFBA and non-IFBA for purposes of identifying which results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of both the IFBA and non-IFBA fuel.

4.3.6 ADDITIONAL SENSITIVITIES

Because of the similarity of the Westinghouse and CE NSSS designs, and because NOTRUMP can be applied in unaltered form, the model sensitivity studies performed and documented in reference 27 for Westinghouse NSSS designs remain applicable for the CE design. Therefore, sensitivity studies performed in reference 27 to demonstrate Appendix K compliance do not need to be repeated for the NOTRUMP EM for CE.

results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of both the IFBA and non-IFBA fuel.

4.3.6 ADDITIONAL SENSITIVITIES

Because of the similarity of the Westinghouse and CE NSSS designs, and because NOTRUMP can be applied in unaltered form, the model sensitivity studies performed and documented in reference 27 for Westinghouse NSSS designs remain applicable for the CE design. Therefore, sensitivity studies performed in reference 27 to demonstrate Appendix K compliance do not need to be repeated for the NOTRUMP EM for CE.

5.0 10 CFR 50.46 ACCEPTANCE CRITERIA

Following is a brief description of the methods for demonstrating compliance with the Acceptance Criteria of 10 CFR 50.46[1] for the proposed 1981 + BART for CE NSSS Evaluation and the NOTRUMP EM for CE.

5.1 PEAK CLADDING TEMPERATURE

Criterion: The calculated maximum fuel element temperature shall not exceed $2200^{\circ}F$.

Peak Clad Temperature is a direct output of the LOCBART code in the BART for CE EM and the LOCTA-IV code in the NOTRUMP EM for CE, and is a reported analysis result.

5.2 MAXIMUM CLADDING OXIDATION

Criterion: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

The LOCBART and LOCTA-IV codes calculate local cladding oxidation throughout the Large Break and Small Break transients respectively. The greatest local cladding oxidation (usually at the hot rod burst location) is a reported analysis result.

5.3 MAXIMUM HYDROGEN GENERATION

Criterion: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated by all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

LOCBART has recently been updated to provide a conservative estimate of hot assembly wide average zirc/water reaction replacing generic values previously reported for core wide zirc/water. A conservative estimate can also be calculated based on LOCTA-IV output. The conservative estimate is compared to the regulatory limit for verification. Upon verification, a value of < 1.0% is reported.

5.4 COOLABLE GEOMETRY

Criterion: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

An accurate geometric representation of the core is modeled in the BART for CE EM and the NOTRUMP EM for CE. This modeling will include predicted alterations in core geometry resulting from a design basis LOCA (hydraulic forces) and/or seismic event as required as a condition of the plant license. It is noted that the BART for CE EM and NOTRUMP EM for CE do not calculate changes in core geometry (other than rod burst), but use information supplied by the NSSS vendor or utility as input to accurately model the expected core geometry. Given an accurate modeling of core geometry, calculation of a PCT not greater than 2200^{o}F confirms that geometry's amenability to cooling.

5.5 LONG TERM COOLING

Criterion: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The Westinghouse commitment for assurance of Long Term Cooling is identified in reference 25. The reactor core is recovered by borated ECCS water having a high enough boron concentration to maintain core shutdown. Following switchover to the recirculation phase, the mixing of the various sources of borated and unborated water (i.e. RCS, SIRWT, SITs, and other sources dumped directly to the containment sump or into the broken RCS) must provide a sufficiently large boron concentration to maintain the reactor core in a subcritical state. Note that this evaluation of long term effects is separate from the short term calculation performed with the BART for CE EM and NOTRUMP EM for CE. Because the ability to maintain the reactor subcritical on boron only is largely related to the specifics of the cycle energy requirements, this evaluation is performed on a cycle-by-cycle basis, independent of the PCT, cladding oxidation, and hydrogen generation results.

Another facet of ensuring long term cooling capability post-LOCA is to preclude the precipitation of boron from the highly borated injection water. Plating out of boron on the fuel rod surface can deteriorate heat transfer, yielding a clad heat-up transient based on the remaining decay heat. To prevent stagnation in the core region for a cold leg break with cold leg SI flow traversing the downcomer tangentially and traveling directly out the break, the recirculation phase is switched from cold leg injection to hot leg injection. The maximum allowable time for this switchover is a function of boron concentration reaching the core from the sump, core initial power, etc. Again, this evaluation of long term effects is separate from the short term calculation performed with the BART for CE EM and the NOTRUMP EM for CE.

Results of the hot leg switchover time calculation generally are included in the plant Emergency Operating Procedures, while results for the minimum boron for subcriticality post-LOCA evaluation is reported merely as post-LOCA $k_{eff} < 1.0$. Because these analyses/evaluations are verified for each fuel cycle and are separate from the short term analysis which is calculated using the BART for CE EM and the NOTRUMP EM for CE, no additional discussion will be devoted to these analyses.

6.0 CONCLUSIONS

This report, in conjunction with the references identified herein, has presented the descriptions, applications, limitations and licensing history for the Westinghouse ECCS Evaluation Models for the Combustion Engineering NSSS.

The Large Break model, the 1981 + BART for CE NSSS Evaluation Model, has been developed by modifying existing Large Break ECCS codes to incorporate features of the CE design. This new code sequence, including the modifications for CE NSSS design, constitutes an EM in compliance with the requirements of 10 CFR 50, Appendix K. This model is acceptable for use in Final Safety Analysis Report (FSAR) Large Break LOCA analyses to demonstrate acceptability of the ECCS for the CE NSSS.

The Small Break model, the NOTRUMP EM for CE NSSS has been previously developed and reviewed for this application. Reference information relating to this model has been updated to reflect current technology. This EM satisfies the requirements of 10 CFR 50, Appendix K, as well as NUREG-0737, Item II.K.3.30. This model is acceptable for use in FSAR Small Break LOCA analyses to demonstrate acceptability of the ECCS for the CE NSSS.

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APPENDIX A

EVALUATION MODEL REVISIONS AUGUST 1990 - MAY 1991

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. 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.

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.

Affected Evaluation Models:

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 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 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.

<u>Change Description:</u>

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.

<u>Affected Evaluation Models:</u>

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

<u>Status:</u>

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.

<u>Change Description</u>

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 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 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 Analy s 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-10266-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.

APPENDIX B

AXIAL POWER DISTRIBUTION STUDY

This Appendix describes the methods, assumptions, and results of an axial power distribution sensitivity study performed using the 1981 + BART for CE NSSS evaluation model. The study is based on the plant geometry, plant operating conditions and technical specification limits for Fort Calhoun Unit 1.

Combustion Engineering plants restrict power distribution by limiting the peak linear heat rate (maximum local PLHR) and the fraction of integral power in the upper portion of the core (Axial Shape Index)[3]. Sensitivity studies have consistently demonstrated that axial power distributions peaked below the core midplane are non-limiting, so this study focuses on center-peaked and top-skewed axial distributions. Unlike Westinghouse methodology, CE plants define a peak linear heat rate independent of core axial location. The LOCA peak linear heat rate used for this study was 15.5 kw/ft.

Four axial distributions were examined. Table B-1 provides a tabulation of the pertinent data related to each power shape - peak power elevation (Z_p), Normalized Z_p, Axial Shape Index, and peak shape profile (Figures).

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3 The Axial Shape Index, ASI is the power level detected by the lower nuclear instrument detectors (L) less the power level detected by the upper nuclear instrument detectors (U) divided by the sum of these power levels:

$$Y = \frac{L-U}{L+U}$$

Table B-1

Shape No.	Zp (ft)	Normalized Zp	- ASI (%)	Profile
1	5.333	0.5	0%	Figure B-1
2	6.33	0.593	9.2%	Figure B-2
3	7.33	0.687	17.2%	Figure B-3
4	8.75	0.820	16%	Figure B-4

Shape #1 is a chopped cosine peaked at the core midplane. Chopped cosine power distributions had traditionally been found to be limiting in earlier Evaluation Models. "Cosine" power distributions, which maximize power near the peak, are generally conservative with respect to actual power shapes having the same peak linear heat rate. Shape #2 is a skewed cosine, peaked approximately one foot above the core midplane. Shape #3 represents the skewed cosine peak with Z_p chosen so as to challenge the ASI limit. While the axial shape index limit for Fort Calhoun is -16%, the ASI for this shape is even more limiting to allow examination of the "cosine" in one foot increments of Z_p. Shape #4 is representative of a more realistic power distribution and is based on sorting of a large database of potential shapes expected over the core lifetime. From among the various shapes, it was chosen because it represents the shape with the highest Z_p which closely matches the Peak Linear Heat Rate and ASI. Minor adjustments were made to the database shape to ensure that the analyzed shape would conservatively challenge PLHR, ASI and integral power, simultaneously. Each of these shapes has the same integral power.

The results of this study are actually a composite of the results of four separate studies or "rounds" of calculations. Each round represents an examination of two or more of the power shapes identified above under the same set of input assumptions. The assumptions, system modeling, code inputs, etc., may vary from round to round. Round 1, for example, represents an early study in which the power distributions were only

altered in the rod heatup calc (LOCBART) and were all based on the same thermal hydraulic transient. Round 4 represents results based on Fort Calhoun Unit 1 input values and model the power distribution in SATAN-VI, WREFLOOD and LOCBART. Rounds 2 and 3 are intermediate calculations performed in the process of verifying the code inputs and EM models. It is noted that these cases are not presented to identify actual Fort Calhoun Unit 1 Large Break results, but to demonstrate the effects of changes in power distribution on Large Break PCT. Results for all four rounds are presented in Figure B-5.

While each round may provide a slightly different sensitivity (slope) for changes in Z_p, the trend of changes in PCT with changes in Z_p is consistent from round to round. From Figure B-5, it is clear that PCT increases with the elevation of the peak linear heat rate. Shape #4 represents the upper limit of Z_p at which the 15.5 Kw/ft Peak Linear Heat Rate can occur under the constraints of operational and Technical Specification limits, and this shape represents the limiting axial power distribution for Fort Calhoun Unit 1.

For the application of the 1981 + BART for CE NSSS EM, this methodology will be employed for determination of the limiting axial power distribution. No attempt will be made to identify a "generically limiting" power distribution assumption for this model. Instead, the application of this model to a CE NSSS shall include the examination of power distribution effects, using the type of methodology identified in this Appendix, for purposes of identifying the power distribution which results in the highest calculated Peak Clad Temperature for that application. Any exception, such as performance of sensitivity studies for small perturbations in plant parameters, or other reasonable exceptions, will include a justification for excluding examination of power distributions. Determination of a generically applicable power distribution, or refinement of this methodology, may be considered at a later time based on knowledge gained from future applications of the model.

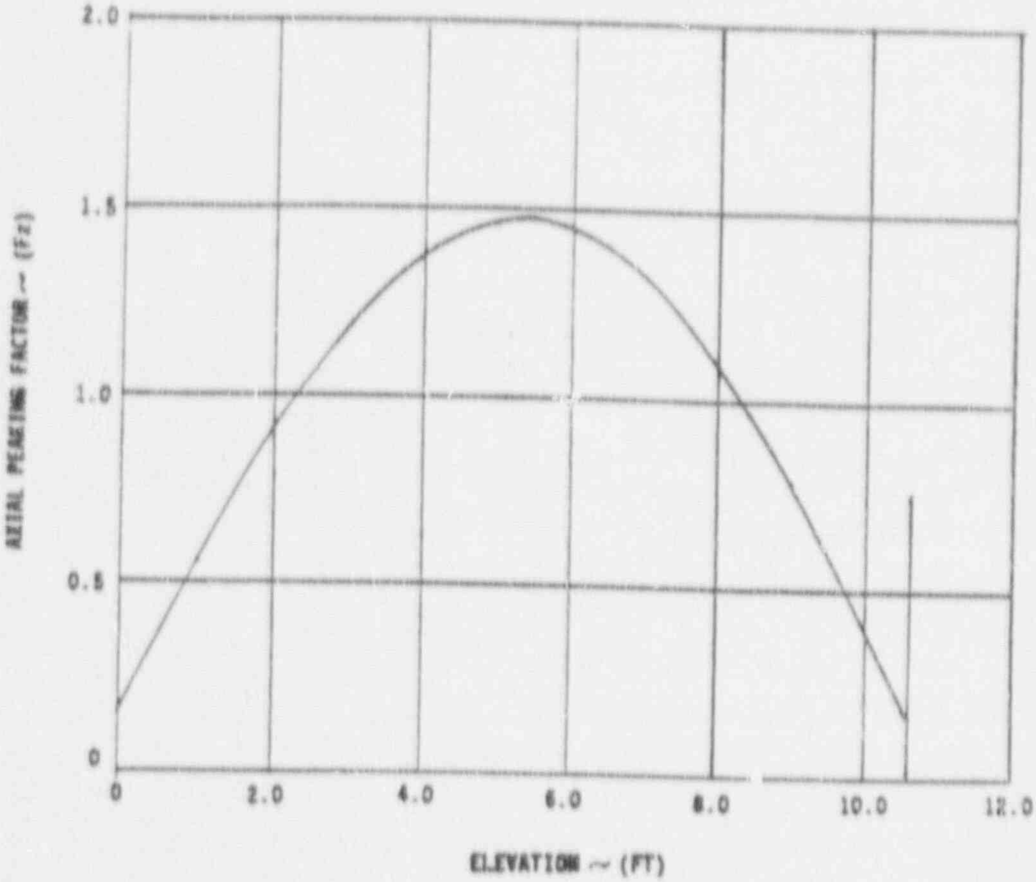


Figure B-1.

Cosine Power Shape, Zp = 5.334 ft., ASI = 0%

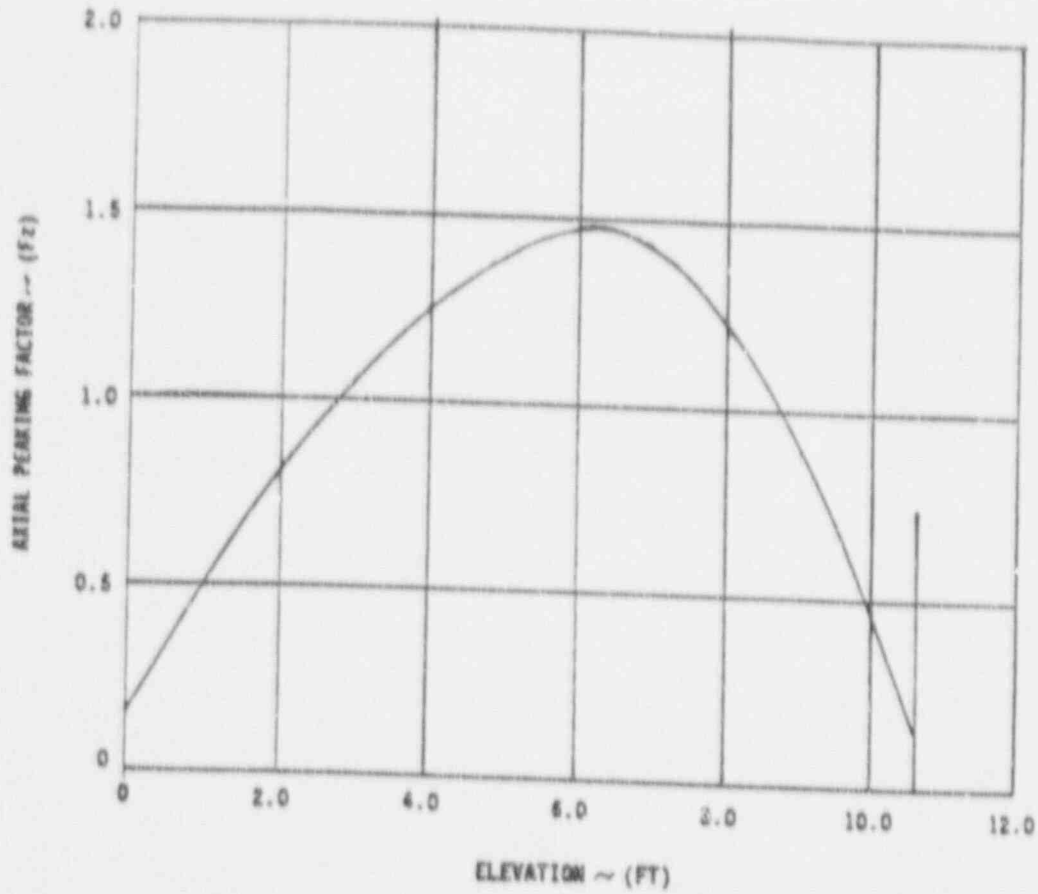


Figure B-2.

Skewed Cosine Power Shape, Zp = 6.33 ft., ASI = 9.2%

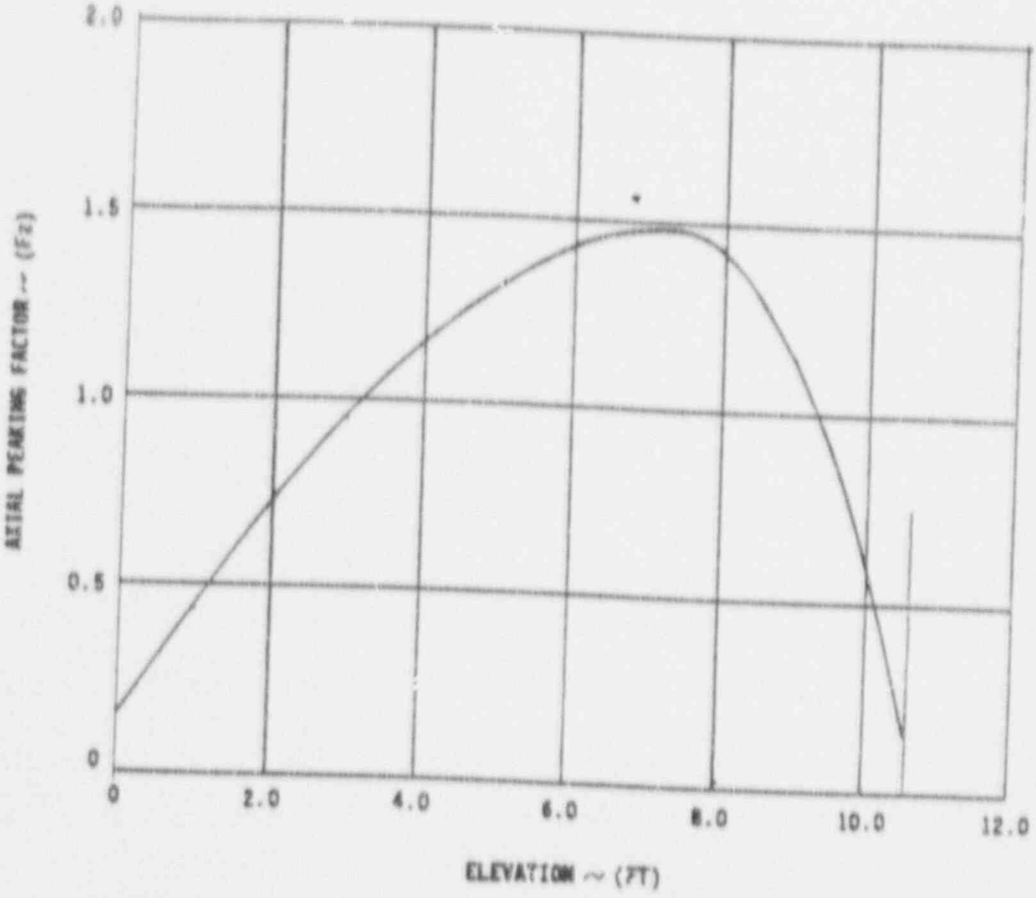


Figure B-3.

Skewed Cosine Power Shape, Zp = 7.33 ft., ASI = 17.2%

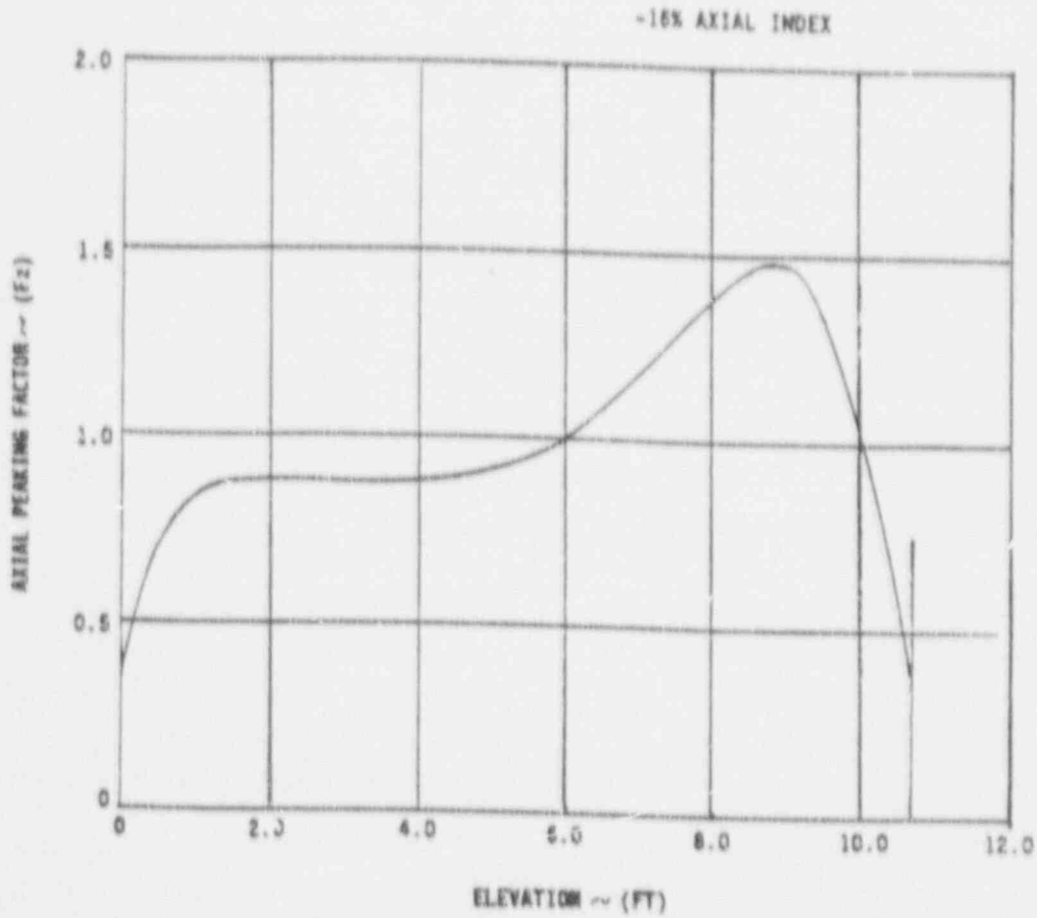
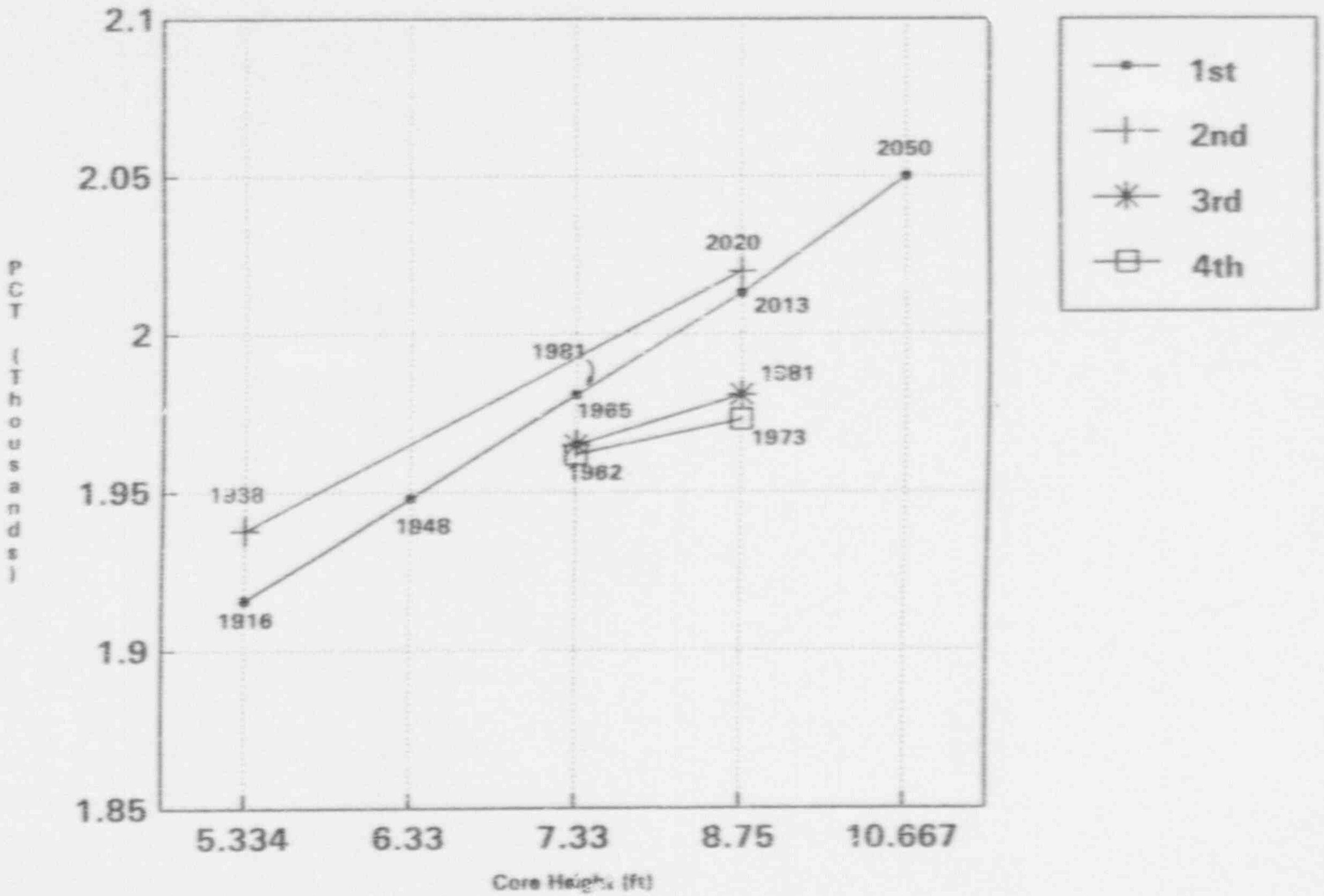


Figure B-4.

Top Skewed Power Shape, Zp = 8.75 ft., ASI = 16%

Figure B-5
Power Shape Study Zpeak vs PCT



APPENDIX C

OTHER SENSITIVITIES

Appendix D of reference 8 identifies 23 parameters, input assumptions, modeling features, etc., for which sensitivity studies are either performed, justified or otherwise explained for the application of an earlier Westinghouse EM to a CE NSSS. These items will also be addressed for the 1981 + BART for CE NSSS EM and are as follows:

1. Power Shapes
2. Core Shroud Region Nodalization
3. Burnup
4. Cold RCS Volume in SATAN
5. Sensitivity to Number of Nodes
6. Artificial Pressure in the SATAN Momentum Equation
7. Effect of Critical Flowchecks on SATAN Momentum with Momentum Flux
8. Cross Flow Effects
9. Reactor Coolant System Thick Metal Heat Release in SATAN
10. Distribution Parameter (C_o) Study
11. Steam Generator Reverse Heat Transfer Effects
12. Accumulator Injection in the Broken Loop During Blowdown
13. LOCTA Pellet Noding
14. LOCTA Time Step Studies
15. Sensitivity of Peak Cladding Temperature to Steam Cooling
16. WREFLOOD Sensitivity to Reactor Coolant Pump Conditions
17. Core Heat Flow Rate during REFLOOD
18. Containment Pressure
19. Single Failure Criterion
20. Nitrogen Gas Injection Impact
21. RCP Assumptions
22. Break Location
23. Flow Blockage Considerations

Item 1 is discussed in Appendix B and in Section 3.3.2.4.

Item 2 was studied explicitly in Appendix D of reference 8. Since this is a modeling change in SATAN-VI and the core/core shroud modeling in SATAN-VI is not changed from that reported in reference 8, the existing sensitivity study applies. This study concluded that a lumped nodalization provided conservative Large Break results. The lumped nodalization is standard in SATAN-VI and will be employed in the BART for CE EM.

Item 3 is discussed in Section 3.3.2.5.

Item 4 is a SATAN-VI sensitivity study [22]. None of the changes to SATAN-VI would be expected to significantly affect the sensitivity results.

Item 5 is also related to SATAN-VI. [

]a,c

Items 4 through 12 are SATAN-VI sensitivity studies [22]. None of the changes to SATAN-VI would be expected to significantly affect the sensitivity results.

Item 12 relates to the noding scheme in the rod heat-up code. The current standard noding in LOCBART consists of [ ]^{a,c} axial nodes to allow flexibility in modeling and ensure that all BART nodes are ≤ 6" in length. The [ ]^{a,c} node axial mesh used in the Westinghouse version of LOCBART[6] will also be used in the 1981 BART for CE NSSS EM.

Item 14 is related to time step size in the rod heat-up calculation. Definition of the iteration scheme used in the approved Westinghouse version of LOCBART is provided in reference 6.

For Item 15, the BART code[3] which is included in LOCBART has sufficient capability to accurately model heat transfer from fuel rod to fluid at all anticipated steam flow rates.

Existing sensitivity studies have consistently demonstrated that the higher resistance resulting from the locked rotor assumption in WREFLOOD yields a higher PCT than the pumps running assumption. The item 16 sensitivity conclusion remains valid.

Core heat flow rate is a relatively insensitive parameter in WREFLOOD [8]. No modifications have been introduced to WREFLOOD to alter this sensitivity so the sensitivity for Item 17 remains valid.

Changes to the containment code, COCO, have not been implemented since this sensitivity was addressed in reference 8. The item 18 sensitivity remains valid.

Item 19, single failure criteria has been previously addressed for Westinghouse plants [22,23], concluding that the limiting single failure for Large Break LOCA is loss of a single Low Pressure Safety Injection pump. Combustion Engineering has confirmed this assumption in similar studies for a CE NSSS design [24]. Consideration of no single failure being the worst case is discussed in Section 3.3.2.3.

The impact of accumulator (SIT) gas injection, Item 20, has been addressed previously [22]. In the proposed model, as in previous Westinghouse Evaluation Models, the effects of nitrogen gas on reflooding will conservatively be neglected.

Item 21 is discussed in Section 3.3.2.2

Item 22 is discussed in section 3.3.2.1.

Item 23 has been discussed previously in this topical. [

] a,c