Westinghouse Electric Company LLC

2000 Day Hill Road
Windsor, CT 06095
USA

9 April, 2001
LD-2001-0024

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

SUBJECT: SUBMITTAL OF "-A" ACCEPTED VERSION OF CENPD-132, SUPPLEMENT 4-P (ENCLOSURE 1 CONTAINS PROPRIETARY INFORMATION)

Reference(s):
1) Letter, S. A. Richards (NRC) to P. W. Richardson (Westinghouse), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model', (TAC No. MA5660)", December 15, 2000
2) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", August 2000

On December 15, 2000 (Reference) the Nuclear Regulatory Commission (NRC) issued its Safety Evaluation Report (SER) for CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (Reference 2). In accordance with the NRC's request, Westinghouse herewith is submitting the "-A" Accepted version of the subject topical report. Enclosures 1 and 2 provide six (6) proprietary and three (3) non-proprietary copies, respectively, for your use.

Westinghouse Electric Company LLC has determined that CENPD-132, Supplement 4-P-A (Enclosure 1) contains information that is proprietary in nature. Consequently, it is requested that the topical report be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that these copies be appropriately safeguarded. The reasons for the classification of this information as proprietary are delineated in the affidavit provided in Enclosure 3.

T008, 1/3

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
Westinghouse Electric Company LLC

Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Enclosure(s): As stated

xc: J. S. Cushing (w/o enclosure, NRC)

WESTINGHOUSE ELECTRIC COMPANY LLC

CENPD-132, SUPPLEMENT 4-P-A
CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

April 2001

WESTINGHOUSE ELECTRIC COMPANY LLC
PROPRIETARY INFORMATION

WESTINGHOUSE ELECTRIC COMPANY LLC

CENPD-132, SUPPLEMENT 4-NP-A
CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

WESTINGHOUSE ELECTRIC COMPANY LLC

CENPD-132, SUPPLEMENT 4-P-A
CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

PROPRIETARY AFFIDAVIT

AFFIDAVIT PURSUANT TO 10 CFR 2.790

I, Philip W. Richardson, depose and say that I am the Licensing Project Manager, Windsor Nuclear Licensing, of Westinghouse Electric Company LLC (WEC), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", March 2001

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by WEC in designating information as a trade secret, privileged or as confidential commercial or financial information. Pursuant to the provisions of 10 CFR 2.790(b)(4) of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure, is owned and has been held in confidence by WEC. It consists of the methodology for the evaluation of LOCA pursuant to 10 CFR 50, Appendix K, comparisons to experimental data for model verification and comparison to the previously approved methodology.
2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to WEC.
3. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public. WEC has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of WEC because:
   a. A similar product is manufactured and sold by major pressurized water reactor competitors of WEC.
   b. Development of this information by WEC required hundreds of thousands of dollars and thousands of man-hours of effort. A competitor would have to undergo similar expense in generating equivalent information.
   c. In order to acquire such information, a competitor would also require considerable time and inconvenience to develop methodology for the evaluation of LOCA pursuant to 10 CFR 50, Appendix K, comparisons to experimental data for model verification and comparison to the previously approved methodology.
   d. The information consists of methodology for the evaluation of LOCA pursuant to 10 CFR 50, Appendix K, comparisons to experimental data for model verification and comparison to the previously approved methodology, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.
   e. In pricing WEC's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of WEC's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.
   f. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on WEC 's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Sworn to before me
This 9 day of April, 2001

Laurie J. White
Notary Public
My commission expires: 8/31/04

CE NUCLEAR POWER LLC

CENPD-132
SUPPLEMENT 4-NP-A

CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

MARCH 2001

CE NUCLEAR POWER LLC

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CE Nuclear Power LLC
2000 Day Hill Road
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CE NUCLEAR POWER LLC

CENPD-132
SUPPLEMENT 4-NP-A

CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

MARCH 2001

CE NUCLEAR POWER LLC

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 9, 2001

Mr. Philip W. Richardson, Manager
Windsor Nuclear Licensing
Westinghouse Electric Company
CE Nuclear Power LLC
2000 Day Hill Road
Windsor, CT 06095

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT CENPD-132, SUPPLEMENT 4-P, REVISION 1, "CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL" NON-PROPRIETARY (TAC NO. MA5660)

Dear Mr. Richardson:

On December 15, 2000, the NRC staff issued proprietary and non-proprietary versions of the safety evaluation (SE) on the subject topical report which was submitted by CE Nuclear Power LLC (CENP). The SE requested that CENP review the non-proprietary version to determine if it contained proprietary information and that we would delay placing the non-proprietary SE in the public document room for a period of ten (10) working days to provide you with the opportunity to comment on the proprietary aspects only. By letter dated December 19, 2000, you stated that the non-proprietary version of the SE did contain some proprietary information and requested that the staff remove the proprietary information before releasing it to the public.

The staff has reviewed the information that you identify as proprietary and agrees that it is proprietary and should not have been included in the non-proprietary version of the SE. The information identified as proprietary has been removed and the non-proprietary version is being reissued and is enclosed. The December 15, 2000, non-proprietary version will not be released to the public. We apologize for any inconvenience this may have caused you.

The subject topical report describes the modifications made to the existing methods for CE Nuclear Power's large break loss-of-coolant (LOCA) accident evaluation model, which is described in CENPD-132, Supplement 3-A, and has been approved by NRC for licensing applications.

The staff has found that CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" is acceptable for referencing in licensing applications for CE designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation. The safety evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure

that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In accordance with procedures established in NUREG-0390, the NRC requests that CE Nuclear Power publish an accepted non-proprietary (-NP) version, within 3 months of receipt of this letter. The non-proprietary version shall incorporate (1) this letter and the enclosed safety evaluation between the title page and the abstract, and (2) an "-A" (designating "accepted") following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, CE Nuclear Power and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision of their respective documentation.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 692

Enclosure: Safety Evaluation (Non-proprietary)

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TOPICAL REPORT CENPD-132, SUPPLEMENT 4-P, REVISION 1

"CALCULATIVE METHODS FOR THE CE NUCLEAR POWER

LARGE BREAK LOCA EVALUATION MODEL"

PROJECT NO. 692

1.0 INTRODUCTION

By letter dated April 30, 1999 (Reference 1), ABB Combustion Engineering Nuclear Power (ABB CENP) submitted for staff review Topical Report CENPD-132, Supplement 4, "Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model." As a result of the initial review of the topical report, the NRC requested additional information, as well as modifications and additions to the topical report to both correct and clarify the technical documentation (Reference 2). Subsequently, Westinghouse Electric Corporation acquired ABB CENP and the company name was changed to CE Nuclear Power LLC (CENP). Therefore, by letters of August 30, 2000 (Reference 3) and September 25, 2000 (Reference 4), respectively, CENP submitted the proprietary and non-proprietary versions of CENPD-132, Supplement 4, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (Reference 5).

CENPD-132, Supplement 4-P, Revision 1, presents modifications to CENP's 1985 evaluation model (EM) (Reference 6), which has been approved by NRC, for analysis of large break loss-of-coolant accidents (LBLOCA). The currently approved EM and the new modified EM through Supplement 4 are abbreviated "1985 EM" and "1999 EM," respectively.

The 1999 EM modifications are organized into three (3) basic categories. The first category of modifications involves changes of the 1985 EM analysis process within the currently NRC-approved methodologies. The second category, the replacement of the Dougall-Rohsenow film boiling heat transfer correlation, is an EM change for compliance of the requirement of the 1988 revision to Appendix K to 10 CFR Part 50. The third category of modifications, called 1999 EM improvements, are model changes designed to reduce unnecessary conservatism in the following models:

- hot assembly fuel rod internal gas pressure
- steam venting reflood thermal-hydraulics
- steam/water interaction during safety injection tank nitrogen discharge
- reflood heat transfer
- hot rod steam cooling heat transfer

The staff review of the 1999 EM modifications is described in the following sections.

2.0 EVALUATION

2.1 Process Changes Within Currently Approved 1985 EM Methodology

Section 2.1 of the topical report (TR) describes the process changes to the LBLOCA emergency core cooling system (ECCS) performance analysis methodologies that remain consistent with the currently approved 1985 EM. These process changes are (1) implementation of an automated/integrated code system, (2) explicit calculation of the NUREG-0630 cladding swelling and rupture model (Reference 7), and (3) consistent modeling of spray and spillage into containment.

2.1.1 Automated/Integrated Code System

The 1985 EM for LBLOCA analysis consists of several computer codes for various aspects of calculations regarding an LBLOCA transient:

- the FATES3B code for fuel performance time-in-life initial conditions calculation,
- the CEFLASH-4A code for blowdown thermal-hydraulics calculation,
- the COMPERC-II code for calculations of refill/reflood thermal hydraulics, containment response, and core reflood heat transfer,
- the HTCOF, FRELAPC, and HPUNCH codes for FLECHT reflood heat transfer coefficient calculation,
- the COMZIRC code for core-wide zircaloy cladding oxidation calculation,
- the PARCH and HCROSS codes for hot assembly blockage and steam cooling calculations, and
- the STRIKIN-II code for hot rod heatup analysis.

These codes were executed one at a time, and required manual transfer of data between codes with or without hand calculation manipulation. In addition, certain user discretionary conservatism was introduced by the analyst during the transfer of interface data from one code to another by deliberately selecting values to conservatively bias the data transfer.

The 1999 EM for LBLOCA calculation combines the majority of the 1985 EM analysis process into an automated/integrated code system (AICS). This allows for an LBLOCA transient case to be executed from start to finish without analyst intervention. The automatic transfer of interface data produces more consistent and accurate transfer of interface parameters from one code to the other, reducing analysis effort as well as discretionary conservatism. As depicted in TR Section 2.1.1 and Figure 2.1-2, this AICS integrates the PARCH and HCROSS codes as subroutines into the STRIKIN-II code, and makes use of the User Controlled Interface (UCI) input file to provide a means for controlling the selection of features, options, and discretionary conservatism on a case-by-case basis. The UCI input file, described in Appendix A of the topical report, is a single file containing input variables that control model options in the CEFLASH-4A, COMPERC-II, STRIKIN-II, PARCH, and HCROSS codes. TR Table A-1 specifies the UCI input variables for the execution of various modes of LBLOCA evaluation from the 1999 EM AICS. Through the use of UCI, the user can execute the 1999 EM AICS in a

manner consistent with the options, features, and approved models of the 1985 EM. This method of execution is referred to as a "1985 EM Simulation."

TR Section 3.5.1 provides a comparison of the LBLOCA analysis results of a typical CENP-designed pressurized water reactor (PWR) analyzed using the 1985 EM simulation of AICS and the reference analysis, designated "base analysis of record," using the 1985 EM standard code system. This 1985 EM simulation uses only the 1999 EM AICS without other process changes made to the 1985 EM described in Sections 2.1.2 and 2.1.3 of this report, and without reduction of discretionary conservatism discussed in TR Section 2.1.1.1. The results of the two calculations are nearly identical, thus demonstrating that the 1999 EM AICS without other process changes produces results equivalent to the 1985 EM. The staff concludes that the 1999 EM AICS represents a procedure change which can be executed without changes to the NRC-accepted 1985 EM thermal-hydraulic models and methodologies.

2.1.2 Explicit Cladding Swelling/Rupture Calculations

The 1985 EM uses the CEFLASH-4A code for the core and hot assembly thermal-hydraulic calculations during blowdown, and the STRIKIN-II code for the limiting fuel rod heatup analysis using blowdown hydraulic boundary conditions calculated by CEFLASH-4A. In compliance with Paragraphs I.C.7 and I.D.5.b of Appendix K to 10 CFR Part 50, the CEFLASH-4A calculation of the hot assembly flow and the STRIKIN-II calculation of the fuel rod heatup take into account flow blockage calculated to occur as a result of cladding swelling or rupture. Both CEFLASH-4A and STRIKIN-II use the NUREG-0630 cladding swelling/rupture model to calculate cladding rupture and flow blockage. The NUREG-0630 cladding swelling/rupture model correlates cladding rupture temperature against heating rate and engineering hoop stress of the clad, and correlates cladding circumferential strain (swelling) and flow area reduction against heating rate and rupture temperature.

In the 1985 EM, the NUREG-0630 swelling/rupture model was implemented external to the CEFLASH-4A and STRIKIN-II codes and through user controlled inputs to the codes. The user controlled input is a cumbersome process with repetitive calculations while iterating on heating rate dependent inputs. An analyst can apply a user discretionary conservatism by using a pre-determined conservative heating rate for the inputs for the calculations of the maximum cladding rupture strain and flow channel blockage.

TR Section 2.1.2 describes a process change implemented in the 1999 EM AICS to explicitly calculate the cladding rupture model directly in CEFLASH-4A and STRIKIN-II. A subroutine was created in each code in accordance with the NUREG-0630 models to perform the cladding rupture temperature calculation and the interpolation for rupture strain and blockage as functions of heating rate and engineering hoop stress in the cladding. This provides an explicit calculation of the time and location of cladding rupture, the amount of pre-rupture plastic strain, rupture strain, and hot assembly blockage based on the actual heating rate and the differential pressure across the cladding calculated in the code. The AICS also maintains options to introduce conservatism through user-specified low heating rate rupture and/or maximum cladding rupture strain and blockage. This process change eliminates the iterative process and preserves the approved NUREG-0630 models in the 1999 EM AICS.

TR Section 2.3.3 and Table 2.3-1 provide a sensitivity study to assess the effects of using the actual heating rate for the cladding rupture and blockage calculations in comparison with the user-controlled conservative fixed heating rate. The results show a small reduction in the peak cladding temperature (PCT) from the 1985 EM calculations as a result of this explicit cladding swelling/rupture calculation, and the elimination of the discretionary conservatism.

The process change of incorporating the NUREG-0630 cladding swelling/rupture and flow blockage models in the 1999 EM AICS continues to use the same models approved for the 1985 EM, and is therefore acceptable.

2.1.3 Consistent Modeling of Spray and Spillage Into Containment

In an LBLOCA analysis, the containment spray provided to condense steam in the containment and the spillage from the reactor vessel out of the break are accounted for in the containment pressure calculation. The containment spray pumps take suction from the refueling water storage tank (RWST). The spillage out of the break includes the overflow from safety injection water from the safety injection tank (SIT) and the RWST through safety injection pumps to the downcomer and the broken discharge leg. The methodology for the calculation of the containment spray and spillage are implemented in the COMPERC-II transient calculation. The containment spray flow from the spray pumps is based on the containment pressure. The water that is spilled out of the break is calculated based on the downcomer liquid level relative to the break location and pressure as described in Section III of COMPERC-II code (Reference 8).

In the 1985 EM, the containment spray and reactor vessel spillage water sources for the containment pressure calculations were typically implemented manually by the analyst through bounding input tables of flow versus time based on conservative estimates. The input for COMPERC-II requires two entries for the temperature of the RWST water, which provides both the containment spray and the pumped safety injection. In the 1985 EM, the RWST temperature was conservatively assumed to have a [ ] value for containment spray and a [ ] value for safety injection.

The 1999 EM AICS utilizes existing portions of the COMPERC-II transient calculation to automatically provide the sources of dispersed water for the containment spray and spillage calculations. The 1999 EM AICS also provides for an input of a table of containment pressure versus flow for containment spray pumps to model the spray pump delivery curves. Also, the SIT discharge model from the CEFLASH-4A is integrated into COMPERC-II to calculate the SIT spillage from the broken discharge leg.

TR Section 3.5.2 and Figure 3.5-16 provide a comparison of the condensation energy removal rate from the steam phase of the containment calculated by the 1985 EM simulation of manually prepared inputs of spray and spillage versus the 1999 EM AICS. The comparison shows that the 1999 EM predicts [ ] condensation heat removal rate from the steam phase of the containment. This is because the use of 1999 EM AICS actually calculates the dispersal of cold water from the spray and spillage sources, and therefore eliminates the conservative bounding estimations of these sources in the manual input used in the 1985 EM.

However, since the COMPERC-II code was approved as a part of 1985 EM, the staff finds the use of the existing COMPERC-II models to calculate the containment spray and spillage to be acceptable since no change has been made to the approved thermal hydraulic models.

2.1.4 Process Changes Evaluation Conclusion

As discussed in the preceding sections, the 1985 EM process changes, including the use of an AICS, the explicit cladding swelling/rupture calculations in the CEFLASH-4A and STRIKIN-II codes, and the consistent modeling of spray and spillage into containment, represent no changes to the NRC-accepted thermal-hydraulic models and methodologies, and is therefore acceptable in the context of the 1999 EM.

TR Section 3.5.1 also demonstrates that the 1999 EM AICS can be executed through the UCI input to produce a 1985 EM simulation that would result in almost the same results of the 1985 EM. However, the use of the 1985 EM simulation for licensing applications would constitute changes in the 1985 EM. This is because 10 CFR Section 50.46, paragraph (c)(2) defines a LOCA EM as the calculational framework that includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure. Therefore, the use of the 1985 EM simulation is a change to the approved 1985 EM.

As discussed in Section 2.2 below, an EM change, or changes that result in a reduction of the PCT by more than 50°F would require the replacement of the Dougall-Rohsenow film boiling correlation currently accepted in the 1985 EM. As discussed in Sections 2.1.2 and 2.1.3 of the topical report and above, the use of the 1999 EM AICS in conjunction of the cladding swelling-rupture calculation and the consistent modeling of spray and spillage into containment could result in significant reduction in the overall conservatism of the EM. As such, use of the 1999 EM AICS with the 1985 EM process improvements would require replacement of the Dougall-Rohsenow correlation with an acceptable film boiling heat transfer correlation. However, if the 1999 EM AICS is used in a strict 1985 EM simulation that results in insignificant reduction in conservatism, the Dougall-Rohsenow correlation need not be replaced. A strict 1985 EM simulation of the 1999 EM AICS involves restrictions to the UCI input to the like of "Standard Input 1985 EM" in Table A-1 of the topical report that would eliminate all model changes and all process changes that could reduce discretionary conservatisms currently used in the 1985 EM. Therefore, the use of the 1999 EM AICS without replacement of the Dougall-Rohsenow correlation for the 1985 EM simulation for licensing applications will require NRC review and approval as to how the AICS will be used.

2.2 Replacement of Dougall-Rohsenow Film Boiling Correlation

In the 1985 EM, the CEFLASH-4A and STRIKIN-II codes used the Dougall-Rohsenow film boiling correlation for blowdown film boiling heat transfer calculation. Though the Dougall-Rohsenow correlation was specified in the original Appendix K to 10 CFR Part 50 as an acceptable correlation for LOCA EMs, the 1988-revised Appendix K, Paragraph I.C.5.c states that EMs that make use of the Dougall-Rohsenow correlation and were approved prior to

October 17, 1988, continue to be acceptable until a change is made to, or an error is corrected in, the EM that results in a significant reduction in the overall conservatism in the EM, i.e., a reduction in the calculated PCT of at least 50°F. At that time continued use of the Dougall-Rohsenow correlation under conditions where non-conservative predictions of heat transfer result will no longer be acceptable. To comply with the revised Appendix K requirement, the 1999 EM replaces the Dougall-Rohsenow correlation with another film boiling correlation.

TR Section 2.2 describes the new film boiling correlation, [ ], to be used in place of the Dougall-Rohsenow correlation. Section 2.2.2 provides the assessment of the new film boiling correlation. The new film boiling correlation was developed from 10 different sources which cover the range of interest in LBLOCA analysis as shown in TR Table 2.2-1 in terms of wall temperature, temperature difference, pressure, mass flux, heat flux, quality, and heat transfer coefficient. NUREG-1230 (Reference 9) states that this correlation has been previously assessed against the Oak Ridge National Laboratory (ORNL) Thermal-Hydraulic Test Facility (THTF) film boiling data, and was found to best predict these data. CENP also implemented this correlation in a special version of CEFLASH-4A and STRIKIN-II to compare with the THTF data. The results shown in TR Figures 2.2-1 and 2.2-2 show that the correlation underpredicted the heat transfer coefficients of a majority of the data except for a few that are slightly overpredicted. TR Section 2.2.3 and Tables 2.2-4 and 2.2-5 provide an assessment of the replacement of the Dougall-Rohsenow correlation on LBLOCA analyses. The results show an increase of the PCT with the new film boiling correlation.

The staff also finds that this same correlation has been approved for use in the LOCA EMs by other PWR vendors. Therefore, the use of this correlation to replace the Dougall-Rohsenow correlation is acceptable.

2.3 CEFLASH-4A Limiting Fuel Assembly Internal Gas Pressure

As described in Section 2.1.2 of this safety evaluation, the 1985 EM uses the cladding swelling/rupture model described in NUREG-0630 for fuel cladding swelling and rupture, and flow blockage calculations in the CEFLASH-4A and STRIKIN-II codes. The NUREG-0630 model correlates the cladding rupture temperature, the amount of swelling (circumferential strain), and flow blockage against the fuel cladding heating rate and engineering hoop stress. The cladding hoop stress is determined from the differential pressure across the cladding, i.e., the difference between the RCS pressure and the fuel rod internal pressure.

In the 1985 EM version of CEFLASH-4A, the internal fuel rod pressure is based on the initial value specified in the initialization of the LOCA transient (using time-in-life dependent fuel performance boundary conditions for the average rod of the hot assembly at operating conditions that correspond to the limiting technical specification for peak linear heat rate), whereas STRIKIN-II calculates the rod internal pressure based on time varying rod temperature and volume. The calculations of the internal rod pressure result in a potential inconsistency in the timing of cladding rupture in which CEFLASH-4A calculates the hot fuel assembly cladding rupture during blowdown, whereas STRIKIN-II calculates the hot rod rupture later in the refill/reflood period.

In the 1999 EM version of CEFLASH-4A, a change is made to calculate the fuel rod internal pressure during the blowdown with the same model used in the STRIKIN-II code for the fuel rod heatup calculation to eliminate inconsistencies in the timing of fuel rod cladding rupture and assembly blockage flow redistribution.

TR Section 2.3.1 describes the fuel rod internal pressure calculation model to be implemented in CEFLASH-4A. The model is the same as the approved model used in STRIKIN-II, and is based on the ideal gas law, where the total moles of gas in the fuel rod are assumed to remain constant during the transient, and on the fuel rod temperature and the gas volume change resulting from cladding and fuel dimensional changes due to thermal and mechanical expansion and contraction, as well as plastic strain of the cladding. The pre-rupture cladding plastic strain calculation for each axial node and the numerical process in CEFLASH-4A are also consistent with those used in STRIKIN-II.

To assess CEFLASH-4A's new fuel rod internal pressure model, TR Figure 2.3-1 provides a comparison of the internal pressures of the hot assembly average rod with the same initial stored energy and internal gas pressure calculated by CEFLASH-4A and STRIKIN-II. The figure shows very similar behavior with [ ] internal pressure calculated by CEFLASH-4A. The internal pressure differences are attributed to the fuel rod axial nodalization, initial fuel rod gas volume, and the dynamic gap conductance calculations on gas temperature in STRIKIN-II.

TR Section 2.3.3 provides an assessment of the impact of the implementation of the new rod internal pressure model on LBLOCA analysis. The assessment is done in three steps, including the step to assess the impact of explicit NUREG-0630 cladding swelling/rupture calculation using the code calculated heat rate (Section 2.1.2), and the step incorporating the new fuel rod internal pressure calculation model in CEFLASH-4A. As shown in TR Table 2.3-1, the results show a reduction in the PCT using the dynamic gap pressure calculation.

The implementation of the dynamic fuel rod internal pressure model in CEFLASH-4A provides a more consistent internal pressure calculation with STRIKIN-II in the LBLOCA analysis. Since this is the same model used in STRIKIN-II, which has been accepted as part of the 1985 EM, the staff concludes that use of this fuel internal pressure model in connection with the NUREG-0630 cladding swelling/rupture model complies with Appendix K requirements, and is therefore acceptable.

2.4 COMPERC-II Steam Venting Reflood Thermal-Hydraulics

During the core refill and reflood periods of an LBLOCA, the two-phase steam-water mixture leaving the core flows through the upper plenum and external loops, with some flowing through the break to the containment, and the rest through the intact loop back to the downcomer annulus. The LBLOCA EM uses COMPERC-II to perform refill/reflood thermal hydraulic calculations including the steam venting through the loops and break. The steam venting flow is calculated based on the flow resistance network constructed, depending on the break location, from the core and upper plenum through both the intact and broken loops, and through the break to the containment. The steam venting flow is also dependent on the pressure

differential (ΔP) between the upper plenum and containment, and the steam temperature and specific volume.

In the 1985 EM, the COMPERC-II calculation of reflood steam venting assumes that the two-phase fluid leaving the core [

.] This SG thermal-hydraulic assumption is overly conservative in light of the observations obtained from the separate effect tests on the model U-tube steam generator conducted as part of the FLECHT-SEASET reflood and natural circulation test program (Reference 10). The FLECHT-SEASET test program showed that (1) the SG tube exit temperature is less than the SG secondary side saturation temperature, (2) the secondary side significantly stratifies with colder liquid in the lower part of the SGs, and (3) the steam exiting the SG tubes is superheated but wet.

For the 1999 EM, a revised steam venting model is implemented into the COMPERC-II code with a new SG thermal-hydraulic model. The new SG model maintains other assumptions in the 1985 EM related to the steam venting calculation, including the liquid entrainment with the core generated steam to the upper plenum, and [ ] de-entrainment in the upper plenum and hot legs. The primary flow at the SG tube inlet is assumed to be [ ] steam [ ]. This results in the primary flow being superheated in the SG based on the heat transfer from the SG secondary side. Compared to the 1985 EM, the overall effect of the 1999 EM SG model is to reduce the degree of steam superheat as it exits the SG tubes, which reduces the steam specific volume and resistance to steam venting through the loops, thus resulting in an increase in the core reflood rate.

TR Section 2.4.1, as amended in the letter of December 1, 2000 (Reference 11), describes the 1999 EM new SG reflood thermal hydraulics model. It includes a [

]. The code solves conservation equations of liquid mass, mixture mass, and mixture energy for the fixed control volume.

The steam venting SG model assumes that the U-tubes are covered by the SG secondary side liquid. It also does not account for the SG wall heat transfer to the SG secondary steam region. In its letter of November 13, 2000 (Reference 12), CENP provided a sensitivity analysis to determine the effects of U-tube uncovery and the inclusion of the wall heat transfer to the secondary side steam on the secondary side pressure and temperature calculations, as well as the effects on peak cladding temperature. The results showed that the effect of U-tube uncovery during the LBLOCA was [ ] in the PCT, and therefore, if the U-tube was partially uncovered during the LBLOCA, not modeling the tube uncovery was [ ]. For the case of neglecting the SG wall heat transfer to the steam region, the sensitivity study showed a [ ] difference in the secondary pressure, and PCT. Therefore, the staff concludes that the SG modeling is acceptable.

In response to a staff question, CENP also provided technical justifications on the use of a specific natural convection heat transfer correlation (Equation 2.4.1.3-1) for the calculation of the SG secondary side heat transfer coefficient, rather than the Eckert-Jackson correlation recommended in NUREG/CR-1534. Comparisons between the two correlations showed that the heat transfer coefficients calculated by the Equation 2.4.1.3-1 are mostly lower than the Eckert-Jackson correlation. In addition, Equation 2.4.1.3-1 is the model used in several instances for modeling free convection in the CENP large and small break EMs, and is used here for consistency. Therefore, the use of the Equation 2.4.1.3-1 natural convection correlation is acceptable.

Steam Venting Reflood Thermal Hydraulic Model Assessment

Except for the implementation of the SG model in the 1999 EM COMPERC-II code, the 1999 EM still maintains several conservative assumptions of the 1985 EM. TR Section 2.4.2 provides assessments of the revised steam venting model.

An assessment was made with simulation of the FLECHT-SEASET SG separate effects tests. The FLECHT-SEASET SG separate effects tests were performed with the objectives of determining the heat transfer characteristics from the larger SG for various known inlet fluid conditions and secondary side conditions. A total of 20 tests were performed covering ranges of pressure, flow, and quality.

The comparisons of the COMPERC-II to the FLECHT-SEASET SG test were done with the objectives of (1) demonstration of the conservatism of the 1999-EM model assumption of [ ] steam at the SG tube inlet, and (2) assessment of margin of conservatism of other assumptions.

To demonstrate the conservatism of the SG model assumption of [ ] steam at the SG inlet, a special version of 1999 EM COMPERC-II code was created to execute only the new SG model for analysis of the FLECHT-SEASET SG separate effect tests. Among the FLECHT-SEASET SG tests, only one test case (Case 22920) has inlet quality of 1.0, all others have inlet quality of 0.8 or lower. TR Section 2.4.2.2 provides comparisons of three FLECHT/SEASET SG test cases run with the 1999 EM COMPERC-II SG model. In response to a staff request, two additional test cases are also chosen so that the five cases cover the pressure and flow ranges typically encountered during the reflood. All five test cases were run with the COMPERC-II SG model assuming [ ] steam at the SG tube inlet, regardless of the test inlet quality conditions. For Test Case 22920, which has the test inlet quality of 1.0, the results show [ ] between the COMPERC-II calculations and the test data in terms of secondary side fluid temperature along the tubes, secondary side exit temperature, primary side steam temperature. This demonstrates that the SG model [

] for this FLECHT-SEASET simulation. For other test cases with inlet quality of about 0.8, the comparisons show that the 1999 EM COMPERC-II SG model [ ] the measured values of the secondary side temperature both at the end of the test and at the tube exit elevation throughout the tests. For the locations near the SG tube inlet, the calculations [ ] the primary side steam temperature during the earlier part of the tests. This is because [

]. For other locations, especially at the SG tube outlet, the calculations [ ] the primary side steam temperature and the secondary side liquid temperature of the tests.

As mentioned earlier, except for the implementation of the new SG model, the 1999 EM COMPERC-II steam venting thermal hydraulic model maintains other assumptions of the 1985 EM, including a liquid entrainment [ ] into the upper plenum, [ ] de-entrainment in the upper plenum and hot legs, and [ ] of the entrained liquid to [ ] steam before entering the SGs. TR Section 2.4.2.3 provides an assessment of margin of conservatism of these assumptions against the realistic conditions of SG tube inlet quality less than 1.0, de-entrainment in the upper plenum and hot legs, and SG tubes exit quality less than 1.0, which are [ ]. This is to demonstrate that the margin of conservatism associated with other models in the 1999 EM COMPERC-II code is much larger than the reduction in PCT that is being credited by the new SG model.

To assess the conservatism of the 1999 EM COMPERC-II model assumption of [ ] steam at the SG tube inlet, a special two-phase SG tube model was implemented into the 1999 EM COMPERC-II code. This special two-phase SG tube model, described in TR Section 2.4.2.3.1, is [ ], but is created for the sole purpose of evaluating the effect on the PCT of entrained liquid into the SGs. TR Section 2.4.2.3.2 provides a comparison of the two-phase SG model calculations against two FLECHT-SEASET SG test cases having an SG inlet quality of about 0.8. The results show that the secondary side fluid temperature at the tube exit is [ ] predicted, but the temperature in the region half way up the tubes is [ ] due to the fact that the special two-phase SG model [ ], whereas the tests showed wet steam at the SG exit. The results also show that the special two-phase SG model predicts [ ] secondary side fluid temperature compared to that calculated with the 1999 EM SG model, which demonstrates the conservatism of the 1999 EM SG model assumption of [ ] steam at the tube inlet.

TR Section 2.4.2.3.3 provides a parametric study of SG inlet quality to assess the effect of the 1999 EM assumption of [ ] steam at the SG inlet on LBLOCA analyses. The results of PCT reduction versus the SG inlet quality show that the assumption of SG inlet [ ] steam of the 1999 EM maintains a conservative margin relative to a more realistic two-phase condition at the SG inlet (e.g., a PCT reduction of about [ ] with a inlet quality of [ ]).

As a part of resolution of the reactor safety issue of water carryover and steam binding with cold leg injection, the 2D/3D Test program showed that, during the reflood of a LBLOCA, water carryover from the core de-entrains in the upper plenum due to the decrease in steam velocity, and that de-entrainment would result in PCT decrease. TR Section 2.4.2.3.4 provides an assessment of the effect of de-entrainment in the upper plenum and hot legs on the reflood rate and PCT compared to the 1999 EM's conservative assumption of [ ] de-entrainment. A special version of COMPERC-II, which [ ], was created with a de-entrainment model in the upper plenum and hot legs for the purpose of assessing the de-entrainment effect. A parametric study was performed with various de-entrainment fractions compared to [ ] de-entrainment fraction of the 1999 EM assumption. The results show a [ ] in PCT as the de-entrainment fraction increases, which is consistent with the TRAC results on de-entrainment performed under the 2D/3D program. The results show the 1999 EM

assumption of [ ] de-entrainment has a significant margin depending on the actual de-entrainment.

Since the 1999 EM assumes [ ] steam at the SG tube inlet, which results in superheated steam in the SG and the SG exit, TR Section 2.4.2.3.5 provides a evaluation of the effect of moisture at the SG tube exit on PCT. The evaluation, using a simple two-phase loop model, shows a [ ] moisture content at the SG exit results in a [ ] in PCT.

TR Section 2.4.2.1 provides a comparison of the system response to a LBLOCA calculated with the 1985 EM and the 1999 EM steam venting model with the new SG model. TR Figures 2.4.2.1-1 through 2.4.2.1-16 show that the trend of the reflood transient remains essentially the same, and that the lower steam temperature calculated by the revised model results in the [ ] loop resistance factor, and [ ]. TR Table 2.4.3-1 shows that the 1999 EM with the new SG model in the COMPERC-II steam venting calculation results in a [ ] in PCT compared to the 1985 EM.

In summary, the 1999 EM COMPERC-II steam venting thermal hydraulic calculation includes a new SG thermal hydraulic model, which results in a reduction of steam superheat in the SG tubes and exit. The steam superheat reduction results in the increase of reflood rate due to lower specific volume of steam and flow resistance in the loop. Therefore, the revised steam venting model results in a [ ] of conservatism margin in the LBLOCA PCT predicted by the 1985 EM. However, the new SG model with the assumption of [ ] steam at the SG tube inlet has been benchmarked and showed conservative predictions against the FLECHT-SEASET SG separate effect tests data. Other conservative assumptions of the 1985 EM, which are maintained in the 1999 EM, such as [ ] de-entrainment of the carryover liquid in the upper plenum and hot legs, also provide a [ ] margin. Therefore, the staff concludes that the revised steam venting model of 1999 EM is acceptable.

2.5 COMPERC-II Steam/Water Interaction During Nitrogen Discharge

In PWRs, the SITs are pressurized with nitrogen. During a large break LOCA, when the water in the SIT attached to each reactor cold leg is depleted, the nitrogen that pressurizes the SITs escapes through the injection piping. Depending on the SIT design, the discharge of nitrogen from the SITs into the primary system occurs shortly after the start of the reflood phase of the LOCA transient. The presence of the non-condensable nitrogen affects the ECCS water-steam interaction during the reflood portion of LBLOCA.

In the 1985 EM COMPERC-II reflood calculation, the effect of nitrogen on the ECCS water-steam interaction was accounted for in the calculation of the injection section pressure differential (ΔP). The calculation of injection section ΔP includes the ΔPs during SIT injection, nitrogen injection, and pump injection periods. As stated in TR Section 2.5, specific bounding ΔP values imposed by the NRC safety evaluation were used in the SIT injection and pump injection periods. The ΔP during nitrogen injection was calculated with a momentum balance model based on the assumptions of [ ] flow of nitrogen/liquid in the safety injection line, and [ ] flow of nitrogen, steam and liquid mixture in the cold leg [ ]. This assumption of [ ] mixture flow resulted in the calculated ΔP

for the nitrogen injection period to be [ ] compared to the ΔPs during SIT injection and pump injection periods, and also resulted in a discontinuity at the end of the nitrogen injection. This [

].

TR Section 2.5 describes the implementation of a change to the steam/water interaction model during nitrogen discharge from the SITs. This new model continues to comply with the bounding values of ΔPs for the SIT injection and pump injection periods in the 1985 EM. The ΔP during the nitrogen injection period is calculated by [

]. The mathematical formulation of the nitrogen injection ΔP calculation is described in TR Section 2.5.1.3. This nitrogen inject ΔP is limited by a control logic that would not allow the calculated value to fall below the bounding values imposed on the ΔPs during the SIT injection and pump injection, respectively. This formulation and control provides the [ ] in the ΔP at the beginning and the end of nitrogen injection, and is [ ] during the nitrogen injection than the limit ΔPs imposed by the NRC for the SIT injection and pump injection, respectively.

The effect of SIT nitrogen discharge on the ECCS behavior during the reflood period was evaluated as a part of resolution of the reactor safety issues in the international 2D/3D program, as described in NUREG/IA-0127 (Reference 13). The evaluation includes the nitrogen discharge tests conducted at the UPTF and the Achilles facilities, respectively, and the TRAC simulation of the UPTF tests and PWR analyses. Section 4.4 of NUREG/IA-0127 summarizes the calculations and observations from the tests and the TRAC analyses. As nitrogen begins to flow into the cold legs, it flows much faster than the preceding water because the pressure losses in the piping are less for lower density gas. The nitrogen quickly pushes emergency core cooling water from the intact cold legs into the reactor vessel downcomer, and pushes water in the top of the downcomer and in the broken cold leg toward the break. The primary system and the injection region is pressurized for a short period until the nitrogen can leave the system. Suppression of steam condensation due to the presence of non-condensable nitrogen causes a further increase of the downcomer pressure. The UPTF test confirmed some phenomena related to SIT nitrogen discharge predicted in TRAC PWR analyses; namely, the pressurization of the downcomer, the dilution of steam in the downcomer and cold legs, and the surge in the core water level. The 2D/3D program concluded that, while the UPTF test did not simulate the effects of nitrogen discharge on core cooling, TRAC PWR analyses suggest that SIT nitrogen discharge and the resulting surge in the core water level are beneficial to core cooling.

The COMPERC-II code [

]. Since the 2D/3D test program shows the opposite behavior in that the nitrogen injection is beneficial to core cooling, it supports the conclusion that the 1999 EM model revision is conservative for modeling steam-water interaction during the period of nitrogen injection. The staff therefore finds the revised nitrogen discharge model acceptable.

2.6 MOD-2C Reflood Heat Transfer Procedure

During the reflood phase of an LBLOCA, the core reflood rate changes as the transient progresses. For the calculation of the reflood heat transfer coefficients (HTC), the COMPERC-II code considers three different reflood rates to approximate the actual reflood process, i.e., the first reflood rate having a large value, a second, long-term reflood rate, and a third reflood rate with the reflood speed less than one inch per second. As described in Appendix G of CENPD-134, the 1985 EM uses the so-called MOD-2C procedure to calculate the reflood HTCs for the various reflood rate periods. The basic heat transfer correlation of the MOD-2C procedure is the MOD-1C FLECHT correlation, which is a modified version of the FLECHT reflood heat transfer correlation for constant reflood rate. Three different HTC curves are calculated using the MOD-1C constant reflood rate heat transfer correlation for three different time intervals of different reflood rates. The MOD-2C procedure provides for a mechanism to calculate the time-dependent reflood HTCs in the interfaces of different reflood rates during the reflood phase.

The MOD-2C procedure contains several aspects related to multiple reflood rate modeling, including mirror imaging and time shifting. The time shifts (Δt) extend the first reflood rate period into the second reflood rate period, and the second period into the third reflood rate period, respectively. The time shifted curve represents a shift to the origin on the time axis so that the MOD-1C FLECHT correlation for constant reflooding rate can be used to represent multiple reflood rates. The equation for the calculation of the time shifting contains two parts, i.e., an empirical elevation-dependent correlation adjustment multiplier (CAM), and a reflood mass integral expressed as a ratio to represent the long term reflooding rate impact inherent in the constant flooding rate heat transfer correlation.

The 1999 EM includes a modification to the MOD-2C procedure for the reflood HTC calculation. The modification does not change the basic MOD-1C FLECHT reflood heat transfer correlation for constant reflood rate, but is made to the MOD-2C procedure through a modification in the empirical elevation-dependent CAM. A revised CAM is described in TR Section 2.6.1. This revised CAM correlation is [

].

TR Section 2.6.2 provides an assessment of the revised CAM correlation. The assessment is made based on comparisons to five tests from the FLECHT-SEASET reflood experiment, and four tests from the CCTF. The CCTF is designed to model a full-height core section and four primary loops of a PWR, and is used to perform reflood heat transfer tests characteristic of a LBLOCA for 15x15 fuel rod design. TR Tables 2.6-1 and 2.6-2, respectively, provide the data summary of the FLECHT-SEASET and CCTF tests. The range of variation in the test data used for the revised CAM correlation is summarized in TR Section 2.6.4.

TR Figures 2.6-2 through 2.6-57 provide comparisons of the HTC as a function of time at various elevations calculated with both the 1985 EM and the 1999 EM revised model against the FLECHT-SEASET and CCTF test data. For both FLECHT-SEASET and CCTF, the HTCs

are calculated from the highest measured temperatures, which assures conservatism in the development of an empirical correlation based on this data. The comparisons show improvement of the reflood HTC predictions of the revised model over the 1985 EM in the comparisons to the data. The comparisons also show that the revised model overpredicts the reflood HTCs of the FLECHT-SEASET forced reflood tests during the period having reflood rate less than 1 in/s at higher elevations. For the FLECHT-SEASET gravity reflood tests with reflood rate greater than 1 in/s, the revised model generally underpredicts the HTCs of the tests. For the CCTF tests, which has reflood rates decrease from about 2 in/s to less than 1 in/s, the revised model underpredicts the HTCs. TR Table 2.6-3 provides a "subjective agreement matrix" for the revised model comparison to the FLECHT-SEASET and CCTF data. Except for non-conservative predictions in the higher elevations of the two FLECHT-SEASET forced reflood tests having a reflood rate of less than 1.0 in/s, which were also overpredicted by the 1985 EM, the comparisons show conservative predictions for the FLECHT-SEASET gravity reflood tests and the CCTF tests. These comparisons demonstrate that on the average, the revised model will predict a lower HTC than measured. For reflood rates less than 1.0 in/s, paragraph I.D.5.b of Appendix K to 10 CFR Part 50 requires the heat transfer calculation to be based on the assumption that cooling is only by steam. Therefore, the higher HTC calculated by the MOD-2C reflood heat transfer correlation would be limited by the HTC based on steam cooling assumption. For typical LBLOCA applications of the CENP LBLOCA EM, the most limiting elevations for PCT occur between 7 and 8 feet elevations, where the revised model conservatively predicts a lower HTC than measured.

TR Figures 2.6-58 through 2.6-61 show that the revised reflood CAM correlation improves the reflood HTCs over the 1985 EM after about 100 to 200 seconds in the reflood period in the 7 to 8 feet elevation. The revised model also results in a [ ] in the PCT calculation.

However, the 1999 EM reflood heat transfer calculation with the revised CAM has been compared to the FLECHT-SEASET and CCTF reflood test data, and shows overall conservatism over these test data except for the FLECHT-SEASET forced gravity tests at high elevations. Since the revised model conservatively underpredicts HTC at the elevations where the limiting PCT occurs for an LBLOCA, the staff concludes that the revised CAM in the MOD-2C reflood heat transfer procedure is acceptable.

2.7 STRIKIN-II Hot Rod Steam Cooling Heat Transfer

In the 1985 EM, the hot rod steam cooling heat transfer calculations are executed in stages with the STRIKIN-II, HCROSS and PARCH codes, and the interface data are transferred manually. The process includes executing the STRIKIN-II hot rod heatup calculation until the initiation of the hot rod steam cooling, transferring the hot channel blockage results to HCROSS for the hot channel flow redistribution calculation at and above the rupture node, and transferring the results to PARCH to calculate the hot rod steam cooling heat transfer coefficients, which are then used by STRIKIN-II for the remainder of the transient calculation.

The 1999 EM AICS integrates the PARCH and HCROSS codes into STRIKIN-II as subroutines, and eliminates manual transfer of interface data. As described in TR Section 2.7, the 1999 EM AICS STRIKIN-II also incorporates the following modifications to improve consistency among these codes for the steam cooling heat transfer calculation:

- Addition of [ ] to PARCH fuel rod model by [ ]. The implementation of the STRIKIN-II [ ], into PARCH improves consistency of the hot rod temperature calculations in STRIKIN-II and PARCH.

- Implementation of the FLECHT reflood heat transfer coefficients calculated by STRIKIN-II to PARCH steam cooling calculation. Transfer of the FLECHT HTCs to PARCH improves the consistency, and satisfies the requirement of using the smaller of the FLECHT HTC for the rupture node and the steam cooling HTC calculated by PARCH.

- [ ]. This improves the PARCH steam channel energy balance at the intermediate HCROSS node elevations and credits all appropriate steam cross-flows to PARCH consistently.

TR Section 2.7.1 describes the implementation of these modifications. A comparison between PCTs with and without the above modifications for steam cooling listed in Table 2.7-1 of the topical report shows only a minor difference in PCT for the steam cooled nodes, and essentially no difference in the PCT calculated by STRIKIN-II. The staff concludes that these modifications are acceptable.

3.0 LBLOCA ECCS PERFORMANCE ANALYSIS

TR Section 3.0 discusses various areas related to the LBLOCA ECCS performance analysis. They are evaluated in the following sections.

3.1 Compliance with Safety Evaluation Report Constraints and Limitations

TR Section 3.2 requested the NRC to address two issues associated with the July 31, 1986, NRC safety evaluation (Reference 14) on the 1985 EM. These two issues, addressed below, are the applicability of the 1985 EM to non-CE manufactured fuel, and the referencing of CENPD-133, Supplement 4.

Applicability of Non-ABB CENP Manufactured Fuel

The NRC's 1986 SER indicated that the application of the CEFLASH-4A to Westinghouse plants, which was discussed in the July 3, 1985, submittal (Reference 15) and supplemented in November 5, 1985, letter (Reference 16), was not reviewed at that time, and will be evaluated in a separate SER at such time that a licensee or applicant declares intent to use that model. The SER "INTRODUCTION" stated that the primary reasons for the CENP requested changes to their large break ECCS evaluation model was to implement revised staff requirements for clad swelling and rupture as described in NUREG-0630. These requirements were developed to account for new data developed subsequent to the approval of the original C-E model. Section 2.1.7 of the SER stated that:

"For reasons mentioned above [i.e., Section 2.1 of that SER], we conclude that the cladding swelling and rupture models described in Reference 1 do not underestimate the degree of swelling or the incidence of rupture thus conforming to the requirements of Appendix K and are acceptable for use in licensing LOCA analyses of CE NSSSs."

The SER "CONCLUSIONS" stated that:

"With the exceptions noted in the report, the model is applicable to all C-E designed PWRs being supplied with C-E manufactured Zircaloy clad fuel."

TR Section 3.2 states that this statement implies the applicability of the 1985 EM is only to CENP manufactured Zircaloy clad fuel, and states that the reasons for NRC implying such a limitation remain unclear, but the licensing issues in question in 1985-86 were related to the implementation of NUREG-0630 cladding rupture and blockage models.

The staff has reviewed the 1986 SER and found no discussion of any reason why the CE model is applicable only to C-E designed PWRs with C-E manufactured Zircaloy clad fuel, except for the statement that the CE cladding models are based on the NRC staff correlations given in NUREG-0630. The staff has reviewed NUREG-0630 with regard to the data base and cladding swelling and rupture models, which were adopted directly by CENP for the calculations of rupture temperature, burst strain, and flow blockage. Because the cladding rupture/swelling data base used in the development of the NUREG-0630 correlations were obtained from test programs that employed (a) pressurized, zircaloy-clad, fuel rod simulators that were internally heated with UO_2 or cartridge heaters, and (b) aqueous atmospheres, they are generically applicable to all fuel design with zircaloy cladding. The staff, therefore, concludes that no limitation is required in the application of these correlations to a specific fuel manufactured by CENP. Therefore, the staff concludes that the constraint specified in the 1986 SER to limit the C-E cladding swelling and rupture models to C-E manufactured fuel should be removed. The statement in the SER "CONCLUSION" shall be revised to read:

"With the exception noted in the report, the model is applicable to all CE-designed PWRs with Zircaloy clad fuel."

It should be noted that the following condition stated in the 1986 SER remains unchanged: "Since no CE NSSS owner has a high-temperature burst application, CE has restricted their use of the NUREG-0630 burst strain and flow blockage correlations to temperature less than 950°C, thus avoiding this high temperature region entirely. Should a cladding rupture temperature greater than 950°C be encountered in any future plant analysis, CE will submit justification for extending their models into this region."

Referencing CENPD-133, Supplement 4

TR Section 3.2 states that the 1986 SER failed to cite in its reference list one of the supplements that comprise the 1985 EM, and that no evidence was found regarding NRC disposition of CEFLASH-4A, Supplement 4 (Reference 17). Since that Supplement, dated April 1977, is a part of the 1985 EM and will comprise the 1999 EM, CENP requests that there be a closure of it.

The CEFLASH-4A code, described in CENPD-133, is used by C-E to analyze the thermal-hydraulic response during the blowdown portion of an LBLOCA. The staff has reviewed CENPD-133, Supplement 4, which documents the changes made to CEFLASH-4A resulting from an NRC requirement concerning blowdown heat transfer and from improvements to the code methodology. The heat transfer logic in CEFLASH-4A is changed to comply with the requirement specified in Section I.C.4.e of Appendix K to 10 CFR Part 50 to prevent nucleate boiling from recurring during blowdown once the CHF is first predicted to occur. CENP, in its letter of November 13, 2000, states that all of the methodology changes described in CENPD-133, Supplement 4, including the change made to conform to the Appendix K requirement on "no return to nucleated boiling," were incorporated into the 1985 EM. The staff concludes that Supplement 4 of CENPD-133 is a part of the 1985 EM.

3.2 Plant Design Data:

TR Table 3.3-1 provides general guidelines for key design input parameters for LBLOCA analyses. The table provides a comparison between the 1985 EM and the 1999 EM in the choices of these input parameters, and provides a qualitative ranking of their impacts. TR Section 3.3 states that the majority of these general characteristics are selected to be bounding from an ECCS performance analysis viewpoint and, therefore, represent a source of conservatism incorporated into the 1999 EM LBLOCA analysis. The table identifies several key parameters that are either changed in the usage for the 1999 EM or substantiated by sensitivity studies. TR Section 3.3 provides sensitivity studies of the ECCS performance results of a reference plant CENP-designed PWR for the following three parameters: (1) refueling water storage tank temperature, (2) SG secondary modeling, and (3) safety injection pump actuation time.

<u>Refueling Water Storage Tank Temperature</u>

Both the ECCS safety injection pumps and the containment spray take suction from the RWST. There are competing effects on the PCT calculation of the RWST water temperature in that a lower temperature containment spray would increase the condensation potential in the containment and result in a lower containment pressure and a lower reflood rate, whereas a lower water temperature of the safety injection water into the downcomer and lower plenum would result in higher reflood rate. In the 1985 EM LBLOCA analysis, conservatism was introduced by the use of different RWST water temperatures, the [ ] temperature for the containment spray and the [ ] for the safety injection flow.

In the 1999 EM, a [ ] will be used for both the safety injection and containment spray. TR Section 3.3.1 provides a sensitivity study of the RWST temperature on the LBLOCA PCT calculation of a reference plant CENP-designed PWR. The results show the use of a [ ] temperature for both safety injection and containment spray gave a higher PCT than the use of a [ ] temperature. This is because the effect of one aspect, such as [ ], is more significant than the other aspect such as [ ], in lowering the reflood rate, which gives a higher PCT for the reference plant CENP-designed plants. Therefore, the sensitivity study provides an acceptable basis to determine the choice of the RWST temperature for the LBLOCA analysis of the reference plant CENP-designed PWR. However, each licensee who uses the 1999 EM must ensure that the

choice of the RWST temperature for the safety injection and containment spray provides a bounding PCT result of the LBLOCA events.

SG Secondary Initial Pressure and Physical Parameters

A parametric study was performed on the SG secondary side initial pressure and other physical parameters such as [ ]. By changing the values of these parameters by about 10 percent, the results show insignificant difference in the calculated PCT. Therefore, TR Table 3.3-1 shows that the 1999 EM will continue to use the 1985 EM inputs for the SG secondary side initial pressure and inventory and SG tube plugging. The staff finds this to be acceptable.

Safety Injection Pump Actuation Time

A parametric study was performed for the impact of the safety injection pump actuation time for three cases: (1) safety injection (SI) actuated during early reflood based on safety injection actuation signal (SIAS) and delay time, (2) SI actuated at the end of blowdown, and (3) SI actuated after SITs empty. The results in TR Table 3.3-4 show the [ ]. Therefore, CENP determines that Case 1 is the preferred for the 1999 EM calculation. The staff finds this to be acceptable because the SI actuation based on the SIAS plus delay time also best represents actual SI actuation design logic.

3.3 Method of Analysis

TR Section 3.4 states that the NRC approved method of analysis for the 1985 EM is unchanged for the 1999 EM, including break spectrum and worst single failure of an ECCS component, etc. TR Section 3.4.3 provides a single failure analysis performed with the 1999 EM for a reference plant CENP-designed PWR among the assumptions of (1) no ECCS component failure, (2) failure of an LPSI pump, and (3) failure of a diesel generator. The results in TR Table 3.4-2 show no failure of an ECCS component produces the highest PCT, which is consistent with the analysis performed with the 1985 EM shown in TR Table 3.4-1. This is because [

]. However, this worst single failure result is not generic. CENP states that other plant configuration combinations of containment size and ECCS design and delivery rates may lead to different conclusions, therefore, the worst single failure will be performed by each applicant using the 1999 EM, including consideration of the most limiting value of the RWST temperature. Therefore, each applicant referencing this TR must perform a plant-specific worst single failure study.

3.4 1999 EM Evaluation

TR Section 3.5 provides a comparison of the results of the LBLOCA analysis of a reference plant CENP-designed PWR analyzed using the 1999 EM and the 1985 EM "simulation" using

the 1999 EM AICS, respectively, and a reference analysis, designated "base analysis of record" (AOR) using the 1985 standard code system. The 1985 EM simulation uses the 1999 EM AICS without the other changes related to explicit NUREG-0630 cladding swelling/rupture model and the reduction of discretionary conservatisms discussed in Section 2.1.1.1 of the topical report. The results provided in TR Table 3.5-1 shows that the 1985 EM simulation produces almost the same result as the AOR.

The analysis using the 1999 EM utilizes all of the proposed 1999 EM improvements including the removal of the Dougall-Rohsenow film boiling correlation. Comparisons of the 1999 EM with the 1985 EM simulation are shown in Figures 3.5-1 through 3.5-15. These comparisons show the replacement of Dougall-Rohsenow film boiling correlation results in [ ] fuel and cladding temperature during blowdown. During reflood, the containment pressure calculated with the 1999 EM becomes [ ] than the 1985 EM simulation. However, overall, the 1999 EM provides [ ] upper plenum pressure and loop flow resistance, and [ ] subcooled level in the core, reflood rate and heat transfer coefficient, and therefore [ ] PCT and cladding oxidation.

TR Section 3.6 presents an LBLOCA break spectrum analysis for a reference plant CENP-designed PWR performed with the 1999 EM with the entire set of model changes described in the topical report. The analysis uses a reference case with a specially selected linear heat generation rate (LHGR) for which the PCT calculated by the 1985 EM simulation is 2199°F. Four break sizes were analyzed ranging from 0.4 double-ended guillotine at pump discharge leg (DEDLG) to 1.0 DEDLG. The analysis results show a reduction in the PCT of about [ ] for the limiting break of 0.6 DEDLG compared to that calculated with the 1985 EM simulation.

TR Section 3.7 provides an assessment of the overall conservatism in the 1999 EM. The purpose of the assessment is to demonstrate that the conservatisms imposed by the Appendix K requirements is large compared to the reduction of the conservatism as a result of 1999 EM model changes from the 1985 EM. The assessment considers three Appendix K requirements: (1) the decay heat of 1.2 times the 1971 ANS standard, (2) use of a locked impeller for the reactor coolant pump resistance to steam venting during reflood, and (3) use of steam cooling for the hot rod at rupture elevation or above when the reflood rate is less than 1 in/s. The analysis was performed with the 1999 EM AICS in the 1985 EM simulation, except that the Dougall-Rohsenow film boiling correlation is replaced. These assessments show an overall margin of conservatism of these three requirements of close to [ ], compared to the use of 1.0 times the 1971 ANS decay heat standard (which is shown in TR Figure 3.7-13 to be [ ] the 1979 ANS standard + 2 sigma), the assumption of a free running pump impeller, and the use of the FLECHT reflood correlation instead of steam cooling assumption. The staff does not necessarily agree that about [ ] of conservatism exists, because there are uncertainties associated with these assessments that are not quantified. However, as listed in TR Table 1.0-1, there are also other sources of conservatism in the 1985 EM. Therefore, the staff concludes that the 1999 EM, with a relatively modest reduction of PCT from the 1985 EM, maintains an appropriate amount of overall model conservatism.

4.0 CONCLUSION

The staff has reviewed the 1999 EM, which is evolved from the 1985 EM with modifications described in Supplement 4-P of CENPD-132. Based on the evaluation discussed above, the staff concludes that the 1999 EM is acceptable for licensing applications for CENP-designed PWRs subject to the limitations discussed in this safety evaluation.

5.0 REFERENCES

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(3) Letter from P. W. Richardson (CE Nuclear Power LLC) (LD-2000-0046) to US Nuclear Regulatory Commission, "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model," August 30, 2000.

(4) Letter from P. W. Richardson (CE Nuclear Power LLC) (LD-2000-0050) to US Nuclear Regulatory Commission, "Submittal of CENPD-132, Supplement 4-NP, Revision 1 -Non Proprietary Version," September 25, 2000.

(5) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," CE Nuclear Power LLC, August 2000.

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(6) CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

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(9) NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," 1988.

(10) NUREG/CR-1534, "PWR FLECHT SEASET Steam Generator Separate Effects Task Data Analysis and Evaluation Report," February 1982.

(11) Letter from P. W. Richardson (CENP) (LD-2000-0060) to US Nuclear Regulatory Commission, "Replacement Pages for CENPD-132, Supplement 4-P, Rev. 1," December 1, 2000.

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(13) NUREG/IA-0127, International Agreement Report, "Reactor Safety Issues Resolved by the 2D/3D Program," July 1993.

(14) Letter from D. M. Crutchfield (USNRC) to A. E. Scherer (CE), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model And Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986.

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Date: January 9, 2001

ABSTRACT

This topical report describes modifications to the CE Nuclear Power LLC Emergency Core Cooling System (ECCS) evaluation model used for the analysis of the large break loss-of-coolant accident (LBLOCA). The modifications include process changes within the currently Nuclear Regulatory Commission (NRC)-accepted evaluation model, the replacement of the Dougall-Rohsenow film boiling correlation, as well as improved models designed to reduce conservatism. Model improvements are made in the areas of (1) cladding swelling and rupture, (2) steam venting reflood thermal hydraulics, (3) steam/water interaction during nitrogen discharge from the safety injection tanks, (4) reflood heat transfer, and (5) hot rod heat transfer to steam. Sensitivity studies and comparisons with experimental data are presented. Input requirements to implement the modifications are defined. The results of a break spectrum analysis for a typical Combustion Engineering designed Pressurized Water Reactor (PWR) employing the improved models are presented and compared to the results obtained using the currently NRC-accepted version of the LBLOCA evaluation model.

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1.0 INTRODUCTION

This topical report describes modifications to the CE Nuclear Power LLC large break loss-of-coolant accident (LBLOCA) Emergency Core Cooling System (ECCS) Evaluation Model (EM). The version of the EM described in this topical report is named the 1999 LBLOCA ECCS Evaluation Model. It is abbreviated as the "1999 EM" in this topical report. As provided in paragraph 10CFR50.46(a)(1)(ii) of Reference 1.0-1, the 1999 LBLOCA ECCS Evaluation Model is an Appendix K ECCS evaluation model.

The currently NRC-accepted EM is named the June 1985 Evaluation Model, which is abbreviated as the "1985 EM." The modifications to the 1985 EM described in this topical report are designed to provide ECCS performance analysis margin by reducing conservatism while remaining within the context of an Appendix K EM. Also, the modifications to the 1985 EM have the objective of improving analysis efficiency and accuracy by automating the code-to-code interfaces, which are currently performed manually.

The 1999 EM modifications are organized into three basic categories. These categories and the list of modifications are as follows:

1) 1985 EM Process Changes Within the Currently NRC-Approved EM
    - Automated/Integrated Code System
    - Explicit NUREG-0630 Cladding Swelling/Rupture
    - Consistent Modeling of Spray and Spillage into the Containment
2) 1999 EM NRC-Required Changes
    - Replacement of Dougall-Rohsenow
3) 1999 EM Changes Requiring Licensing Review and NRC Approval
    - Hot Assembly Fuel Rod Internal Gas Pressure
    - Steam Venting Reflood Thermal-Hydraulics
    - Steam/Water Interaction During Nitrogen Discharge
    - Reflood Heat Transfer
    - Hot Rod Steam Cooling Heat Transfer

Section 2.0 of this topical report presents a detailed description of the modifications listed above for each of the three categories of change.

Section 3.0 presents the results of a LBLOCA ECCS performance break spectrum analysis performed for a typical Combustion Engineering designed PWR using the 1999 EM. It includes a comparison to the results calculated using the 1985 EM. Section 3.0 also includes an assessment of the overall conservatism of the 1999 EM. The assessment compares the peak cladding temperature (PCT) results of the 1999 EM to the PCT margin associated with removal of three EM features that are required by Appendix K to 10CFR50:

(i) the conservative multiplier on the decay heat model,

(ii) the conservative "locked rotor" hydraulic resistance to steam venting, and

(iii) the conservative use of heat transfer from steam only on the hot rod heatup calculation under conditions where the core reflood rate is less than one inch per second.

Appendix A of this topical report identifies the computer code input requirements necessary to implement the modifications introduced in the 1999 EM.

After this topical report was originally issued, Westinghouse Electric Corporation acquired ABB Combustion Engineering Nuclear Power and the company name was changed to CE Nuclear Power LLC. Also, during the licensing review process, the NRC requested additional information (RAI) and suggested that a number of modifications and additions to the topical report should be made to both correct and clarify the technical documentation. Also, with training and usage, the Automated/Integrated Code System was enhanced, which lead to additional documentation changes to incorporate the latest capabilities and user-guidance material. Therefore, prior to completion of the licensing process, Revision 1 of this topical report was prepared for the following reasons:

1. Change the company name
2. Incorporate the NRC RAI and the responses to the RAI
3. Incorporate corrections and clarifications identified by NRC during the licensing review process
4. Provide updated user guidance material associated with the content and usage of the Automated/Integrated Code System.

After completion of the licensing process, this revised topical report will be reissued in conformance with any NRC instructions included in the Safety Evaluation Report (SER). Following normal practice, this will include changing the document designation to indicate NRC acceptance as follows: CENPD-132, Supplement 4-P-A. (Note that this accepted version of the document will not be designated Revision 1.)

1.1 Summary Description of the LBLOCA EM

The original CE Nuclear Power LLC LBLOCA EM methodology was accepted by NRC in June 1975, (Reference 1.0-2). Several improvements to the original EM were made between December 1975 and September 1978. The last major modification was licensed in July 1986, which is referred to as the 1985 EM. This is the currently NRC-accepted EM for LBLOCA.

The 1985 EM is an Appendix K EM, which implies that it contains major sources of conservatism. Table 1.0-1 lists selected sources of conservatism in four categories. These categories are as follows:

(1) regulatory conservatism from 10 CFR 50, Appendix K,
(2) SER constraints or limitations imposed by the NRC as part of the licensing process,
(3) model aspects and assumptions introduced during the model development, and
(4) discretionary conservatism included as part of the process for conducting the ECCS performance analyses.

In addition, plant design data, used to provide analysis inputs, are conservatively biased in accordance with the LBLOCA EM to produce conservatism in the calculated limiting PCT and limiting peak cladding local oxidation percentage (PLO). In some cases, conservative biases have been selected that in reality cannot mutually co-exist.

In the 1985 EM, the CEFLASH-4A computer code (Reference 1.0-3) is used to perform the blowdown hydraulic analysis of the reactor coolant system (RCS) and the COMPERC-II computer code (Reference 1.0-4) is used to perform the refill/reflood hydraulic analysis and to calculate FLECHT-based reflood heat transfer coefficients. The HCROSS (Reference 1.0-5) and PARCH (Reference 1.0-6) computer codes are used to calculate steam cooling heat transfer coefficients. The PCT and PLO are calculated by the STRIKIN-II computer code (Reference 1.0-7). Core-wide cladding oxidation is calculated using the COMZIRC computer code (Appendix C of Supplement 1 of Reference 1.0-4). The initial steady state fuel rod conditions used in the analysis are determined using an NRC-approved fuel performance computer code, FATES3B (Reference 1.0-8).

1.2 Summary Description of Modifications to the LBLOCA EM

The following is a brief summary description of the modifications to the LBLOCA Evaluation Model for the 1999 EM. The modifications are organized into the three categories of changes described earlier.

1.2.1 1985 EM Process Changes Within the Current EM

There are three process changes to the LBLOCA ECCS performance analysis methodology that maintain consistency with the currently NRC-accepted EM. These 1985 EM process changes are included in this topical report in the context of the 1999 EM, but they do not require NRC review since they do not represent any change to the NRC-accepted EM. These process changes are as follows:

- Automated/Integrated Code System

  The automated/integrated code system for the 1985 EM and the 1999 EM is a process change that combines the various computer codes of the 1985 EM into an integrated code system. With the automated/integrated code system, the analysis process can be executed from start to finish without analyst intervention. The automated/integrated code system can be executed with selected computer codes, models, options, and features that fully represent the methodologies that comprise the currently NRC-accepted 1985 EM. This is referred to as "1985 EM Simulation." The modifications to the 1985 EM that are included in the 1999 EM are activated through options in the User Control Interface (UCI) file, which is part of the automated/integrated code system. The benefit of this 1985 EM process change is to reduce the introduction of discretionary conservatism that is commonly used to avoid repetitive case running and to eliminate interface hand calculations involved in manual transfer of data from one code to the next.

- Explicit NUREG-0630 Cladding Swelling/Rupture

  In the 1985 EM, the NUREG-0630 cladding swelling and rupture models are implemented through user controlled inputs. This process often leads to repetitive calculations while the analyst iterates on heating rate dependent inputs. The 1999 EM automated/integrated code system explicitly calculates the NUREG-0630 model components without analyst intervention or iteration in a manner fully consistent with the approach described in the NRC-approved documentation for the 1985 EM. Therefore, this 1985 EM process change improves numerical precision, eliminates iterative case running, and remains consistent with currently approved methodology.

- Consistent Modeling of Spray and Spillage into the Containment

  The 1999 EM automated/integrated code system utilizes existing portions of the COMPERC-II transient calculation to automatically provide the sources of dispersed water for the containment spray and spillage calculation. The analysis methodology for doing this transfer was already incorporated in COMPERC-II when the code was first written. But the linkage of the calculations from the RCS to the containment was implemented manually by the analyst in the 1985 EM. These manual methods are based typically on conservative estimates of spillage from the RCS along with inconsistent but conservative assumptions for the enthalpy of the spilled coolant. The 1999 EM code system approach for consistent modeling of spray and spillage into the containment replaces the manual methods currently used by the analyst.

1.2.2 1999 EM NRC-Required Changes

The 1988 revision to Appendix K requires the removal of the Dougall-Rohsenow film boiling correlation the first time a change is made to the EM. This replacement is described as follows:

- Replacement of Dougall-Rohsenow

  The 1985 EM uses the Dougall-Rohsenow film boiling correlation during the blowdown period of the LBLOCA transient. The 1999 EM removes the Dougall-Rohsenow correlation and replaces it with the[ ]correlation. The [ ]correlation is conservative compared to test data, is applicable for the range of applicability required for LBLOCA analysis, and is endorsed by the NRC in NUREG-1230 (Reference 1.0-9).

1.2.3 1999 EM Changes Requiring Licensing Review and Approval

The following summary describes those modifications to the 1985 EM that require NRC licensing review and approval prior to use. These 1999 EM changes represent new or updated capabilities designed to provide LBLOCA analysis margin by removing excessive

conservatism. The overall conservatism of the 1999 EM is maintained through the numerous sources described in Section 1.1, which are not affected by these changes. Further analytical demonstration of the overall conservatism of the 1999 EM is given in Section 3.7.

- Hot Assembly Fuel Rod Internal Gas Pressure

  The 1999 EM introduces the NRC-approved dynamic model for fuel rod internal gap pressure into the hot assembly thermal-hydraulic calculation during blowdown. This modification replaces the 1985 EM conservative model assumption of constant fuel rod internal pressure. This change produces a blowdown calculation that is more consistent with the hot rod heatup calculation and avoids conservatism often introduced in the 1985 EM.

- Steam Venting Reflood Thermal-Hydraulics

  The 1999 EM implements steam generator secondary side thermal-hydraulic modeling to improve the primary side steam venting process during LBLOCA reflood. The 1985 EM conservatively models the steam generator secondary [ ] The 1999 EM retains conservatism associated with an Appendix K approach to modeling by [ ] The 1999 EM is justified by comparison to steam generator heat transfer tests. The degree of conservatism associated with the 1999 EM model assumptions is quantified by comparison to calculations using special models for realistic reflood phenomena.

- Steam/Water Interaction During Nitrogen Discharge

  The 1999 EM introduces an improvement to the steam/water interaction model used during nitrogen discharge. The improvement is designed to remove conservatism in the 1985 EM, which is based on [ ] The 1985 EM conservatively produces [ ] injection section pressure drops during nitrogen discharge that are [ ] bounding pressure drops imposed by licensing constraints

during SIT injection and ECCS pump injection. The 1999 EM resolves momentum pressure changes in a manner consistent with the licensed bounding pressure drop requirements. Conservatism is maintained in the 1999 EM through the use of the conservative upper bounds in pressure drop during SIT and ECCS pump injection and through conservative momentum formulations during nitrogen discharge.

- Reflood Heat Transfer

  The 1999 EM introduces an improvement to the MOD-1C method for applying the FLECHT heat transfer correlation during reflood. The model change improves the "time-shift" equation for variable flooding rates by utilizing FLECHT-SEASET and CCTF test data. The model revision maintains overall conservatism relative to the data.

- Hot Rod Steam Cooling Heat Transfer

  Through the automated/integrated code system, the 1999 EM introduces several modifications to the steam cooling heat transfer calculation. These modifications are designed to improve model consistency between the hot rod heatup calculation and the steam cooling heat transfer calculation. The model change[

  ] Conservatism in the 1999 EM is maintained through the conservative assumptions of the 1985 EM, which are unchanged in this model update.

Benchmarking comparisons are used to validate and justify the model bases for the following 1999 EM improvements, that were described above, and are discussed in detail in Section 2: (1) the replacement of the Dougall-Rohsenow film boiling correlation, (2) the improvement of the steam venting reflood thermal-hydraulics model, and (3) the improvement to the MOD-1C reflood heat transfer model. In particular, benchmarking with special model changes is used to justify the conservatism of several modeling assumptions related to the improved steam

venting model. The analysis approach used for the benchmarking exercises is characterized by the following guidelines:

- The 1999 EM methods for nodalizing, specifying options for models or correlations, and selecting physical design input are applied to the test comparisons in a manner fully compliant with the methods used for calculating PWR plant response, without modification or adjustment unique to the benchmarking exercise.
- Initial and boundary conditions for the benchmarking exercises are represented with best available values to minimize any bias that selecting these conditions may have on the outcome of the test comparison.
- There are no differences in analysis assumptions between the benchmarking exercise and the PWR analysis unless specifically noted for the purpose of showing the conservatism in the 1999 EM as the outcome of the test comparison.
- Special versions of the computer codes and special model modifications are used to facilitate the benchmarking exercises if necessary; otherwise, the 1999 EM computer versions and models are used for the test comparison. The use of these special versions or special models is clearly documented. Even though the versions and models may be different, it is the intention of these exercises to apply the analysis methods without inconsistencies regarding the approach for calculating the LBLOCA PWR plant response.
- The use of these special versions and special models in licensing analyses is not intended and only the option selections documented in Appendix A are to be used with the 1999 EM computer code versions.

1.3 References

1.0-1 Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

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CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

1.0-7 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

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CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

1.0-8 CENPD-139-P-A, "C-E Fuel Evaluation Model," July 1974.

CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

1.0-9 NUREG-1230, R4, "Compendium of ECCS Research for Realistic LOCA Analysis," April 1987.

Table 1.0-1
1985 EM Major Sources of Conservatism

Category of Conservatism	List of Sources of Conservatism
10 CFR 50, Appendix K	1971 ANS Decay Heat with 1.2 Multiplier Reactor Coolant Pump Locked Rotor Resistance to Steam Venting Less than 1 in/sec reflood steam cooling Baker-Just oxidation kinetics No return to nucleate boiling
SER Constraints/Limitations	Worst axial power shape and worst time-in-life fuel performance Worst single failure of ECCS component confirmed for every submittal Highest injection section pressure drop loss coefficients NRC uniform plastic strain modeled NRC prescribed hot wall delay for ECCS refilling the vessel Worst rod-to-rod radiation enclosure for hot rod heatup Heat transfer coefficients from steam cooling model must be no greater than FLECHT
Model Aspects/Assumptions	Injected ECCS removed from vessel at end of blowdown Infinite steam generator secondary heat source/heat sink modeling during reflood Steam cooling based on core average steam flow All reflood liquid carryover evaporated to minimize steam venting
Discretionary	Bounding plant parameter or analysis inputs Combine worst conditions for containment response with different worse conditions for core response Forced cladding rupture elevation to maximize PCT Maximize cladding swelling and blockage

2.0 DESCRIPTION OF MODIFICATIONS TO THE LBLOCA EVALUATION MODEL

This section describes the details of the modifications to the LBLOCA EM that comprise the process changes to the 1985 EM and the improvements introduced in the 1999 EM. Section 2.1 describes the 1985 EM process changes to the LBLOCA ECCS performance analysis methodology that remain consistent with the currently NRC-accepted EM. To comply with the requirements of the 1988 revision to Appendix K, Section 2.2 describes the replacement of the Dougall-Rohsenow film boiling correlation. Sections 2.3 through 2.7 describe the modifications to the 1985 EM that require NRC licensing review and approval prior to implementation. These changes represent new or updated capabilities for the 1999 EM designed to provide LBLOCA analysis margin by removing unnecessary conservatism. Section 2.8 and Appendix A provide a description of the computer code input options that are used to model the required features of the 1999 EM.

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2.1 Process Change within the Currently NRC-Accepted EM

This section describes three process changes to the LBLOCA ECCS performance analysis methodology that remain consistent with the currently NRC-accepted 1985 EM. These process changes are presented here in the context of the 1999 EM but they do not require NRC review since they do not represent any change to the NRC-accepted 1985 EM. These changes are provided herein for completeness of the 1999 EM description. These three process changes are the following:

1. Automated/Integrated Code System
2. Explicit NUREG-0630 Cladding Swelling/Rupture
3. Consistent Modeling of Spray and Spillage into the Containment

In the 1985 EM, the analyst may introduce conservatism in certain parameters in order to eliminate repetitive case running and excessive interface hand calculations required to transfer data from one code to the next. That is, the analyst may control the manner in which interface data is transferred from one code to the next by deliberately selecting values to conservatively bias the data transfer process. The purpose of these three 1985 EM process changes is to reduce this type of discretionary conservatism and bring more consistency to the analysis process and its results. These 1985 EM process changes represent no change to the NRC-accepted methodology. Conservatism is maintained through the many conservative aspects of the 1985 EM and through the other discretionary conservatisms listed in Section 2.1.1.1.

2.1.1 1999 EM Automated/Integrated Code System

The 1985 EM for LBLOCA (Reference 2.1-1) consists of the following computer codes:

Computer Code	Functional Purpose	Reference
CEFLASH-4A & 4APUNCH	Blowdown Thermal-Hydraulics	2.1-2
COMPERC-II, COMZIRC, & HTCOF (a subprogram of COMPERC-II)	Refill/Reflood Thermal-Hydraulics Containment Response Core Reflood Heat Transfer Methodology (MOD-1C & MOD-2C) Core-wide Zircaloy Oxidation Calculation Methodology (COMZIRC)	2.1-3
PARCH/REM & HCROSS	Hot Assembly Blockage and Steam Cooling Methodology	2.1-4 & 5
STRIKIN-II	Hot Rod Heatup Analysis	2.1-6
FATES3B	Hot Rod and Average Rod Time-in-Life Fuel Performance Analysis	2.1-7
FRELAPC & HPUNCH	Utility Codes for Processing FLECHT Heat Transfer Coefficients	2.1-8 & 9
The 1985 EM also includes several utility codes that serve to process inputs and may be used to create input data, namely, FLASHIN, AXPSHP, HEATSNK, RADENCL, and RELBOT.		

Figure 2.1-1 provides a flowchart for the LBLOCA EM that illustrates the manner and direction of data transfer between the various computer codes. The figure does not show the input processing utility codes or the two minor programs, 4APUNCH, and HPUNCH, which are used to facilitate data transfer between the major codes. The overall process for executing the ECCS performance analysis using the 1985 EM for LBLOCA consists of the following principal elements:

(1) Execution of each of the major computer codes listed above and shown in Figure 2.1-1, with base decks, and case-dependent input;

(2) Execution of the minor utility codes listed above for data processing, with base decks, and case-dependent input;

(3) Hand calculations to prepare data to be transferred from one code to the next on a case-by-case basis; and

(4) Routine manual data transfer between codes without calculational manipulation including, for example, the fuel performance data from the FATES3B computer code.

The 1999 EM for LBLOCA combines the majority of these elements of the 1985 EM analysis process into an automated/integrated code system. This allows a specific LBLOCA transient case to be executed from start to finish without analyst intervention.

Figure 2.1-2 illustrates the LBLOCA computer codes and interface files that comprise the 1999 EM automated/integrated code system. This figure does not show the input and output files that are processed or the various utility codes related to input preparation or processing.

Figure 2.1-2 introduces the User Controlled Interface (UCI) file, which provides a means for controlling the selection of features, options, and discretionary conservatism on a case-by-case basis in the 1999 EM automated/integrated code system. Appendix A, Table A-1, describes the contents of the UCI file.

Even though the interface with the fuel performance methodology (FATES3B computer code, Reference 2.1-7) is indicated in Figure 2.1-2 along with its interface files for STRIKIN-II, actually, the fuel performance analysis is performed separately from the LBLOCA analysis and is not part of the 1999 EM automated/integrated code system. The time-in-life fuel performance calculations from the fuel performance methodology provide boundary conditions for the LBLOCA analyses that are accessed by the 1999 EM automated/integrated code system.

An important capability of the 1999 EM automated/integrated code system is that it can be executed in a manner consistent with the options, features, and approved models of the 1985 EM. In this way, an analysis can be conducted in a manner that remains fully compliant with the currently NRC-accepted methodology. This method of execution is referred to as a "1985 EM Simulation." A comparison of the 1985 EM results to the 1985 EM Simulation using the 1999 EM automated/integrated code system is given in Section 3.5 for a typical Combustion Engineering designed PWR. In the example given in Section 3.5, the results of the two calculations are nearly identical thus demonstrating that the 1999 EM automated/integrated code system is procedurally equivalent to the 1985 EM. The benefits of the 1999 EM automated/integrated code system are that the analysis results using the 1999 EM code system are produced with (1) greater numerical precision, (2) significantly reduced analysis effort, and (3) with the same overall conservatism inherent in the 1985 EM.

The process for creating the 1999 EM automated/integrated code system was designed to execute the existing LBLOCA methodology with minimum coding modifications. The following groundrules were applied during the development process:

- Maintain consistency with the NRC accepted ECCS LBLOCA EM.
- Maintain all currently existing options and features of the codes.
- Automatic linkage of codes is activated by the presence of interface files created by upstream calculations.
- If the interface file is not present during execution, operation of the code defaults to fixed inputs with existing base decks.
- Develop clean interfaces, which simulate the current transfer process with higher precision.
- Allow operation in auto-mode with existing 1985 EM base decks requiring no input changes. The UCI file controls process changes.

These groundrules preserve the possibility that the 1999 EM automated/integrated code system may be executed one computer code at a time, i.e., in stand-alone mode. In this case, the automatic linkage of codes with its defined interfaces may be overruled by analyst actions in the same manual manner that the 1985 EM is currently executed.

2.1.1.1 Interfaces

The 1999 EM automated/integrated code system contains no new or modified inputs for the existing base decks for 1985 EM Simulation. A number of inputs to the base decks are no longer needed if the LBLOCA analysis is run in the auto-mode; however, dummy input values must still be provided to satisfy input reading code logic. In auto-mode, the interface file between codes takes precedence over code defaults, user fixed inputs, or dummy inputs. Appendix A, Section A.6, describes the impact of the interface variables contained in the auto-generated interface files on the existing base decks.

A User Controlled Interface (UCI) file allows user control of the options and features of the 1999 EM automated/integrated code system. The UCI file provides control of the three 1985 EM process changes described in Section 2.1 as well as the 1999 EM improvements described

in Sections 2.2 through 2.7. The UCI file was also designed to allow the continued use of several forms of discretionary conservatism through specially selected interface variables. If elected by the analyst, the 1999 EM automated/integrated code system can be used to explicitly control the amount of discretionary conservatism in the LBLOCA analysis through these UCI interface variables. The list of 1985 EM discretionary conservatisms that are controlled by the UCI file is as follows:

[

]

2.1.1.2 Consistent and Accurate Transfer of Data from One Code to the Next

Figure 2.1-2 illustrates the 1999 EM automated/integrated code system interface files. Each of the interface files is designed to accurately transfer data from one code to the next and remove the burden from the analyst of manually transferring and/or calculating the interface parameters that exist in the current methodology. The contents of each of the interface files are given in Appendix A, Section A.4.

Section 3.5.1 presents results showing that the impact of improved accuracy and consistency of the transfer of interface data changes PCT[ ] The benefits of the automatic transfer of interface data are that it allows improved control of discretionary conservatism, decreases analysis turnaround time, reduces analysis documentation, and shifts analysis emphasis from processing data to processing results.

2.1.1.3 Integration of PARCH and HCROSS into STRIKIN-II

The PARCH (Reference 2.1-5) and HCROSS (Reference 2.1-4) codes, which in the 1985 EM are stand-alone codes, are integrated as subroutines into the 1999 EM version of the STRIKIN-II code (Reference 2.1-6). The PARCH routines are now called from STRIKIN-II every time step after the later of the time of cladding rupture and the time that the reflood rate drops below one inch per second. The HCROSS routine is called from PARCH once during the first execution of PARCH to calculate the steam flow redistribution as a function of elevation in the blocked channel. The implementation of the PARCH / HCROSS routines into the 1999 EM version of STRIKIN-II significantly improves the efficiency and accuracy of the transfer of data from PARCH / HCROSS to STRIKIN-II. This integration of the codes allows transferring data every time step, and calculating the HCROSS steam flow distribution directly from the actual blockage calculated in STRIKIN-II. By contrast, the 1985 EM analysis methodology requires the construction of a discrete table of heat transfer coefficients

versus time (no more than 50 points) for a user-specified blockage and hot rod cladding rupture elevation. This table of heat transfer coefficients is then manually incorporated into the STRIKIN-II input deck.

For the 1999 EM automated/integrated code system, the execution control of each of the three modules (STRIKIN-II, PARCH, or HCROSS) is determined by the existence of an input file for the corresponding module. When the input file for a module exists, then the code system will execute the linked module.

2.1.2 Explicit NUREG-0630 Cladding Swelling/Rupture

Another process change implemented in the 1999 EM automated/integrated code system is the explicit calculation of the NUREG-0630 cladding swelling and rupture model, Reference 2.1-10. In the 1985 EM, the NUREG-0630 models are implemented through user controlled inputs in CEFLASH-4A and STRIKIN-II, Reference 2.1-4. This process is often cumbersome and leads to repetitive calculations while iterating on heating rate dependent inputs. The 1999 EM versions of CEFLASH-4A and STRIKIN-II explicitly calculate the NUREG-0630 model components and produce outputs in a manner consistent with the methods described in the previous 1985 EM licensing documentation and SER approvals. This process change (1) improves numerical precision, (2) eliminates iterative, repetitive case running, and (3) remains fully in compliance with currently approved methodology.

In the 1985 EM methodology, the NUREG-0630 cladding rupture model (Reference 2.1-10) for CEFLASH-4A and STRIKIN-II is conservatively implemented external to the code with iteration required by the analyst to account for cladding rupture as a function of heating rate. For the 1999 EM, implementing the cladding rupture model directly in CEFLASH-4A and STRIKIN-II eliminates this iterative process and provides an explicit calculation of the time of cladding rupture. Also, the amounts of pre-rupture plastic strain, rupture strain, and hot assembly blockage are explicitly determined to be consistent with the calculated cladding rupture condition.

The goals of this 1985 EM process change are summarized as follows:

- Introduce dynamic heating rate dependent plastic strain and rupture, rather than use pre-determined conservative maximum representations
- Introduce dynamic coolant channel blockage, rather than use a pre-determined conservative maximum
- Eliminate iterative case running and inconsistency between the actual calculated heating rate and a pre-determined conservative maximum
- Preserve the use of maximum strain and blockage as an option for user-controlled discretionary conservatism
- Eliminate the need for NRC review of this explicit implementation by making no change in methodology or analysis approach from that already reviewed and accepted by NRC.

This 1985 EM process change is contained in three basic parts, which are described as follows:

1) Cladding Heating Rate: Calculation of the cladding heating rate for each node at each time step is required for both CEFLASH-4A and STRIKIN-II. The NRC-accepted cladding heating rate model is defined in Reference 2.1-4, page 3-4. The heating rate used to determine the cladding rupture conditions is averaged over a time interval defined to be the smaller of either ten seconds or the time that it takes for the cladding temperature to increase 200°F prior to reaching the rupture temperature. This restriction on the maximum change in cladding temperature excludes periods of cooling during blowdown and refill prior to rupture. If negative heating rates are calculated then zero heating rate is assumed. This method yields an average heating rate prior to rupture that is equivalent to or conservatively lower than the instantaneous or initial heating rate employed in the experiments referenced in NUREG-0630 (Reference 2.1-10).

2) Cladding Rupture: Calculation of heating rate dependent rupture using NUREG-0630 is performed for each node at each time step for both CEFLASH-4A and STRIKIN-II. The NRC-accepted cladding rupture model is given in Reference 2.1-10 or in Reference 2.1-4, Appendix B. Explicit implementation of the cladding rupture model allows for precise heating rate input to the calculation.

Optionally, the user may specify the heating rate to be used for the calculation. The engineering hoop stress needed in the rupture temperature model is explicitly determined by the code-calculated differential pressure between the coolant channel and the fuel rod gap and by using the cladding dimensions provided in the required code inputs. The cladding rupture strain is determined by interpolating the values given in NUREG-0630 on heating rate and rupture temperature. Optionally, the user may specify that maximum rupture strain be used for the calculation.

3) Hot Assembly Blockage: Calculation of the hot assembly blockage conditions following rupture is based on the NUREG-0630 blockage model. The NRC-accepted assembly blockage model is given in Reference 2.1-10 or in Reference 2.1-4, Appendix B. The assembly blockage is determined by interpolating the values given in NUREG-0630 on heating rate and rupture temperature. Optionally, the user may specify that maximum blockage be used for the calculation. As described in the previous section, in the 1999 EM automated/integrated version of STRIKIN-II this blockage is transferred to the HCROSS program for calculating the steam flow redistribution and to the PARCH program for calculating the steam heat transfer coefficients. In CEFLASH-4A, this blockage is used to define the hot assembly flow redistribution to the nearest neighbors and remainder of the core due to rupture of the fuel rods in the hot assembly.

NRC has accepted each of the above elements of the 1985 EM process change described above in previous licensing submittals and SER documentation. The NUREG-0630 cladding rupture and blockage model and also the cladding plastic strain model are actually NRC developed models that have been integrated into the EM in response to previous NRC SER directives, (References 2.1-11 & 12). The explicit implementation of NUREG-0630 into CEFLASH-4A and STRIKIN-II is consistent with the methodology described in the NRC-accepted submittal, Reference 2.1-4, and needs no additional model justification than that previously provided to NRC. Reference 2.1-4 describes how the models are included in

CEFLASH-4A and STRIKIN-II. Implementation of these models into the 1999 EM automated/integrated code system followed the prescriptions given in Reference 2.1-4.

Section 2.3 describes a model change for the 1999 EM version of CEFLASH-4A to dynamically calculate the internal fuel rod gas pressure in a manner consistent with the STRIKIN-II methodology. This model change in combination with the explicit determination of cladding behavior with NUREG-0630 will provide better consistency between CEFLASH-4A and STRIKIN-II calculations. This improved fuel rod gas pressure model is not part of the 1985 EM process change described here and will require NRC review and approval.

The explicit implementation of NUREG-0630 in the 1999 EM automated/integrated code system gives exactly the same result as the externally user-prepared calculation for the same rupture boundary conditions of cladding heating rate and fuel rod differential pressure. However, through the dynamic calculation of the heating rate for each node at each time step, the explicit cladding behavior models impact the calculated PCT. The impact of this 1985 EM process change is described below in Section 2.1.3.2 and discussed in further detail in Section 2.3 in combination with the improved CEFLASH-4A fuel rod internal gas pressure model.

To explicitly introduce the NUREG-0630 cladding rupture and blockage models into CEFLASH-4A and STRIKIN-II, a subroutine was created in each code. This routine performs the rupture temperature calculation and the interpolation for rupture strain and blockage as functions of heating rate and differential pressure across the cladding, in accordance with the NUREG-0630 prescribed models. This subroutine preserves the methodology described in Reference 2.1-4. Appendix A describes the options available to the user for this methodology. These options are controlled through the User Controlled Interface File.

As described above, the 1999 EM automated/integrated code system explicitly calculates the heating rate dependent fuel rod rupture and blockage characteristics represented by the NUREG-0630 cladding swelling and rupture model. The options to introduce conservatism

through the modeling of low heating rate rupture and/or the modeling of maximum cladding rupture strain and blockage remain available to the analyst, are described in Reference 2.1-4. (Appendix A lists the available input options for the 1999 EM.) Reference 2.1-4 also states that the option is reserved to model cladding rupture and blockage with the actual heating rate, if the analysis conditions warrant. The 1999 EM automated/integrated code system explicitly represents this provision of Reference 2.1-4 as a user-controlled option.

Section 2.3.3 presents the results of analyses using the 1985 EM explicit NUREG-0630 model in the 1999 EM automated/integrated code system. The results indicate that the use of actual heating rate dependent calculations of cladding rupture[ ] the PCT by[ ] The use of actual hot assembly blockage instead of maximum blockage[ ] the PCT by an additional [ ]

2.1.3 Consistent Modeling of Spray and Spillage into the Containment

The spray and spillage calculation for the 1985 EM LBLOCA analysis is done manually by preparing bounding tables of flow versus time for the following sources of dispersed water in the containment:

[ ]

The 1999 EM automated/integrated code system utilizes existing portions of the COMPERC-II transient calculation to automatically provide each of these sources of containment spray and spillage. The analysis methodology for doing this transfer was already incorporated in COMPERC-II when the code was first written. But the linkage of the calculations from the RCS to the containment was implemented manually by the analyst. The 1999 EM code system approach to modeling spray and spillage replaces the manual methods currently used by the analyst. These manual methods are based on conservative estimates of spillage from the RCS along with inconsistent but conservative assumptions for the enthalpy of the spilled coolant.

Section 3.5.2 and in particular Figure 3.5-16 provide a comparison of the condensation energy removal rate in the containment for the manually prepared inputs of spray and spillage versus the 1999 EM automated/integrated code system calculations. The results show that the condensation energy removal rate in the containment is[ ]by the improvements of the 1999 EM during the blowdown and refill time periods.[

]

Section 3.4.2 presents results of a 1985 EM simulation worst single failure analysis, where the automatic spray and spillage feature is utilized along with other 1985 EM process improvements. The results show that the 1985 EM process improvements[ ]PCT by [ ]for a case with no failure of an ECCS component.

Other improvements to the plant design inputs have been made with the 1999 EM automated/integrated code system. In order to model the containment spray pump delivery curves, a table of pressure vs. flow for the containment spray pumps was added as part of the new input for the 1999 EM code system, see Appendix A, Section A.2. Also the SIT discharge model from CEFLASH-4A was integrated into COMPERC-II to do the calculation of the SIT spillage from the broken discharge leg automatically instead of generating the spillage data using the RELBOT utility code. This SIT calculation is also supported with additional inputs described in Appendix A, Section A.2.

2.1.4 References

2.1-1 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

2.1-2 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

2.1-3 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

2.1-4 Enclosure 1-P-A to LD-81-095, "C-E ECCS Evaluation Model Flow Blockage Analysis," December 1981.

2.1-5 CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

2.1-6 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

2.1-7 CENPD-139-P-A, "C-E Fuel Evaluation Model," July 1974.

CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

2.1-8 LOCA-76-349, "FRELAPC – An Analytical FLECHT Rod Elevation and Power Correction Program," T. C. Kessler, May 26, 1976.

2.1-9 SA-75-53, "HPUNCH-HTCOF Selective Card Punching Program," C. M. Molnar, February 3, 1975.

2.1-10 NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analyses," D. A. Powers and R. O. Meyer, April 1980.

2.1-11 O. D. Parr (NRC) to F. M. Stern (C-E), June 13, 1975.

2.1-12 D. M. Crutchfield (NRC) to A. E. Scherer(C-E), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986.

Figure 2.1-1

LBLOCA EM Computer Code Flow Chart

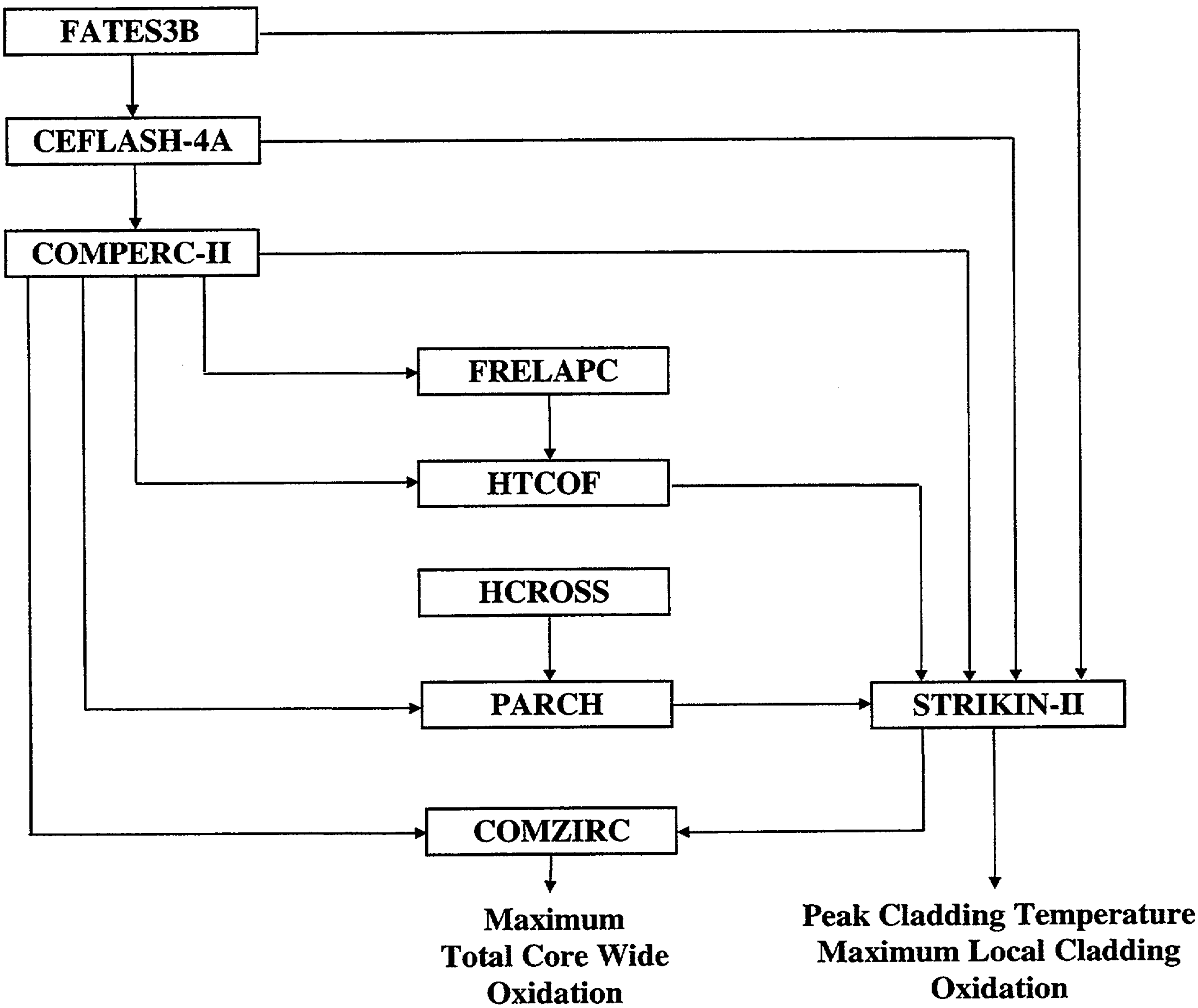


Figure 2.1-2

Automated/Integrated LBLOCA Computer Codes and Interface Files

2.2 Replacement of Dougall-Rohsenow Film Boiling Correlation

CEFLASH-4A (Reference 2.2-1) and STRIKIN-II (Reference 2.2-2) use the Dougall-Rohsenow film boiling correlation for film boiling conditions during the blowdown portion of a LBLOCA transient. The 1988 revision to Appendix K (Federal Register, Vol. 53, No. 180, September 16, 1988) requires the removal of the Dougall-Rohsenow film boiling correlation the first time a change to the EM is made.

This section describes the replacement of the Dougall-Rohsenow film boiling correlation with the [ ] film boiling correlation (References 2.2-3 and 2.2-4). The [ ] compared to test data, is applicable for the range of applicability required for LBLOCA analysis, and has been endorsed by the NRC (Reference 2.2-5, page 6.2-6).

2.2.1 Model Description

The [ ] correlation (References 2.2-3 and 2.2-4) is the following:

[ ] (2.2.1-1)

where

[

]

2.2.2 Model Assessment

2.2.2.1 Assessment of the Correlation

The [

] In the development of the correlation, heat transfer was assumed to

[

]

[

]

2.2.2.2 Assessment of the Correlation against THTF Data

Film boiling correlations, including [

] against data from a series of 22 steady state film boiling experiments at the Oak Ridge National Laboratory (ORNL) Thermal Hydraulic Test Facility (THTF). The correlation assessment in Reference 2.2-4 also included evaluations against three transient film boiling tests. The experimental data were obtained from high pressure, high temperature, steady state or transient tests with water flowing upward through an 8x8 rod bundle. The test conditions for each steady state run are shown in Table 2.2-2. [

]

while Figure 54 shows that the Dougall-Rohsenow film boiling correlation overpredicts most THTF data.

[ ] as implemented in the 1999 EM automated/integrated code system, was [ ] The resultant calculated heat transfer coefficient was then compared to the measured value.

Figures 2.2-1 and 2.2-2 show the comparisons of the [ ] film boiling correlation calculated by CEFLASH-4A and STRIKIN-II against the THTF data. Furthermore, comparison of Figures 2.2-1 and 2.2.-2 with Figure 58 in Reference 2.2-4 demonstrates [ ] correlation in CEFLASH-4A and STRIKIN-II with the one in Reference 2.2-4.

These benchmarking comparisons show that the [ ] to lower than actual heat transfer coefficient values over the full spectrum of THTF cases. This is acceptable performance for a film boiling correlation in the 1999 EM Appendix K ECCS model.

2.2.3 Application to LBLOCA Analysis

The implementation of the [ ] film boiling correlation in CEFLASH-4A and STRIKIN-II affects the energy removed from the fuel rod whenever film boiling is calculated to occur during the blowdown period of a LBLOCA. Since the [ ] with respect to Dougall-Rohsenow as documented in Reference 2.2-4, the smaller energy removed from the rod during the blowdown by the [ ]

The impact of this change in the film boiling correlation is [

] These results are shown in Tables 2.2-4 and 2.2-5. Table 2.2-5 shows that the hot rod PCT during the late reflood [ ]

2.2.4 Applicability to LBLOCA Analysis

The [ ] film boiling correlation is [ ]

2.2.5 Model as Coded

2.2.5.1 CEFLASH-4A Code

The [ ] correlation is implemented in the CEFLASH-4A code in a heat transfer calculations subroutine. The correlation is implemented in the form given in Equation (2.2.1-1) with the following [

]

2.2.5.2 STRIKIN-II Code

The [ ] correlation is implemented in the STRIKIN-II code by an extrinsic function relationship. The correlation is implemented in the form given in Equation (2.2.1-1) with the following [

]

2.2.6 References

2.2-1 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

2.2-2 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

2.2-3 [ ]

2.2-4 "Dispersed Flow Film Boiling in Rod Bundle Geometry – Steady State Heat Transfer Data and Correlation Comparisons," G. L. Yoder et al, NUREG/CR-2435 (ORNL-5822).

2.2-5 NUREG-1230, R4, "Compendium of ECCS Research for Realistic LOCA Analysis," April, 1987.

Table 2.2-1

Range of Parameters for [ ] Correlation

[

]

Table 2.2-2

THTF Steady State Upflow Film Boiling Data Range (Reference 2.2-4, Page 3)

Test	Pressure (psia)	Mass Flux (10^5 lbm/hr ft^2)	Heat Flux (10^5 Btu/hr ft^2)	Quality at CHF
B	1849.5	5.25	2.9	0.373
C	1805.3	2.46	1.8	0.684
D	1847.3	3.81	2.2	0.541
E	1908.0	4.37	2.3	0.482
F	1829.6	1.88	1.2	0.886
G	1817.6	1.84	1.0	0.929
H	1288.9	1.89	1.3	0.870
I	1331.5	2.67	1.8	0.881
J	1937.4	5.40	2.4	0.406
K	634.8	1.66	1.4	0.883
L	1202.5	3.88	2.5	0.794
M	1242.1	4.84	2.8	0.742
N	1234.2	5.94	3.0	0.662
O	866.7	2.26	1.7	0.874
P	874.3	3.83	2.6	0.771
Q	946.7	2.40	1.8	0.838
R	952.4	2.68	2.0	0.846
T	1723.0	1.77	1.0	0.967
U	1694.4	1.78	1.0	0.94
V	1744.6	1.84	1.0	0.957
W	1819.5	1.89	1.2	0.882
X	871.5	2.54	1.9	0.856

Table 2.2-3

THTF Transient Film Boiling Test Data Range (Reference 2.2-4)

Test	Pressure (psia)	Mass Flux (lbm/hr ft^2)	Heat Flux (Btu/hr ft^2)	Quality
3.03.6AR	700 to 1500	1.0×10^5 to 3.7×10^5	$5. \times 10^4$ to 3.2×10^5	23% to 100%
3.08.6C	950 to 1700	2.4×10^5 to 8×10^5	$5. \times 10^4$ to 3.5×10^5	35% to 100%
3.06.6B	875 to 1900	1.0×10^5 to 4.5×10^5	$5. \times 10^4$ to 2×10^5	5% to 100%

Table 2.2-4

Effect of Removing the Dougall-Rohsenow Correlation from CEFLASH-4A which Analyzes the Hot Assembly Average Rod during the Blowdown Portion of the LBLOCA Transient

[

]

Table 2.2-5

Effect of Removing the Dougall-Rohsenow Correlation from STRIKIN-II which Analyzes the Hot Rod During the Entire LBLOCA Transient

[

]

Figure 2.2-1

[

]

Figure 2.2-2

[

]

2.3 CEFLASH-4A Fuel Rod Internal Pressure

The 1985 EM version of CEFLASH-4A (Reference 2.3-1) models the core flow distribution of the hot assembly during blowdown as required by Appendix K to 10CFR50.46. The methodology includes cross flow between adjacent assemblies and any flow blockage calculated to occur during blowdown as a result of cladding swelling and rupture. Cladding swelling and rupture of the fuel rods in the hot assembly are calculated using the NRC-specified NUREG-0630 rupture and blockage models (Reference 2.3-2). These component models for cladding damage during LBLOCA are correlated against the differential pressure across the cladding and the fuel rod heating rate. The implementation of the NUREG-0630 models into CEFLASH-4A is described in Reference 2.3-3. The 1985 EM version of CEFLASH-4A conservatively utilizes a constant internal fuel rod pressure in the calculation of the differential pressure. This pressure is specified at the initialization of the LOCA transient using time-in-life dependent fuel performance boundary conditions for the average rod of the hot assembly at operating conditions that correspond to the technical specification for peak linear heat generation rate.

This section describes the implementation of a dynamic fuel rod internal pressure model in the 1999 EM version of CEFLASH-4A so that the CEFLASH-4A methodology for cladding rupture during blowdown will be more consistent with the fuel rod heatup calculation in the STRIKIN-II computer code (Reference 2.3-4). The STRIKIN-II computer code is used to calculate PCT and peak local cladding oxidation (PLO) percentage on the hot rod using the CEFLASH-4A hot assembly coolant flow rate and enthalpy distributions as boundary conditions during blowdown. It is the STRIKIN-II hot rod heatup analysis that determines the need for cladding rupture and assembly blockage modeling in CEFLASH-4A during blowdown. However, the assumption of constant internal gas pressure in CEFLASH-4A occasionally leads to premature rupture in the hot assembly compared to the STRIKIN-II hot rod calculation.

The proposed change from a constant fuel rod internal pressure to a dynamic calculation based on time-varying fuel rod temperature and volume is designed to minimize

inconsistencies in the timing of fuel rod cladding rupture and assembly blockage flow redistribution. Significant inconsistency may exist with the 1985 EM versions of CEFLASH-4A and STRIKIN-II, especially for higher burnup time-in-life boundary conditions. With the 1985 EM, the occurrence of assembly blockage flow redistribution in CEFLASH-4A when fuel rod cladding rupture is not calculated during blowdown in STRIKIN-II may produce [ ] therefore, this inconsistency is overly conservative. The implementation of the dynamic fuel rod internal pressure model in the 1999 EM will reduce the extent of conservatism created by inconsistent calculations between CEFLASH-4A and STRIKIN-II.

For the 1999 EM automated/integrated code system, a dynamic fuel rod internal pressure model in CEFLASH-4A along with the explicit implementation of the NUREG-0630 Cladding Rupture and Blockage model in both CEFLASH-4A and STRIKIN-II, described in Section 2.1.2, produces an analysis process that is more consistent and designed to achieve results without unnecessary conservatism.

2.3.1 Model Description

The fuel rod internal gas pressure model implemented in CEFLASH-4A is the same model used in STRIKIN-II, documented in Reference 2.3-4, Section II.7-i, and previously accepted by NRC for the hot rod heatup calculation, Reference 2.3-5. The gas pressure model is based on the ideal gas law, where the total moles of gas in the fuel rod are assumed to remain constant during the transient. Based on first principles, the model includes the effect of the dynamic gas volume change during the transient resulting from cladding and fuel dimensional changes due to thermal and mechanical expansion and contraction. Using NRC-approved models, the fuel and cladding dimensional changes are based on the model described in Reference 2.3-6, Equations 3.4.1-6 and 3.4.1-7. The dimensional changes are calculated for each of the CEFLASH-4A fuel rod nodes in the radial and axial core model (Reference 2.3-7, Section III.C.7) for each time step. For consistency between CEFLASH-4A and STRIKIN-II, the fuel rod internal gas pressure is updated for each radial core region at each time step in response to pre-rupture cladding plastic strain, which is dynamically determined for each axial

node by the explicit implementation of the NRC Coffman plastic strain model (Reference 2.3-4, Supplement 2). This computational scheme is consistent with the NRC-accepted numerical process, Reference 2.3-8. The method was implemented to be consistent with the model already used and NRC-accepted in STRIKIN-II.

$$PV = nRT \tag{2.3-1}$$

where

P = Pressure (psia)

V = Volume (ft^3)

T = Temperature (°R)

R = Gas Constant

n = Total number of moles of gas

For the transient the total moles of gas in the fuel rod are assumed constant.

Also, the gas pressure in the fuel rod is[ ]

Summing Equation (2.3 - 1) over all nodes provides the following relationship.

$$P \sum_{i=1}^{N} \frac{V_i}{T_i} = R \sum_{i=1}^{N} n_i = CVOT \tag{2.3-2}$$

where

CVOT = constant determined at initialization

Therfore during the transient, as V_i and T_i change with time, Equation (2.3 - 2) becomes

$$P = CVOT / \sum_{i=1}^{N} (V_i / T_i) \tag{2.3-3}$$

In CEFLASH-4A, the summation in Equation (2.3-3) is over the gas plena (upper and lower) and the gap between the fuel pellet and the cladding for each axial node of a given radial region of the core. The gas plena are conservatively assumed to have constant volume and temperature. The CEFLASH-4A radial fuel rod heat conduction model explicitly calculates the fuel rod gas gap temperature, and the volume is dynamically updated for mechanical and thermal expansion of the cladding and fuel pellet. Each fuel node gas gap has an average transient temperature and volume for each time step.

2.3.2 Model Assessment

To assess the CEFLASH-4A dynamic fuel rod internal pressure model, the time-varying pressure is compared to the STRIKIN-II calculated value, for a special case with CEFLASH-4A and STRIKIN-II initializing the average rod with the same stored energy and internal gas pressure. Figure 2.3-1 shows the STRIKIN-II hot assembly average rod transient internal gas pressure compared to the CEFLASH-4A hot assembly average rod calculation. As expected, the comparison shows that the[

]

2.3.3 Application to LBLOCA Analysis

The explicit implementation of the NUREG-0630 models with dynamic calculation of the cladding heating rate and fuel rod differential pressure for each fuel rod node at each time step using the dynamic fuel rod internal pressure model will affect the LBLOCA ECCS performance analysis and PCT through the following calculated mechanisms:

[

]

Also as described above, through the use of the dynamic fuel rod internal gas pressure model in CEFLASH-4A, the occurrence of blowdown rupture in the hot assembly is made consistent with the more detailed STRIKIN-II calculation for the hot rod. This prevents the condition of a CEFLASH-4A blowdown rupture when STRIKIN-II calculates that rupture occurs later.

The impact of the explicit implementation of NUREG-0630 and the dynamic internal gas pressure model is assessed in the following three steps:

1. Compare the user-controlled cladding rupture approach of the 1985 EM to the automated/integrated explicit implementation of NUREG-0630 in the 1999 EM code system. For this comparison, the CEFLASH-4A dynamic gas pressure model is not utilized and the degree of hot assembly blockage is maximized through the use of the correlation limit.
2. Compare the results of Step 1, which used the correlation limit for hot assembly blockage, to a case using the NUREG-0630 explicitly calculated hot assembly blockage.
3. Compare the results of Step 2, which used a constant internal gas pressure in CEFLASH-4A, to a case using the 1999 EM dynamic internal gas pressure model.

Table 2.3-1 provides results of these cases using the 1999 EM automated/integrated code system. The results in the table are organized by each of the major computer codes. The following is a detailed discussion of the results for each of the three steps.

- <u>Step 1</u>: The NUREG-0630 model for rupture temperature and rupture strain was used in CEFLASH-4A and STRIKIN-II with the explicit (actual) calculated heating rate instead of the fixed heating rate, externally prepared, user input model. Table 2.3-1 shows that the PCT is [ ] for this case. This performance is characterized by the following elements:
    i. The user prepared input for CEFLASH-4A was for a conservatively low 45°F/sec cladding heating rate rupture. The input also represented a NUREG-0630 fast-ramp maximum cladding rupture strain of 60% and

attempted to simulate the NUREG-0630 fast-ramp maximum blockage of 46%. However, CEFLASH-4A conservatively produced 48.1% blockage in the actual run with the user prepared inputs.

ii. By comparison, the explicit implementation of NUREG-0630 with the actual heating rate option produced a CEFLASH-4A rupture at a cladding heating rate of 78°F/sec. This is a value greater than the correlation limit of 50°F/sec and greater than the user-selected condition. This resulted in a maximum rupture strain of 60% and a maximum blockage of roughly 46%, corresponding to the NUREG-0630 fast-ramp limit.

iii. Similarly, the user-prepared input for STRIKIN-II was for a heating rate rupture of 30 °F/sec. A rupture strain of 76.7% and a maximum hot channel blockage of 71% were modeled.

iv. By comparison, the explicit implementation of NUREG-0630 with the actual heating rate option produced a STRIKIN-II rupture at a cladding heating rate of 60.8°F/sec. A rupture strain of 60% was calculated, which would have utilized the maximum blockage of 46% for that heating rate. However, as an example of user-controlled discretionary conservatism, this analysis used the NUREG-0630 maximum slow-ramp assembly blockage of 71.5%. This blockage was used for the PARCH/HCROSS calculated steam flow redistribution and the steam cooling heat transfer coefficients for the less than 1 in/sec reflood period.

v. Table 2.3-1 shows the results of Step 1. The table compares the explicit implementation of NUREG-0630 to the user-controlled input case. The STRIKIN-II hot rod rupture occurred[ ]and the STRIKIN-II PCT was[ ] for the explicit implementation of NUREG-0630.

vi. Since the blockage related hydraulics were[ ]in both cases, the [ ]PCT for the explicit NUREG-0630 case was the result of the actual heating rate calculation in STRIKIN-II, which produced[

]

vii. In this case, CEFLASH-4A predicted blowdown rupture at 5.3 seconds, which is not consistent with the refill period cladding rupture predicted by STRIKIN-II at 25 seconds.

- Step 2: The actual NUREG-0630 assembly blockage was explicitly determined by the 1999 EM automated/integrated code system and automatically incorporated into the CEFLASH-4A flow diversion calculation during blowdown, and in the PARCH/HCROSS steam cooling calculation during reflood. For PARCH/HCROSS, the calculated assembly blockage was 44.9% compared to the conservative maximum of 71.5% used in Step 1. In this case, Table 2.3-1 shows that the STRIKIN-II PCT was [ ] on the limiting FLECHT cooled node and by [ ] PARCH/HCROSS steam cooled node. In this case the blockage modeling impact [

  ]

- Step 3: The fuel rod gas pressure model was utilized in CEFLASH-4A to allow the calculation of cladding rupture consistent with the more detailed hot rod analysis performed by STRIKIN-II. Table 2.3-1 shows that the STRIKIN-II PCT was [

  ] during blowdown on the CEFLASH-4A calculated thermal-hydraulic conditions that are imposed on the STRIKIN-II hot rod heatup calculation.

In summary, the combined impact in the LBLOCA analysis of both the explicit implementation of the NUREG-0630 cladding rupture and assembly blockage model and the dynamic fuel rod internal pressure model changes was a [ ] of the STRIKIN-II PCT of [ ]

2.3.4 Applicability to LBLOCA Analysis

The 1985 EM and the 1999 EM utilize the NUREG-0630 and Coffman models to calculate cladding deformation during the LBLOCA transient. The use of these models in LBLOCA analyses is required by NRC, therefore the applicability of these models to LBLOCA analysis is not an issue. The change to a dynamic internal gas pressure model in CEFLASH-4A eliminates overly conservative inconsistencies in calculating cladding deformation that result with use of constant gas pressure. Overall conservatism is assured in this 1999 EM improvement through the conservative nature of the NUREG-0630 and Coffman swelling and rupture models, as required by Appendix K.

The 1999 EM CEFLASH-4A dynamic gas pressure model is based on the ideal gas law and is the same model as used in the STRIKIN-II analyses for hot rod PCT and PLO. Its use in STRIKIN-II is NRC-approved. This model change achieves better consistency between CEFLASH-4A and STRIKIN-II for the LBLOCA transient.

2.3.5 Model as Coded

The addition of the STRIKIN-II fuel rod internal gap pressure model to CEFLASH-4A is made in the subroutine for calculating assembly blockage. The logic is inserted at the end of the subroutine and is controlled through the user controlled interface file (UCI). The volume of the gap is updated every time step for the effects of thermal expansion of the pellet, thermal expansion of the cladding, mechanical expansion of the cladding due to the differential pressure, and plastic strain of the cladding. These calculations are taken from the logic of the approved versions of the STRIKIN-II and PARCH codes and contain the same temperature dependence and cladding dimensional dependence as modeled in these hot rod heatup calculations. The internal fuel rod gas pressure is updated at every time step then used in the next time step to determine pre-rupture cladding plastic strain and cladding rupture. The CEFLASH-4A gap pressure model parameters and the stored energy of the hot assembly are initialized by the 1999 EM automated/integrated code system using interface data produced

from the STRIKIN-II average channel initialization that is based on the time-in-life dependent FATES3B fuel performance data for the average rod of the hot assembly.

2.3.6 References

2.3-1 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

2.3-2 NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analyses," D. A. Powers and R. O. Meyer, April 1980.

2.3-3 Enclosure 1-P-A to LD-81-095, "C-E ECCS Evaluation Model Flow Blockage Analysis," December 1981.

2.3-4 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

2.3-5 O. D. Parr (NRC) to F. M. Stern (C-E), June 13, 1975.

2.3-6 CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

2.3-7 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

2.3-8 D. M. Crutchfield (NRC) to A. E. Scherer(C-E), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986.

Table 2.3-1

Results Showing the Impact of the Explicit Implementation of NUREG-0630 and the CEFLASH-4A Fuel Rod Internal Pressure Model

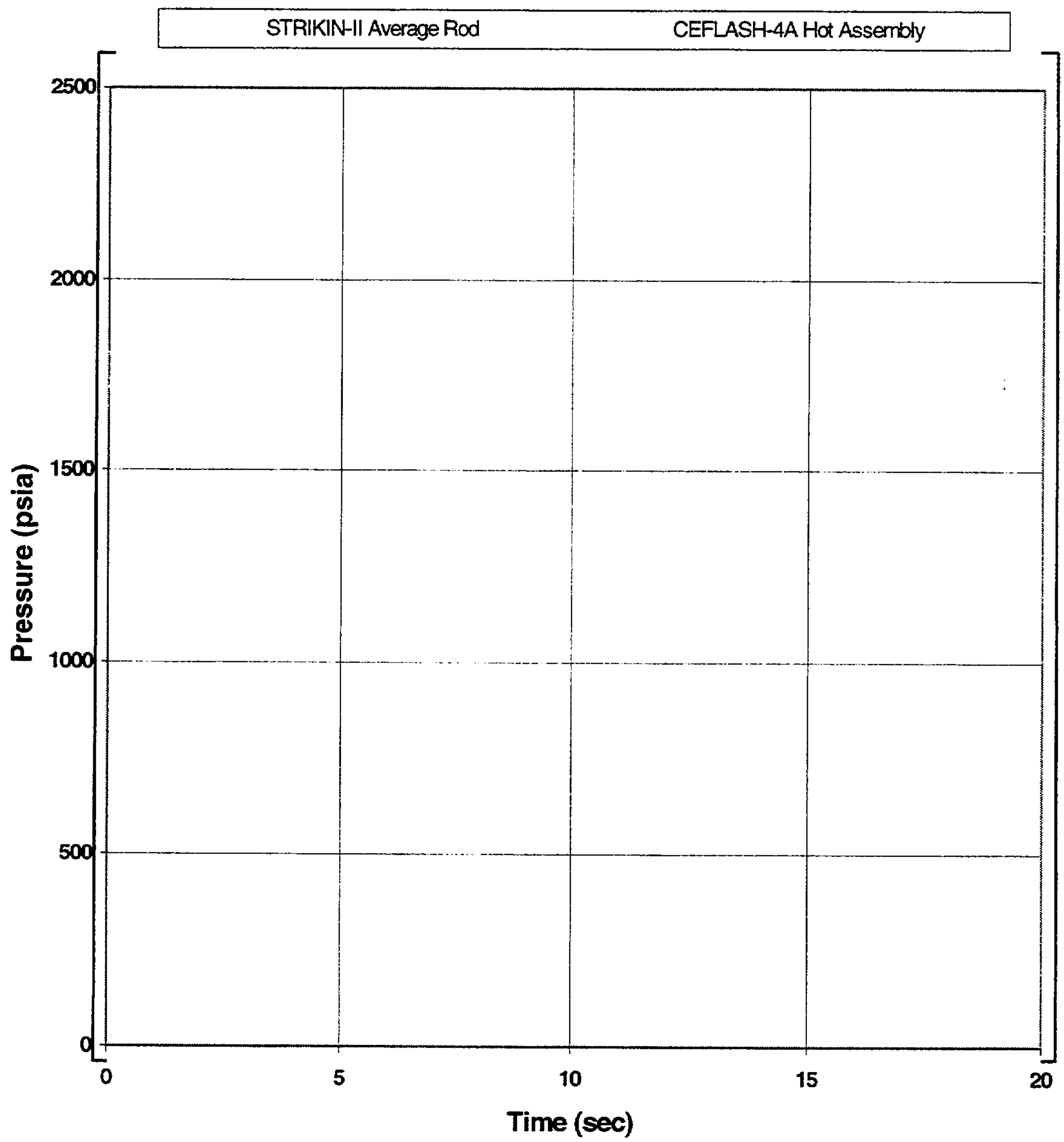
[

1999 EM Automated/Integrated Code System Results For Each Computer Code	1985 EM Simulation User-Controlled Rupture & Blockage	Step 1 NUREG-0630 Actual Heating Rate Rupture & Max Blockage	Step 2 NUREG-0630 Actual Heating Rate Rupture & Blockage	Step 3 CEFLASH Dynamic Gap Pressure

]

* In step 3, with the fuel rod gas pressure model utilized in CEFLASH-4A, cladding rupture and assembly blockage were not calculated during the blowdown period.

Figure 2.3-1
Comparison of CEFLASH-4A and STRIKIN-II Hot Assembly Average Fuel Rod Internal Gas Pressure



2.4 COMPERC-II Steam Venting Reflood Thermal-Hydraulics

The 1985 EM COMPERC-II code (Reference 2.4-1) assumes that the two-phase fluid leaving the core during reflood[

]

Separate effect tests on a model U-tube steam generator conducted as part of the FLECHT SEASET reflood and natural circulation test program (References 2.4-2 and 2.4-3), have provided test data which shows:

- The steam generator tube exit temperature is less than the steam generator secondary side saturation temperature.
- The secondary side significantly stratifies with colder liquid in the lower part of the steam generators.
- The steam exiting the steam generator tubes is superheated but wet (a fine liquid mist exits the tubes).

For the 1999 EM,[

] This improvement was made consistent with the EM model formulation and requirements and at the same time was made to be more consistent with the trend of the experimental data. The following[

]

Implementation of the 1999 EM [

]

2.4.1 Model Description

2.4.1.1 Phenomenological Description of Steam Generator Heat Transfer During Reflood and Assumptions of the 1999 EM

During the reflood period of a postulated LBLOCA the core generates a two-phase mixture as it refloods and quenches. The two-phase mixture is swept into the reactor upper plenum and travels down the hot leg into the steam generator inlet plenum. The steam generators, which are assumed to be isolated for the LBLOCA transient, are dry on the primary side and the secondary side collapsed level covers the tubes. As the two-phase mixture enters the tubes, the two-phase mixture evaporates and removes heat from the secondary side inventory in contact with the tubes along a quenched region at the bottom of the tubes. The cooler secondary side fluid becomes stratified and forms cooler thermal layers which mix horizontally by thermally driven natural convection and cover both the inlet and outlet regions of the tubes.

Above the quench front, the primary side steam is superheated and entrained liquid droplets are evaporated by conduction heat transfer from the hotter secondary side. At the exit of the tubes, the primary side steam is cooled by the stratified cooler layers of secondary side fluid. The reduction in steam temperature exiting the steam generator tubes compared to the 1985 EM reduces the loop steam specific volume. This in turn reduces the loop pressure losses allowing more steam to vent and an improved core reflood transient.

FLECHT-SEASET test data (References 2.4-2 and 2.4-3) shows that not all entrained liquid droplets are vaporized in the steam generator tubes. The presence of droplets at the exit of the steam generator tubes further decreases the loop specific volume and increases further the steam venting capability.

The thermal-hydraulic phenomena described above are quite complex. The 1999 EM that is described in this section is [

]

The following is a list of the basic assumptions for the 1999 EM revised steam venting model. The first four assumptions that follow are different from the 1985 EM methodology. The last two assumptions are the same. The assumptions are as follows:

[

]

These assumptions are conservative and are justified in later discussions and evaluations.

2.4.1.2 Non-Equilibrium Secondary Side Model

The secondary side of the steam generator is represented by [

] The thermal-hydraulic formulation is the [

] The control volume represents the mass and energy of the system and a fixed volume. The code solves conservation equations of liquid mass, mixture mass, and mixture energy for the control volume. The conservation equations are the following:

i. <u>Conservation of Mixture Mass</u>

[ ]

ii. <u>Conservation of Liquid Mass</u>

[

]

iii. Conservation of Mixture Energy

[

]

iv. Calculation of Node Pressure

This section describes the selection criteria and logic for the secondary side pressure model. A single pressure that applies to both the liquid and steam phases and to all the layers in the liquid phase (see Section 2.4.1.3) is calculated with this model. As described above, the 1999 EM version of COMPERC-II dynamically determines the thermal state of the steam generator secondary side during the transient using a single control volume with separated liquid and steam phases in thermal equilibrium or non-equilibrium. Two cases are described below, which determine if the steam generator secondary side is in a state of thermal equilibrium or non-equilibrium. Upon initialization at TAD, COMPERC-II assumes that the secondary side is in thermal equilibrium at the saturation temperature at TAD. The initial saturation conditions are calculated by the code from the steam generator secondary side pressure at TAD.

For each time step during the transient, COMPERC-II integrates the mass and energy conservation equations for the control volume, and calculates the secondary side thermal conditions (pressure and thermal state of the liquid and steam phases) assuming the thermal

non-equilibrium conditions described in Case 1, below. If a consistent solution is found, i.e., the iterative process converges and calculates [ ] then the solution is accepted. If the iterative process converges [ ] then the Case 1 calculations are rejected, [ ] are assumed to occur in the secondary side, and the pressure is recalculated from the mass and energy using the method described in Case 2, below.

If the iterative process fails to converge for Cases 1 or 2, then the COMPERC-II execution is terminated with an error message.

[

]

v. Wall Heat Model

[

]

2.4.1.3 Sectionalized Steam Generator Model

[

]

i. Sectionalized Secondary Side Temperature Model

[

]

ii. Steam Generator Tube Temperature Model

[

]

Here section i ranges from 1 to 2N.

iii. Steam Generator Tube Primary Side Temperature Model

[

]

Here section i ranges from 1 to 2N. Note that the delta temperature definition at the inlet and outlet of axial section i, follows from the assumption that the tube metal temperature is constant along the length of each section (see Subsection ii).

iv. Primary Side Heat Transfer Coefficients

The primary side heat transfer coefficients for each axial section in the steam generator tubes are calculated with the[

]

All fluid properties are evaluated at the bulk temperature for section i.

The overall heat transfer coefficient on the primary side is calculated as follows for each axial section of the tubes:

[

]

v. Secondary Side Heat Transfer Coefficients

The heat transfer rate for each axial section on the steam generator tubes secondary side (i ranging from 1 to 2N) is calculated using the equation

$$Q_{sec,i} = H_{sec,i} A_{sec,i} (T_{tube,i} - T_{sec,j})$$

where

$Q_{sec,i}$	Steam generator tube heat transfer rate (secondary side) (Btu/sec) (section i)
$H_{sec,i}$	Overall secondary side heat transfer coefficient (Btu/sec ft^2 °F) (section i)
$A_{sec,i}$	Secondary side tube heat transfer area (ft^2) (section i)
$T_{tube,i}$	Steam generator tube temperature (°F) (section i)
$T_{sec,j}$	Secondary side layer temperature (°F) (layer j)
j	Secondary side layer in contact with section i of the tubes

The heat transfer coefficients for each axial section on the steam generator tubes secondary side are calculated with the[

]

The overall heat transfer coefficient on the secondary side of the tubes for each section is calculated as follows:

[

]

2.4.2 Model Assessment

2.4.2.1 Steam Generator Model Performance for LBLOCA Analysis

Implementation of the 1999 EM steam generator model to de-superheat the steam as it exits the steam generator tubes reduces the resistance to steam venting through the loops, and results in an increase of the core reflood rate. An evaluation of the system response to the implementation of the secondary side model on a COMPERC-II LBLOCA calculation is shown in Figures 2.4.2.1-1 through 2.4.2.1-16. In these figures, the performance of the 1985 EM is compared to the 1999 EM. These comparisons are intended to illustrate the effect and

performance of the improved models relative to the current models. Detailed overall comparisons for PWR applications and an assessment of the overall conservatism of the 1999 EM are presented in Sections 3.5 and 3.7, respectively. Note that the COMPERC-II plots shown here have two different time scales. The containment pressure is shown from the beginning of the transient (Time After Break). The remaining plots are shown starting with time zero at contact time. Contact is the time when the reflood coolant contacts the bottom of the core (typically of the order of 30 seconds after the beginning of the transient). The zero elevation in Figures 2.4.2.1-2 through 2.4.2.1-4 is the bottom of the core. Figures 2.4.2.1-3 and 2.4.2.1-4 show the core subcooled liquid level and core mixture level, respectively.

These figures show that[

] Note that Figure 2.4.2.1-7 shows the steam flow exiting the core. Figure 2.4.2.1-8 shows the steam flow in the primary piping loops, which is equal to the steam flow exiting the upper plenum plus liquid entrained from the core region converted to steam. Further note that the equivalent factor in Figure 2.4.2.1-11 is the equivalent flow resistance coefficient between the upper plenum and the containment (Reference 2.4-1, pg. 24). It is a single phase flow resistance coefficient and does not include a two-phase friction multiplier.

Parameters describing the response of the steam generator secondary side and steam generator tube are shown in Figures 2.4.2.1-13 through 2.4.2.1-16.

Figure 2.4.2.1-13 shows the secondary side pressure and a comparison to [
]

Figure 2.4.2.1-14 shows the primary side steam temperature at the tube exit and a comparison to[ ]

Figure 2.4.2.1-15 shows the secondary side fluid temperature at the elevation of the steam generator tube sheet and a comparison to the[

]

Figure 2.4.2.1-16 shows the steam generator tube exit steam specific volume and a comparison to the specific volume calculated by the 1985 EM.

2.4.2.2 Simulation of the FLECHT-SEASET Steam Generator Tests - Saturated Steam U-Tubes Inlet Quality

i. The FLECHT-SEASET Steam Generator Tests

The FLECHT-SEASET steam generator test facility is a separate effects test loop, which is described in Reference 2.4-2, Section 3. The major components in the loop are a boiler, a water supply tank, steam/water mixer, steam generator, steam separator, and a containment tank. The steam generator consists of 32 unplugged tubes, which preserve the flow area scaling relationship. The test facility is designed to supply the steam generator with a steady state two-phase mixture. The test loop and steam generator responses are essentially steady state except for the secondary water, which cools down slowly.

Details of the test facility instrumentation, data acquisition and facility operation are described in Reference 2.4-2 Section 3. The summary report of the tests is Reference 2.4-3.

Table 2.4.2.2-1 lists the FLECHT-SEASET steam generator tests and the initial and boundary conditions. The tube boundary conditions shown in the table were held constant for the duration of the test, whereas, the secondary side initial conditions were the starting point conditions of the test. All of the tests listed in the table except Test 21121 are approximately 1500 second tests. The test cases can be categorized as follows:

- High pressure (~60 psia), mid pressure (~40 psia), low pressure (~20 psia)
- High flow (~0.99 lbm/sec), mid flow (~0.75 lbm/sec), low flow (~0.49 lbm/sec)
- High quality (1.), mid quality (~0.8), low quality (~0.7 and below)

The following three tests were selected for the assessment of the performance of the 1999 EM COMPERC-II model:

- Test No. 20904 (high pressure, low flow, mid quality)
- Test No. 22213 (high flow, mid pressure, mid quality)
- Test No. 22920 (high quality, mid pressure, low flow)

The three tests are shown in Table 2.4.2.2-1. The FLECHT-SEASET tests are separate effect tests, which model only the steam generator tubes and steam generator secondary side. Thus, a special version of 1999 EM COMPERC-II code was created, which executes only the steam generator primary and secondary side models. The boundary and initial conditions for the tests are specified through input. [

]

The 1999 EM COMPERC-II model[ ]
(Section 2.4.1.1). Thus, the FLECHT-SEASET test comparisons were run[

] has the following three effects:

[

]

A summary of the results for these three test comparisons follows.

ii. Test Case 20904 (Simulation with Inlet Quality equal to 1)

Test Case 20904 is a high pressure, low flow, mid quality test. Test comparisons using the 1999 EM COMPERC-II calculations are shown in Figures 2.4.2.2-1 through 2.4.2.2-4.

Figure 2.4.2.2-1 shows the primary side steam generator tube temperature response as a function of distance from the inlet every 150 seconds.

Figure 2.4.2.2-2 shows the secondary side fluid temperature along the steam generator tube as a function of distance from the inlet every 150 seconds. With the exception of the flat low temperature regions of the data at the inlet and outlet of the steam generator tubes the results show a[

]in the 1999 EM COMPERC-II calculation.

Figure 2.4.2.2-3 shows the calculated secondary side fluid temperature along the steam generator tubes at 1500 seconds against the test results from Reference 2.4-3, Page 20904-17. Figure 2.4.2.2-3 shows that as expected the 1999 EM COMPERC-II model[

]

Figure 2.4.2.2-4 shows the transient response of the secondary side fluid temperature at the steam generator tube exit, and shows that the 1999 EM COMPERC-II model[

] The reference data is again plotted from the data in the figures in Reference 2.4-3, Page 20904-17.

iii. Test Case 22213 (Simulation with Inlet Quality equal to 1)

Test Case 22213 is a mid-pressure, high flow, mid-quality test. Test comparisons using the 1999 EM COMPERC-II calculations are shown in Figures 2.4.2.2-5 through 2.4.2.2-8.

The comparison of the COMPERC-II results to the test data (Reference 2.4-3, Page 22213-17) shows [

]

The results of this test comparison show again as expected that [

]

iv. Test Case 22920 (Simulation with Inlet Quality equal to 1)

Test Case 22920 is a mid-pressure, low flow, high quality test. Test comparisons using the 1999 EM COMPERC-II calculations are shown in Figures 2.4.2.2-9 through 2.4.2.2-12. This test is a pure steam test simulation, and thus it provides [

] Consistent with the test boundary conditions, the inlet temperature was superheated at 314°F (Reference 2.4-3, Page 22920-1).

Figures 2.4.2.2-11 and 2.4.2.2-12 show [

]

2.4.2.3 Assessment of 1999 EM Steam Generator Model Conservative Assumptions

[

] The purpose of this section is to demonstrate that [

]

In order to evaluate the effect [

]

In order to evaluate the effect [

]

In order to evaluate the effect on [

]

2.4.2.3.1 Implementation of a Special Two-Phase Steam Generator Tube Model

A special two-phase steam generator tube model fully consistent with the superheated steam model described in Section 2.4.1.1 was implemented into the 1999 EM COMPERC-II code. As stated above, this model [

]

The steam generator primary side is axially sectionalized as described in Section 2.4.1.3. The special steam generator tube two-phase model was implemented consistent with the steam temperature model described in Section 2.4.1.3 by overlapping the following three effects:

[

]

i. Heat Transfer to the Steam Phase

The heat transfer from the steam generator tubes to the steam phase is calculated as described in Section 2.4.1.3, [

]

ii. Heat Transfer to the Liquid Phase

Reference 2.4-2, Section 2-2, gives a phenomenological description of steam generator heat transfer during reflood based on FLECHT-SEASET steam generator test data. The heat transfer to the liquid phase is characterized by the occurrence of a quench front, which is the boundary layer between a dry wall tube (above the wetting temperature) and a quenched tube. Below the quench front, the liquid film on the tube keeps the tube wall in nucleate boiling or forced convection at the saturation temperature of the primary side. Above the quench front,

film boiling occurs in the primary tube and the heat transfer is from the superheated tube wall to superheated steam to the entrained liquid droplets.

The heat transfer from superheated steam to the entrained liquid droplets is described in the following subsection (iii).

[

]

iii. Heat Transfer from the Steam to the Steam Generator Tubes Liquid Phase

The heat transfer from the steam to the liquid phase is calculated assuming heat transfer from superheated steam to saturated liquid droplets. The calculation is done for each axial section in the steam generator tubes. [

]

The critical steam flow is an expression of a critical superficial vapor velocity necessary to carry liquid droplets of a specified size or Weber number. From a force balance on a liquid drop, where the drag force, characterized by a drag coefficient C_D, is set equal to the body force [

]

The special model was implemented using [

]

The droplet heat transfer area is calculated assuming a uniform distribution of spherical droplets in the tube region. For a void fraction α, the number of droplets per volume Vol is

$$N = 6\ Vol\ (1-\alpha) / (\pi\ d^3)$$

Thus the heat transfer area for a volume Vol is

$$A_{surf} = 6\ Vol\ (1-\alpha) / d$$

where

d = droplet diameter (ft)

The void fraction in the tubes is calculated assuming slip flow with slip ratio equal to [

]

$$\alpha = W_{stm} / (W_{stm} + W_{liq}\ Slip\ \rho_v / \rho_l)$$

where

W_{stm} = Steam flow (lbm/sec)

W_{liq} = Liquid flow (lbm/sec)

Slip = Slip ratio

The evaporation rate for each axial region is

$$W_{evap} = Q_{drop} / h_{fg}$$

where

W_{evap} = Evaporation rate of the entrained liquid (lbm/sec)

Q_{drop} = Heat transfer rate from superheated steam to entrained droplets (Btu/sec)

h_{fg} = $h_g - h_f$ (Btu/lbm)

The liquid and steam flows are calculated at the end points of the steam generator tubes sections and are equal to

$$W_{liq,i+1} = W_{liq,i} - W_{evap,i}$$

$$W_{stm,i+1} = W_{stm,i} + W_{evap,i}$$

iv. De-superheating of Steam due to Interphase Heat Transfer to the Liquid Phase

The de-superheating of the steam due to the interphase heat transfer to the liquid phase is calculated by evaluating the axial steam delta temperature at the tube axial interfaces as follows:

$$\Delta T_{stm,qdrop} = -\, Q_{drop} / (W_{stm}\, Cp)$$

where Cp is the steam specific heat (Btu/lbm-°F).

v. De-superheating of the Steam due to Steam Generated by Evaporation of the Liquid Droplets

The de-superheating of the steam due to the generation of saturated steam due to evaporation of the liquid droplets is calculated by mixing the local steam flow at the local steam temperature with the evaporated steam at the saturation temperature at the tube axial interfaces as follows:

$$T_{stm,post} = (W_{stm,pre}\, Cp_{pre}\, T_{stm,pre} + W_{evap}\, Cp_{sat}\, T_{sat}) \;/\; (W_{stm,pre}\, Cp_{pre} + W_{evap}\, Cp_{sat})$$

where

$T_{stm,post}$ = Steam temperature after mixing (°F)

$W_{stm,pre}$ = Steam temperature prior to mixing (°F)

Cp_{pre} = Steam specific heat prior to mixing (Btu/lbm-°F)

$T_{stm,pre}$ = Steam temperature prior to mixing (°F)

W_{evap} = Steam evaporation rate (lbm/sec)

Cp_{sat} = Saturated steam specific heat (Btu/lbm-°F)

T_{sat} = Saturation temperature (°F)

2.4.2.3.2 *Simulation of the FLECHT-SEASET Steam Generator Tests - Steam Generator Inlet Quality Less than One*

The simulation of the FLECHT-SEASET steam generator tests described in Section 2.4.2.2, [

]

i. FLECHT-SEASET Test No. 20904 (Simulation with Inlet Quality equal to 0.798)

Test Case 20904 is a high pressure, low flow, mid quality test. The test comparison is run with the following boundary conditions (Reference 2.4-3, Page 20904-1 or Reference 2.4-2, Table 4-2A):

Boundary Condition	Value
SG Tubes Pressure (psia)	60.4
SG tubes total flow (lbm/sec)	0.494
SG tubes inlet quality	0.798
SG pressure (secondary side, psia)	850.
Secondary side void fraction	0.074

Test comparisons using the special COMPERC-II calculations are shown in Figures 2.4.2.3.2-1 through 2.4.2.3.2-4.

Figure 2.4.2.3.2-1 shows the primary side steam generator tube temperature response as a function of distance from the inlet every 150 seconds.

Figure 2.4.2.3.2-2 shows the secondary side fluid temperature along the steam generator tubes as a function of distance from the inlet every 150 seconds. The results are [ ] with test data (Reference 2.4-3, Page 20904-17). The steam generator secondary side fluid temperature at the tube exit [

] Measurements during the FLECHT-SEASET experiments showed that the steam exits the steam generator tubes wet.

Figure 2.4.2.3.2-3 shows the comparison of the calculated secondary side fluid temperature along the steam generator tubes at 1500 seconds against the test results from Reference 2.4-3,

Page 20904-17. Figure 2.4.2.3.2-3 shows that [

]

Figure 2.4.2.3.2-4 shows the transient response of the secondary side fluid temperature at the steam generator tube exit, and shows that the [

] The reference data is again plotted from the data in the figures of Reference 2.4-3, Page 20904-17.

Comparison of the COMPERC-II results [

]

ii. FLECHT-SEASET Test No. 22213 (Simulation with Inlet Quality equal to 0.797)

Test Case 22213 is a mid pressure, high flow, mid quality test. The test comparison is run with the following boundary conditions (Reference 2.4-3, Page 22213-1 or Reference 2.4-2, Table 4-2A):

Boundary Condition	Value
SG Tubes Pressure (psia)	40.0
SG tubes total flow (lbm/sec)	0.991
SG tubes inlet quality	0.797
SG pressure (secondary side, psia)	841.
Secondary side void fraction	0.037

Test comparisons using the special COMPERC-II calculations are shown in Figures 2.4.2.3.2-5 through 2.4.2.3.2-8.

The conclusions from the simulation of Test No. 22213 are [

]

Comparison of the special COMPERC-II results calculated here and the corresponding ones in Section 2.4.2.2 (results with inlet quality equal to one) against test data shows that [

]

2.4.2.3.3 Assessment of the Effect of Steam Generator Inlet Quality Equal to One Assumption on LBLOCA Analysis

The models of the 1999 EM COMPERC-II code are conservative EM component models. As such, the code simulates reflood thermal-hydraulics by implementing models with conservative assumptions. For example, the 1999 EM COMPERC-II code [

]

The LBLOCA cases were run with the 1999 EM COMPERC-II code activating the steam generator[

]

The following four cases were run to evaluate the effect[ ] with the 1999 EM automated/integrated code system and the secondary side model active with inlet quality less than or equal to one:

[

]

The inlet quality was held constant during the transient at the indicated values for each of these calculations. No energy balance calculation was done. The delta PCT results are given in the following table.

[

]

[

]

2.4.2.3.4 *Implementation of a De-entrainment Model in COMPERC-II*

The 2D/3D Test Program (Reference 2.4-7, Section 4.8) evaluated the effect on PCT of de-entrainment in the upper plenum and hot legs during the reflood of a LBLOCA, and concluded that PCT decreases as the de-entrainment increases. The results were calculated with TRAC using data from the UPTF carryover and steam binding tests as part of the evaluation of the effect of water carryover and steam binding during the reflood period of a LBLOCA. Figure 4.8-7 in Reference 2.4-7 (reproduced here as Figure 2.4.2.3.4-1) summarizes the results and shows:

- The best estimate percent carryover to the steam generator tubes is 20 to 30% (70 to 80% de-entrainment).
- There is 300°F conservatism in PCT between Appendix K modeling (100% carryover) and best estimate modeling (20 to 30% carryover).

Since the 1999 EM COMPERC-II code[

]

A[

]

The special de-entrainment model in COMPERC-II was implemented consistent with the 1999 EM COMPERC-II formulation and models. The special model is optionally activated through input. The following[

]

i. Downcomer and Lower Plenum Mass and Energy Equations.

The downcomer and lower plenum mass (Reference 1, Equation III.A.1-1) and energy equations (Reference 1, Equation III.A.2-1) were modified as follows:

[

]

The terms that were added to the 1999 EM conservation equations are the [

]

ii. Calculation of the De-Entrainment Flow

The de-entrainment flow in the upper plenum and hot legs is calculated by partitioning the total upper plenum flow into the loop flow and de-entrained flow. The total upper plenum flow for the 1999 EM version of COMPERC-II is

[

]

iii. Effect on PCT of the Special COMPERC-II De-Entrainment Model

Although [

] with the TRAC results on de-entrainment described in Reference 2.4-7. The study was performed with the following steps:

[

]

This special evaluation was made [

]

A summary of the delta PCT results calculated for the automated/integrated test cases is shown in Figure 2.4.2.3.4-2. The automated/integrated code system calculates [

]

Although there is no direct comparison between the 1999 EM automated/integrated test cases and the TRAC results, for purposes of evaluating trends, the TRAC delta PCT values described above are [

]

2.4.2.3.5 *Implementation of a Special Entrained Liquid Model at the Steam Generator Tube Outlet*

A special steam generator tube exit moisture model, which assumes user specified two-phase conditions at the steam generator tube exit, and is consistent with the superheated steam model described in Section 2.4.1.1, was implemented into the 1999 EM COMPERC-II code. As stated above, the[

]

For a user specified moisture content at the steam generator tubes outlet, the code calculates the specific volume of the mixture (for the frictional loop calculations), and the enthalpy of the mixture (for the mass - energy transfer to the containment) as follows:

$$v_{mix} = (W_{stm}\, v_{steam} + W_{liq}\, v_f) / (W_{stm} + W_{liq})$$

$$h_{mix} = (W_{stm}\, h_{steam} + W_{liq}\, h_f) / (W_{stm} + W_{liq})$$

where

v_{mix} = Mixture specific volume (ft^3/lbm)

v_{steam} = Steam specific volume (ft^3/lbm)

v_f = Saturated liquid specific volume (ft^3/lbm)

W_{stm} = Steam flow (lbm/sec)

W_{liq} = Liquid flow (lbm/sec)

h_{mix} = Mixture enthalpy (Btu/lbm)

h_{steam} = Steam enthalpy (Btu/lbm)

h_f = Saturated liquid enthalpy (Btu/lbm)

The moisture exiting the steam generator tubes [

]

The effect of assuming the presence of moisture at the steam generator exit was tested by activating the moisture model in COMPERC-II with a user-specified exit quality. The results of the comparison are shown in the following table, which shows that [

]

[

]

Although the margin on conservatism on PCT associated with the moisture content at the steam generator tubes outlet calculated by the 1999 EM COMPERC-II code [

]

2.4.3 Application to LBLOCA Analysis

The effect of implementing the steam generator model for LBLOCA calculations is obtained by comparing a reference test case using the 1985 EM and a case which activates the 1999 EM steam generator model. The 1999 EM is activated [

]

Table 2.4.3-1 shows that [

]

2.4.4 Applicability to LBLOCA Analysis

Justification of the applicability of the revised models to LBLOCA analyses is provided only for the model improvements which are explicitly implemented as part of the 1999 EM.

i. Applicability of the Steam Generator Secondary Side Model to LBLOCA Analysis

Section 2.4.1.1 provided a phenomenological description of the steam generator heat transfer during reflood of a LBLOCA. The FLECHT-SEASET steam generator tests provided test data, which supported this phenomenological description of the thermal-hydraulics for the primary and secondary side responses.

The 1999 EM steam generator model, which was implemented for the secondary side, [

]

The 1999 EM steam generator tube model, which was implemented, [

]

ii. Applicability of FLECHT-SEASET Test Data to LBLOCA Conditions during Reflood

Table 2.4.4-1 shows a comparison of the fluid conditions covered in the FLECHT-SEASET steam generator tests and the corresponding fluid conditions occurring during reflood for two typical Combustion Engineering designed PWRs. The table compares the steam velocity at the steam generator tube inlet, assuming [

]

Table 2.4.4-1 shows that the FLECHT-SEASET fluid test conditions replicate the calculated steam velocity conditions during the reflood of a LBLOCA for a PWR. Thus, the COMPERC-II analyses and comparisons of the FLECHT-SEASET tests support the conclusion that the 1999 EM COMPERC-II steam generator model [

]

iii. Applicability of the Steam Generator Tubes Heat Transfer Correlations to LBLOCA Analysis

a. The [ ] Correlation

The [ ] correlation was derived from the experimental data obtained by Weise and Saunders for short vertical plates (Reference 2.4-5). The correlation fits the data presented by the author to within 5% and is given as follows:

[

]

Minkowycs and Sparrow (Reference 2.4-6) obtained analytical solutions for flow over vertical cylinders when the above criteria is not met. They have shown that for[

]

b. The[ ]Correlation

The [ ] correlation is used to calculate heat transfer to the steam phase in the steam generator tubes. The correlation was developed from data for heating and cooling in tubes in Reference 2.4-4. Its applicability to the turbulent region (Re > 6000) has been confirmed by experiments to within ±25% (Reference 2.4-11).

2.4.5 <u>Model as Coded</u>

The steam generator tube model coding follows the description of the model in Sections 2.4.1.3 (iii) and 2.4.2.3.1. The steam generator secondary side model coding follows the description of the model in Sections 2.4.1.2 and 2.4.1.3 ((i) and (ii)). The [ ] correlation (Section 2.1.4.3 (v)) and the[ ]correlation (Section 2.1.4.3 (iv)) are implemented as described. The steam generator resistance to steam venting is corrected for density as described in the COMPERC-II topical report, Reference 2.4-1, Page E-2. For the 1999 EM, the density used is the inverse of the steam generator tubes average specific volume, which is calculated as follows:

[

]

2.4.6 References

2.4-1 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

2.4-2 "PWR FLECHT-SEASET Steam Generator Separate-Effects Task, Data Analysis and Evaluation Report," EPRI NP-1461, NUREG/CR-1534, WCAP-9724, February 1982.

2.4-3 "PWR FLECHT-SEASET Steam Generator Separate-Effects Task, Data Report," NRC/EPRI/Westinghouse Report No. 4.

2.4-4 Dittus, F. W., and Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," Pub. in Eng., Vol. 2, n. 13, Univ. of California, pp. 443 - 461, 1930.

2.4-5 McAdams, W. H., "Heat Transmission," 3d, McGraw-Hill Book Company, Inc., 1954.

2.4-6 Karlekar, B. V., and Desmond R. M., "Engineering Heat Transfer," West Publishing Company, 1977.

2.4-7 "Reactor Safety Issues Resolved by the 2D/3D Program," NUREG/IA-0127, GRS-101, MPR-1346, July 1993.

2.4-8 "Small Break LOCA Realistic Evaluation Model," Volume 1, Part 1, Calculational Models, CEN 420-P Vol. 1, Pt. 1, February 1993.

2.4-9 "Reflood Heat Transfer THERM: A Thermal-Hydraulic Emergency Reflood Model," CENPD-228-P, January 1977.

2.4-10 Wallis, Graham B., "One Dimensional Two-Phase Flow," McGraw-Hill Book Co., New York, 1969.

2.4-11 Kreith F. and Bohn M. S., "Principles of Heat Transfer," 4th edition, New York, Harper and Row, 1986.

2.4-12 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

Table 2.4.2.2-1

FLECHT-SEASET Steam Generator Tests (Reference 2.4-2, Table 4-1A)

	Test Tube Conditions (1)			SG Secondary Side Initial Conditions (2)	
Test No.	**Total Flow lbm/sec**	**Inlet Quality**	**Pressure psia**	**Temperature °F**	**Level ft**
20904	0.494	0.798	60.4	525	32.4
21001	0.499	0.779	40.0	525	32.5
21121	0.488	0.801	40.1	267	32.1
21711	0.494	0.800	40.9	520	32.4
21806	0.500	0.200	41.0	520	32.7
21909	0.946	0.105	40.5	525	33.9
22010	0.503	0.801	40.0	525	33.6
22112	0.496	0.799	40.3	523	34.0
22213	0.991	0.797	40.0	524	33.7
22314	0.499	0.495	40.1	526	34.2
22415	0.614	0.345	40.0	524	33.6
22503	0.494	0.798	19.9	525	33.6
22608	0.495	0.796	40.0	525	7.7
22701	0.495	0.798	40.0	524	33.0
22920	0.493	1.000	39.8	523	32.5
23005	0.754	0.671	41.3	522	33.3
23207	0.504	0.801	39.9	400	32.2
23315	0.495	0.201	39.9	523	32.3
23402	0.989	0.799	40.0	523	32.1
23605	0.494	0.496	39.8	523	32.6

Notes:

(1) Tube boundary conditions held constant for the duration of the test

(2) Secondary side starting point conditions

Table 2.4.3-1

Effect of Activating the 1999 EM Steam Generator Model in COMPERC-II

[

]

Table 2.4.4-1

Comparison of Steam Velocity during Reflood for CE PWRs and FLECHT-SEASET Tests

CE PWRs					
	Pressure psia	Steam Loop Flow lbm/sec	Steam Specific Volume ft^3/lbm	SG Flow Area ft^2	Steam Velocity ft/sec
FLECHT-SEASET Tests					
20904	60.4	0.494	7.13	0.1048	33.6
22213	40	0.991	10.5	0.1048	99.3
22920	39.8	0.493	10.55	0.1048	49.6

Figure 2.4.2.1-1
CONTAINMENT PRESSURE

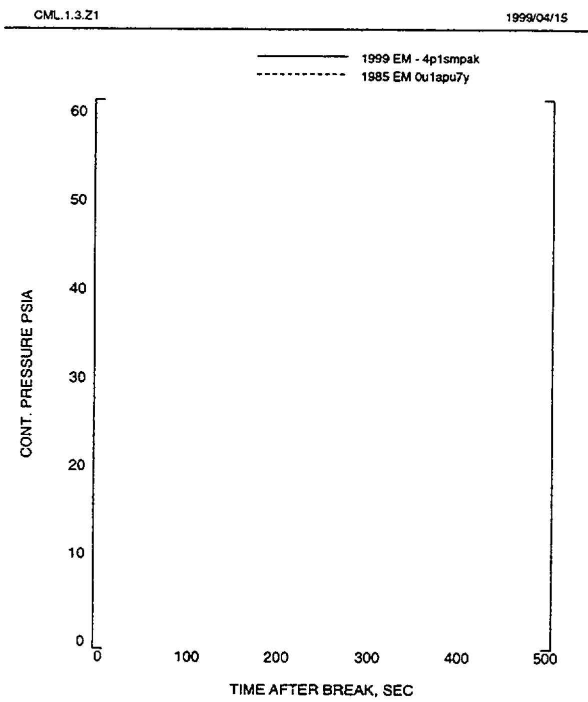


Figure 2.4.2.1-2
LEVEL IN DOWNCOMER

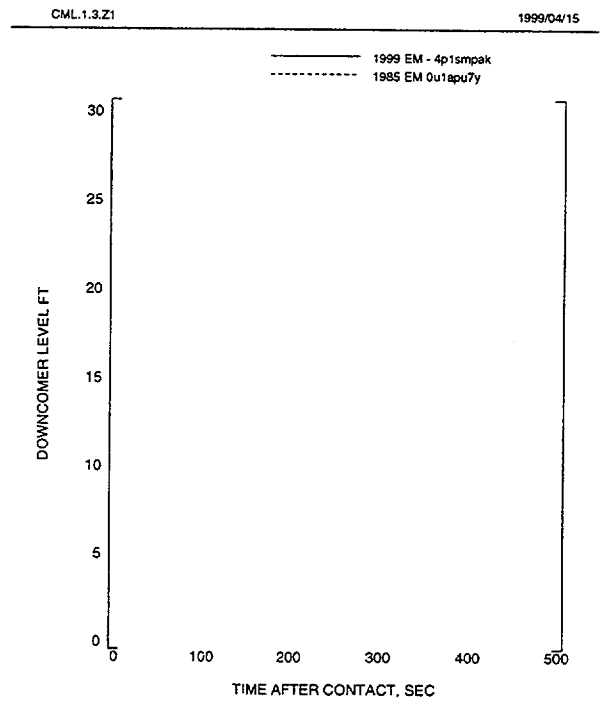


Figure 2.4.2.1-3
SUBCOOLED LIQ. LEVEL

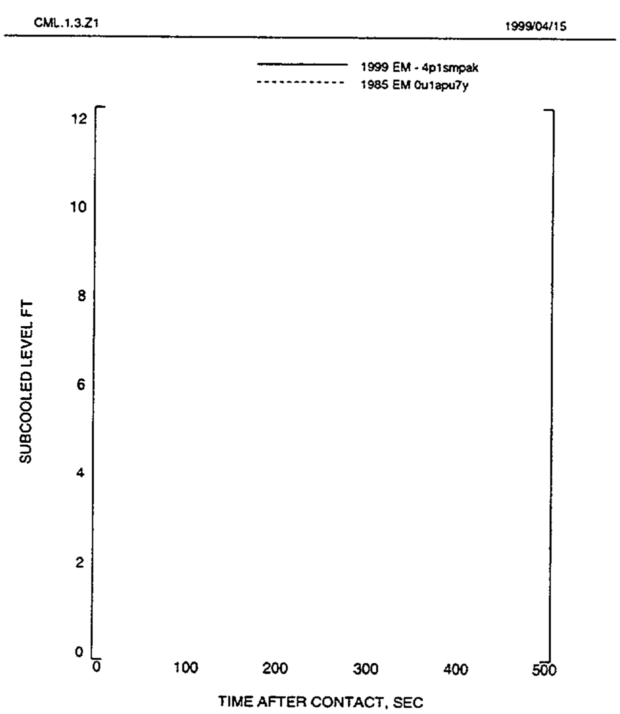


Figure 2.4.2.1-4
TWO PHASE LEVEL

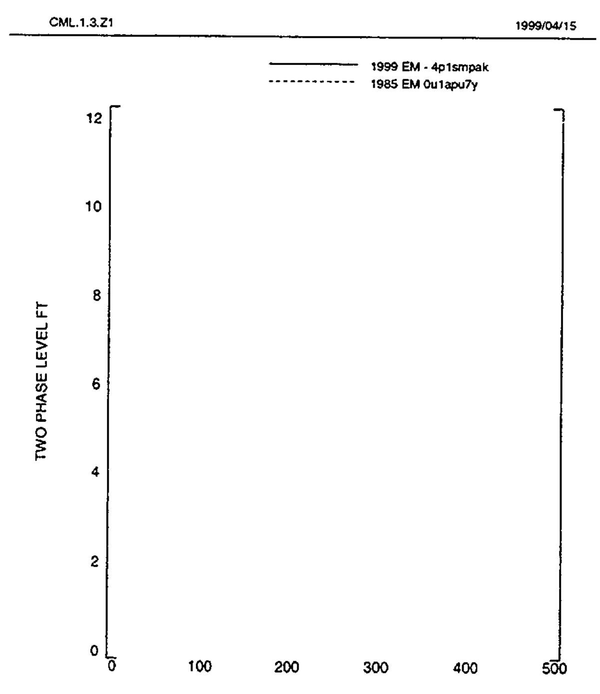


Figure 2.4.2.1-5
UPPER PLENUM PRESSURE

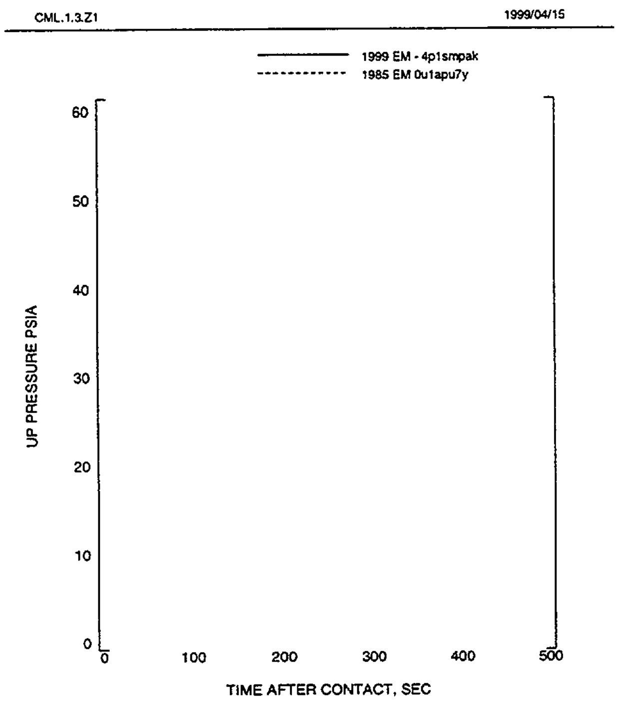


Figure 2.4.2.1-6
REFLOOD LIQ. MASS

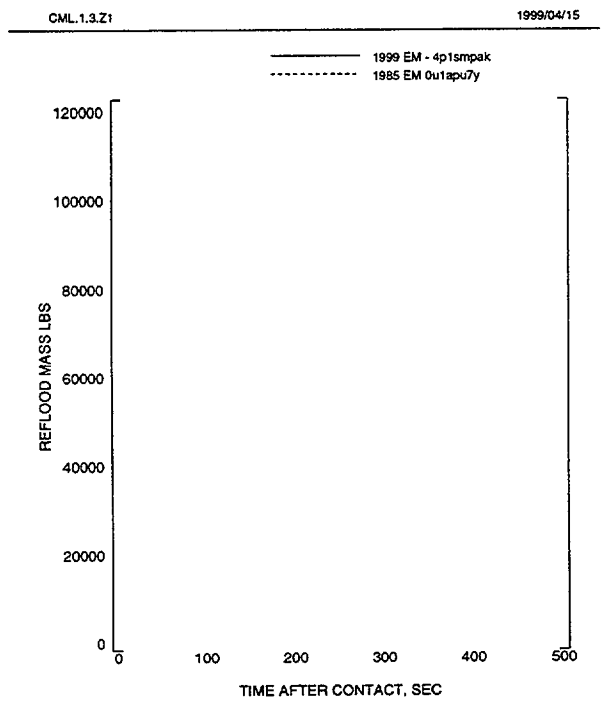


Figure 2.4.2.1-7
STEAM FLOW FROM U.P.

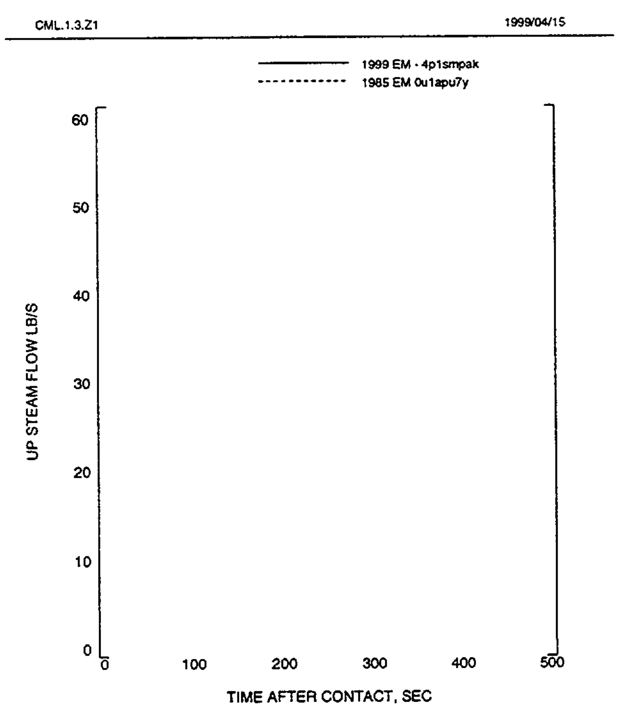


Figure 2.4.2.1-8
STEAM FLOW IN HOT LEG

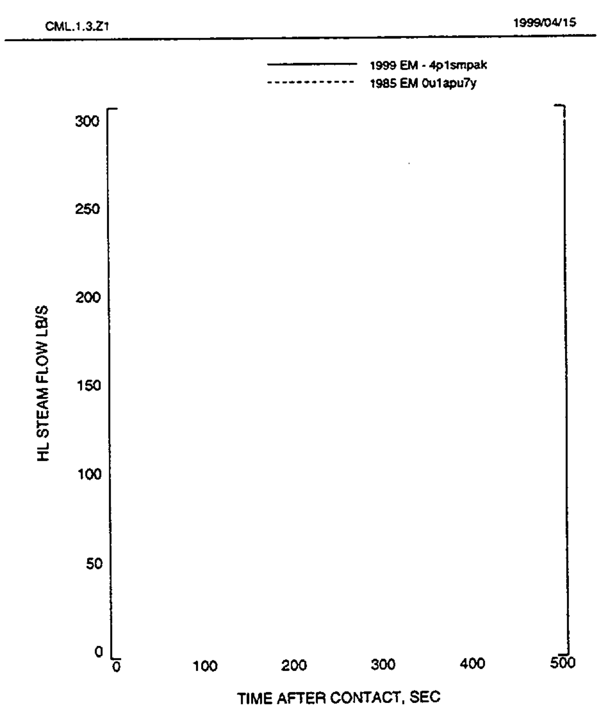


Figure 2.4.2.1-9
INTEGRAT. STEAM FLOWIN HOT LEG

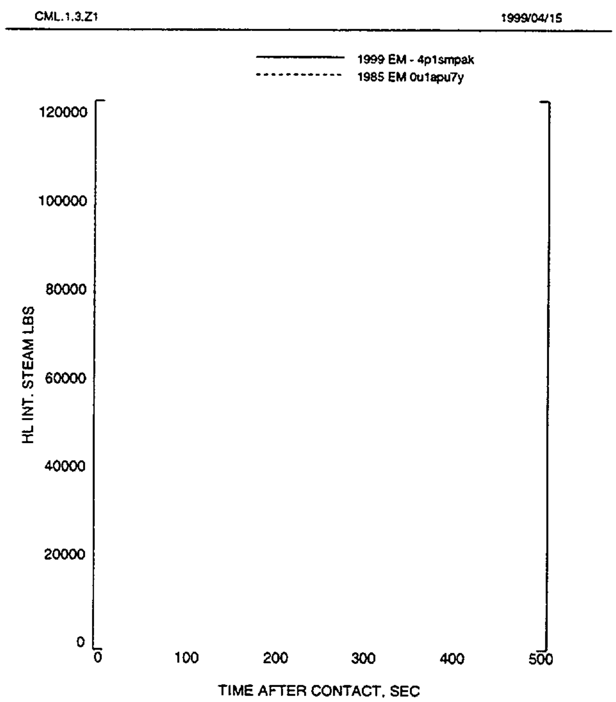


Figure 2.4.2.1-10
INTEGRAT. STEAM FLOWFROM U.P.

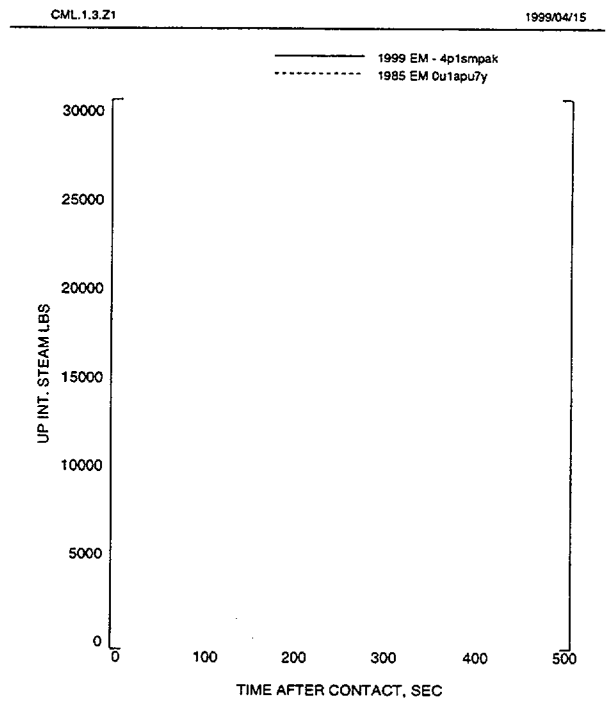


Figure 2.4.2.1-11
EQUIVALENT K-FACTOR

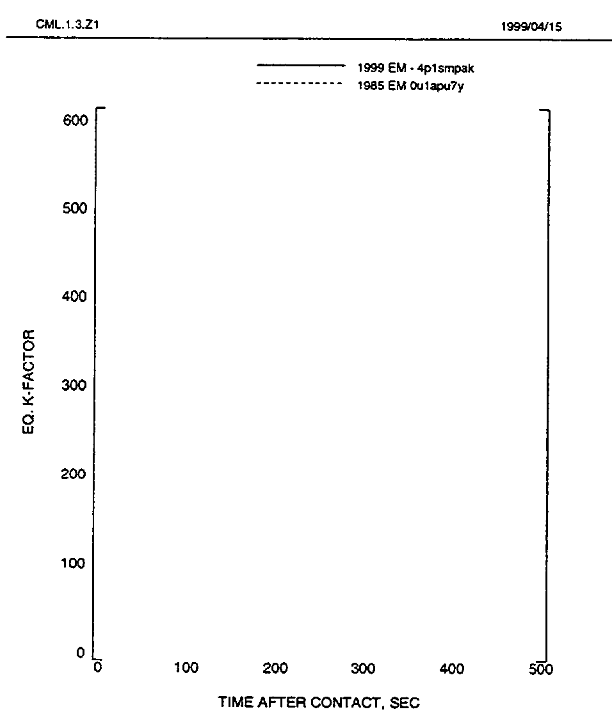


Figure 2.4.2.1-12
INSTANTANEOUS REFLOOD RATE

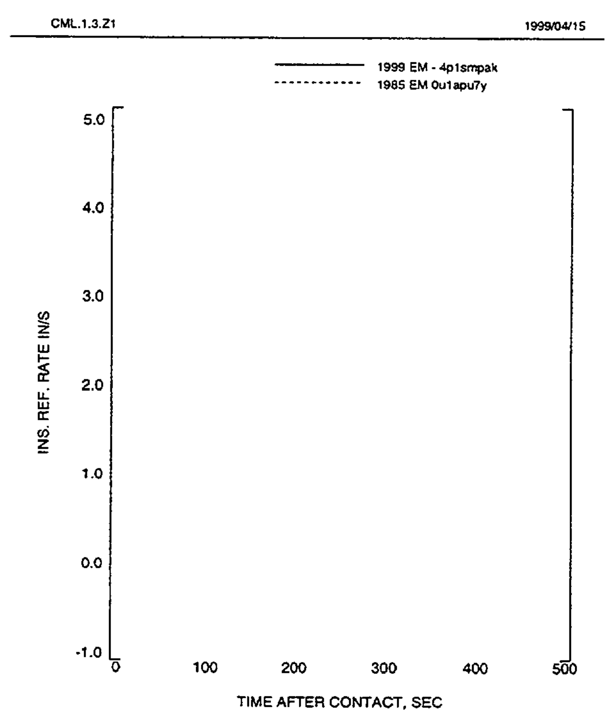


Figure 2.4.2.1-13

SG Secondary Side Pressure

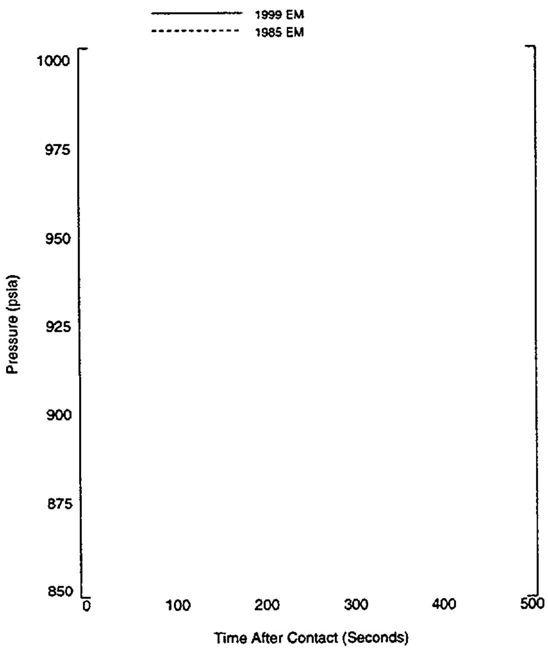


Figure 2.4.2.1-14

Primary Side SG Tube Exit Temperature

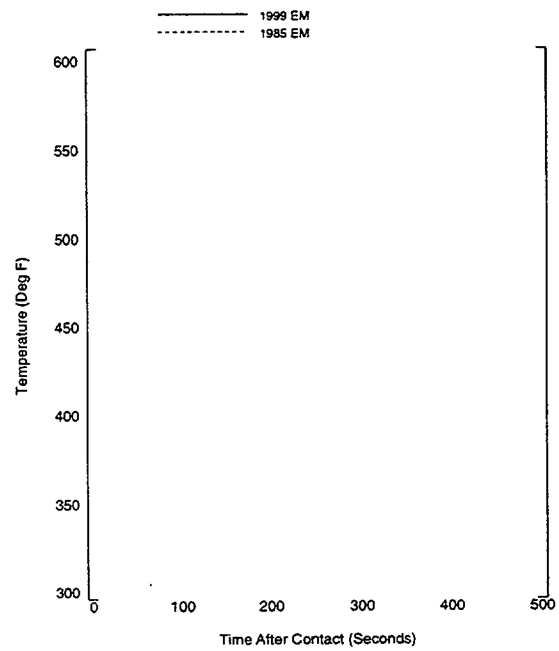


Figure 2.4.2.1-15

SG Sec Side Fluid Temperature at Tube Sheet

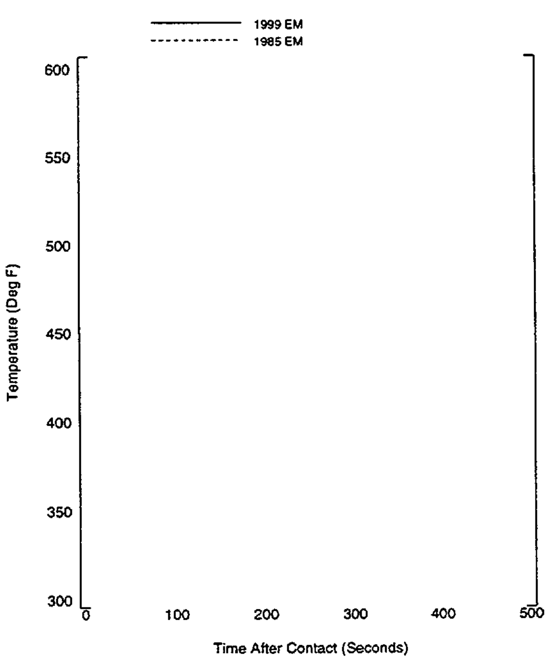


Figure 2.4.2.1-16

SG Tube Exit Steam Specific Volume

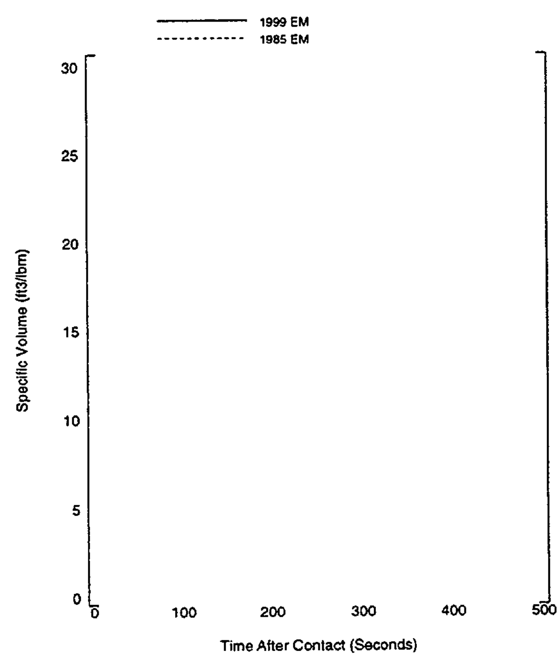


Figure 2.4.2.2-1
FLECHT-SEASET Test 20904
Primary Side SG Tube Axial Temperature
COMPERC-II Calculation (Every 150 seconds)

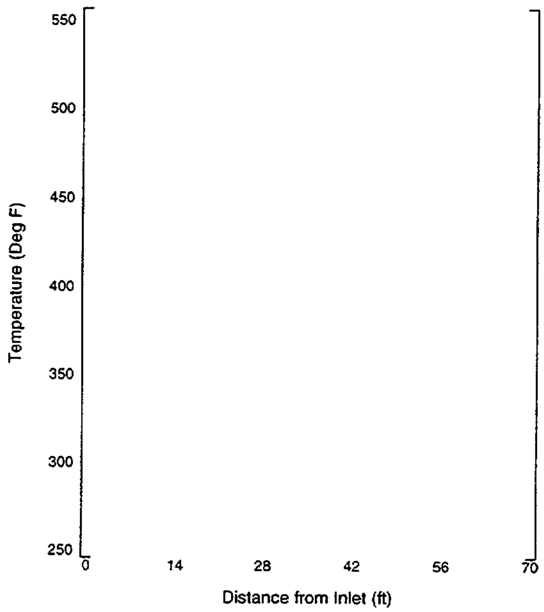


Figure 2.4.2.2-2
FLECHT-SEASET Test 20904
Secondary Side Fluid Temperature Along SG Tube
COMPERC-II Calculation (Every 150 seconds)

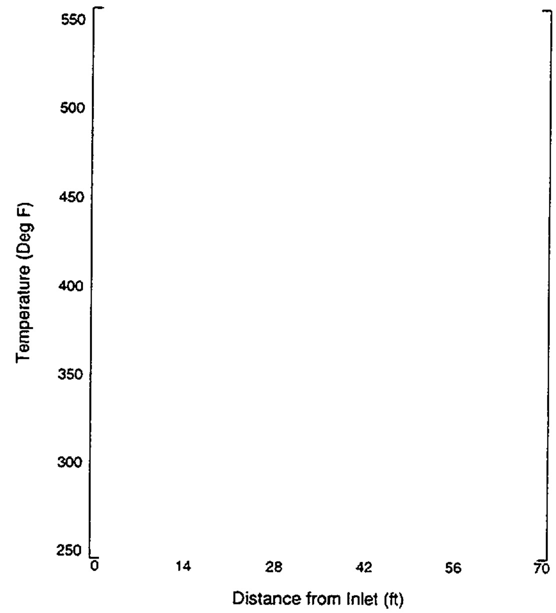


Figure 2.4.2.2-3
FLECHT-SEASET Test 20904
Secondary Side Fluid Temperature along SG Tube
Comparison to COMPERC-II Calculation (1500 Seconds)

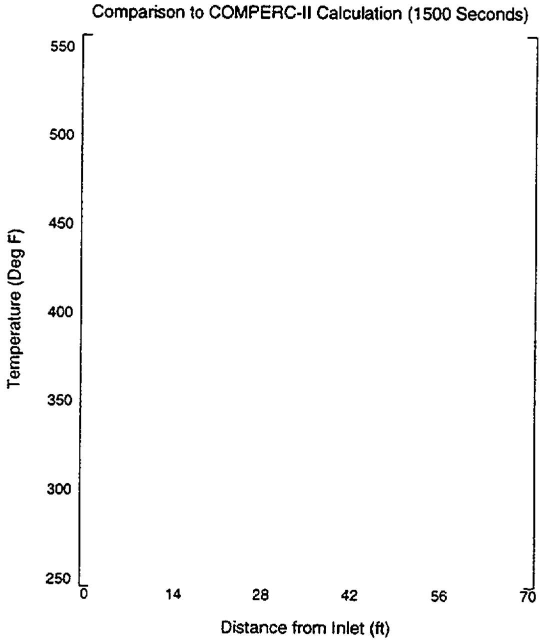


Figure 2.4.2.2-4
FLECHT-SEASET Test 20904
Secondary Side Fluid Temperature (at tube exit)
COMPERC-II Calculation

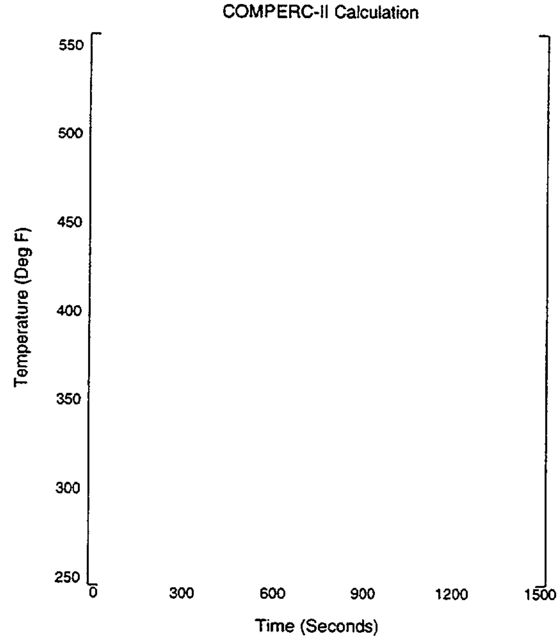


Figure 2.4.2.2-5
FLECHT-SEASET Test 22213
Primary Side SG Tube Axial Temperature
COMPERC-II Calculation (Every 150 seconds)

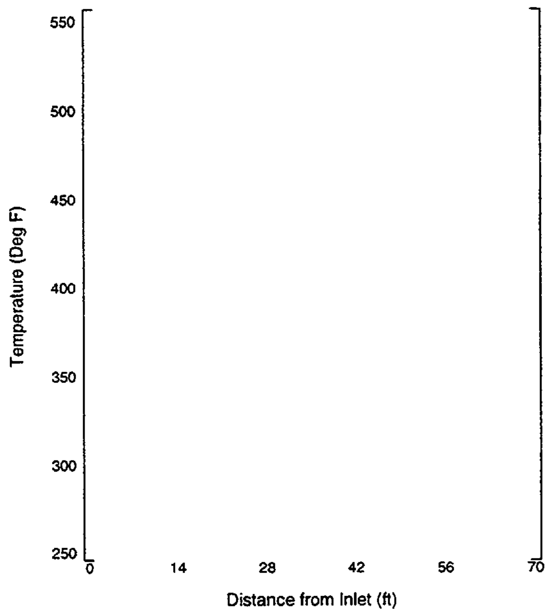


Figure 2.4.2.2-6
FLECHT-SEASET Test 22213
Secondary Side Fluid Temperature Along SG Tube
COMPERC-II Calculation (Every 150 seconds)

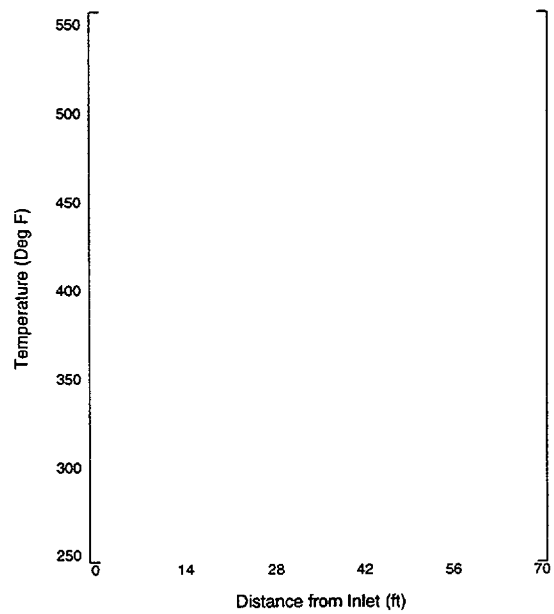


Figure 2.4.2.2-7
FLECHT-SEASET Test 22213
Secondary Side Fluid Temperature along SG Tube
Comparison to COMPERC-II Calculation (1500 Seconds)

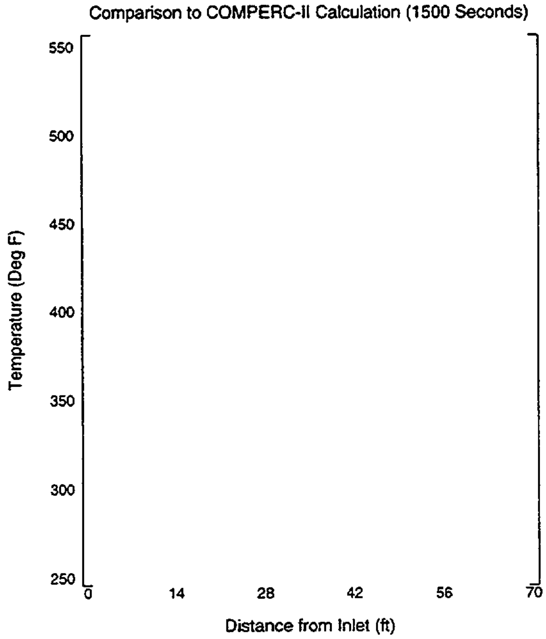


Figure 2.4.2.2-8
FLECHT-SEASET Test 22213
Secondary Side Fluid Temperature (at tube exit)
COMPERC-II Calculation

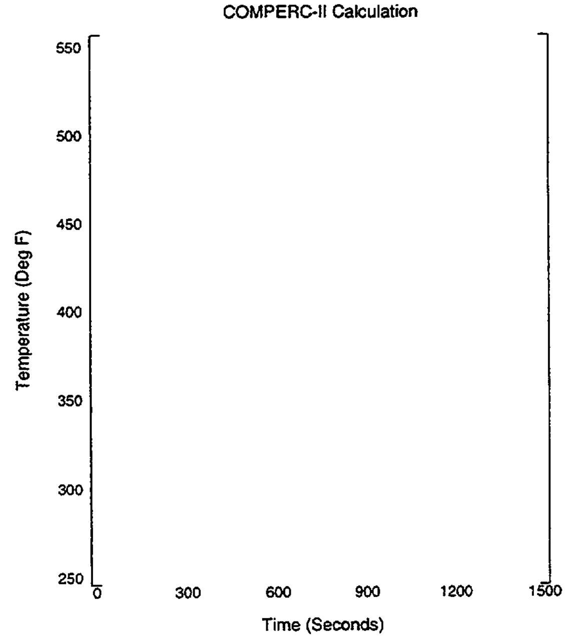


Figure 2.4.2.2-9

FLECHT-SEASET Test 22920

Primary Side SG Tube Axial Temperature

COMPERC-II Calculation (Every 150 seconds)

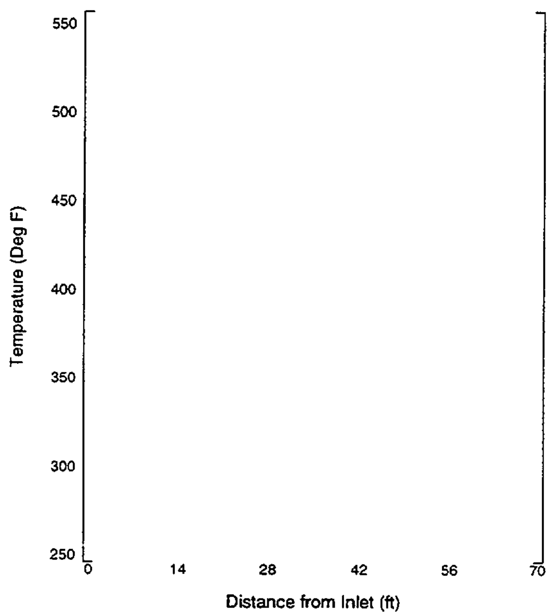


Figure 2.4.2.2-10

FLECHT-SEASET Test 22920

Secondary Side Fluid Temperature Along SG Tube

COMPERC-II Calculation (Every 150 seconds)

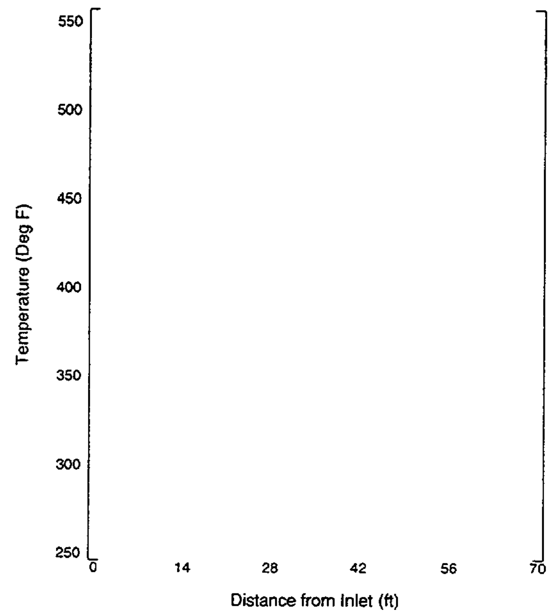


Figure 2.4.2.2-11

FLECHT-SEASET Test 22920

Secondary Side Fluid Temperature along SG Tube

Comparison to COMPERC-II Calculation (1500 Seconds)

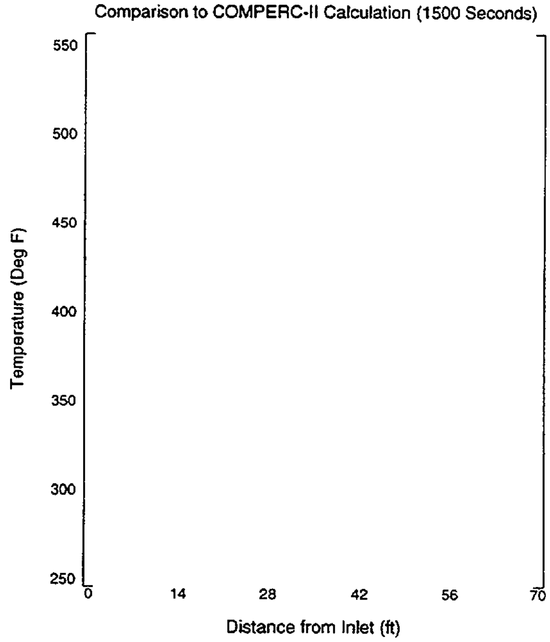


Figure 2.4.2.2-12

FLECHT-SEASET Test 22920

Secondary Side Fluid Temperature (at tube exit)

COMPERC-II Calculation

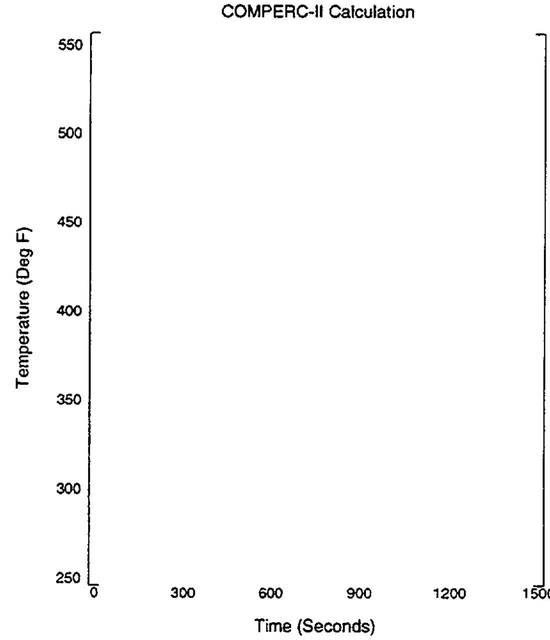


Figure 2.4.2.3.2-1

FLECHT-SEASET Test 20904 Xin (two-phase)

Primary Side SG Tube Axial Temperature

COMPERC-II Calculation (Every 150 seconds)

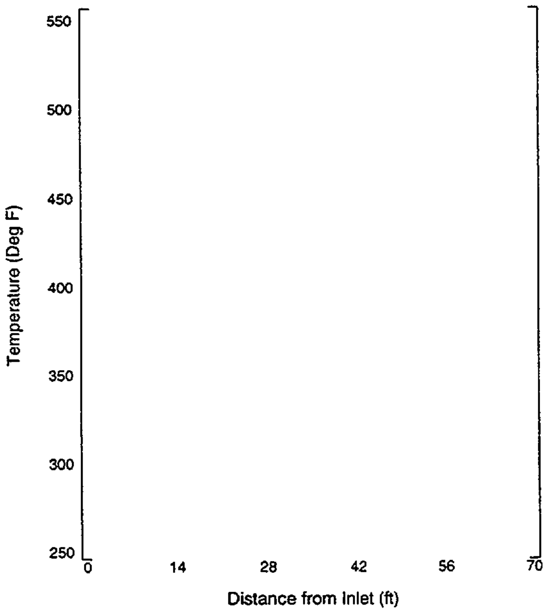


Figure 2.4.2.3.2-2

FLECHT-SEASET Test 20904 Xin (two-phase)

Secondary Side Fluid Temperature Along SG Tube

COMPERC-II Calculation (Every 150 seconds)

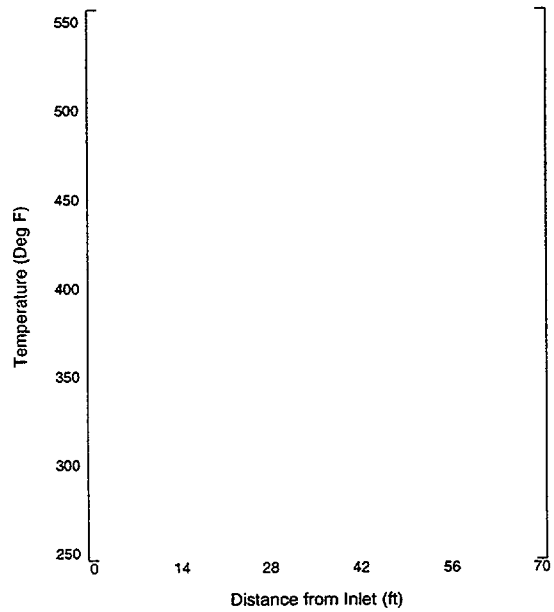


Figure 2.4.2.3.2-3

FLECHT-SEASET Test 20904 Xin (two-phase)

Secondary Side Fluid Temperature along SG Tube

Comparison to COMPERC-II Calculation (1500 Seconds)

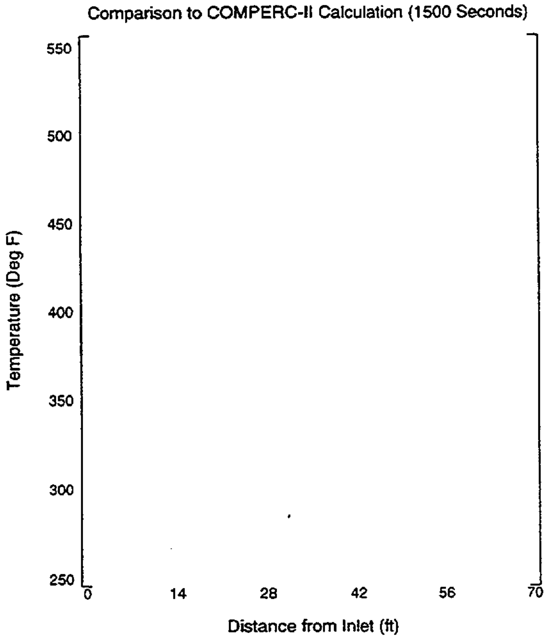


Figure 2.4.2.3.2-4

FLECHT-SEASET Test 20904 Xin (two-phase)

Secondary Side Fluid Temperature (at tube exit)

COMPERC-II Calculation

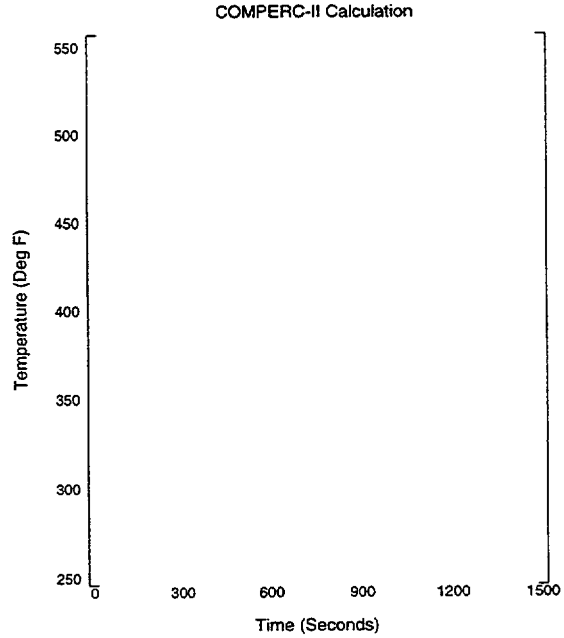


Figure 2.4.2.3.2-5
FLECHT-SEASET Test 22213 Xin (two-phase)
Primary Side SG Tube Axial Temperature
COMPERC-II Calculation (Every 150 seconds)

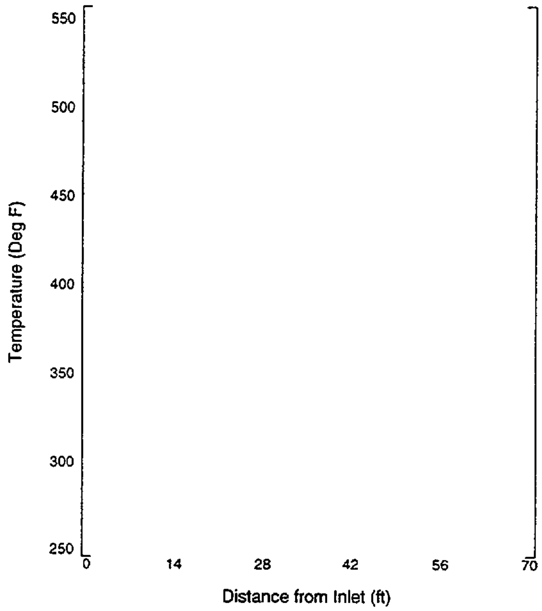


Figure 2.4.2.3.2-6
FLECHT-SEASET Test 22213 Xin (two-phase)
Secondary Side Fluid Temperature Along SG Tube
COMPERC-II Calculation (Every 150 seconds)

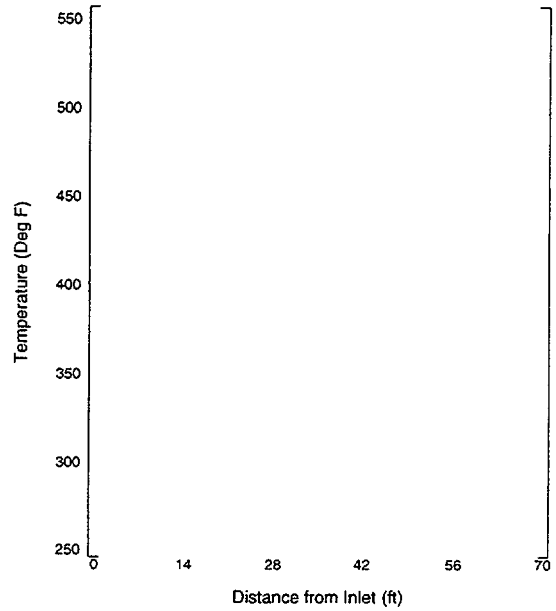


Figure 2.4.2.3.2-7
FLECHT-SEASET Test 22213 Xin (two-phase)
Secondary Side Fluid Temperature along SG Tube
Comparison to COMPERC-II Calculation (1500 Seconds)

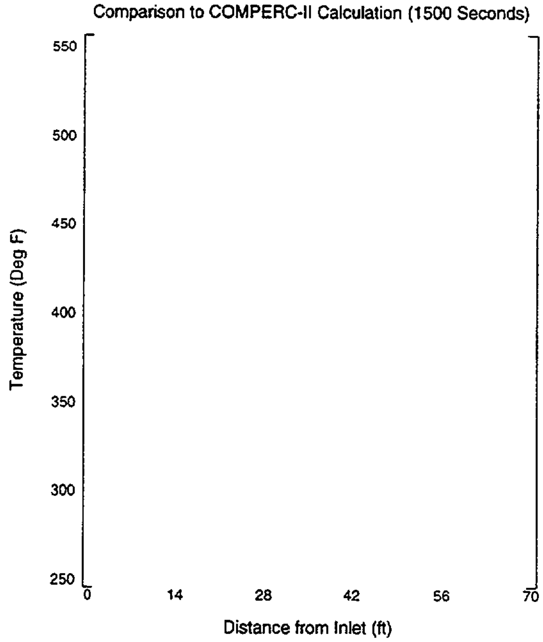


Figure 2.4.2.3.2-8
FLECHT-SEASET Test 22213 Xin (two-phase)
Secondary Side Fluid Temperature (at tube exit)
COMPERC-II Calculation

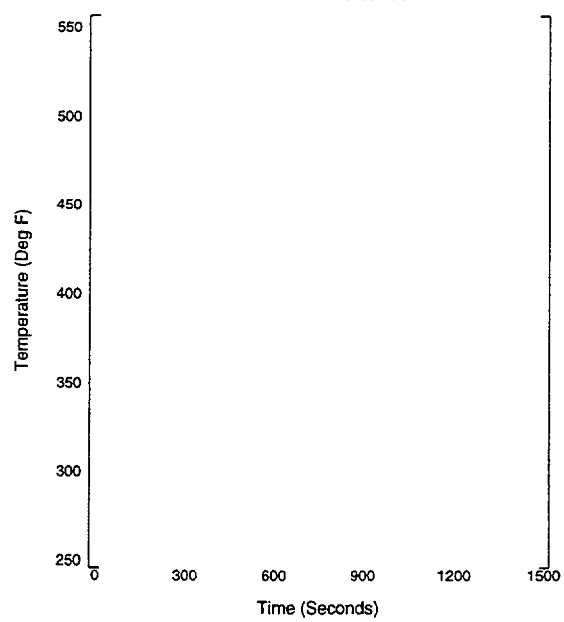


Figure 2.4.2.3.4-1

Effect of Entrainment on PCT (TRAC)

Delta PCT vs. Percent carryover to SG Tubes

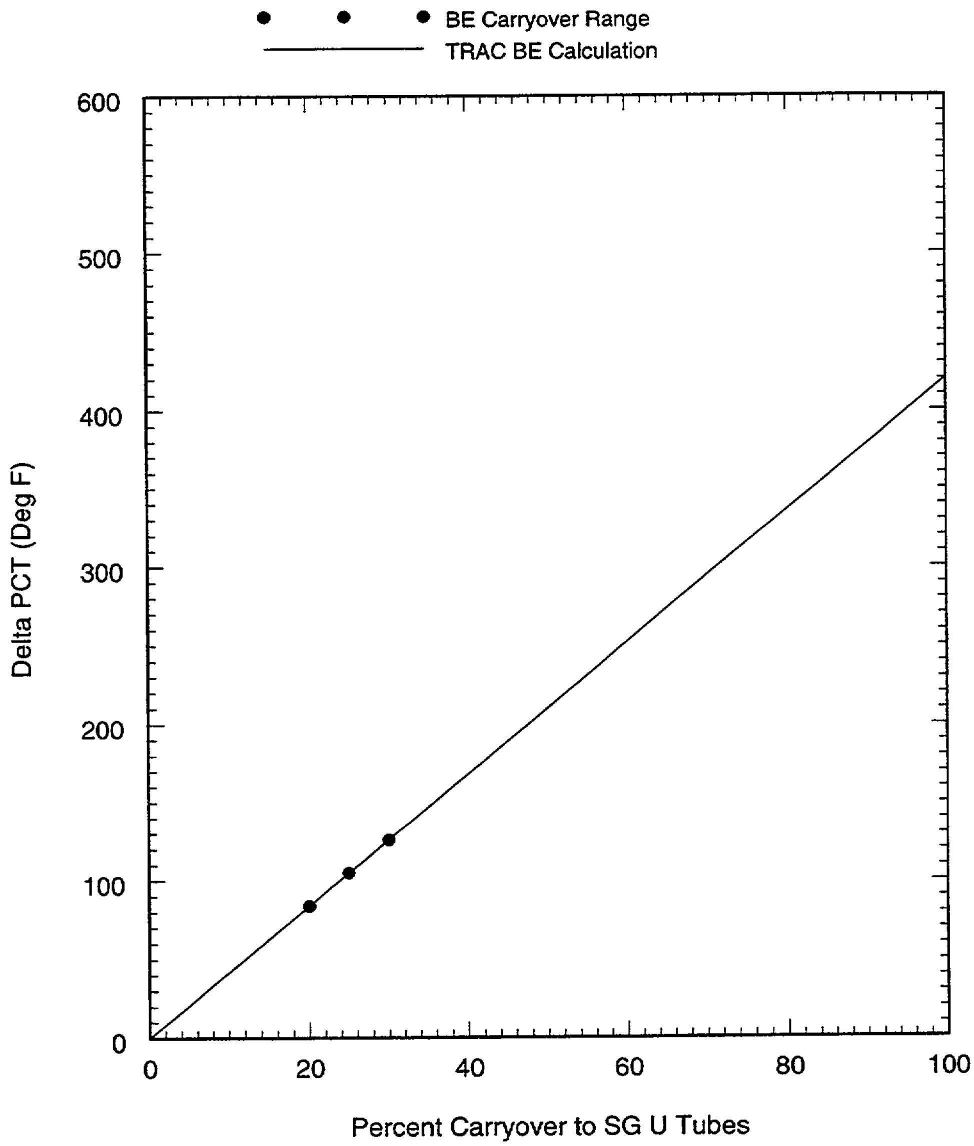
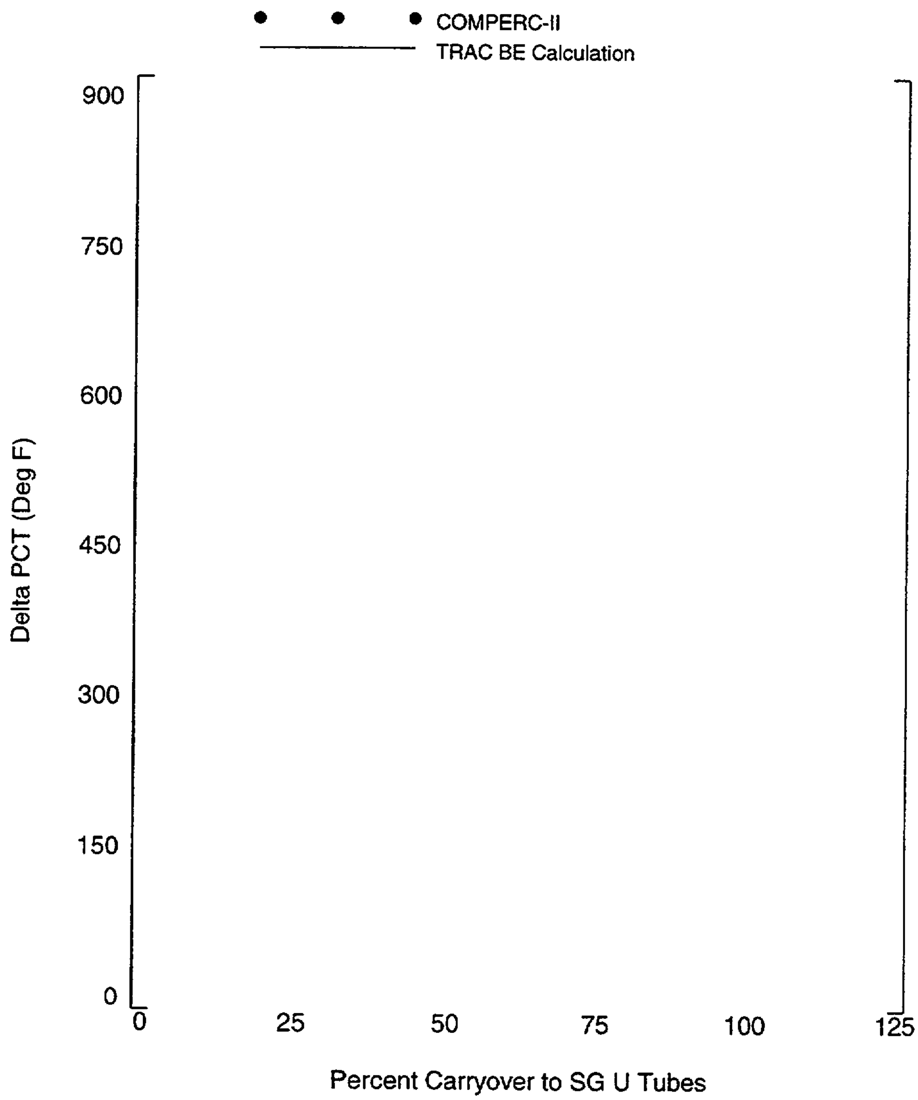


Figure 2.4.2.3.4-2

Effect of Entrainment on PCT

COMPERC-II vs. TRAC BE Calculation



2.5 COMPERC-II Steam/Water Interaction during Nitrogen Discharge

This section describes the implementation of improvements to the steam/water interaction model used during nitrogen discharge. The improvements are designed to remove conservatism in the 1985 EM, which is based on [

]

The 1985 EM COMPERC-II model (Reference 2.5-1) for the calculation of the injection section delta pressure during the reflood portion of a LBLOCA is as follows:

[

]

This model conservatively calculates [

.] The 1999 EM modifies the momentum delta pressure calculation during nitrogen injection in a manner consistent with the licensing bounding delta pressure drop requirements during SIT discharge and ECCS pump injection.

2.5.1 Model Description

2.5.1.1 Historical Background

The 1985 EM for steam interaction with ECCS water during the reflood period of a LBLOCA has been extensively reviewed by the NRC. The model is documented in CENPD-132, Volume 1 (Section III.D.5), Supplement 1 to CENPD-132 (Section III.D.5), and Supplement 2 to CENPD-132 (Supplement to Section III.D.5), Reference 2.5-2. Responses from the NRC

are documented in NRC SERs (References 2.5-3 and 2.5-4). The COMPERC-II implementation of the model for steam interaction with ECCS water during the reflood period of a LBLOCA is documented in Reference 2.5-1, Appendix J.

CENPD-132 Volume 1, Section III.D.5 (Reference 2.5-2) included test results from one-fifth and one-third scale models of the cold leg piping, and showed that[

]

For SIT discharge, Supplement 1 to CENPD-132, Section III.D.5.a.1 (Reference 2.5-2) provided additional analytical arguments to support[

]

Supplement 2 to CENPD-132 (Page S/2 III.D.5-2) showed that[

] Supplement 2 to CENPD-132 implemented a partial condensation slip model and demonstrated that[

]

[

]

2.5.1.2 Evaluation of the Delta Pressure Calculated by the 1985 EM

As described earlier, the 1985 EM model for calculation of the differential pressure for the ECCS water - steam interaction during the refill/reflood periods of a LBLOCA is calculated in COMPERC-II as follows:

[

]

The delta pressure calculated by the 1985 EM during the period of nitrogen discharge is calculated by the following momentum balance (Reference 2.5-1, Appendix J). The equation is rewritten to support the evaluation of the delta pressure, which follows:

[

]

The liquid and nitrogen are assumed to flow [ ] in the safety injection line upstream of the cold leg. Liquid, steam and nitrogen are assumed to flow [

.]

The total injection section delta pressure, which is calculated by the 1985 EM for a typical reflood case, is shown in Figure 2.5-1. Nitrogen release begins at [

] This figure shows the following analysis result characteristics:

[

•

]

The reason for the 1985 EM response noted above can be seen from the liquid, steam, and nitrogen velocity values during the period of nitrogen blowdown displayed in Figure 2.5-2. This figure shows the following characteristics:

[

]

The steam velocity is relatively flat during this period of time. The mixture velocity changes more but it is also relatively flat. The nitrogen flow decreases significantly as the nitrogen blowdown progresses while the ECCS pump liquid flow remains essentially constant. Since, the mixture of liquid and nitrogen enters the cold legs[ ] the 1985 EM requires the nitrogen / liquid mixture to be accelerated from smaller velocities (due to the smaller nitrogen flows) to an essentially large constant mixture velocity. Thus the change in the liquid momentum is[ ]at the end of the nitrogen blowdown.

The [

]

2.5.1.3 Evaluation of the Delta Pressure Calculated by the 1999 EM

As described in Section 2.5.1.2, the differential pressure (psid) calculated by the 1985 EM during the period of nitrogen blowdown is [ ] Since delta pressure values measured in experimental tests [ ] in delta pressure calculated by the 1985 EM during the period of nitrogen blowdown, the 1999 EM calculation for the delta pressure during the period of nitrogen blowdown is revised to be consistent with [ ] The 1999 EM revision is made by [ ] The delta pressure is calculated as follows:

[ ]

The transition from the delta pressure defined during the period of SIT discharge to the delta pressure calculated during nitrogen discharge is required to be [ ] Therefore, the model includes the requirement that the initial injection section delta pressure during nitrogen discharge be [ ] as the delta pressure value used during SIT discharge.

The cold leg mixture velocity used in Equation (2.5-1) is calculated as follows:

$$Vel_{mix} = W_{mix}\, sv_{mix} / A \quad (2.5\text{-}2)$$

where

$$W_{mix} = W_{liq} + W_{stm} + W_{nit} \quad (2.5\text{-}3)$$

$$sv_{mix} = W_{stm}*sv_{stm}/W_{mix} + W_{nit}*sv_{nit}/W_{mix} + W_{liq}*sv_{liq}/W_{mix} \quad (2.5\text{-}4)$$

A = cold leg flow area (ft^2)

In these equations,

W = Mass flow rate (lbm/sec)

sv = Specific volume (ft^3/lbm)

mix = Mixture (liquid, steam and nitrogen)

liq = Liquid

stm = Steam

nit = Nitrogen

The nitrogen / liquid velocity used in Equation (2.5-1) in the safety injection line is calculated as follows:

$$Vel_{liq,nit} = W_{liq,nit} * sv_{liq,nit} / A_{SI} \quad (2.5\text{-}5)$$

where

$$W_{liq,nit} = W_{liq} + W_{nit} \quad (2.5\text{-}6)$$

$$sv_{liq,nit} = W_{liq}*sv_{liq}/W_{liq,nit} + W_{nit}*sv_{nit}/W_{liq,nit} \quad (2.5\text{-}7)$$

A_{SI} = safety injection line flow area (ft^2)

The total injection section delta pressure which is calculated by the 1999 EM for a typical reflood calculation is shown in Figure 2.5-1 (solid line). [

]

The resulting equivalent k-factor for the 1999 EM for a typical LBLOCA reflood calculation is shown in Figure 2.5-3. The 1999 EM calculates a k-factor that is [ ]

The primary piping steam flow calculated with the 1999 EM is shown in Figure 2.5-4. The significance of these results is [ ]

2.5.2 <u>Model Assessment</u>

The following is a series of observations and conclusions that support the performance of the 1999 EM revision of the injection section delta pressure during nitrogen blowdown for Appendix K ECCS performance analyses:

- Results documented in CENPD-132 and in Supplements 1 and 2 to CENPD-132 (Reference 2.5-2) for one-third and one-fifth scale experimental data, and for EPRI / Westinghouse 1/14 experimental tests, showed that [ ]
- As described in Section 2.5.1.3, the 1985 EM calculates [ ]
- In addition, as noted in Section 2.5.1.3, the delta pressure calculation by the 1985 EM [ ]
- The 1999 EM model is [ ]
- The 1999 EM formulation is [ ]

- The effect of nitrogen discharge on the reflood of a LBLOCA was addressed in the 2D/3D Test Program and in the Achilles Program. The 2D/3D test program ran UPTF tests (Test C1-15 and 27A) to simulate nitrogen discharge during reflood (Reference 2.5-6, Section 4.4). Also the 2D/3D program included TRAC simulations of the tests, and PWR LBLOCA TRAC analyses to evaluate the effect of nitrogen injection in ECCS performance calculations. The Achilles Test program ran Test Run A1B105 (Reference 2.5-7).

Observations and conclusions from the 2D/3D test program based on test data and on the TRAC code simulation of the tests relevant to the effect of nitrogen discharge during reflood in a LBLOCA are discussed in Reference 2.5-6, Section 4.4. These calculations and observations from the TRAC analyses show that as nitrogen begins to flow into the cold legs it flows much faster than the preceding water because the pressure losses in the piping are less for the lower density gas. The nitrogen quickly pushes ECCS water from the intact cold legs into the reactor vessel downcomer. Also water in the top of the downcomer and in the broken cold leg is pushed towards the break. The primary system is locally pressurized, for a short period until the nitrogen can leave the system. Suppression of steam condensation due to the presence of nitrogen causes a further increase of the downcomer pressure.

The nitrogen pressurization of the downcomer forces water in the downcomer into the lower plenum, displacing lower plenum water into the core. As water surges into the core additional steam is produced there. The increased steam production increases the pressure in the upper plenum, which stops the rise in core water and forces some of the water out of the core and back into the lower plenum. However more water remains in the core than was present before the nitrogen-induced surge. That is, the net effect of the nitrogen discharge process is a net addition of water into the core (see Table 4.4-3 and Figure 4.4-8 in Reference 2.5-7).

Although the UPTF tests did not simulate the effects of nitrogen discharge on core cooling (the one test malfunctioned and the other was terminated early), TRAC PWR analyses suggest that nitrogen discharge and the resulting surge in the core water level are beneficial to core cooling (Reference 2.5-7, Section 4.4).

In the Achilles test, the surge of water into the core also enhanced core cooling and temporarily increased steam generation. Manometer oscillations between the core and the downcomer occurred for about 50 seconds.

COMPERC-II [

] Since the 2D/3D test program shows the opposite behavior (nitrogen addition results in a net increase of water into the core), then the 2D/3D program supports the conclusion that the 1999 EM model revision is [

]

2.5.3 Application to LBLOCA Analysis

The effect of the 1999 EM revision for steam/water interaction during nitrogen discharge was evaluated by running a LBLOCA for two plants with different reflood thermal-hydraulic characteristics. The effect on PCT is shown in Table 2.5-1.

[

]

[

]

2.5.4 Applicability to LBLOCA Analysis

- The discussion in CENPD-132 Volume 1, Section III.D.5 (Reference 2.5-2) includes results from one-fifth and one-third scale models of the cold leg piping, and showed that ECCS injection can be described[

  ]

- The 2D/3D Test Program (Reference 2.5-6), and the discussion in Section 2.5.2, show that the realistic observed effect of the nitrogen discharge process is a net addition of water into the core. Since the 1999 EM injection section delta pressure model reduces the amount of water entering the core during the time of nitrogen injection, the model [ ]

2.5.5 Model as Coded

The COMPERC-II nitrogen blowdown delta pressure is calculated as described. The nitrogen and liquid mixture velocity in the safety injection line is calculated using Equations (2.5-5), (2.5-6) and (2.5-7). The liquid, steam and nitrogen mixture velocity in the cold legs is calculated using Equations (2.5-2), (2.5-3) and (2.5-4).

The differential pressure across the injection line is calculated using Equation (2.5-1). To ensure conformance to regulatory limits on the injection section differential pressure, the updated code logic subjects the results of Equation (2.5-1) to the following limitations:

[

]

2.5.6 References

2.5-1 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

2.5-2 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

2.5-3 SER for 1974 Evaluation Model, letter O.D. Parr (NRC) to F.M. Stern (CE), June 13, 1975.

2.5-4 SER for CENPD-132, Supplement 2, letter O.D. Parr (NRC) to A.E. Scherer (CE), December 9, 1975.

2.5-5 Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

Code of Federal Regulations, Title 10, Part 50, Appendix K, "ECCS Evaluation Models."

2.5-6 "Reactor Safety Issues Resolved by the 2D/3D Program," NUREG/IA-0127, GRS-101, MPR-1346, July 1993.

2.5-7 "ISP-25, Boundary Conditions and Experimental Procedure for ACHILLES Run A1B105," M. K. Dehnam and P. Dore, AEEW, SESD Note 495, September 1988.

Table 2.5-1

Effect of the 1999 EM Nitrogen Release Flow Resistance Model on PCT

Case	Reduction in PCT °F	Comparison of time of one inch/sec (1985 EM vs. 1999 EM) Seconds	Time of end of nitrogen blowdown (1985 EM and 1999 EM) Seconds
[			
			]

Figure 2.5-1

Injection Section Delta Pressure During Nitrogen Blowdown

1985 EM vs 1999 EM

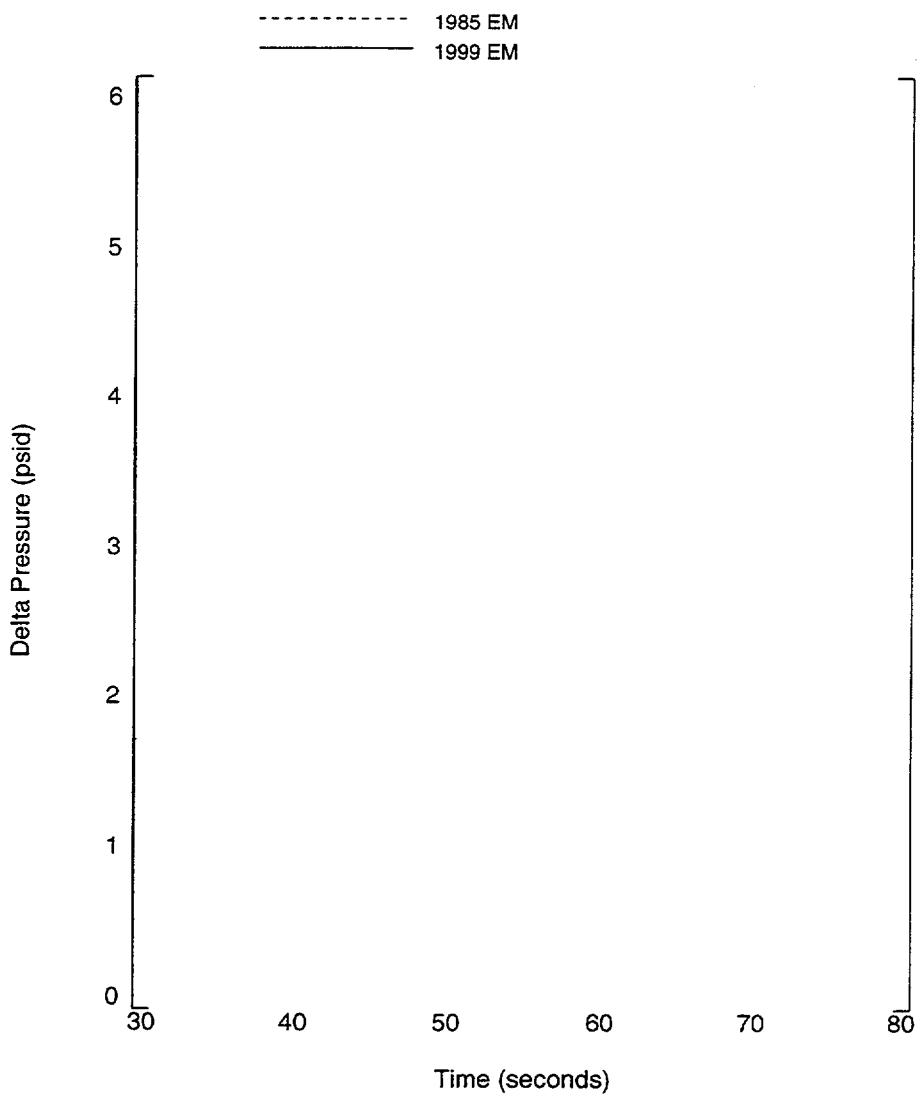


Figure 2.5-2

Velocities during Nitrogen Blowdown

1985 EM COMPERC-II

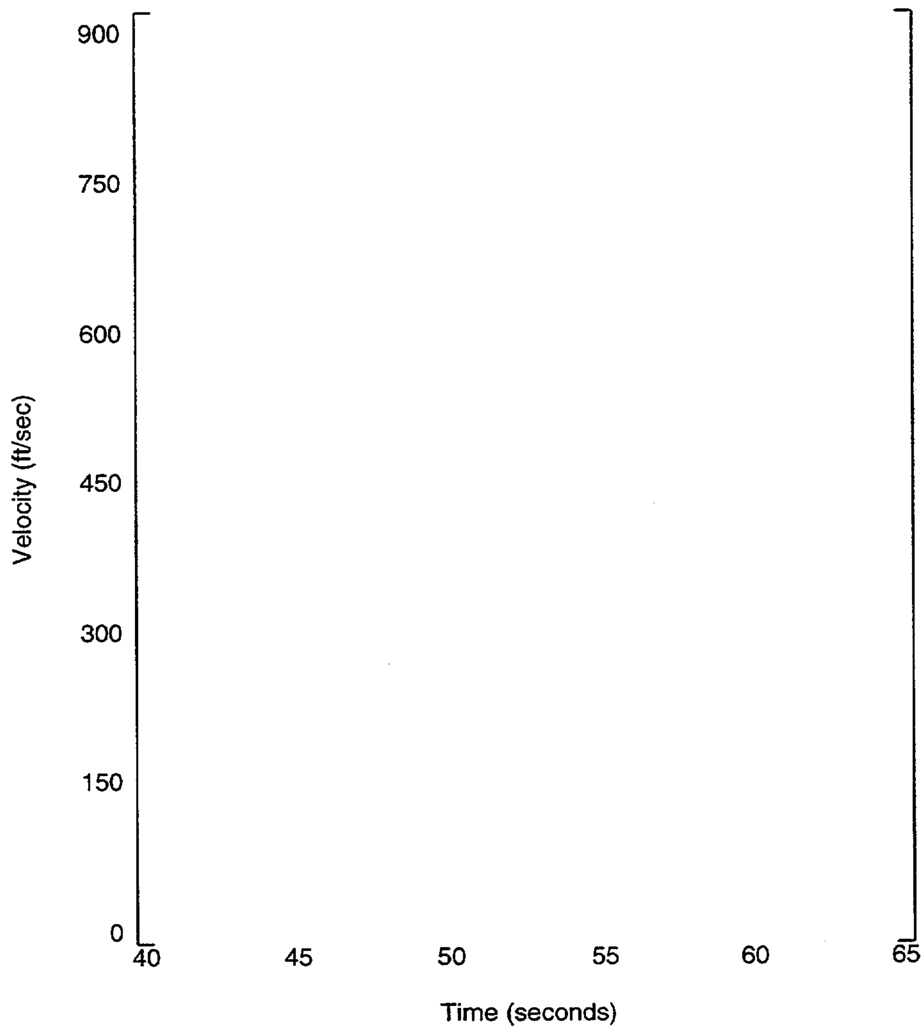


Figure 2.5-3
EQUIVALENT K-FACTOR

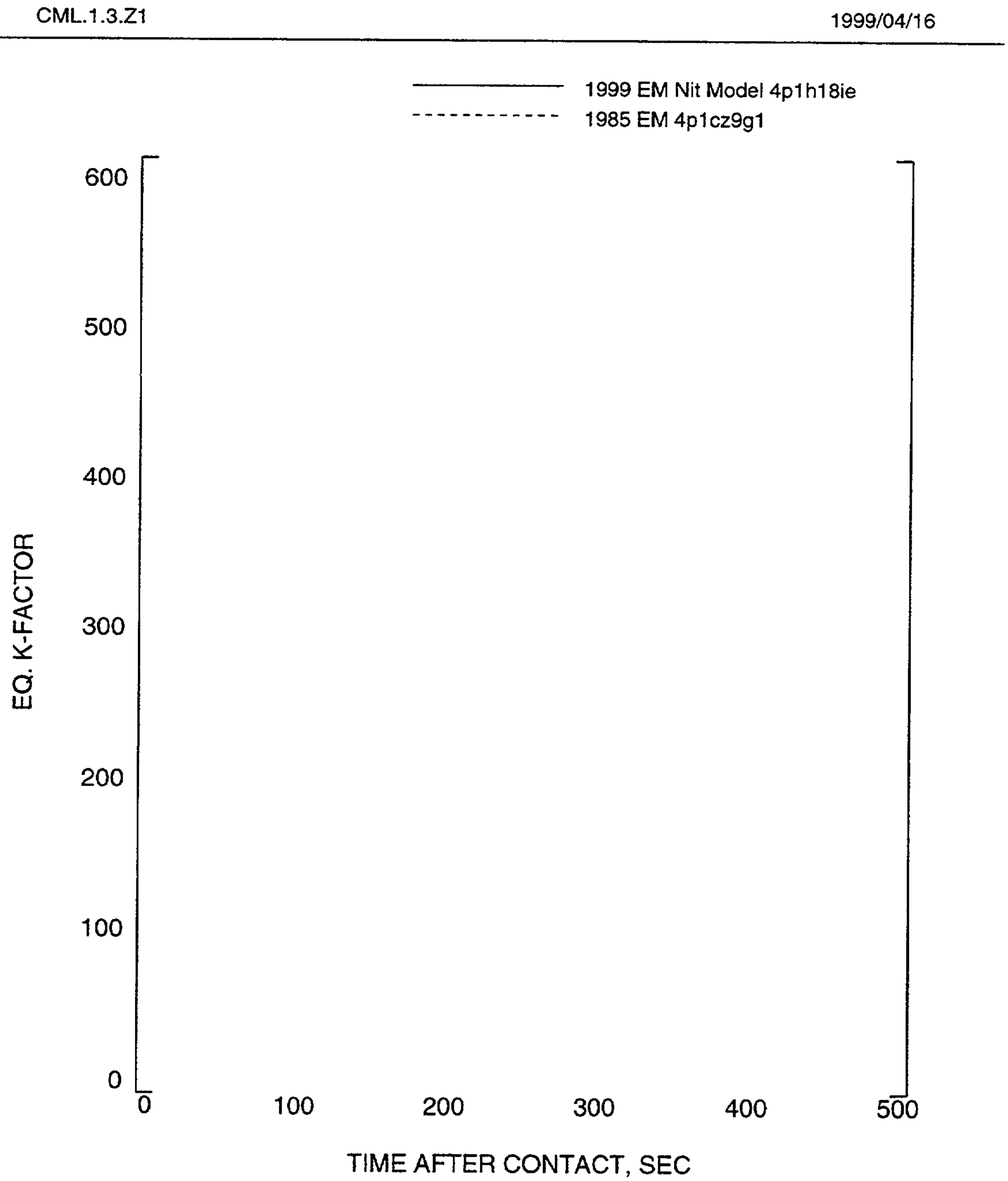


Figure 2.5-4
STEAM FLOW IN HOT LEG

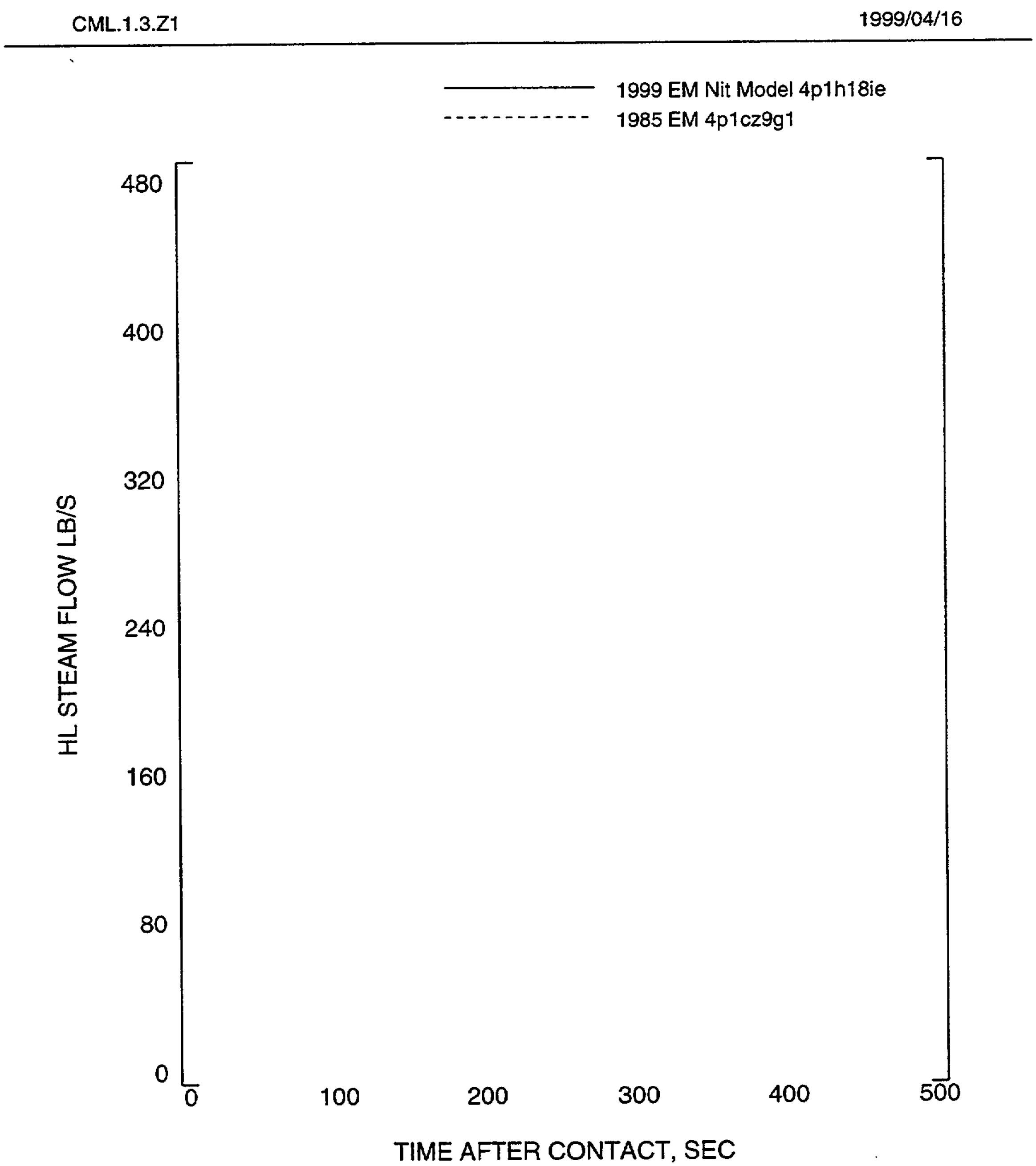
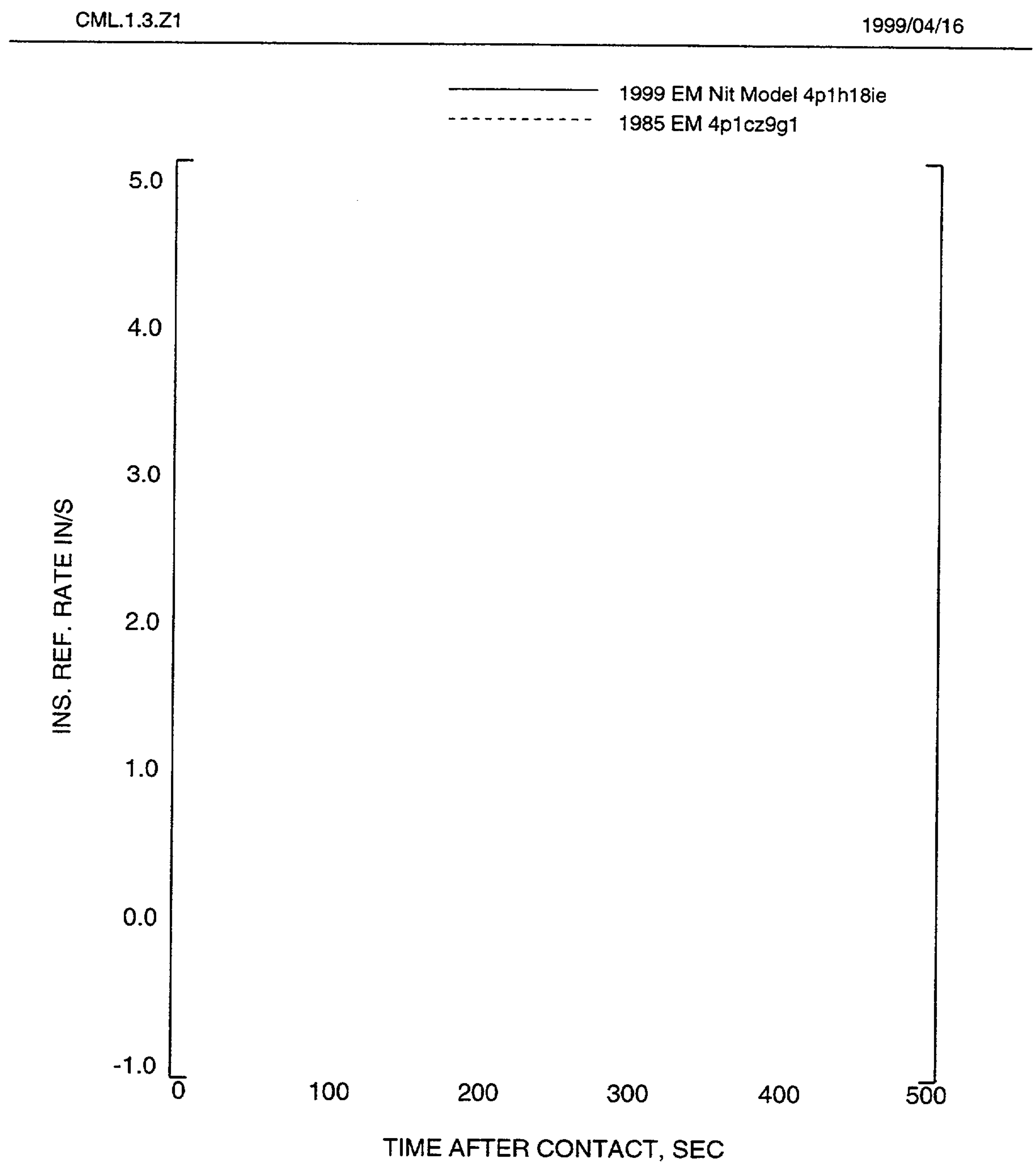


Figure 2.5-5
INSTANTANEOUS REFLOOD RATE



2.6 MOD-1C Reflood Heat Transfer

This section describes the implementation of an improved MOD-1C reflood heat transfer methodology in the 1985 EM version of COMPERC-II for LBLOCA (Reference 2.6-1). As required by Appendix K to 10CFR50, the 1985 EM version of COMPERC-II calculates reflood heat transfer, by utilizing the modified FLECHT correlation, which is demonstrated to be conservative by comparison to FLECHT data for the range of parameters consistent with the LBLOCA transient. The 1999 EM version of COMPERC-II with the improved MOD-1C reflood heat transfer methodology utilizes the same modified FLECHT correlation and maintains overall model conservatism relative to data, but improves the method of representing multiple reflood rate conditions over the course of the transient.

The 1985 EM LBLOCA method for determining reflood heat transfer coefficients, particularly for reflood rates less than one inch per second, conservatively underpredicts 15x15 FLECHT data. This method was first validated as conservative for 14x14 fuel (Reference 2.6-1) and then subsequently validated as conservative for use with 16x16 fuel (Reference 2.6-2). The objective of the 1999 EM change is to improve the 1985 EM methodology using test data that was not available at the time of the original model development, namely, the 17x17 FLECHT-SEASET data (Reference 2.6-3) and the 15x15 Cylindrical Core Test Facility (CCTF) data (Reference 2.6-5).

The goals of this model change are summarized as follows:

- Using the 17x17 FLECHT-SEASET data for multiple reflood rates, develop an improvement to the 1985 EM methodology for representing fuel rod heat transfer coefficients.
- Verify the model improvement by comparison to the data of the FLECHT-SEASET and the CCTF test programs.
- Minimize any change in methodology or analysis approach from that already reviewed and accepted by NRC. In addition, based on initial feedback from the

NRC (meeting held 11/17/98), maintain realistic, physical, and conservative representation of the major elements of the correlation.

2.6.1 Model Description

The basic FLECHT correlation for constant flooding rates that is contained within the MOD-1C methodology is unchanged from Reference 2.6-1, Appendix G. The model element selected for improvement is the part of the MOD-1C methodology used to construct the heat transfer coefficients for multiple reflood rates. This element is called the MOD-2C procedure.

The MOD-2C procedure contains several aspects related to multiple reflood rate modeling including, mirror imaging and time shifting. The focus of the 1999 EM improvement is the time shifting aspect of the MOD-2C procedure. The time shifting aspect of the procedure is contained in one of the equations from the reflood heat transfer model, which has been previously accepted for use by the NRC. This equation is referred to as the "time-shifted" curve, and represents a shift to the left of the origin on the time axis so that the FLECHT correlation for constant flooding rate can be used to represent multiple reflood rates. The equation for time shifting contains two parts, (1) an empirical elevation dependent correlation adjustment multiplier, and (2) a reflood mass integral expressed as a ratio to represent the long term flooding rate impact inherent in the constant flooding rate heat transfer correlation. The 1985 EM equation with these two parts is as follows:

$$\Delta t = CAM\int_0^{tr}[(V_{r1} - V_{r2}) / V_{r2}]dt$$

$$CAM = (0.214z - 0.386)$$

where

t = time, (sec)

Δt = time shift to left, (sec)

CAM = Correlation Adjustment Multiplier

z = elevation from bottom of fuel rod, (ft)

V = reflood rate, (lbms / sec)

$r1$ = designates prior reflood rate period

$r2$ = designates reflood rate period for time shifted calculation

tr = time of start of r2 reflood rate period, (sec)

An improvement to the MOD-2C procedure is made through a modification in the empirical elevation dependent Correlation Adjustment Multiplier (CAM). This change is shown in Figure 2.6-1 and is given analytically as follows:

[

]

As Figure 2.6-1 illustrates, the correlation change is[

]

2.6.2 Model Assessment

The development and justification of the 1999 EM reflood heat transfer model change described above is based on comparisons to the 17x17 FLECHT-SEASET data (Reference 2.6-3). The FLECHT-SEASET experimental test facility is designed to model a full-height core section of electrically heated fuel rod simulators representative of the 17x17 assembly design. The facility was used to perform reflood heat transfer tests characteristic of a LBLOCA for both forced reflooding and gravity feed. The core consisted of 161 full length fuel rods in a cylindrical cross-section with a radially uniform power distribution and a 1.66 cosine axial power shape.

The comparisons made to the FLECHT-SEASET test data in this model assessment process encompass five of the six multiple reflood rate tests in the unblocked test series documented in Reference 2.6-3 (see Table 2.6-1). There were two forced reflood tests with variable flooding rates and three gravity reflood tests, one of which simulated a different initial axial temperature distribution. (A fourth gravity reflood test was not considered because the power history boundary condition in the test was faulted.)

Heat transfer coefficient data is reported in Reference 2.6-3 for each of these five FLECHT-SEASET tests at eight elevations between the 6.49 ft elevation and the 11.61 ft elevation. Two elevations at the top of the fuel rod simulators for two of the gravity tests showed no heatup due to liquid fallback. Therefore, the model comparisons to heat transfer coefficient data are made for 36 separate transient profiles.

The revised model documented in this supplement is independently justified against a second test program, the cylindrical core test facility (CCTF), see Reference 2.6-5. The CCTF is an experimental facility designed to model a full-height core section and four primary loops and their components of a PWR. The facility is used to perform reflood heat transfer tests characteristic of a LBLOCA for 15x15 type fuel rod design. The core consists of thirty-two 8x8 fuel assemblies arranged in a cylindrical configuration. The core is subdivided into three power regions to achieve a desired radial power profile.

Four CCTF tests (see Table 2.6-2) from the Core-II test series were selected for the comparisons, Reference 2.6-5. This selection is based on availability of data and adequacy of test conditions for the range of conditions for LBLOCA. Heat transfer coefficient data is reported for each of the tests at four elevations between the 3 ft elevation and the 10 ft elevation.

For both FLECHT-SEASET and CCTF, the test measurements documented in the test reports and the heat transfer coefficients calculated from the measurements represent the heat transfer coefficients from the highest measured temperatures. That is, the data represents the hottest fuel rod simulators over the axial extent of the hottest conditions. This assures conservatism in the development of an empirical correlation based on this data. The tests used to justify this model change including the as-run test conditions and the test results are given in Tables 2.6-1 and 2.6-2.

The comparisons of the 1985 EM methodology from Reference 2.6-1 and the 1999 EM methodology to the selected test data are given in Figures 2.6-2 through 2.6-57. These comparisons are produced in a two step process.

- The first step in the comparison is to apply a procedure for translating the heat transfer methodology from a FLECHT based geometry for 15x15 fuel assembly designs to (1) a FLECHT-SEASET based geometry for 17x17 fuel assembly designs or (2) a CCTF based geometry for 15x15 fuel assembly designs. This translation is performed using the NRC accepted FRELAPC computer code procedure, Reference 2.6-4. This procedure was developed for use with (1) CE Nuclear Power LLC fuel designs and (2) axial power shapes that are different from the FLECHT fuel design and the FLECHT axial power shape.
- The second step in the comparison is to execute the 1985 and 1999 EM heat transfer methodologies (the HTCOF option in the COMPERC-II computer code) using the measured boundary conditions from the tests. This produces the graphical comparisons given in Figures 2.6-2 through 2.6-57.

The comparisons show that the proposed model change improves the comparisons to the data, while still maintaining overall conservatism. The degree of conservatism is subjectively determined in Table 2.6-3. An agreement scale number is assigned to each elevation for each test based on the average difference between the test measurement and the model prediction for the heat transfer coefficient over the major portion of the transient response. The average value of the subjective scale reveals that the revised model is [ ] relative to the data for FLECHT-SEASET and [ ] for CCTF. That is, on the average the revised model will predict a [ ] This is particularly true for the comparisons to the CCTF tests.

2.6.3 Application to LBLOCA Analysis

The reduction in PCT during ECCS performance analyses from this 1999 EM reflood heat transfer model change will be varied depending on the timing and magnitude of the calculated reflood rates and on the fuel rod elevation being examined. The largest effects of the 1999 EM model change occur for the [ ] of the fuel rod (see Figure 2.6-1), where the improved time-shift function [ ] from the 1985 EM function. The impact of the change [ ] For typical LBLOCA applications, the most [.] Reductions in calculated PCT for these limiting elevations using the 1999 EM reflood heat transfer model are as follows:

[

Model Assessment Results for NSSS LBLOCA 1999 EM Reflood Heat Transfer		

]

[

]

The limiting steam cooled node above the rupture location, Node 16, also has a[ ]PCT of[ ] This occurs for several reasons:

[

]

The 1985 EM and 1999 EM comparison was also performed for a[

]

Therefore, for the most limiting case (rupture Node 15), the impact of the 1999 EM on the limiting PCT is a[ ] Depending on the hot rod elevation and on the reflood hydraulic conditions associated with the rupture location, the impact of the 1999 EM methodology ranges from[ ] Figures 2.6-58 through 2.6-61 show comparisons of the 1999 EM heat transfer coefficients to the 1985 EM values for the limiting nodes of the Rupture Node 15 case discussed above. In these figures, the time of PCT relative to contact is between 170 and 200 seconds. These figures illustrate the magnitude of the improvement in heat transfer coefficient with the 1999 EM for NSSS LBLOCA limiting conditions.

2.6.4 Applicability to LBLOCA Analysis

As described in Reference 2.6-6, Section III.D.6, the NRC-approved MOD-1C FLECHT heat transfer correlation is valid over the range of parameter variation shown in the second column of the following table. The variable flooding rate test data available to justify the 1999 EM improvement to the FLECHT correlation covers the parameter range of variation given in the third column of the following table (see Tables 2.6-1 and 2.6-2, also).

Parameter	MOD-1C FLECHT Correlation Valid Range of Variation	Range of Variation in the FLECHT-SEASET and CCTF Data Used for 1999 EM Improvement
Flooding Rate (V_{in})	0.4 – 10 in/sec	0.62 – 13 in/sec
Pressure (P)	15 – 90 psia	20 – 40 psia
Inlet Coolant Subcooling (ΔT_{sub})	16 – 189 °F	100 – 140 °F
Initial Clad Temperature (T_{init})	1200 – 2200 °F	359 – 1631 °F
Peak Power Density (Q'_{max})	0.69 – 1.40 kw/ft	0.7 – 0.82 kw/ft
Elevation (z)	4 – 8 ft	3.8 – 11.61 ft
Percent Blockage (see Note)	0 – 75%	0%

Note: As required by Appendix K, only FLECHT data for unblocked cores is utilized in that portion of the reflood heat transfer methodology using the FLECHT correlation.

2.6.5 Model as Coded

For the 1999 EM, the option to control the use of this improved reflood heat transfer methodology is included in the user controlled interface file (UCI). In the 1999 EM version

of COMPERC-II, the calculation of the correlation adjustment multiplier is implemented as described in Section 2.6.1.

One limitation programmed in the subroutine logic has been added to prevent the [

]

2.6.6 References

2.6-1 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

2.6-2 CENPD-213-P, "Application of FLECHT Reflood Heat Transfer Coefficients to C-E's 16x16 Fuel Bundles," January 1976.

2.6-3 NUREG/CR-1532, EPRI NP-1459, WCAP-9699, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Test Data Report," NRC/EPRI/W Report No. 7, Volumes 1 & 2, June 1980.

2.6-4 LOCA-76-349, "FRELAPC – An Analytical FLECHT Rod Elevation and Power Correction Program," T. C. Kessler, May 26, 1976.

2.6-5 JAERI-M 85-026, "Evaluation Report on CCTF Core-II Reflood Test C2-4 (Run 62) – Investigation of Reproducibility," Tsutomu Okubo, et. al., March 1985.

JAERI-M 85-025, "Evaluation Report on CCTF Core-II Reflood Test Second Shakedown Test C2-SH2 (Run 54) – Effect of Core Supplied Power on Reflood Phenomena," Tadashi Iguchi, et. al., March 1985.

JAERI-M 86-185, "Evaluation Report on CCTF Core-II Reflood Test C2-3 (Run 61) – Investigation of Initial Downcomer Water Accumulation Velocity Effects," Tsutomu Okubo, et. al., January 1987.

JAERI-M 87-001, "Evaluation Report on CCTF Core-II Reflood Test C2-8 (Run 67) – Effect of System Pressure," Hajime Akimoto, et. al., January 1987.

2.6-6 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

Table 2.6-1
FLECHT-SEASET Unblocked Bundle Reflood Test Data Summary

Test No.	Run No.	As-Run Test Conditions: Upper Plenum Pressure (psia)	Rod Initial T_{clad} at 72 in. (°F)	Rod Peak Power (kw/ft)	Flooding Rate (in/sec)	Coolant Temp. (°F)	Radial Power Distribution	Results: Hottest Rod T/C and Elevation (in)	Initial Temp (°F)	Maximum Temp (°F)	Temp Rise (°F)	Turn Around Time (sec)	Quench Time (sec)	Bundle Quench Time (sec)
Forced Reflood Tests with Variable Flooding Rate														
33	32333	40	1631	0.70	6.36 5 sec 0.82 onward	125	Uniform	6L-76	1550	2099	549	131	337	639
35	32235	20	1630	0.70	6.53 5 sec 0.98 200 sec 0.62 onward	88	Uniform	6K-78	1514	2096	582	142	546	964
Gravity Reflood Tests					Injection Rate (lbm/sec)									
36	33436	39	1611	0.70	12.8 15 sec 1.73 onward	125	Uniform	10H-70	1636	1670	34	4	121	174
38	33338	40	1600 (hot) 1096 (cold)	0.70 (hot) 0.40 (cold)	13 15 sec 1.78 onward	125	Hot/Cold channels	10H-70	1664	1697	33	6	76	181
Gravity Reflood Test with Axial Temperature Distribution														
44	33644	39	359 (0 to 3) 1610	0.70	12.8 15 sec 1.76 onward	125	Uniform	7D-76	1623	1705	82	9	104	250

Table 2.6-2
CCTF Reflood Test Data Summary

Test No.	Run No.	Description	Upper Plenum Pressure (psia)	Rod Initial Maximum Temperature (°F)	Initial Subcooling (°F)	Flooding Rate (in/sec)
C2-4	62	CCTF-II Base Case	37.7	1470.2	111.5	2.089 until 10 sec 1.377 until 45 sec 0.752 onward
C2-SH1	53	CCTF-II Base Case	37.7	1455.8	110.2	1.972 until 7.1 sec 1.630 until 33.3 sec 0.702 onward
C2-3	61	High Rate of Downcomer Water Accumulation	37.7	1481.0	139.0	2.507 until 4.76 sec 1.406 until 42.9 sec 0.719 onward
C2-8	67	Low Pressure	29.0	1472.0	100.7	2.089 until 10 sec 1.755 until 20 sec 0.702 onward

Table 2.6-3
Subjective Agreement Matrix for the 1999 EM Model Comparison to FLECHT-SEASET and CCTF Data

[

Elevation, (ft)	F/S Test No. 32235	F/S Test No. 32333	F/S Test No. 33436	F/S Test No. 33338	F/S Test No. 33644

Elevation, (ft)	CCTF Test No. C2-4/62	CCTF Test No. C2-SH1/53	CCTF Test No. C2-3/61	CCTF Test No. C2-8/67

]

Figure 2.6-1

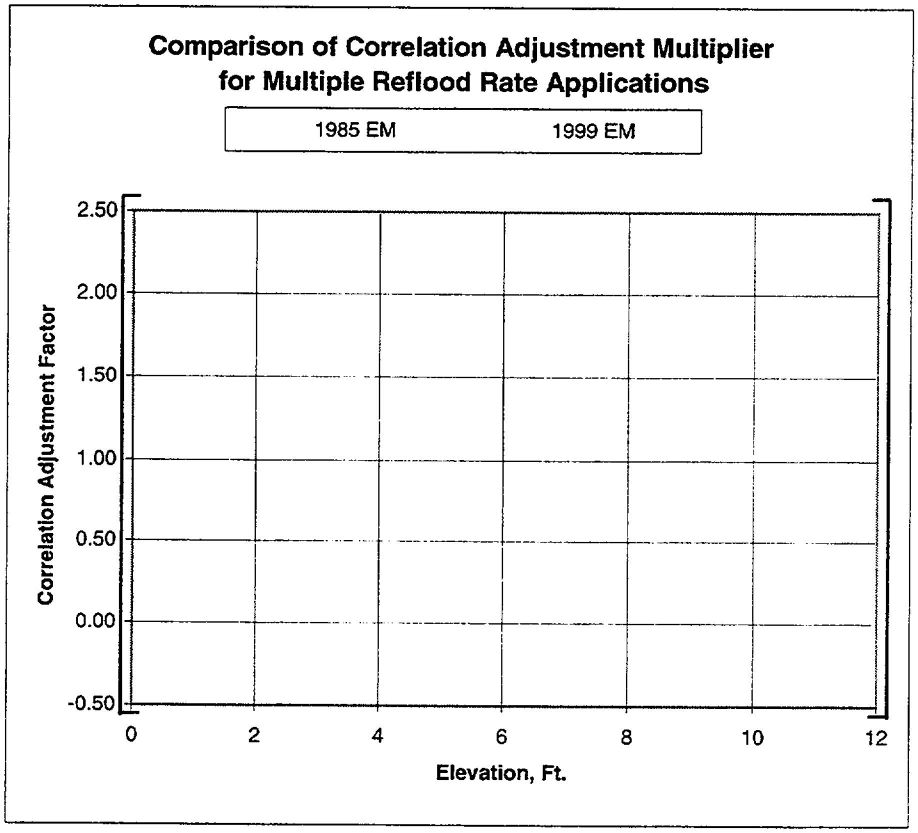

Comparison of Correlation Adjustment Multiplier
for Multiple Reflood Rate Applications
1985 EM
1999 EM
Correlation Adjustment Factor
2.50
2.00
1.50
1.00
0.50
0.00
-0.50
0
2
4
6
8
10
12
Elevation, Ft.

Figure 2.6-2
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV= 6.49

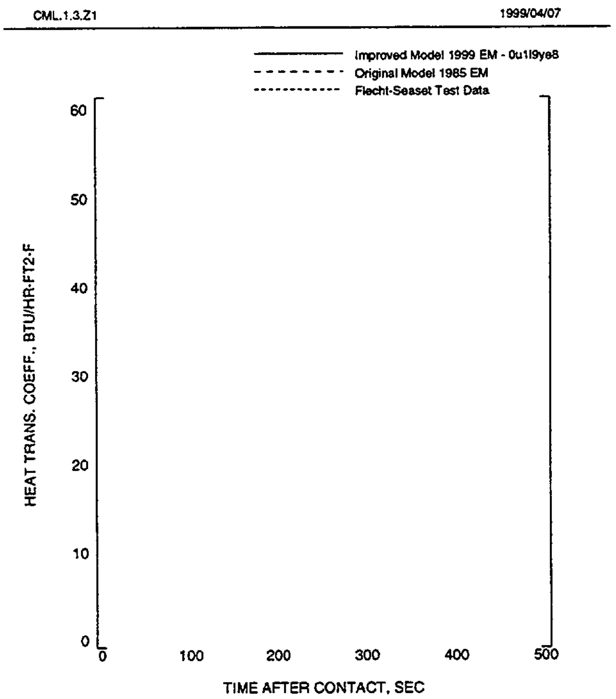


Figure 2.6-3
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV= 6.95

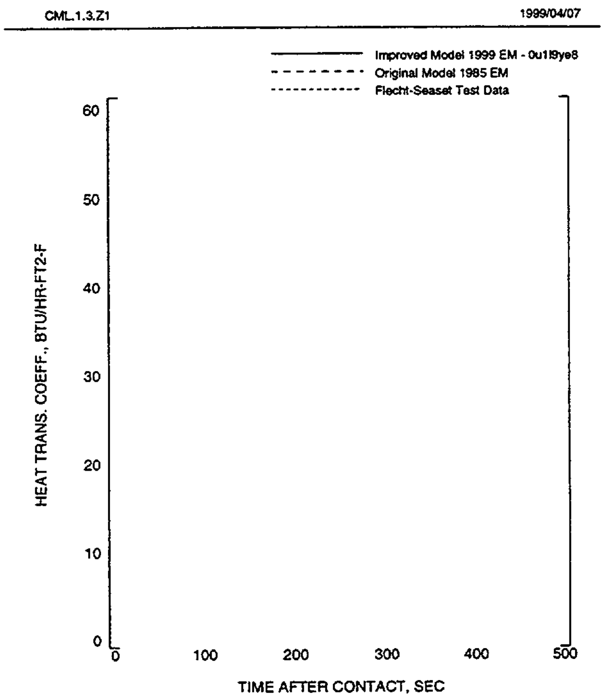


Figure 2.6-4
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV= 7.83

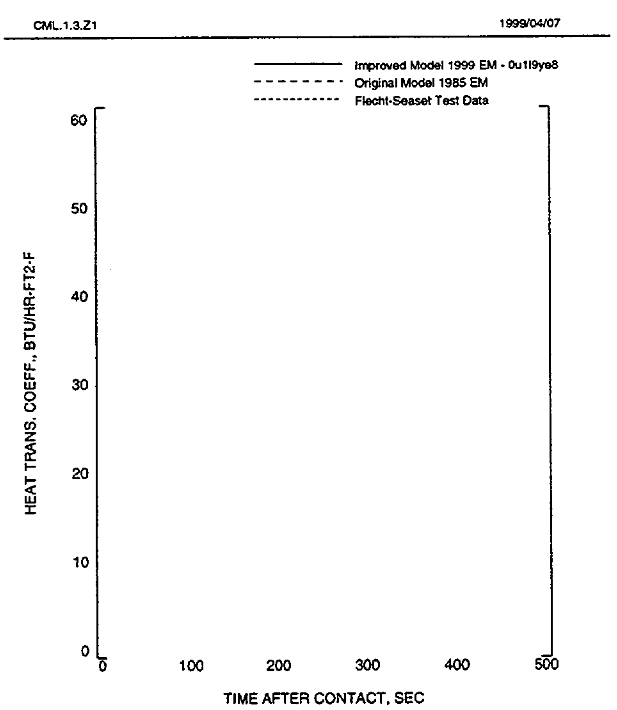


Figure 2.6-5
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV= 8.34

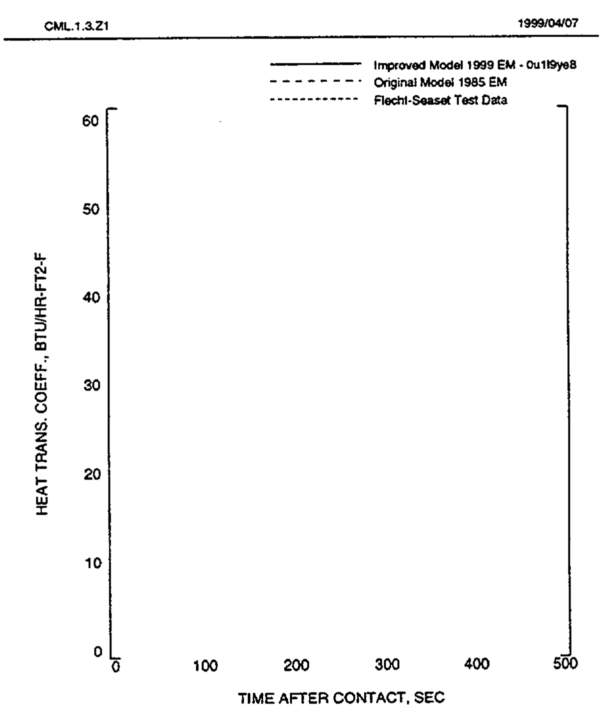


Figure 2.6-6
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV= 9.50

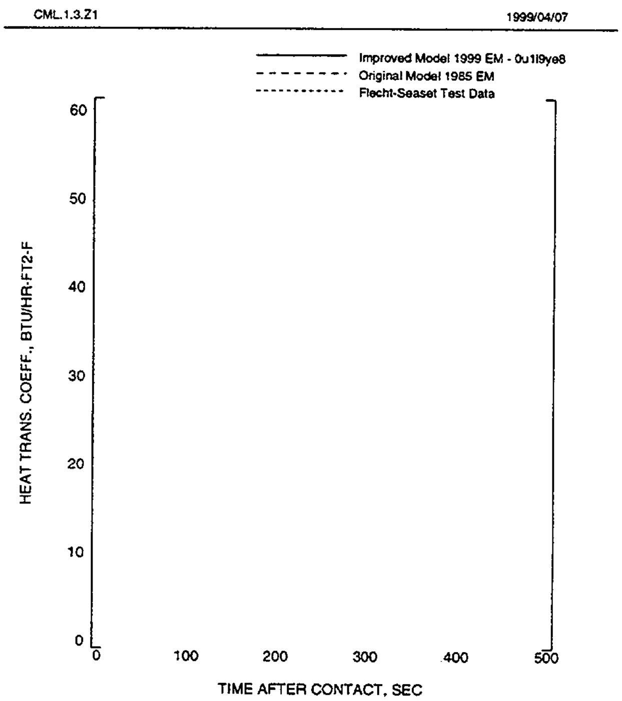


Figure 2.6-7
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV=10.11

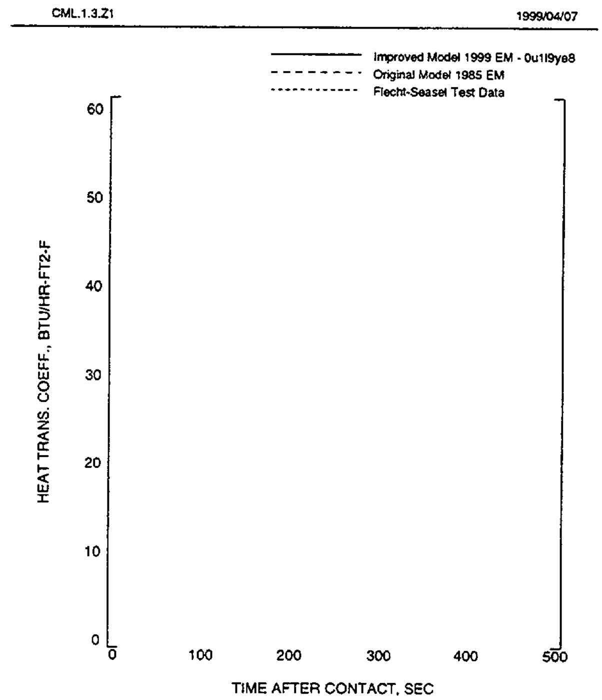


Figure 2.6-8
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV=11.11

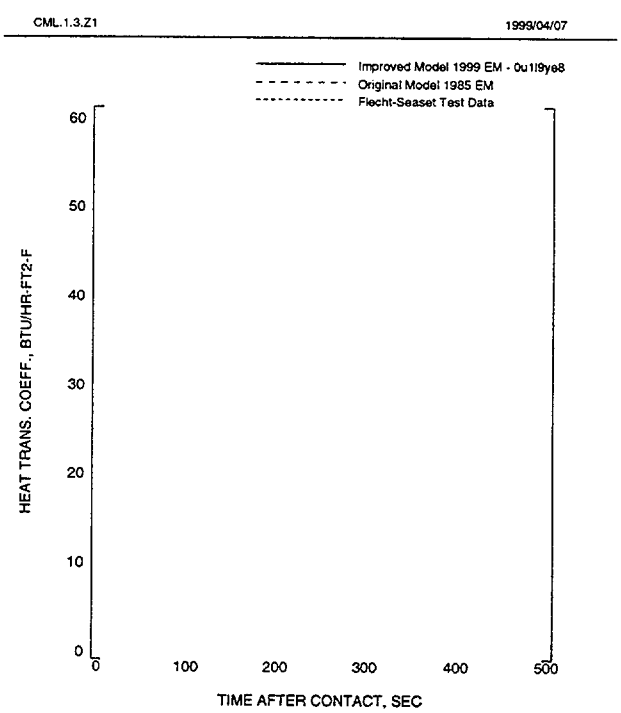


Figure 2.6-9
FLECHT-SEASET TEST COMPARISON NO. 32235
HEAT TRANS. COEFF.
ELEV=11.61

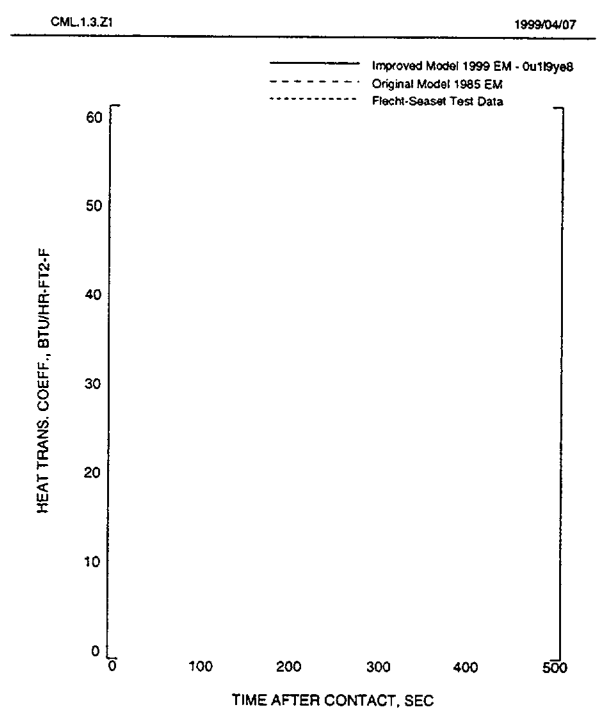


Figure 2.6-10
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV= 6.49

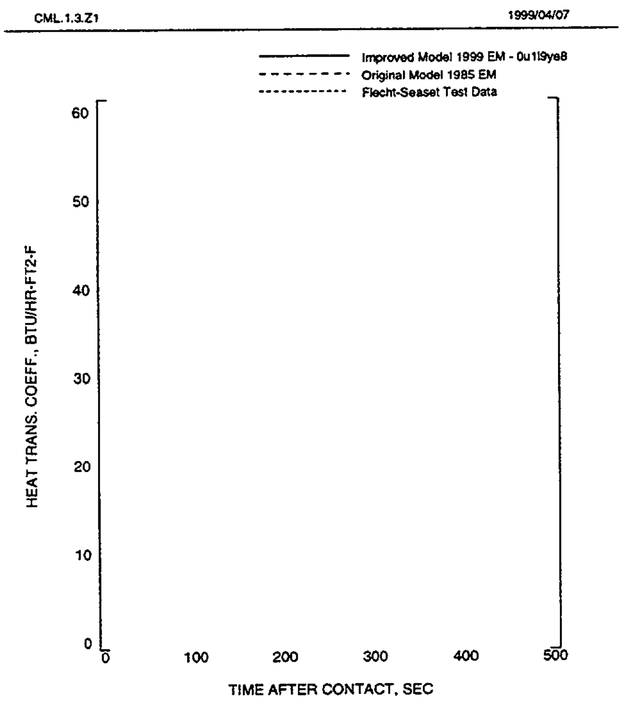


Figure 2.6-11
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV= 6.95

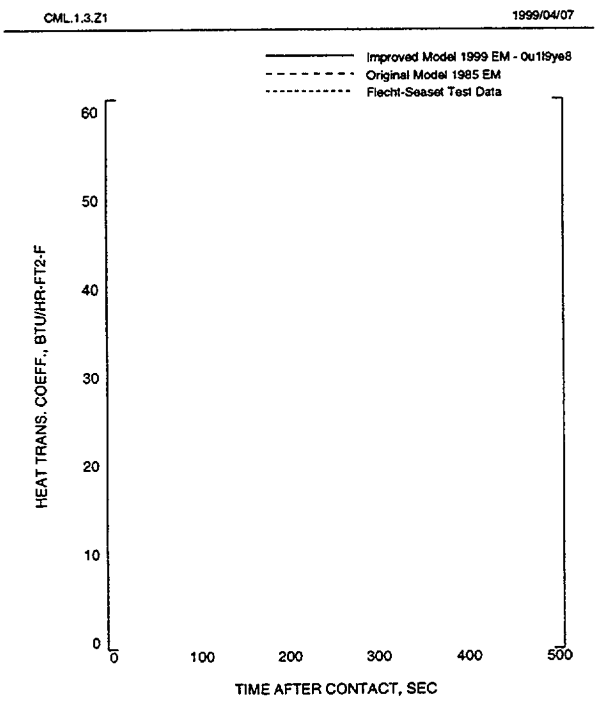


Figure 2.6-12
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV= 7.83

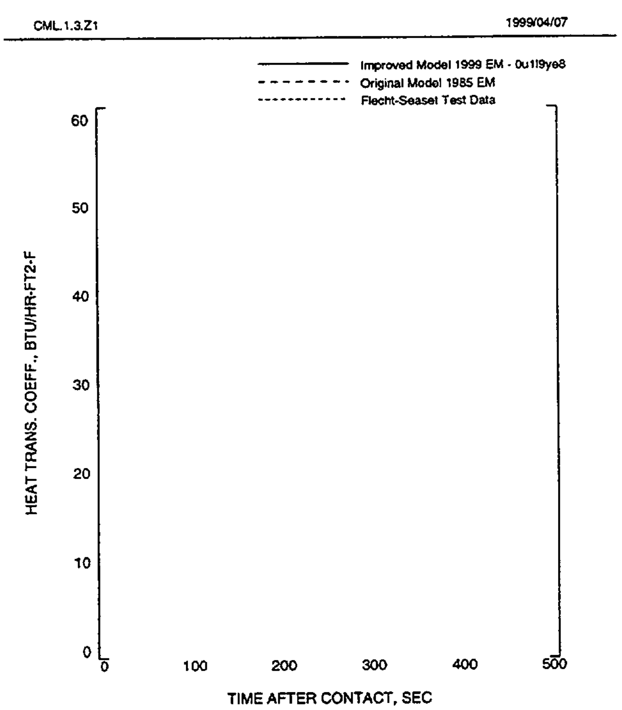


Figure 2.6-13
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV= 8.34

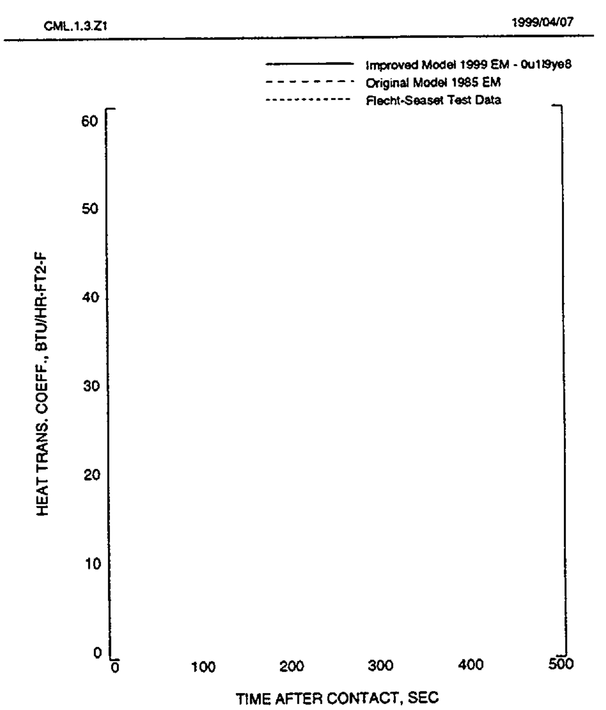


Figure 2.6-14
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV= 9.50

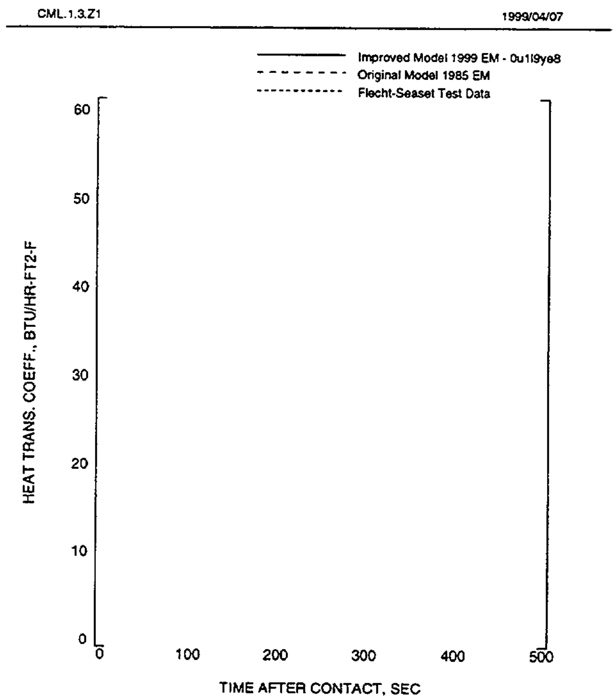


Figure 2.6-15
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV=10.11

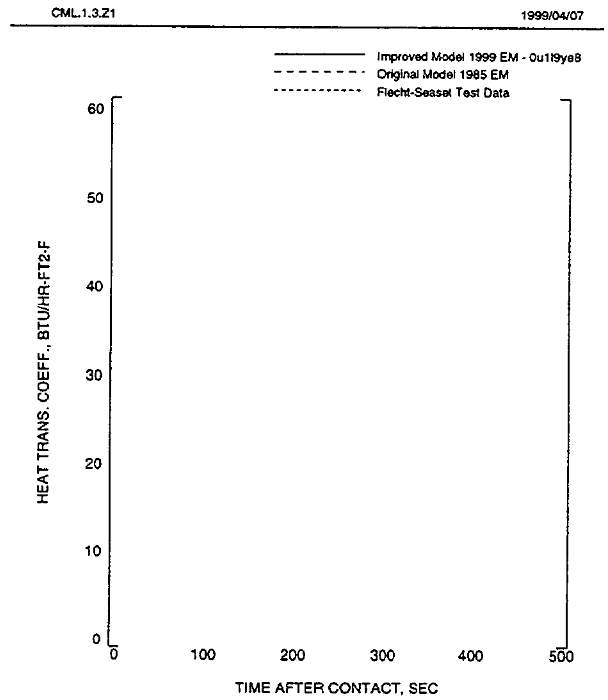


Figure 2.6-16
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV=11.11

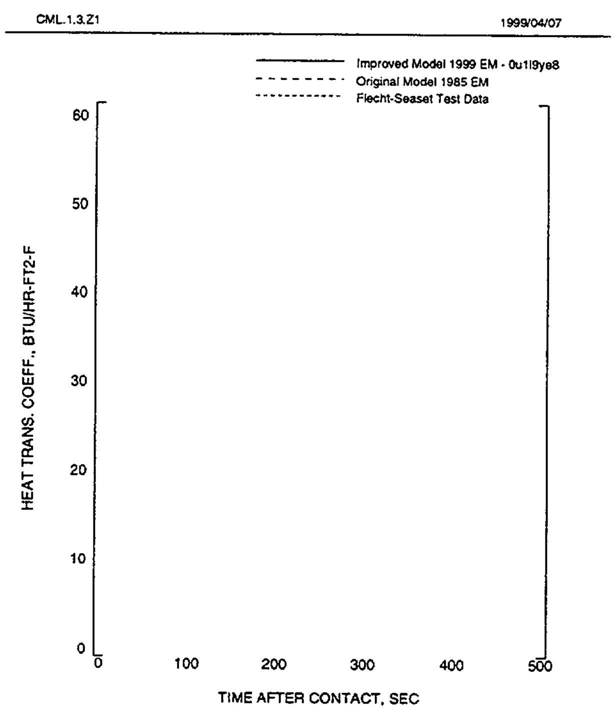


Figure 2.6-17
FLECHT-SEASET TEST COMPARISON NO. 32333
HEAT TRANS. COEFF.
ELEV=11.61

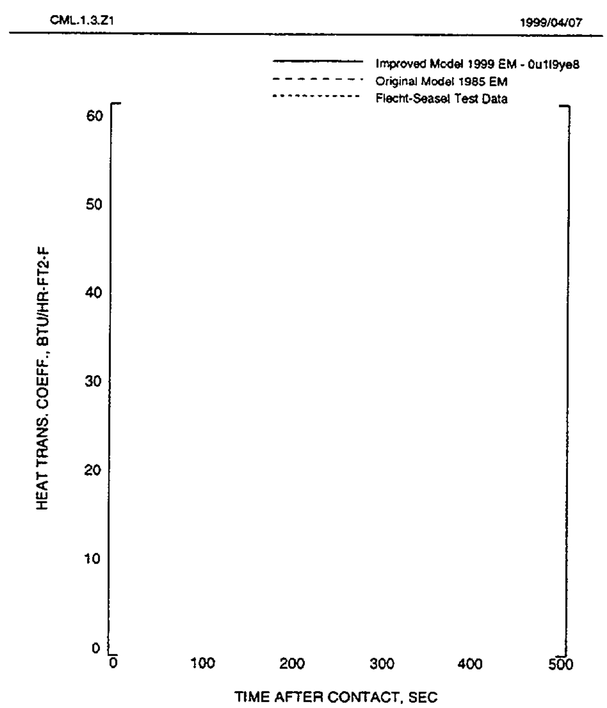


Figure 2.6-18
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV= 6.49

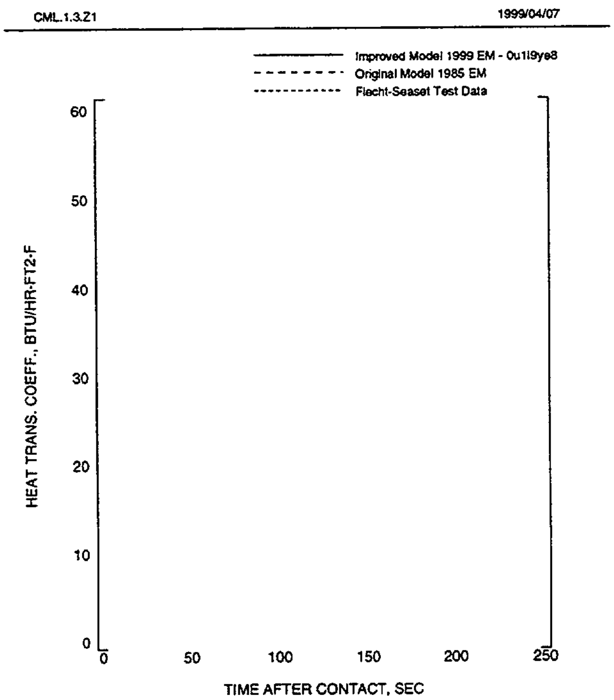


Figure 2.6-19
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV= 6.95

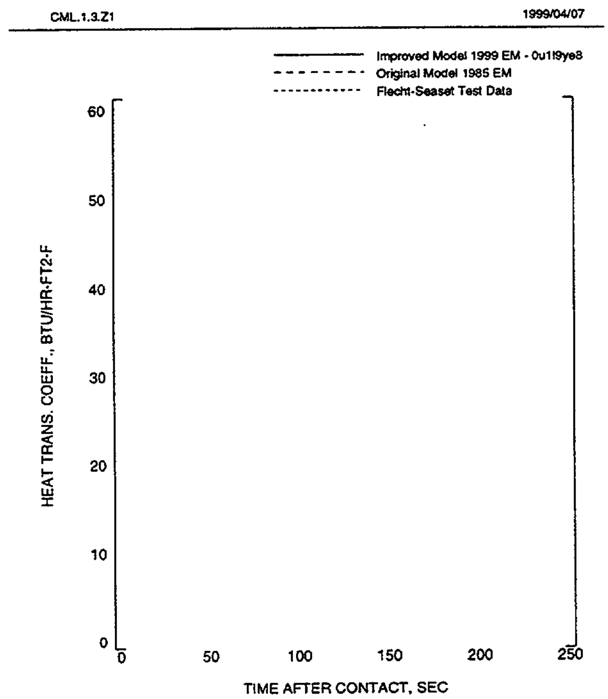


Figure 2.6-20
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV= 7.83

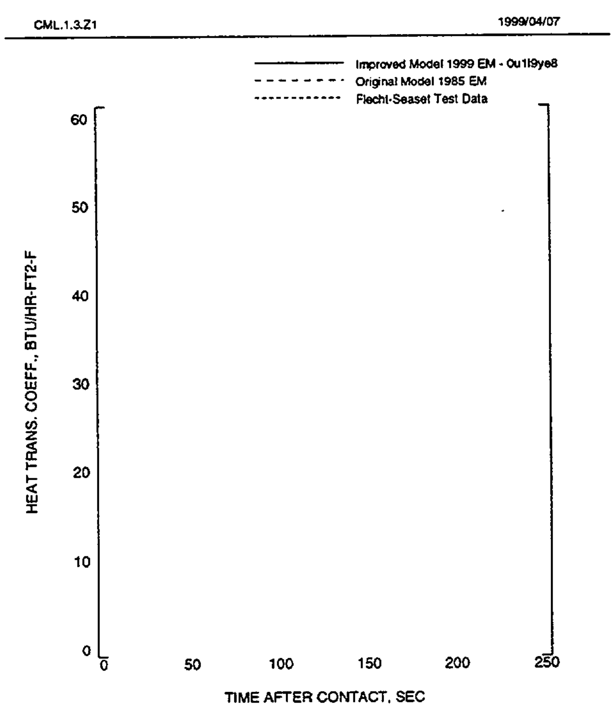


Figure 2.6-21
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV= 8.34

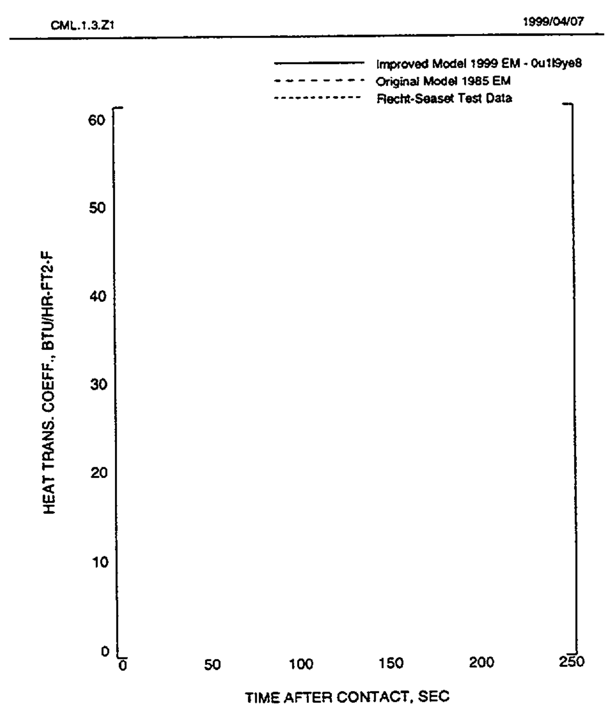


Figure 2.6-22
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV= 9.50

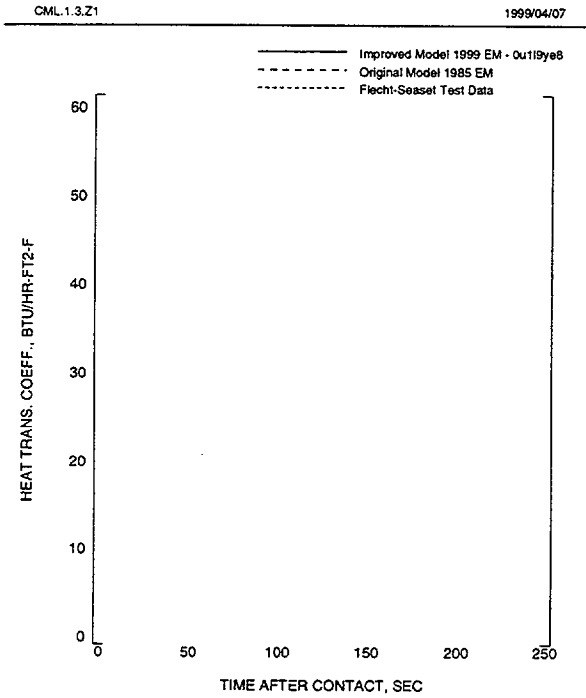


Figure 2.6-23
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV=10.11

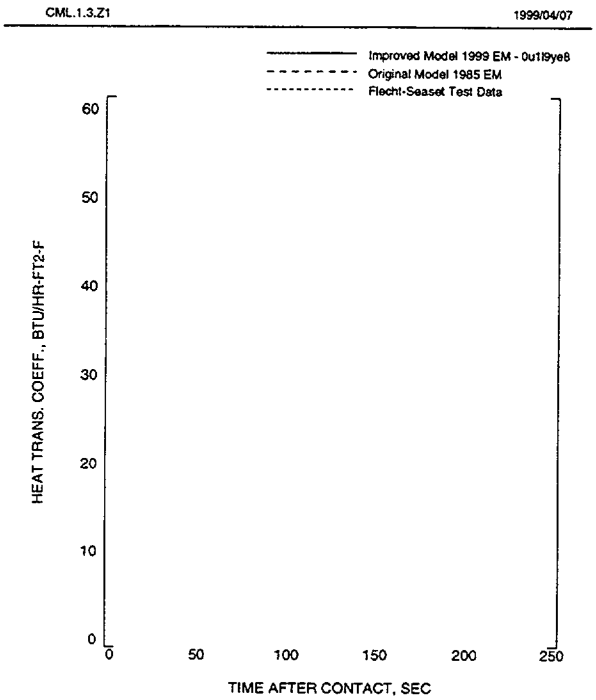


Figure 2.6-24
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV=11.11

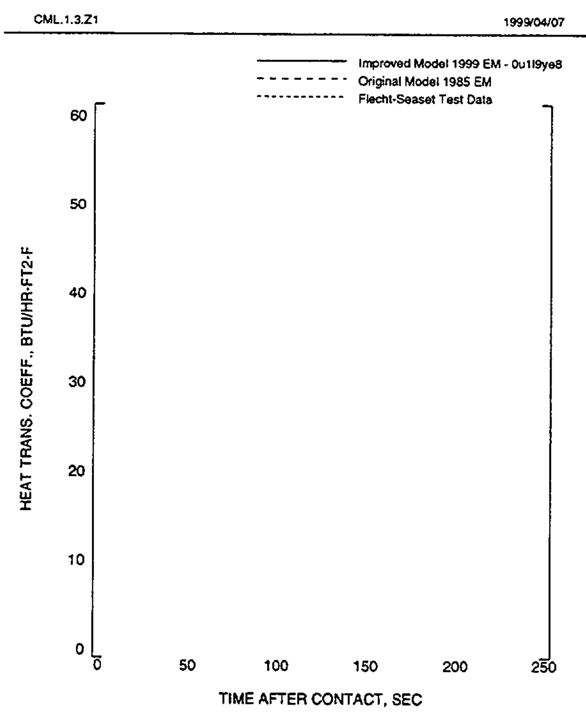


Figure 2.6-25
FLECHT-SEASET TEST COMPARISON NO. 33436
HEAT TRANS. COEFF.
ELEV=11.61

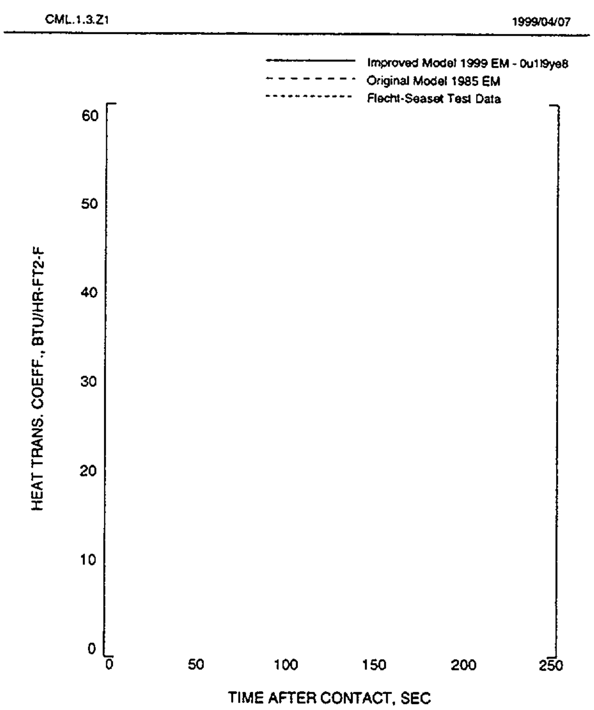


Figure 2.6-26
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV= 6.49

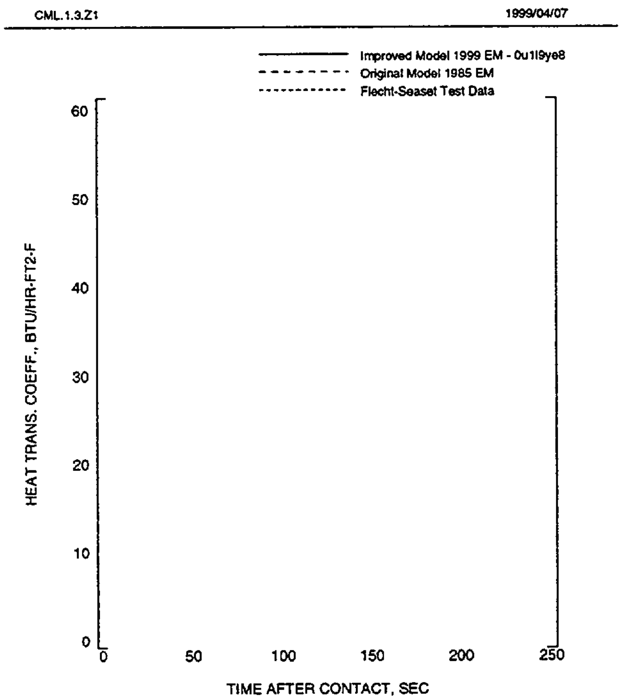


Figure 2.6-27
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV= 6.95

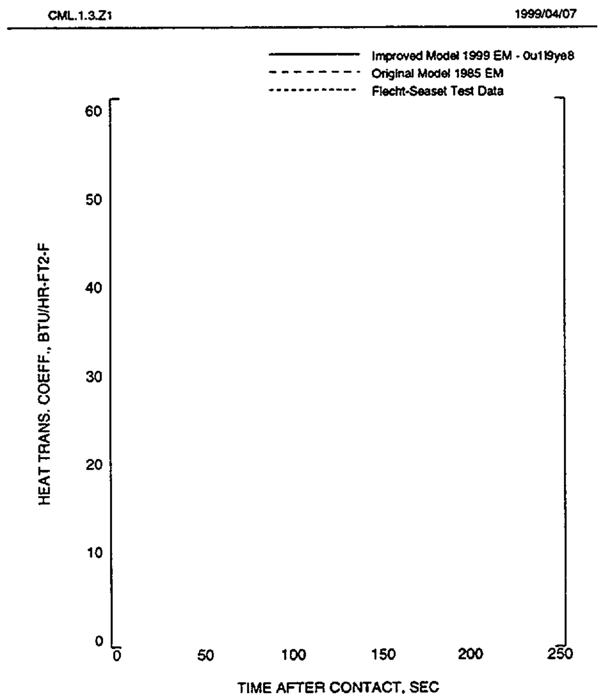


Figure 2.6-28
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV= 7.83

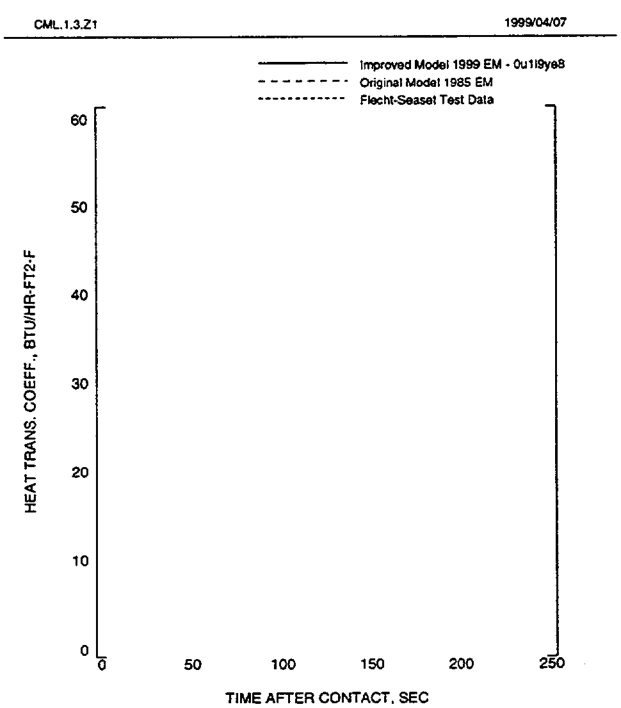


Figure 2.6-29
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV= 8.34

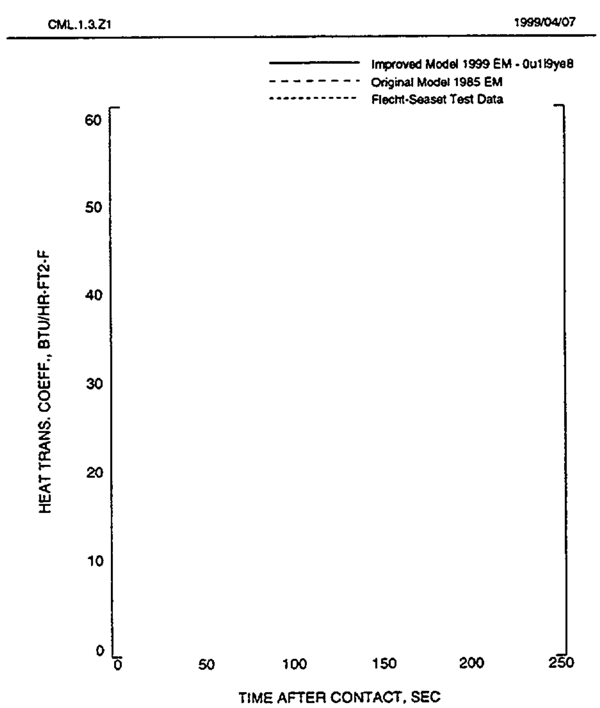


Figure 2.6-30
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV= 9.50

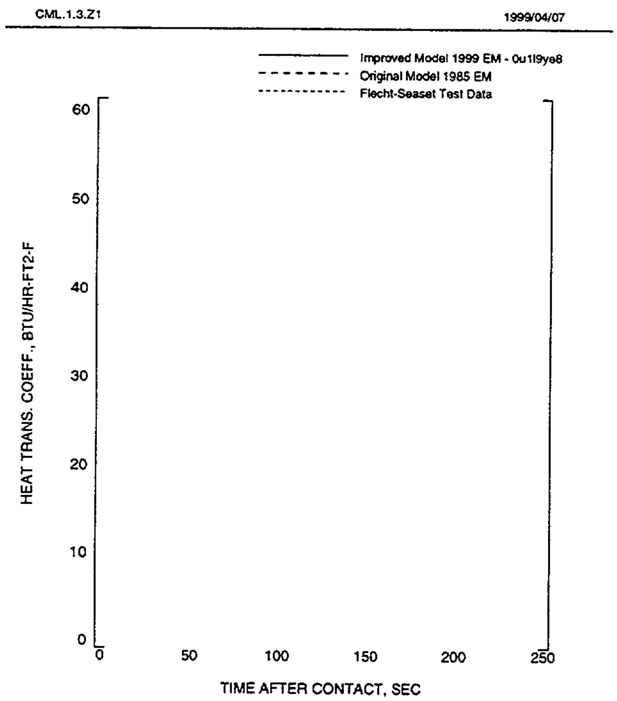


Figure 2.6-31
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV=10.11

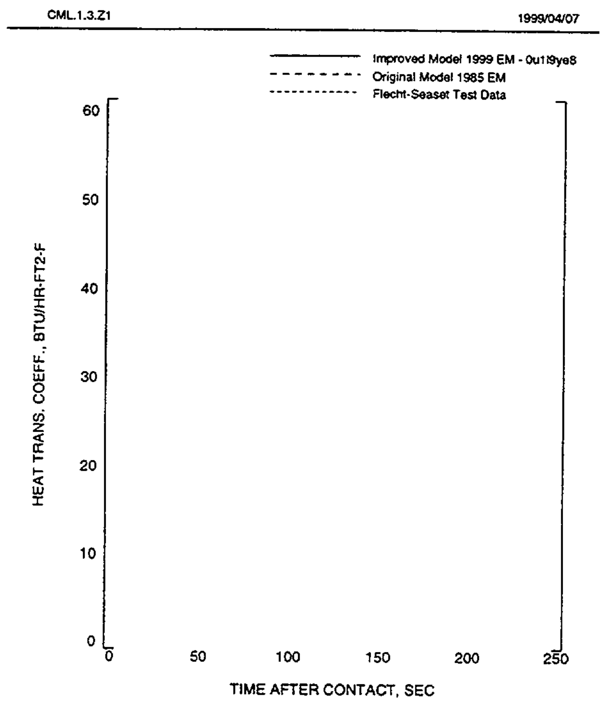


Figure 2.6-32
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV=11.11

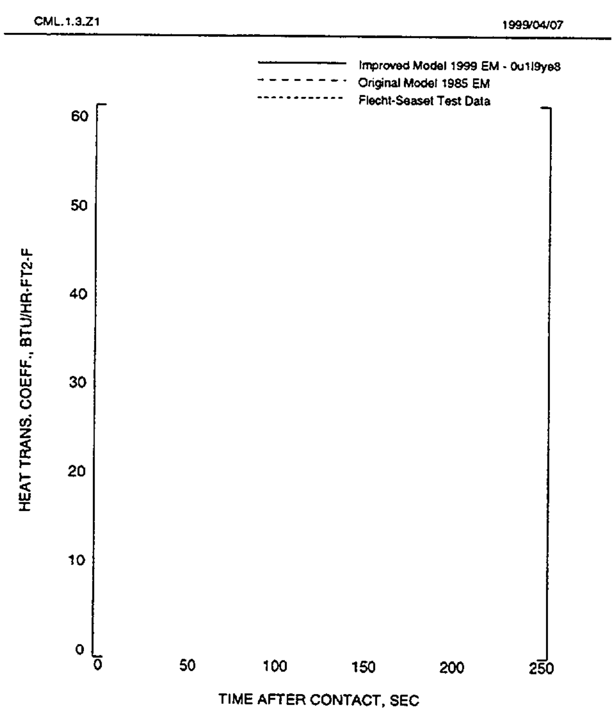


Figure 2.6-33
FLECHT-SEASET TEST COMPARISON NO. 33338
HEAT TRANS. COEFF.
ELEV=11.61

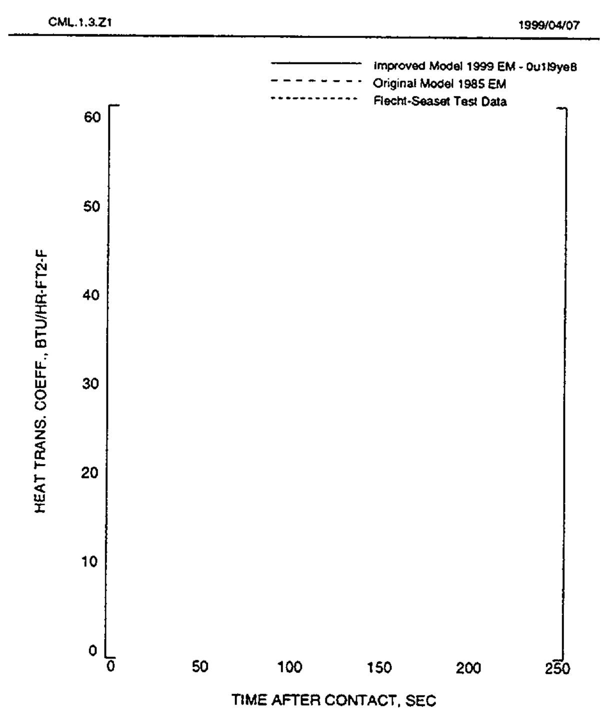


Figure 2.6-34
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV= 6.49

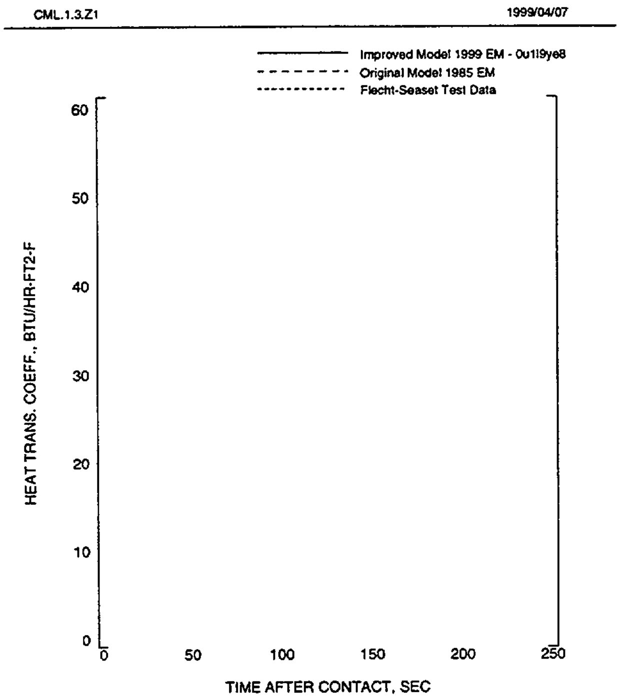


Figure 2.6-35
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV= 6.95

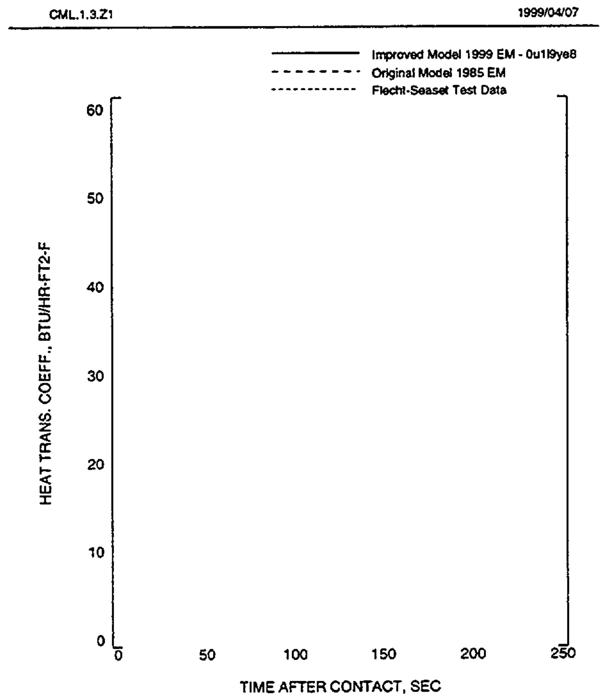


Figure 2.6-36
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV= 7.83

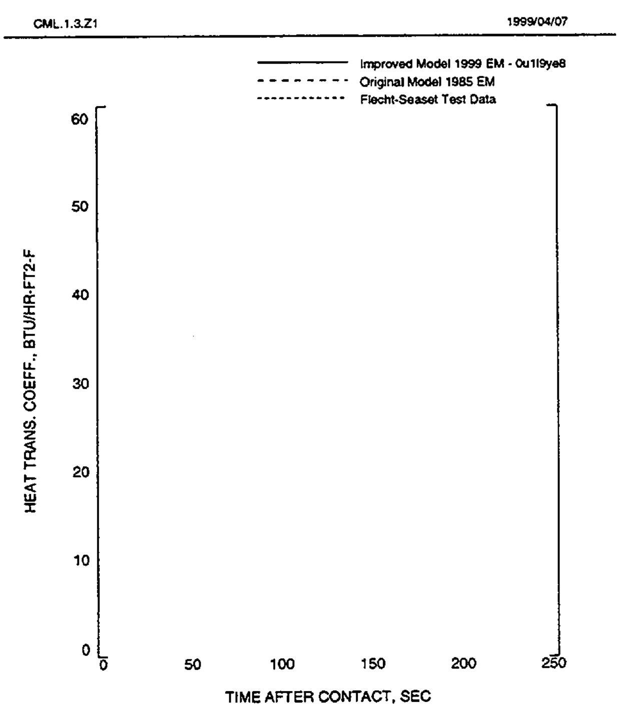


Figure 2.6-37
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV= 8.34

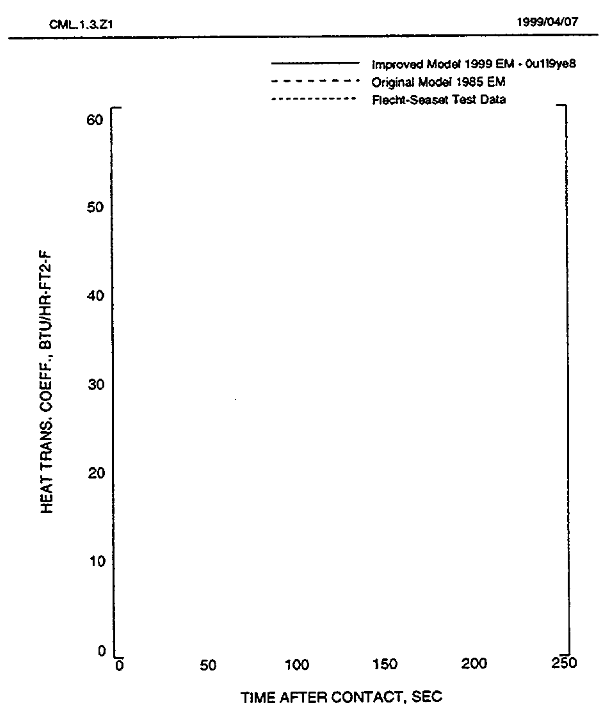


Figure 2.6-38
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV= 9.50

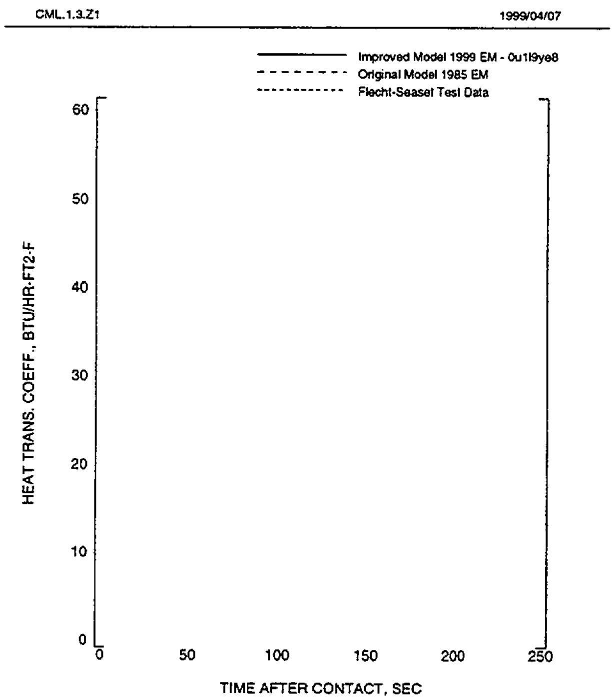


Figure 2.6-39
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV=10.11

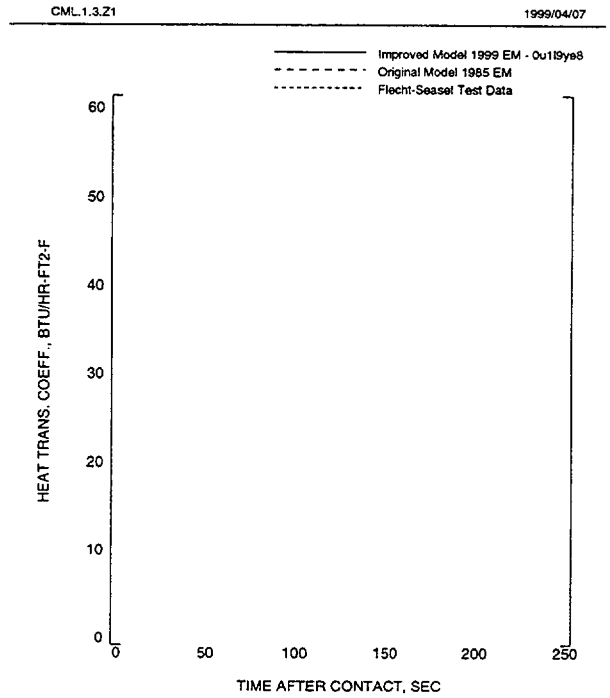


Figure 2.6-40
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV=11.11

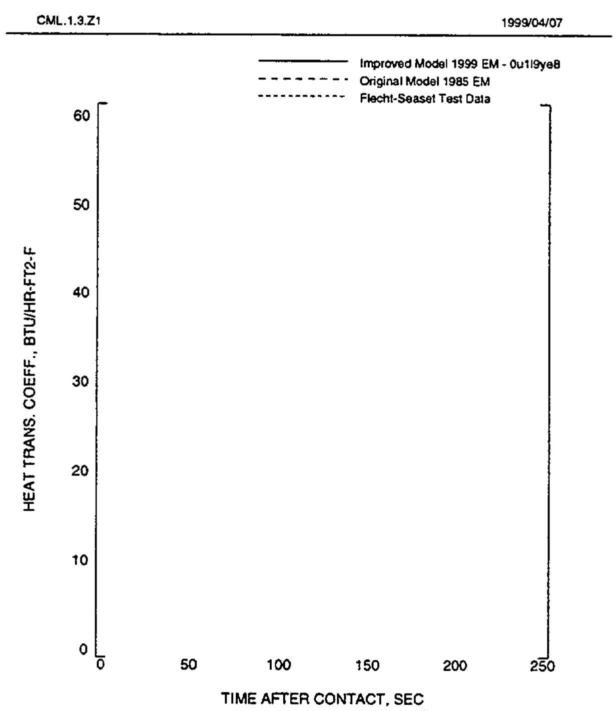


Figure 2.6-41
FLECHT-SEASET TEST COMPARISON NO. 33644
HEAT TRANS. COEFF.
ELEV=11.61

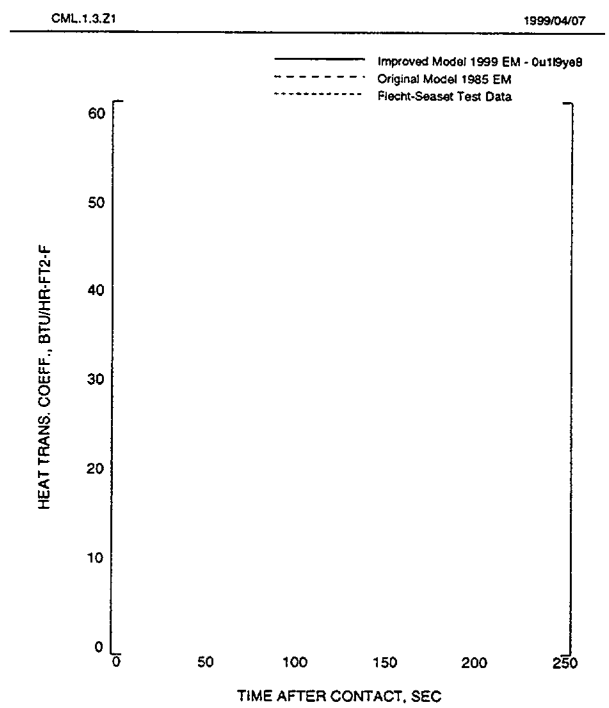


Figure 2.6-42
CCTF CORE-II TEST COMPARISON NO. C2-4/62
HEAT TRANS. COEFF.
ELEV= 3.80

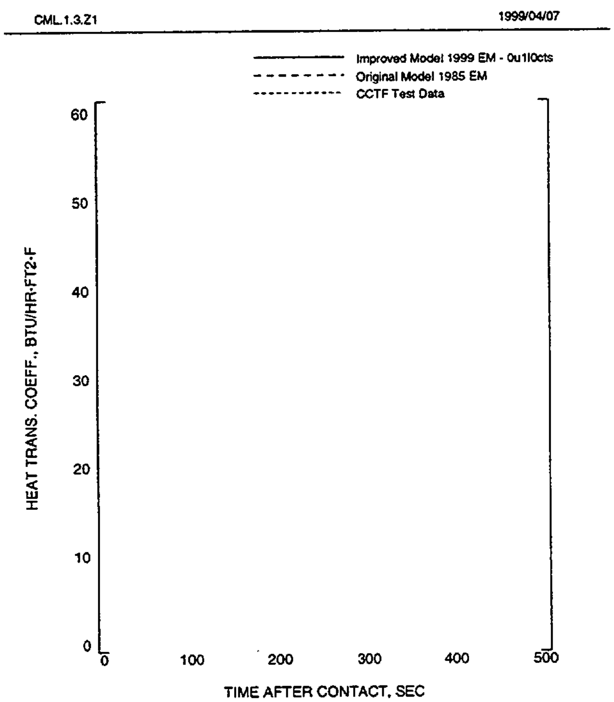


Figure 2.6-43
CCTF CORE-II TEST COMPARISON NO. C2-4/62
HEAT TRANS. COEFF.
ELEV= 6.57

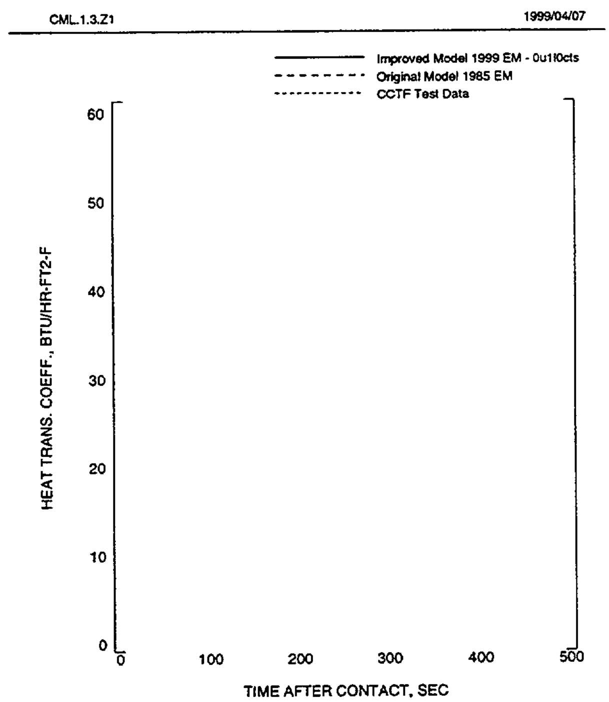


Figure 2.6-44
CCTF CORE-II TEST COMPARISON NO. C2-4/62
HEAT TRANS. COEFF.
ELEV= 7.96

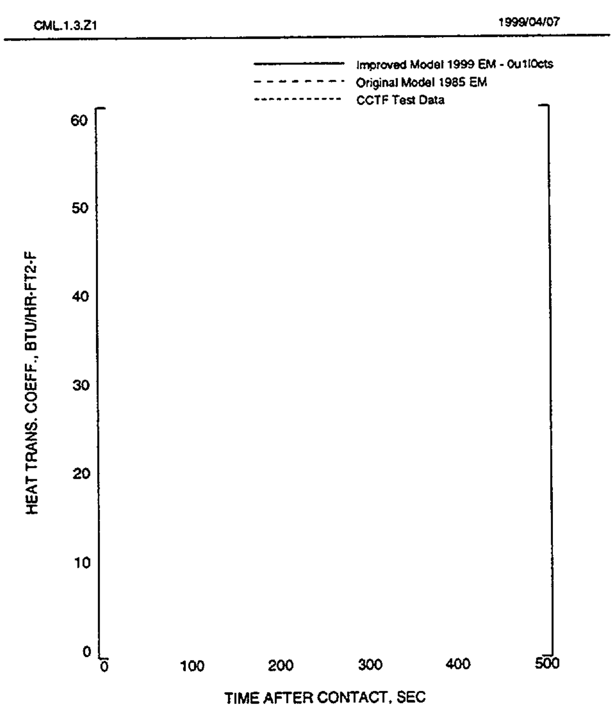


Figure 2.6-45
CCTF CORE-II TEST COMPARISON NO. C2-4/62
HEAT TRANS. COEFF.
ELEV= 9.44

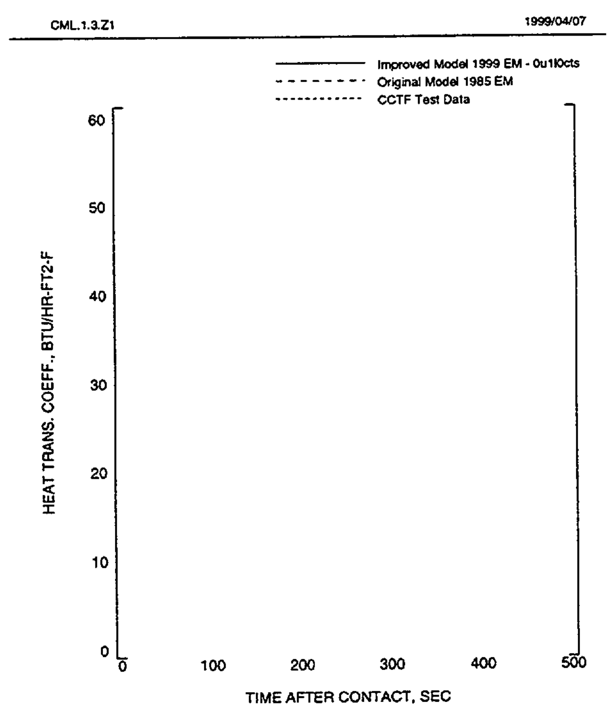


Figure 2.6-46
CCTF CORE-II TEST COMPARISON NO. C2-SH1/53
HEAT TRANS. COEFF.
ELEV= 3.80

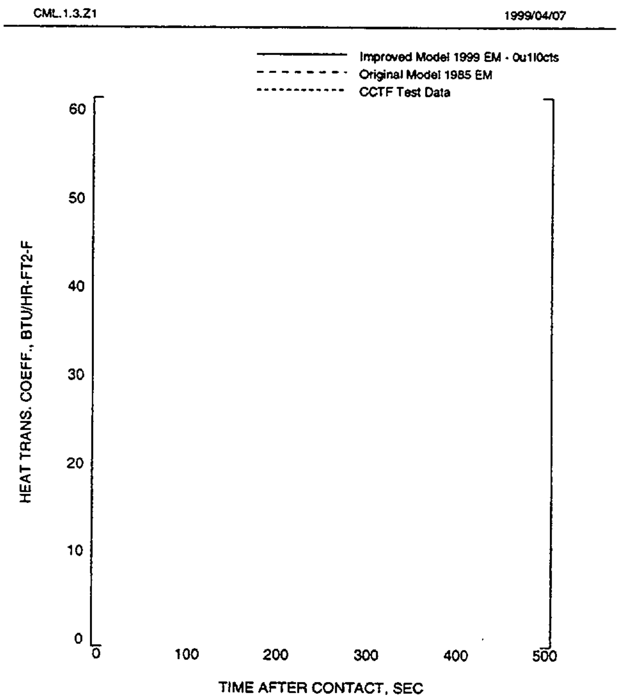


Figure 2.6-47
CCTF CORE-II TEST COMPARISON NO. C2-SH1/53
HEAT TRANS. COEFF.
ELEV= 6.57

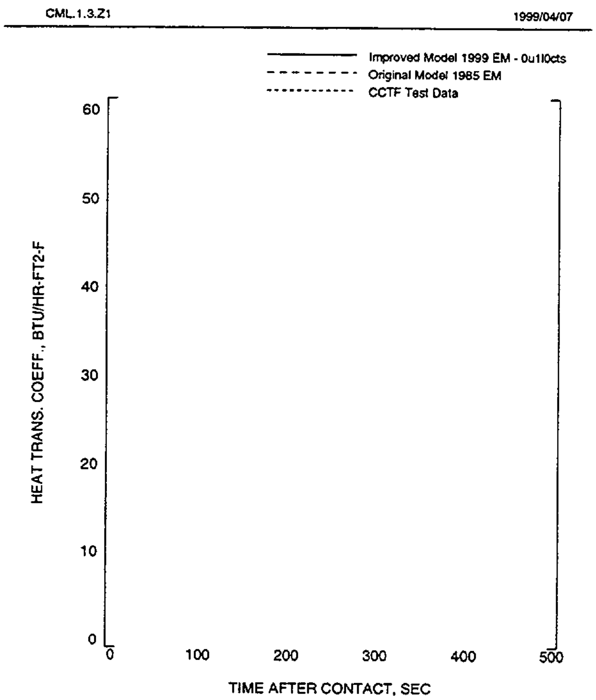


Figure 2.6-48
CCTF CORE-II TEST COMPARISON NO. C2-SH1/53
HEAT TRANS. COEFF.
ELEV= 7.96

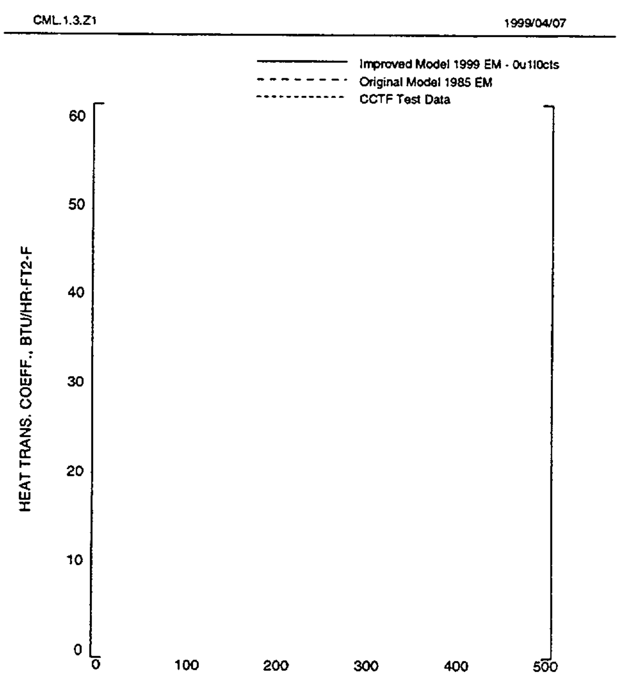


Figure 2.6-49
CCTF CORE-II TEST COMPARISON NO. C2-SH1/53
HEAT TRANS. COEFF.
ELEV= 9.44

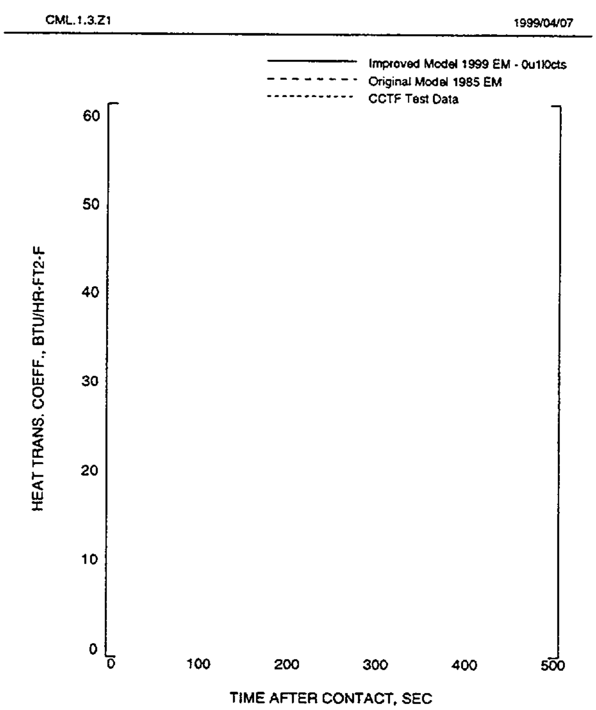


Figure 2.6-50
CCTF CORE-II TEST COMPARISON NO. C2-3/61
HEAT TRANS. COEFF.
ELEV= 3.80

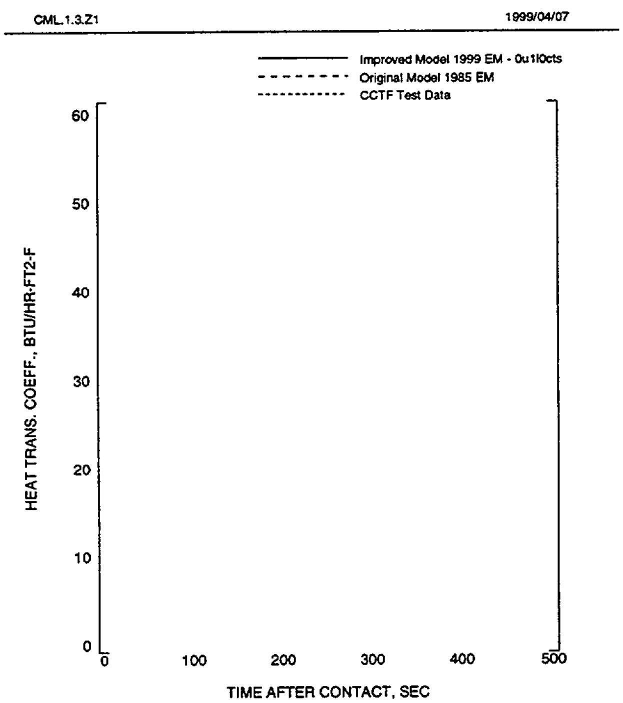


Figure 2.6-51
CCTF CORE-II TEST COMPARISON NO. C2-3/61
HEAT TRANS. COEFF.
ELEV= 6.57

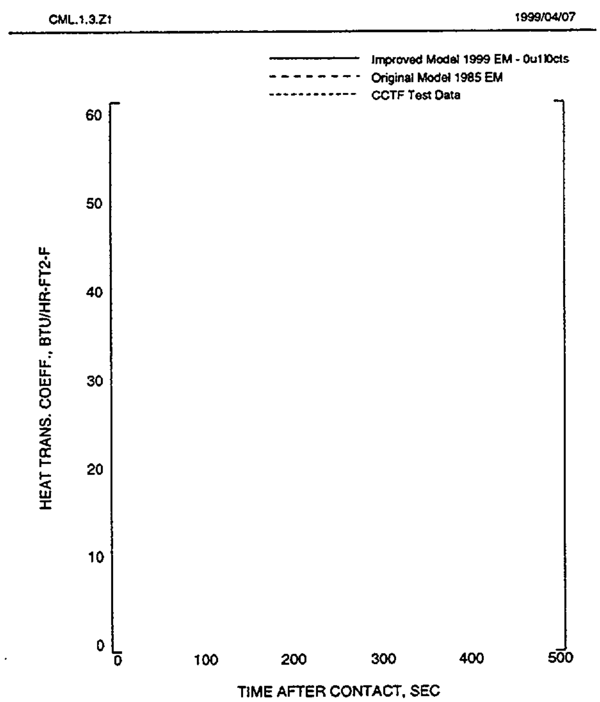


Figure 2.6-52
CCTF CORE-II TEST COMPARISON NO. C2-3/61
HEAT TRANS. COEFF.
ELEV= 7.96

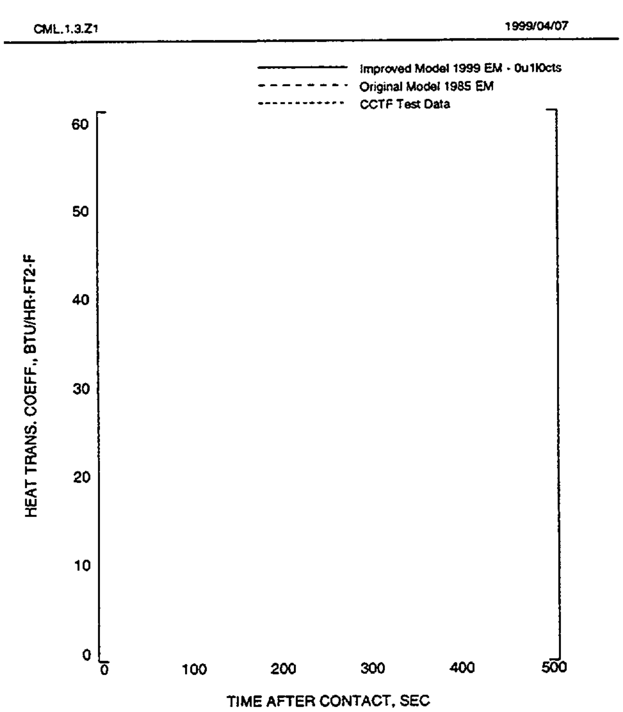


Figure 2.6-53
CCTF CORE-II TEST COMPARISON NO. C2-3/61
HEAT TRANS. COEFF.
ELEV= 9.44

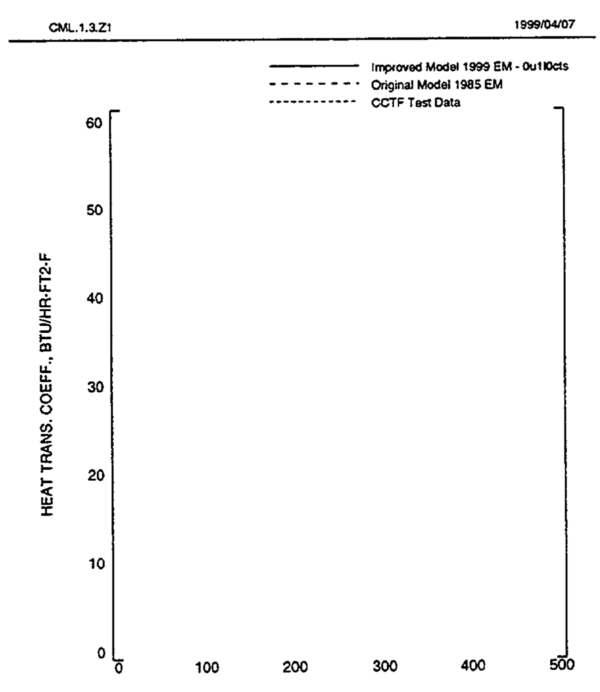


Figure 2.6-54
CCTF CORE-II TEST COMPARISON NO. C2-8/67
HEAT TRANS. COEFF.
ELEV= 3.80

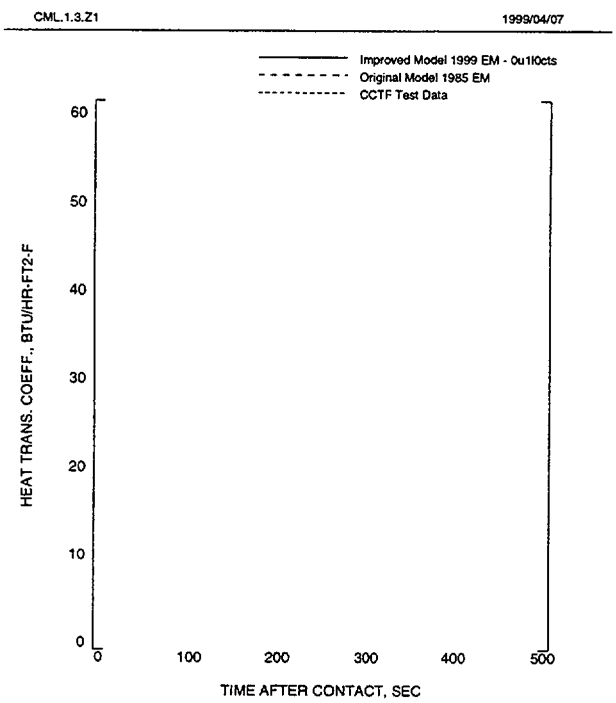


Figure 2.6-55
CCTF CORE-II TEST COMPARISON NO. C2-8/67
HEAT TRANS. COEFF.
ELEV= 6.57

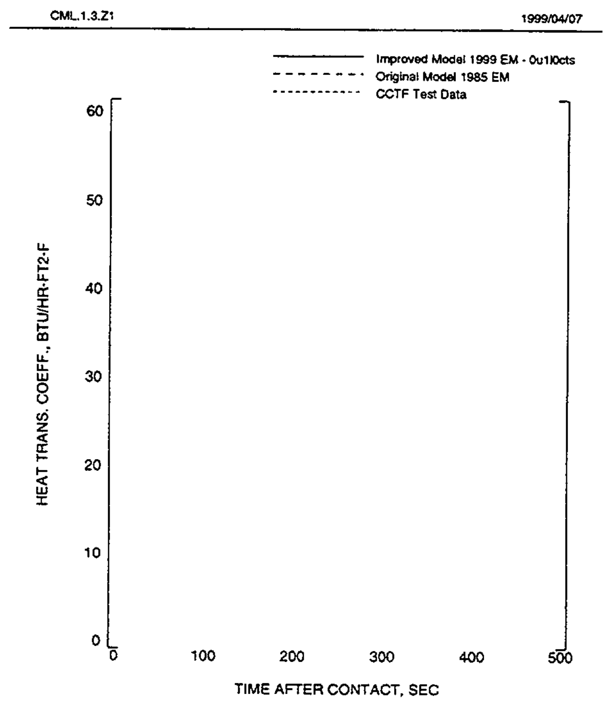


Figure 2.6-56
CCTF CORE-II TEST COMPARISON NO. C2-8/67
HEAT TRANS. COEFF.
ELEV= 7.96

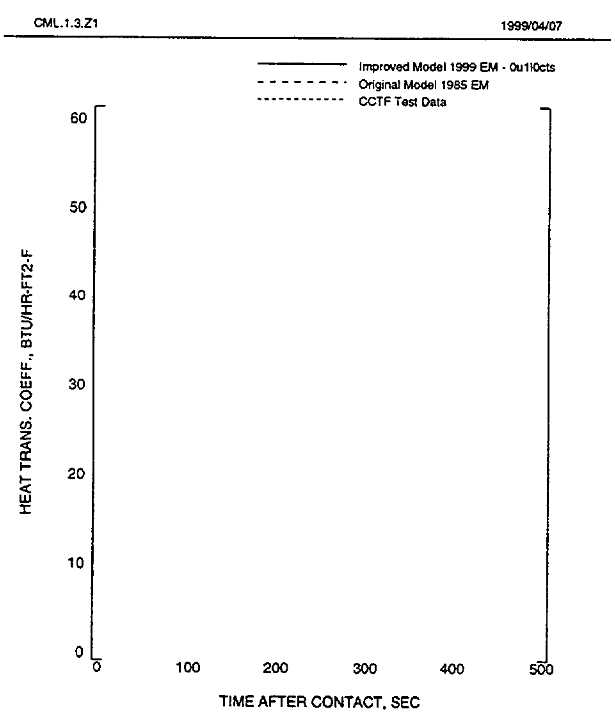


Figure 2.6-57
CCTF CORE-II TEST COMPARISON NO. C2-8/67
HEAT TRANS. COEFF.
ELEV= 9.44

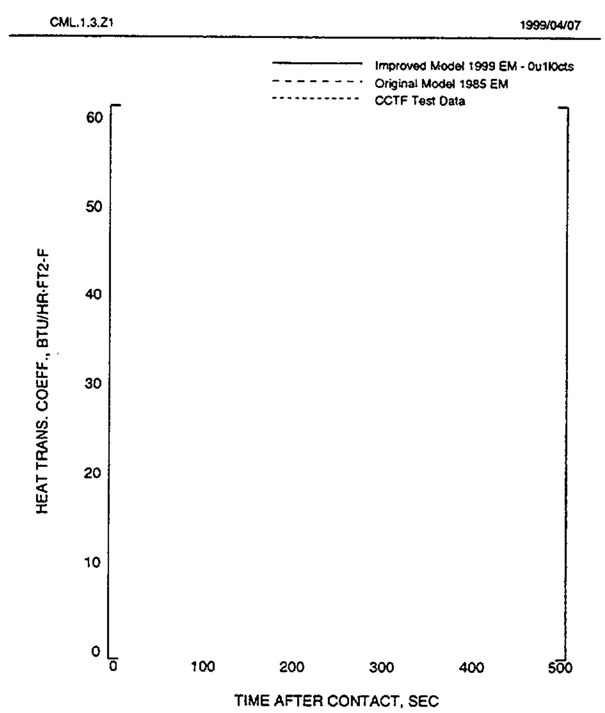


Figure 2.6-58
COMPARISON OF 1999 EM VERSUS 1985 EM
HEAT TRANS. COEFF.
ELEV= 7.16

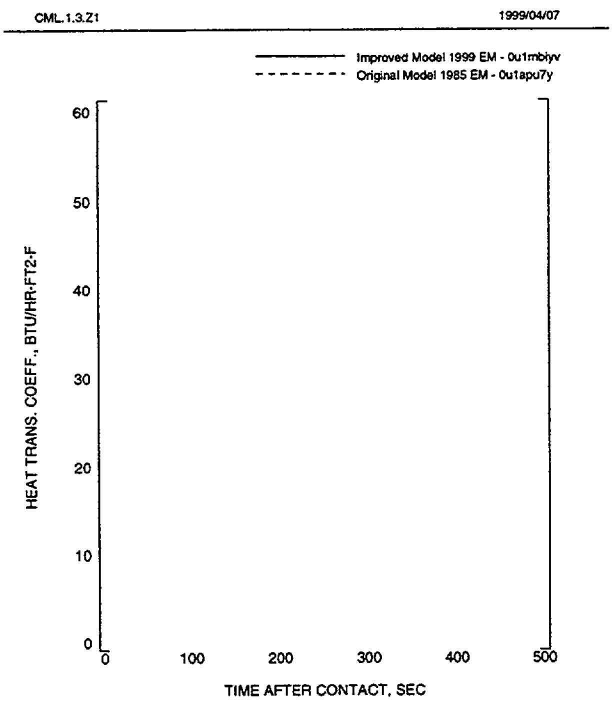


Figure 2.6-59
COMPARISON OF 1999 EM VERSUS 1985 EM
HEAT TRANS. COEFF.
ELEV= 7.52

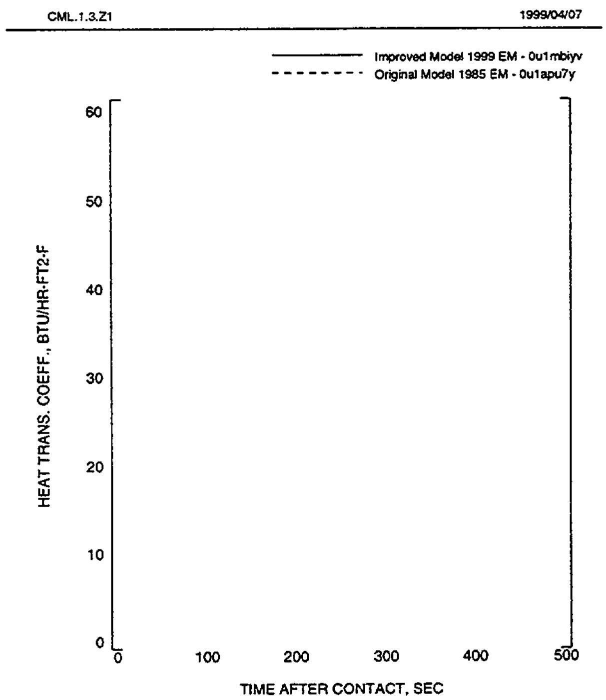


Figure 2.6-60
COMPARISON OF 1999 EM VERSUS 1985 EM
HEAT TRANS. COEFF.
ELEV= 7.87

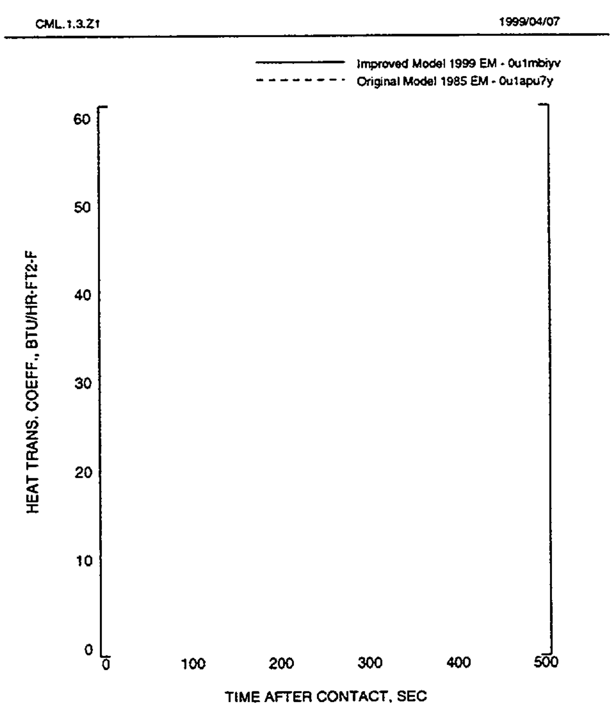
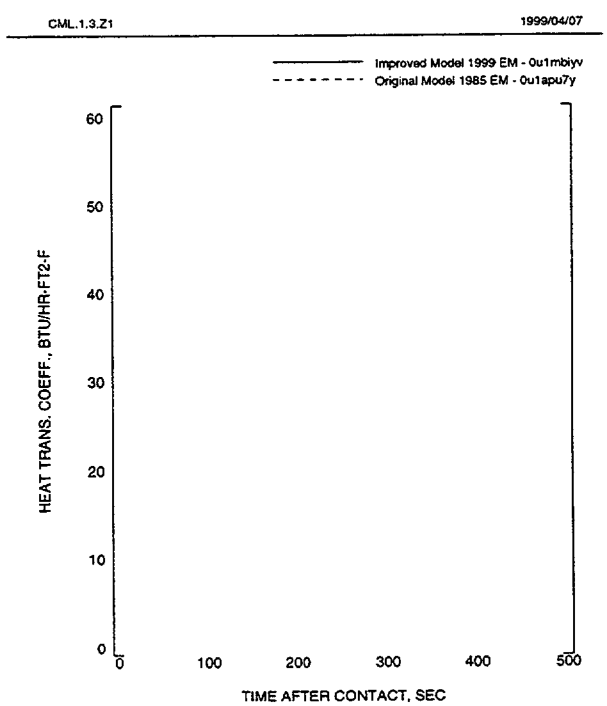


Figure 2.6-61
COMPARISON OF 1999 EM VERSUS 1985 EM
HEAT TRANS. COEFF.
ELEV= 8.18



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2.7 STRIKIN-II Hot Rod Steam Cooling Heat Transfer

The 1985 EM implements the hot rod steam cooling heat transfer calculations by executing in stages STRIKIN-II (Reference 2.7-1), PARCH (Reference 2.7-2), and HCROSS (Reference 2.7-3) and then by manually transferring data between these codes. The execution of these 1985 EM codes is as follows:

- Execute STRIKIN-II until the time of initiation of the hot rod steam cooling heat transfer calculations (maximum of rupture time and time of the start of less than one inch/sec core reflood)
- Manually transfer the hot channel blockage from STRIKIN-II to HCROSS, then execute HCROSS and calculate the hot channel flow redistribution at and above the rupture node.
- Manually transfer the rupture node and the hot rod axial temperature distribution at the time of the initiation of the steam cooling heat transfer calculations from STRIKIN-II to PARCH, then initialize the PARCH steam flow redistribution at and above the rupture node using the flow profile calculated by HCROSS, and execute PARCH to calculate the hot rod steam cooling heat transfer coefficients at and above the rupture node.
- Manually transfer to STRIKIN-II the steam cooling heat transfer coefficients calculated by the PARCH code and execute STRIKIN-II for the remainder of the transient.

Through the 1999 EM automated/integrated code system (Section 2.1), the new version of STRIKIN-II now performs the[

]

In addition, the 1999 EM version of STRIKIN-II incorporates several[

]

The 1999 EM changes are the following:

[

]

2.7.1 Model Description

2.7.1.1 Addition of Rod-to-Rod Radiation to the PARCH Fuel Rod Model

The 1985 EM fuel rod model in PARCH[

]

The STRIKIN-II rod-to-rod radiation model (Reference 2.7-1, Appendix L) calculates the radiation heat flux for the hot rod using a[ ]
This calculation is performed every time step at every axial node. [

]

2.7.1.2 Transfer of FLECHT Heat Transfer Coefficients for the Rupture Node to PARCH

Because of the large blockage fraction for the rupture node, PARCH usually calculates large steam cooling heat transfer coefficients (larger than the FLECHT heat transfer coefficients calculated in STRIKIN-II). The STRIKIN-II hot rod heatup analysis is required (by SER limitation) to use the smaller FLECHT heat transfer coefficient rather than the PARCH generated steam cooling value. In order to[

] The coefficient in PARCH is corrected for steam temperature as follows:

$$h_{PARCH} = h_{STRIK} (T_{clad} - T_{sat}) / (T_{clad} - T_{stm})$$

where

h_{PARCH} = PARCH heat transfer coefficient (Btu/ft^2-hr-°F)

h_{STRIK} = STRIKIN-II heat transfer coefficient (Btu/ft^2-hr-°F)

T_{clad} = Cladding temperature (°F)

T_{sat} = Saturation temperature (°F)

T_{stm} = Steam temperature (°F)

2.7.1.3 Transfer of all HCROSS Steam Cross-Flows to the PARCH Steam Channel

The 1985 EM coolant flow blockage and steam flow redistribution model is implemented by running the HCROSS and PARCH codes. Since HCROSS is run with[

]The 1999 EM steam channel energy balance was modified fully consistent with the 1985 EM methodology described in Reference 2.7-3.

2.7.2 Model Assessment

2.7.2.1 Implementation of the STRIKIN-II Rod-to-Rod Radiation Model into PARCH

The 1985 EM STRIKIN-II rod-to-rod radiation model is an approved NRC model. Implementation of the STRIKIN-II radiation heat flux into PARCH[

] The 1985 EM STRIKIN-II rod-to-rod radiation model is[ ] for application to LBLOCA, through the radiation enclosure selection process, which by an SER constraint, is selected to produce the [ ]

2.7.2.2 Transfer FLECHT Heat Transfer Coefficient for the Rupture Node to PARCH

Transfer of the FLECHT heat transfer coefficients to PARCH for the rupture node[ ] Use of the minimum of the FLECHT and steam cooling heat transfer coefficients in STRIKIN-II is a requirement of the SER for the hot rod temperature calculations.

2.7.2.3 Transfer of all HCROSS Steam Cross-Flows to the PARCH Steam Channel

Transfer of all the HCROSS steam cross-flows to PARCH is[ ]

2.7.3 Application to LBLOCA Analysis

Table 2.7-1 shows that the implementation of the hot channel steam cooling heat transfer modifications described in this section produces[ ]

2.7.4 Applicability to LBLOCA Analysis

- The STRIKIN-II rod-to-rod radiation model is an approved NRC model for LBLOCA analysis. The STRIKIN-II calculated heat flux is implemented into PARCH and is applied fully consistent with the STRIKIN-II application.
- The use of the minimum of the FLECHT and the steam cooling heat transfer coefficients in the PARCH code for the rupture node is consistent with the SER requirement on the STRIKIN-II calculation.
- The HCROSS steam cross-flow model for the calculation of flow blockage and flow redistribution is also a NRC approved model for LBLOCA calculations.

2.7.5 Model as Coded

2.7.5.1 Implementation of the STRIKIN-II Rod-to-Rod Radiation Heat Flux into PARCH

The rod-to-rod radiation heat flux is calculated in STRIKIN-II. The radiation heat flux is transferred to PARCH at each time step and each STRIKIN-II axial node through the heat flux interface variable in units of Btu/hr-ft^2.

The PARCH program transposes the radiation heat flux array from the STRIKIN-II nodalization into an array for the PARCH nodalization by[

]

To implement the radiation heat flux into the integration of the fuel rod temperature equations in PARCH, the term b_3 in Equation (3.2.1-21) in CENPD-138, Reference 2.7-2, is modified as follows:

[

]

2.7.5.2 Transfer of FLECHT Heat Transfer Coefficients for the Rupture Node to PARCH

STRIKIN-II calculates the heat transfer coefficients for all nodes during the reflood period. For the hot rod nodes equal to or above the rupture node elevation, the required heat transfer coefficient is equal to [

] In PARCH, the interface heat transfer coefficient is temperature corrected using the steam temperature, and is used to bound the new steam cooling heat transfer coefficient.

2.7.5.3 Transfer of all HCROSS Steam Cross-Flows to the PARCH Steam Channel

The steam cross-flows are calculated in HCROSS as axial flow fractions. The PARCH code reads the HCROSS flow fractions and defines the PARCH code flow fraction variables. The steam flow at each elevation is defined at each HCROSS axial node. This flow rate is [

] The PARCH coolant energy balance is calculated using the same methodology as the 1985 EM with a finer nodalization.

2.7.6 References

2.7-1 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

2.7-2 CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

2.7-3 Enclosure 1-P-A to LD-81-095, "C-E ECCS Evaluation Model Flow Blockage Analysis," December 1981.

Table 2.7-1

Effect of the 1999 EM Steam Cooling Heat Transfer Changes on PCT

	STRIKIN-II PCT °F	Steam Cooled PCT °F
[		
		]

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2.8 Required Input Options for the 1999 EM

Appendix A lists the new input, new input options, and new output options, which are required to execute the 1999 EM.

The 1999 EM options are controlled through the User Control Interface (UCI) file. This is a generic file that is used to control the model options in all of the component codes, which constitute the 1999 EM automated/integrated code system.

A new input file is defined for the CEFLASH-4A code (Reference 2.8-1) and the COMPERC-II code (Reference 2.8-2). This input file contains new input data to execute CEFLASH-4A and COMPERC-II as part of the automated/integrated code system. The file provides new input for the steam generator secondary side modeling, new input for the automatic spray and spillage model, and new input to calculate the time of safety injection pump actuation.

References

2.8-1 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

2.8-2 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

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3.0 LBLOCA ECCS PERFORMANCE BREAK SPECTRUM ANALYSIS

3.1 Introduction

This section presents the results of a LBLOCA ECCS performance break spectrum analysis using the 1999 EM for a typical Combustion Engineering designed PWR. The analysis is performed (1) to demonstrate typical results using the improved models of the 1999 EM, and (2) to compare the results to those obtained from a 1985 EM analysis. This section also presents an assessment of the overall conservatism of the 1999 EM by comparing the reduction in peak cladding temperature (PCT) produced by the 1999 EM modifications to the reduction produced by a non-EM case with selected Appendix K requirements removed.

Both PCT and peak local cladding oxidation percentage (PLO) results are tabulated and plotted for these comparisons. However, the descriptions of calculated differences are characterized primarily in terms of PCT, which is the more limiting ECCS performance criterion for these analyses. Core-wide cladding oxidation percentage is commonly reported in break spectrum analyses as ≤0.99%, a value just below the ECCS performance criterion. This ECCS performance indicator is not normally limiting for the LBLOCA analysis; therefore, reporting this value gives maximum flexibility for future evaluations. The 1999 EM improvements in the[ ]described in this topical report will[ ]the calculation of core-wide cladding oxidation compared to current evaluations.

Section 3.2 discusses two open licensing issues related to the 1985 EM, which impact the implementation of the 1985 EM process improvements and the 1999 EM. These two open issues are (1) applicability to non-CE Nuclear Power LLC manufactured fuel, and (2) referencing CENPD-133 Supplement 4-P. With the exception of the 1999 EM improvements described in Sections 2.2 through 2.7, which require NRC acceptance, the 1999 EM automated/integrated code system remains in compliance with all SERs relevant to the 1985 EM and is fully compliant with all current SER constraints or limitations for the 1985 EM.

Section 3.3 summarizes the plant design data and general characteristics used in LBLOCA ECCS performance analyses and presents the results of parametric studies that analyze the impact of using the 1999 EM with variations in selected parameters. Section 3.4 reviews the LBLOCA method of analysis in view of the improvements incorporated in the 1999 EM. Section 3.4 also presents the results of worst single failure analyses using both the 1985 EM simulation with its process improvements and the 1999 EM. The results and conclusions presented in Section 3.0 showing the impact of the 1999 EM for the plant characteristics and analysis methods given in Sections 3.3 and 3.4 are representative of the LBLOCA behavior of all Combustion Engineering designed PWRs.

Section 3.5 presents the results of a LBLOCA analyzed with the 1985 EM compared to the results analyzed with the 1999 EM using the same set of plant design inputs. One of the comparisons shows that with the same set of inputs the automated/integrated code system, when executed in a manner simulating the 1985 EM, produces[ ]PCT result.

Section 3.6 presents the results of the LBLOCA spectrum analysis using the 1999 EM. That is, this section presents results of an application analysis in which all of the modifications to the 1985 EM designed to reduce PCT are fully implemented in accordance with the CE Nuclear Power LLC ECCS 1999 EM performance analysis methodology.

Section 3.7 presents an assessment of the overall conservatism of the 1999 EM. The assessment compares PCT results of the LBLOCA analysis using the 1999 EM to the PCT margin associated with the removal of three conservative Appendix K model requirements.

3.2 Compliance with SER Constraints and Limitations

With the exception of the 1999 EM improvements described in Section 2.0, the 1999 EM automated/integrated code system remains in compliance with all SERs relevant to the 1985 EM and is fully compliant with all current SER constraints or limitations for the 1985 EM. While the model improvements incorporated into the 1999 EM represent model changes that require NRC acceptance, their introduction does not change or invalidate existing model requirements as previously accepted by NRC.

However, there are two issues related to the current licensing basis of the 1985 EM that need to be addressed with this topical report submittal.

1. **Applicability to Non-CE Nuclear Power LLC Manufactured Fuel**
   One SER constraint from Reference 3.0-1 implies the applicability of the 1985 EM is only to CE Nuclear Power LLC manufactured Zircaloy clad fuel. The licensing issues in question in 1985-1986 related to the implementation of the NRC-prescribed NUREG-0630 cladding rupture and blockage models. The model changes implemented in the 1985 EM at that time were generic to the models described in NUREG-0630, and the implementation of those models made no distinction or exception regarding fuel manufacturer. The reasons for NRC implying such a limitation remain unclear today. Because of the wording of Reference 3.0-1, clarification is needed in order to apply the 1985 EM and subsequently, the 1999 EM to LBLOCA ECCS performance analyses with non-CE Nuclear Power LLC manufactured fuel assemblies. Such conditions may occur from non-CE Nuclear Power LLC manufactured fuel assemblies in co-resident core configurations during transition operation from one fuel vendor to another. It is concluded from the studies conducted during the development and justification of the 1999 EM, that no impediments or restrictions exist regarding the applicability of the NUREG-0630 models to non-CE Nuclear Power LLC manufactured fuel assemblies using Zircaloy cladding as implemented in the 1985 EM and in the 1999 EM automated/integrated code system. It is important to note

that no other SER for the 1985 EM or earlier versions of the EM implied any such similar limitation. Moreover, CE Nuclear Power LLC believes that the 1985 EM and the 1999 EM are in fact applicable to analysis of mixed core configurations.

Resolution of this issue is also important because of the acquisition of CE Nuclear Power LLC by Westinghouse. This means that the fuel for Combustion Engineering PWRs will be manufactured at a Westinghouse facility and could therefore be considered Westinghouse manufactured fuel, not CE manufactured fuel.

2. Referencing CENPD-133 Supplement 4-P

The SER for the 1985 EM, Reference 3.0-1, inadvertently failed to cite in its reference list one of the topical report supplements that comprise the 1985 EM; and which will also comprise the 1999 EM. During our research for the 1999 EM model changes documented in this submittal, no evidence was found regarding NRC disposition of CEFLASH-4A Supplement 4. It is possible that the supplement did not require an SER since it impacted PCT by less than the 20°F limit imposed in the 1970's. Furthermore, the exact model change had been made earlier to the STRIKIN-II code and was approved by the staff. Nevertheless, there should be closure to this issue since the submitted change is made in order to conform to the Appendix K requirement on "no return to nucleate boiling." The 1999 EM automated/integrated code system licensing basis includes the referenced supplement since it is part of the 1985 EM. All of the methodology changes described in CENPD-133 Supplement 4-P, including the change made to conform to the Appendix K requirement on "no return to nucleate boiling," were incorporated into the 1985 EM.

3.3 Plant Design Data

Table 3.3-1 presents a summary of the key plant design data utilized in typical LBLOCA analyses. The majority of these general characteristics of the plant are selected to be bounding from an ECCS performance analysis viewpoint and, therefore, represent a source of conservatism incorporated into the 1999 EM LBLOCA analysis. As Table 3.3-1 indicates, all of the key plant design data with high impact (greater than ±25°F) or medium impact (greater than ±10°F) are represented with analysis maximum or minimum values to account for variations related to the parameter. Since ECCS performance sensitivities and plant parameter values may vary greatly from plant design to design, the characterizations shown in Table 3.3-1 are general in nature and are not meant to be quantitatively precise.

Table 3.3-1 compares the sensitivity of each key parameter in the 1985 EM to its sensitivity in the 1999 EM. With the exception of the ECCS component parameters (SITs and Pumps), the sensitivity of these key plant design parameters is not affected by the 1999 EM changes. Section 3.4 describes the impact of various ECCS worst single failure analysis conditions on analyses using the 1985 EM process improvements and the 1999 EM.

Table 3.3-1 identifies several key parameters that are either changed in usage for the 1999 EM and/or substantiated by sensitivity studies. These parameters for the 1999 EM are marked with **Bold** in the table and are listed as follows:

- Steam generator initial pressure
- Steam generator secondary side initial inventory
- Steam generator secondary side metal heat capacity
- Safety injection pump flow rates (worst single failure, see Section 3.4)
- Refueling water storage tank (RWST) temperature
- Safety injection pump actuation signal setpoint
- Safety injection pump delivery delay
- Containment spray pump number (worst single failure, see Section 3.4)

The following three parameter studies were conducted with the 1999 EM to establish the sensitivity of the ECCS performance results for typical variations in (1) RWST temperature, (2) steam generator secondary modeling, and (3) safety injection pump actuation time. The analyses utilized all of the 1999 EM improvements including the automated/integrated code system. The ECCS delivery was represented for the configuration of no failure or maximum delivery.

3.3.1 Refueling Water Storage Tank Temperature

This parameter study analyzed the impact of using minimum and maximum RWST temperature in the reflood hydraulics portion of the LBLOCA transient. The input for COMPERC-II in the 1985 EM requires two entries for this temperature, one entry for the safety injection pumps and one for the containment spray pumps. As shown in Table 3.3-1, for conservatism, the 1985 EM uses [ ] temperature for safety injection [ ] and [ ] temperature for containment spray [ ]

The 1985 EM process change includes automatic spray and spillage, which is designed to consistently represent the spray and spillage components. Therefore, the 1999 EM includes modeling the RWST with [ ]

Table 3.3-2 compares the results of the analysis using minimum and maximum RWST temperatures. [ ]

3.3.2 Steam Generator Secondary Initial Pressure and Physical Parameters

The 1999 EM model improvement for steam venting reflood thermal-hydraulics [ ] the calculated primary side steam superheat by [ ] of the steam generator secondary side. This parameter study analyzed the impact of increasing the steam generator secondary side initial pressure by 50 psia and increasing the physical characteristics of the steam generator secondary side by 10%. Increasing the physical characteristics by 10%, [ ]

Table 3.3-3 shows that this magnitude of variation in the key parameters representing the secondary side conditions in the 1999 EM changes the PCT by [ ]

3.3.3 Safety Injection Pump Actuation Time

This parameter study analyzed the impact of three different times for safety injection pump actuation time. The three cases analyzed were for the following:

(1) Safety injection actuated during early reflood (based on SIAS and delay time),
(2) Safety injection actuated at the end of blowdown, that is, at the time of annulus downflow (TAD), and
(3) Safety injection actuated after the SITs empty.

The third case is the 1985 EM method used in CE Nuclear Power LLC ECCS analyses. The first and second cases allow earlier actuation, which acts to [ ]

This early actuation feature was introduced into the 1985 EM version of the COMPERC-II code for non-Combustion Engineering plant design applications. Its use in the 1985 EM is contingent on an accompanying worst single failure analysis. This aspect of the 1985 EM remains unchanged for the 1999 EM.

Table 3.3-4 provides the results of the three cases analyzed in this parameter study. Earlier actuation of safety injection pump delivery, Cases (1) and (2), is[

]

Table 3.3-1
General Guidelines for LBLOCA Design Input Parameters for the 1999 EM
(1999 EM items in **Bold** have supporting sensitivity results)

System Parameter	*1985 EM*	*Impact **	*1999 EM*	*Notes*

[

]

Table 3.3-2

1999 EM Parameter Study for RWST Temperature

[

Result	1999 EM RWST Min 35°F	1999 EM RWST Max 110°F

]

Table 3.3-3

1999 EM Parameter Study

Steam Generator Secondary Physical Parameters and Initial Pressure

[

Result Physical Parameters: Initial Pressure:	1999 EM Nominal 800 psia	1999 EM Increase 10% 800 psia	1999 EM Nominal 850 psia

]

Table 3.3-4

1999 EM Parameter Study

Safety Injection Pump Actuation Time

[

Result	Case (1) 1999 EM SI on Early Reflood	Case (2) 1999 EM SI on at TAD	Case (3) 1999 EM SI on after SITs Empty

]

3.4 Method of Analysis

The spectrum analyses described in Section 3.6 were performed for a typical range of break sizes used in CE Nuclear Power LLC LBLOCA analysis methodology. All of these breaks in the spectrum are double-ended discharge leg guillotine breaks (DEDLG), with varying break size multipliers. As demonstrated in previous licensing submittals (Reference 3.0-2), slot breaks are not limiting for the CE Nuclear Power LLC Appendix K ECCS EM, and are therefore not analyzed. The discharge leg guillotine break is limiting because both the core flow rate during blowdown and the core reflood rate are minimized for this location and type of pipe break.

Four break sizes are analyzed in the break spectrum analyses: 1.0xDEDLG, 0.8xDEDLG, 0.6xDEDLG, and 0.4xDEDLG. The spectrum analysis results and conclusions presented in Section 3.6 showing the impact of the 1999 EM are representative of all break sizes and break locations used for LBLOCA analysis of ECCS performance.

The NRC-approved method of analysis for the 1985 EM is unchanged for the 1999 EM. This method of analysis covers the following ancillary studies, which are normally part of each full application of the EM:

[

]

Each of these studies pertain to aspects of modeling the hot rod except the worst single failure study. Hot rod modeling and selection of the limiting conditions to be used in the spectrum analysis are not changed by any of the modifications to the 1985 EM that comprise the 1999

EM. However, the standard worst single failure study has been affected by the following 1985 EM simulation process improvements and the 1999 EM improvements: (1) the automated/integrated code system with automatic spray and spillage calculations, (2) the improved models for COMPERC-II steam venting reflood thermal-hydraulics, and (3) steam/water interaction during Nitrogen discharge. Therefore, the following results demonstrate the impact of the 1985 EM and 1999 EM improvements on the worst single failure study for a typical Combustion Engineering designed PWR application analysis. Tables 3.4-1 and 3.4-2 summarize the ECCS performance results for the worst single failure cases. The worst single failure cases consist of the following:

1. No failure of an ECCS component with maximum delivery of safety injection inventory to the RCS from 2 Low Pressure Safety Injection (LPSI) pumps, 2 High Pressure Safety Injection (HPSI) pumps, 4 Safety Injection Tanks (SITs), and maximum containment spray from two spray pumps
2. Loss of a LPSI pump with minimum delivery of safety injection inventory to the RCS and maximum containment spray from two spray pumps
3. Loss of a diesel generator with minimum delivery of safety injection inventory to the RCS and maximum containment spray from one spray pump. In the following studies, this case of a loss of a diesel generator is analyzed for the 1999 EM only.

3.4.1 <u>1985 EM Simulation</u>

This analysis establishes a reference point for comparison (used here and in Section 3.6 for the spectrum analysis). This 1985 EM analysis is performed by deliberately selecting a PLHGR that produces a PCT very close to the ECCS performance acceptance criterion limit of 2200°F. This analysis for a typical PWR includes the 1985 EM representation of no failure of an ECCS component in the manually prepared spray and spillage tables that are input to the COMPERC-II reflood hydraulics code. These tables also represent maximum delivery of containment spray from two spray pumps. The RWST temperature used for the spray and spillage tables is conservatively specified to be the[ ]temperature. In the 1985 EM representation, even though spray and spillage is based on no ECCS failure, safety injection pump delivery to the RCS is represented with[ ] The RWST temperature used for the pumped safety injection is conservatively specified to be the[ ]temperature. Safety injection pump actuation is modeled to occur after the SITs empty. For this 1985 EM simulation analysis, the process

changes for the 1985 EM automated/integrated code system, including the automatic spray and spillage model are not utilized. Table 3.4-1 shows that the PCT for this case is 2199°F. Figures 3.4-1 through 3.4-4 show the transient response of this 1985 EM simulation case.

Note that the graphical output shown in this topical report for PWR applications consists of plots from three computer codes: CEFLASH-4A, COMPERC-II, and STRIKIN-II. The computer code associated with the plot is identified in the header of each graph. The CEFLASH-4A plots show the transient response from the time of the break to the time of the end of blowdown (typically on the order of 25 seconds). The COMPERC-II plots utilize two different time scales. The containment pressure graph shows the transient response from the time of the break to the end of reflood (typically on the order of 500 seconds). The remaining COMPERC-II graphs show the transient response starting with time zero at contact time, that is, the time when the reflood coolant contacts the bottom of the core (typically on the order of 30 seconds after the break). The STRIKIN-II plots show the response from the time of the break to the end of the transient.

3.4.2 1985 EM Simulation Worst Single Failure Analysis

This section presents the results of two worst single failure analyses using the 1999 EM automated/integrated code system run in 1985 EM simulation mode with the process improvements described in Section 2.1. The two worst single failure cases are (i) no failure of an ECCS component, and (ii) the loss of a LPSI pump. These cases utilized the 1999 EM automated/integrated code system executed with options, features, and approved models selected to be fully consistent with the 1985 EM, including the process improvements related to explicit NUREG-0630 cladding rupture and reduction of discretionary conservatisms described in Section 2.1. Appendix A provides a description of the required code inputs for this mode of analysis. The process changes analyzed included the following:

- Explicit NUREG-0630 swelling and rupture with actual calculated heating rate dependent rupture and blockage.
- Automatic calculation of spray and spillage during the reflood thermal-hydraulics analysis
- A [ ] value for the refueling water storage tank (RWST) temperature (see discussion and parameter study in Section 3.3.1)
- Steam generator secondary side wall heat included in the blowdown thermal-hydraulics analysis (see discussion and parameter study in Section 3.3.2)

i. 1985 EM Simulation with Process Improvements and No ECCS Failure

This case represented no failure of an ECCS component. That is, maximum ECCS delivery to the RCS from two LPSI and two HPSI pumps is represented. In the 1985 EM, delivery to the

broken cold leg is spilled directly to the containment. Also, the safety injection pumps were actuated during early reflood. This parameter representation of the timing of safety injection pump actuation in the 1985 EM and in the 1999 EM is discussed and analyzed with a parametric study in Section 3.3.3.

Table 3.4-1 shows that the specially designed process changes for the 1985 EM simulation resulted in a PCT of [ ] which is a [ ] in PCT of [ ] relative to the 1985 EM simulation case.

Figures 3.4-1 through 3.4-4 show comparisons of the 1985 EM simulation with process improvements and with no failure of an ECCS component to the previous case for the 1985 EM simulation. Figures 3.4-1 and 3.4-2 show reflood thermal-hydraulics characteristics as calculated by the COMPERC-II code. [

]

ii. 1985 EM Simulation with Process Improvements and Loss of a LPSI Pump

This analysis utilizes the 1985 EM with process improvements including the automated/integrated code system and the automatic spray and spillage model. ECCS delivery is represented by the minimum injection to the cold legs from one LPSI pump and one HPSI pump. Safety injection pump actuation is modeled to occur prior to the SITs emptying during early reflood. Maximum delivery of containment spray from two spray

pumps is represented. Table 3.4-1 shows that the PCT for this case is [ ] which is a [ ] relative to the case with no ECCS component failure.

Figures 3.4-1 through 3.4-4 show this case compared to the previous case, the 1985 EM with no failure of an ECCS component. [

]

In the gravity reflood process, the downcomer is refilled by delivery from the SITs and remains filled during LPSI and HPSI pump discharge. The loss of a LPSI has [ ] on the downcomer level and on the gravity reflood process except through the effects described above.

3.4.3 <u>1999 EM Worst Single Failure Analysis</u>

This section presents the results of three worst single failure analyses using the 1999 EM automated/integrated code system. The three worst single failure cases are (i) no failure of an ECCS component, (ii) the loss of a LPSI pump, and (iii) the loss of a diesel generator. These cases utilized all of the proposed 1999 EM improvements including the automated/integrated code system with the automatic spray and spillage model. This includes modeling the refueling water storage tank (RWST), which supplies inventory for both the containment spray pumps and the safety injection pumps, with a [

] as described in Section 3.3. Appendix A provides a description of the required code inputs for this mode of analysis.

i. 1999 EM with No Failure of an ECCS Component

For this case of no failure of an ECCS component, maximum ECCS delivery to the RCS from two LPSI and two HPSI pumps is represented. As in the 1985 EM, delivery to the broken cold leg is spilled directly to the containment. Safety injection pump actuation is modeled to occur prior to the SITs emptying. Maximum delivery of containment spray from two spray pumps is represented. Table 3.4-2 shows that the PCT for this case is [ ] which is a [ ] compared to the 1985 EM simulation case. Figures 3.4-5 through 3.4-8 show the transient response for this 1999 EM case with no ECCS component failure.

ii. 1999 EM with Failure of a LPSI Pump

This analysis utilizes all of the proposed 1999 EM improvements including the automated/integrated code system with the automatic spray and spillage model. ECCS delivery is represented by the minimum injection to the cold legs from one LPSI pump and one HPSI pump. Safety injection pump actuation is modeled to occur prior to the SITs emptying. Maximum delivery of containment spray from two spray pumps is represented. Table 3.4-2 shows that the PCT for this case is [ ] which is a [ ] relative to the case with no ECCS component failure.

Figures 3.4-5 through 3.4-8 show this case compared to the previous case, the 1999 EM with no failure of an ECCS component. [

]

iii. 1999 EM with Failure of a Diesel Generator

This analysis utilizes all of the proposed 1999 EM improvements including the automated/integrated code system with the automatic spray and spillage model. ECCS delivery is represented by the minimum injection to the cold legs from one LPSI pump and one HPSI pump. Containment spray delivery is represented with one spray pump, and for conservatism, maximum delivery is assumed. Table 3.4-2 shows that the PCT for this case is [ ]which is a[ ]relative to the case with no ECCS component failure.

Figures 3.4-5 through 3.4-8 show the impact of the failure of a diesel to the previous cases with no failure and loss of a LPSI. [

]

3.4.4 Summary of Worst Single Failure Study

In summary, this worst single failure analysis shows that no failure of an ECCS component produces the highest PCT for the 1999 EM. Consistent representation of ECCS injection to the cold legs and spillage of ECCS inventory to containment using the automatic spray and spillage model produces a worse single failure response that is slightly different than that calculated in the 1985 EM reference case, which is characterized by a conservative representation of these effects. The PCTs for the 1999 EM worst single failure cases were [ ] with the largest difference being [ ] Other plant configuration combinations of containment size and ECCS delivery rates, may lead to a different conclusion, therefore, the worst single failure analysis will be performed for each application of the 1999 EM. The worst single failure of an ECCS component must include consideration of the most limiting value of the RWST temperature as described in Section 3.3.1.

Table 3.4-1

1985 EM Worst Single Failure Analysis

[

Result	1985 EM Simulation	1985 EM No Failure	1985 EM Loss of LPSI

]

Table 3.4-2

1999 EM Worst Single Failure Analysis

[

Result	1999 EM No Failure	1999 EM Loss of LPSI	1999 EM Loss of Diesel

]

Figure 3.4-1
1985 EM WORST SINGLE FAILURE ANALYSIS COMPERC-II
CONTAINMENT PRESSURE

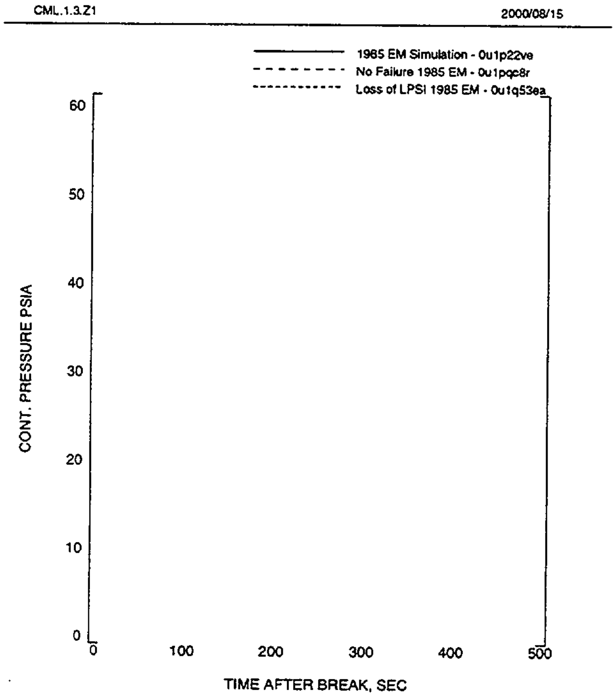


Figure 3.4-2
1985 EM WORST SINGLE FAILURE ANALYSIS COMPERC-II
REFLOOD LIQ. MASS

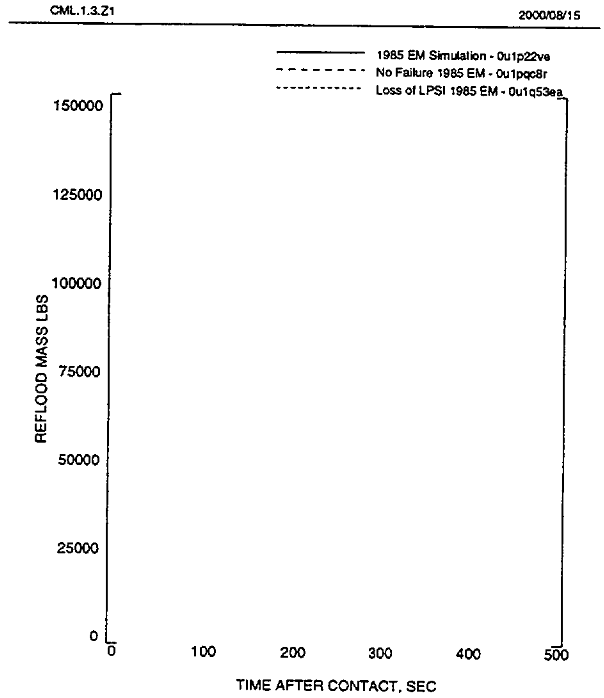


Figure 3.4-3
1985 EM WORST SINGLE FAILURE ANALYSIS HTCOF
HEAT TRANS. COEFF. NODE 14
ELEV= 7.74

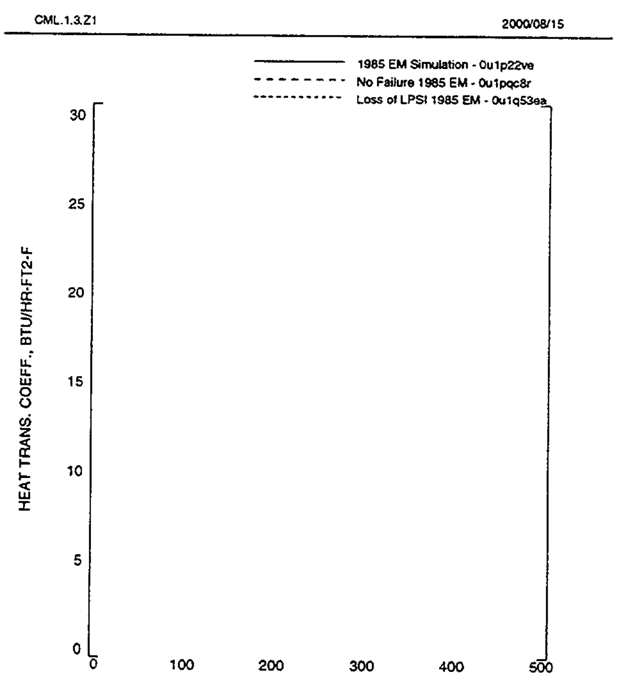


Figure 3.4-4
1985 EM WORST SINGLE FAILURE ANALYSIS STRIKIN-II
CLADDING TEMP NODE 14
HOT CHANNEL

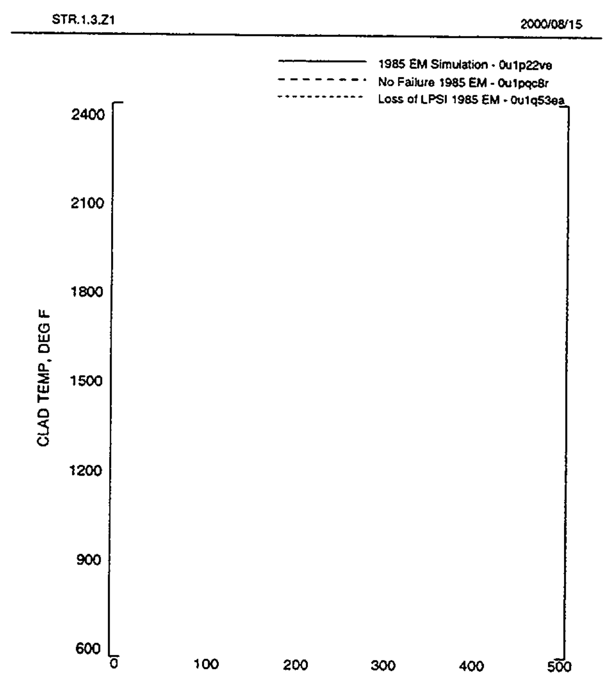


Figure 3.4-5
1999 EM WORST SINGLE FAILURE ANALYSIS COMPERC-II
CONTAINMENT PRESSURE

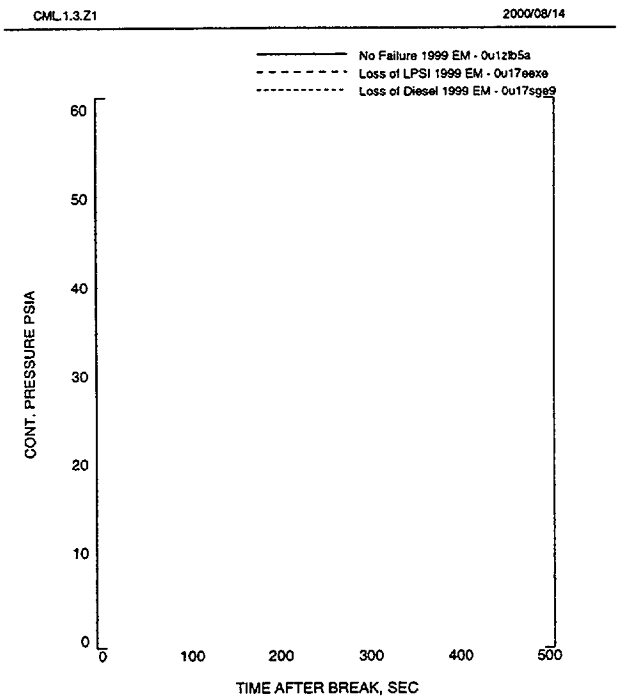


Figure 3.4-6
1999 EM WORST SINGLE FAILURE ANALYSIS COMPERC-II
REFLOOD LIQ. MASS

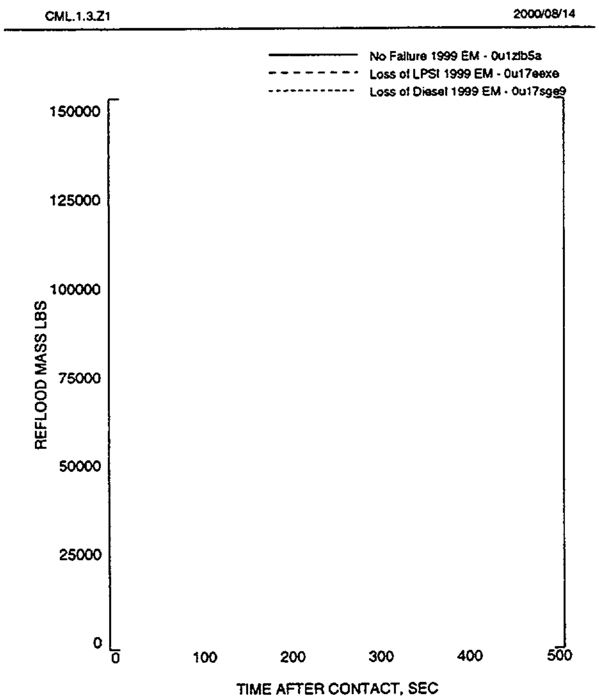


Figure 3.4-7
1999 EM WORST SINGLE FAILURE ANALYSIS HTCOF
HEAT TRANS. COEFF. NODE 14
ELEV= 7.74

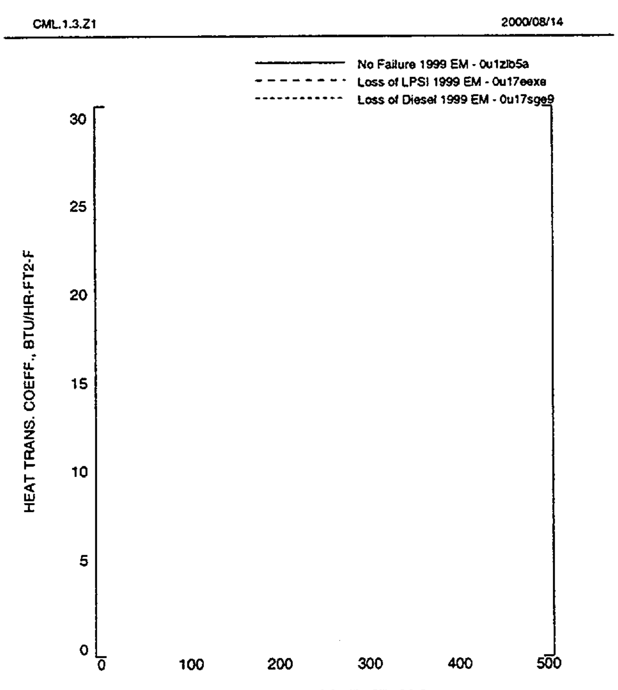
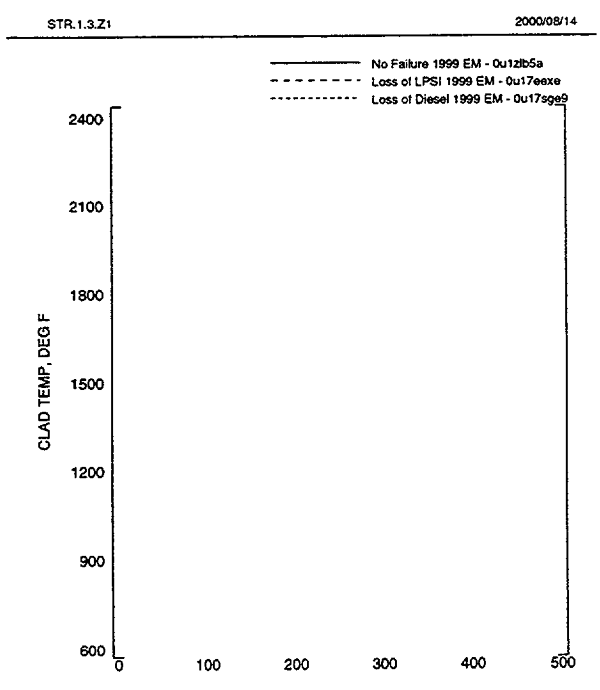


Figure 3.4-8
1999 EM WORST SINGLE FAILURE ANALYSIS STRIKIN-II
CLADDING TEMP NODE 14
HOT CHANNEL



3.5 Results Comparing the 1985 EM and the 1999 EM

This section presents the results of analyses comparing the 1999 EM automated/integrated code system to the results of analyses executed in the 1985 EM Simulation mode. The reference analysis is designated as the "Base AOR" (Analysis of Record) in Table 3.5-1. The PCT for this reference analysis was 2188°F, which was determined using the 1985 EM standard code system for a typical Combustion Engineering designed PWR. The following cases compare the Base AOR results to the results using the revised methodology:

3.5.1 1985 EM Simulation

This case utilized the 1999 EM automated/integrated code system executed with options, features, and approved models selected to be fully consistent with the 1985 EM. The process changes made to the 1985 EM within the currently NRC-accepted EM for explicit NUREG-0630 cladding swelling and rupture and reduction of discretionary conservatisms were not used. The 1999 EM automated/integrated code system was used to provide the interface data between the various codes, which in the Base AOR were manually prepared. Appendix A provides a description of the required code inputs for this mode of analysis.

The results in Table 3.5-1 show that the PCT for this 1985 EM simulation is 2189°F, only 1°F different from the Base AOR. The table of comparisons shows that the two calculations are nearly identical. This demonstrates that the automated/integrated code system is procedurally equivalent to the 1985 EM regarding the manner of transferring the interface data from one code to the next. The benefits of the automated/integrated code system are that the analysis results using the 1999 EM code system are produced with (1) greater numerical precision, (2) significantly reduced analysis effort, and (3) with the same overall conservatisms inherent in the 1985 EM.

3.5.2 1999 EM Analysis

This analysis utilized all of the proposed 1999 EM improvements including the removal of the Dougall-Rohsenow film boiling correlation. This analysis included the process changes described above, the changes in key design plant input parameter representations described in Section 3.3, the worst single failure conclusion in Section 3.4, and all of the proposed model improvements described in Section 2.0. This includes the use of a[ ] Appendix A provides a description of the required code inputs for this mode of analysis.

Table 3.5-1 shows that the full implementation of the 1999 EM[ ]the PCT from 2188°F for the Base AOR to[ ]which is a[ ] This result is comparable to the[ ]that was shown in Section 3.4 and in Table 3.4-1 for the worst single failure analysis comparison.

Figures 3.5-1 through 3.5-16 show comparisons of the calculations using the 1999 EM to the 1985 EM simulation designed to match the Base AOR. Figures 3.5-1 through 3.5-5 show blowdown thermal-hydraulic characteristics that are calculated by the CEFLASH-4A code. Figures 3.5-3 and 3.5-4 show the impact of the replacement of Dougall-Rohsenow film boiling heat transfer model with [ ]

Figures 3.5-6 through 3.5-11 and Figure 3.5-16 show reflood thermal-hydraulics characteristics that are calculated by the COMPERC-II code. [

]

Figures 3.5-13 through 3.5-15 show the PCT, the steam cooled limiting node above the rupture node, and the peak local cladding oxidation calculated by the STRIKIN-II code. Each of these figures shows the improvements calculated with the 1999 EM compared to the 1985 EM simulation.

Figure 3.5-16 shows that the condensation energy removal rate in the containment is[

]

In the 1985 EM, spray and spillage inputs to COMPERC-II are manually prepared in accordance with the approved methodology and are shown in Figure 3.5-16 as the solid line. In the 1999 EM automated/integrated code system, the COMPERC-II code automatically calculates the condensation energy removal from the steam phase of the containment due to the dispersal of cold water from several sources using actual code generated state dependent physical properties. These are the same sources used by the analyst in the manual process and are the sources approved for the 1985 EM, but are based on actual calculated conditions rather than on bounding conservative estimations. The sources of liquid spray and spillage in the 1985 EM are[

]

The data plotted in Figure 3.5-16 are the summations from each of these sources of the liquid flow rate times the enthalpy difference between saturated liquid at the containment steam partial pressure and the dispersed liquid. The dispersed liquid enthalpy may be the [

]

Table 3.5-1

Results Comparing the 1985 EM Simulation with the 1999 EM

Result	Base AOR	1985 EM Simulation	1999 EM

Figure 3.5-1
1985 EM VERSUS 1999 EM CEFLASH-4A
PRESSURE IN CENTER OF HOT ASSEMBLY
NODE 13

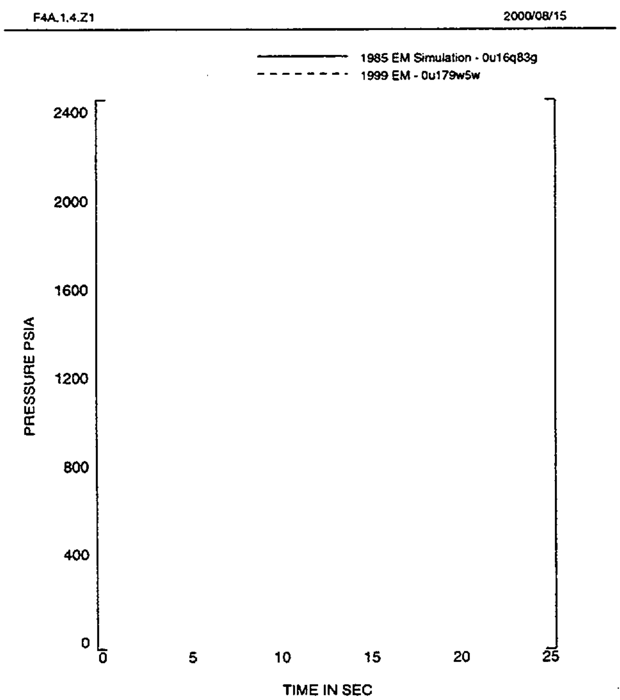


Figure 3.5-2
1985 EM VERSUS 1999 EM CEFLASH-4A
PRESSURE IN STEAM GENERATOR SECONDARY
NODE 52

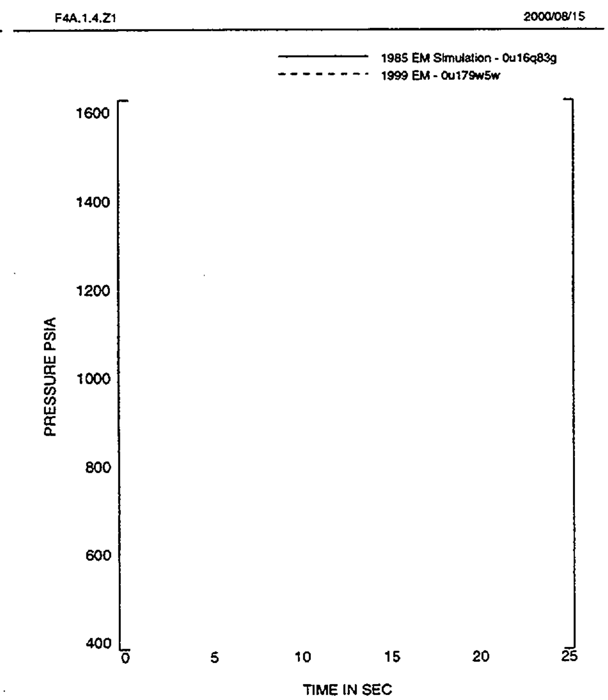


Figure 3.5-3
1985 EM VERSUS 1999 EM CEFLASH-4A
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
NODE 14

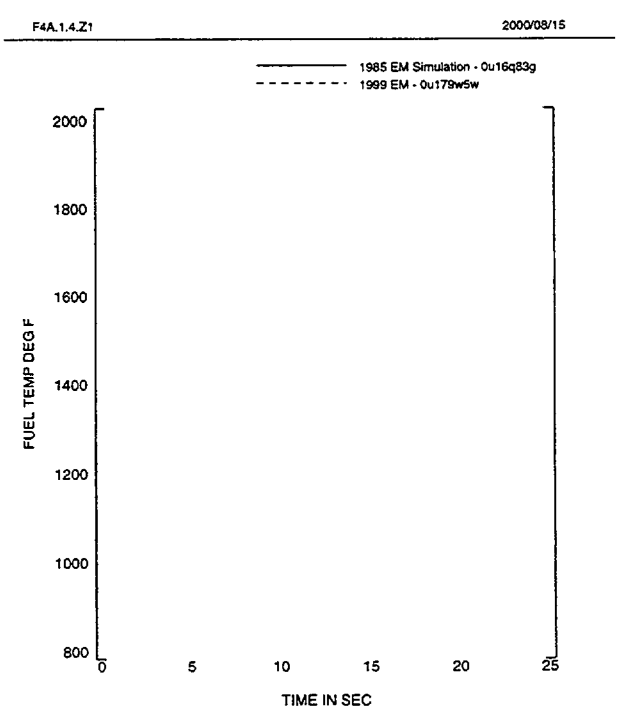


Figure 3.5-4
1985 EM VERSUS 1999 EM CEFLASH-4A
HOT ASSEMBLY CLADDING TEMP HOT SPOT
NODE 14

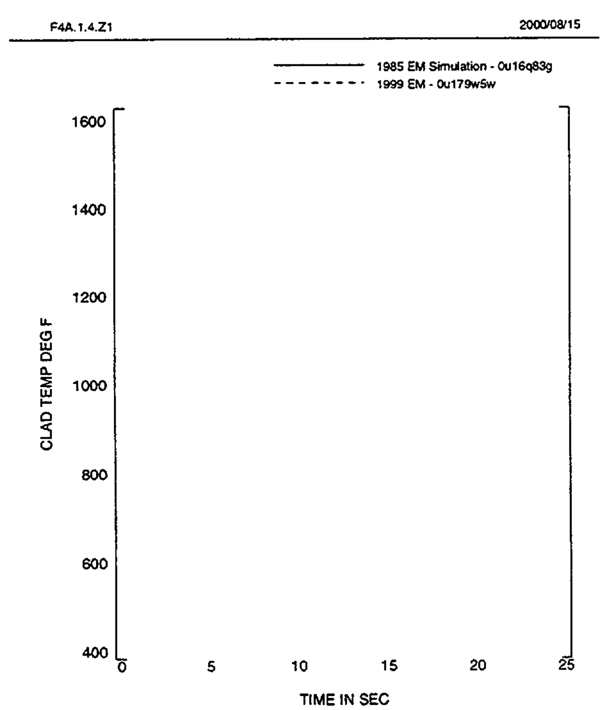


Figure 3.5-5
1985 EM VERSUS 1999 EM CEFLASH-4A
HOT ASSEMBLY FLOW RATE NEAR HOT SPOT
PATH 16

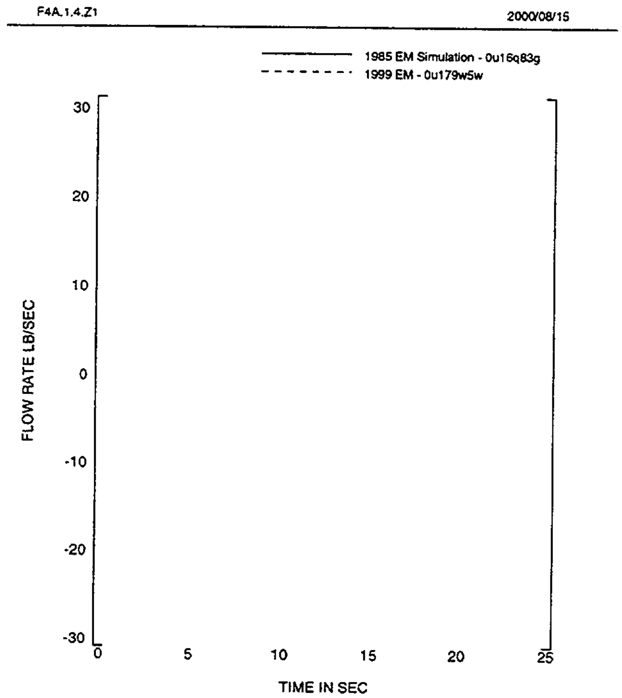


Figure 3.5-6
1985 EM VERSUS 1999 EM COMPERC-II
CONTAINMENT PRESSURE

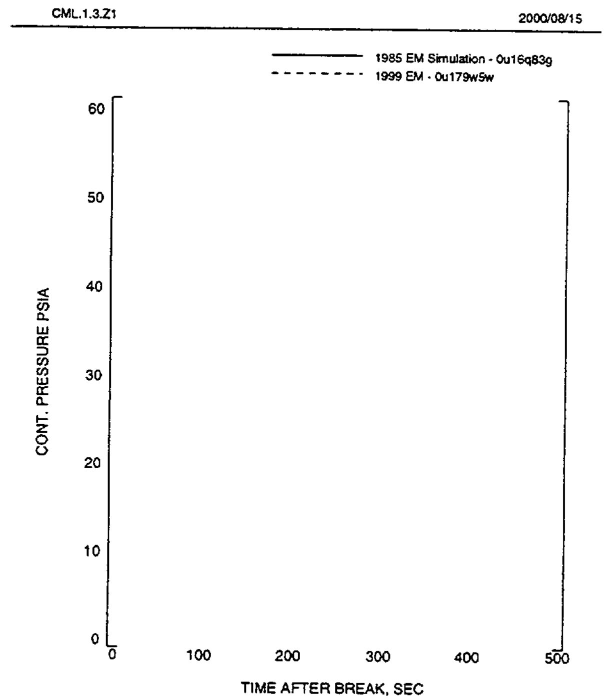


Figure 3.5-7
1985 EM VERSUS 1999 EM COMPERC-II
SUBCOOLED LIQ. LEVEL

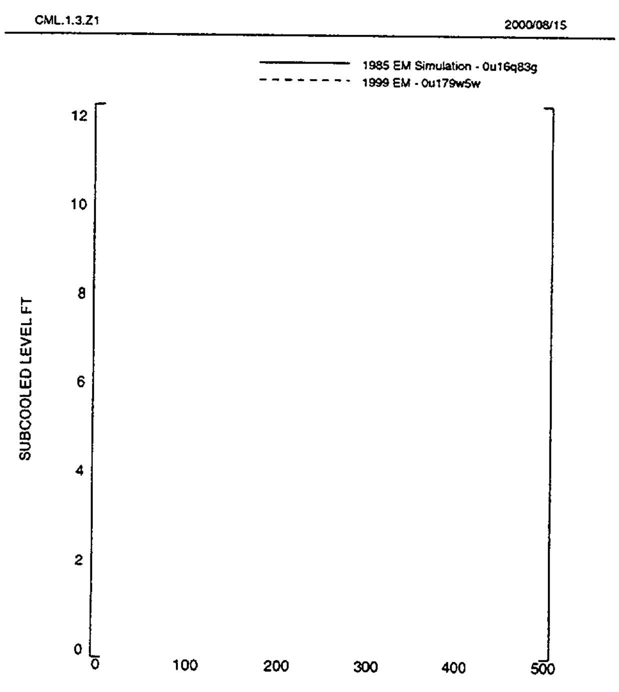


Figure 3.5-8
1985 EM VERSUS 1999 EM COMPERC-II
UPPER PLENUM PRESSURE

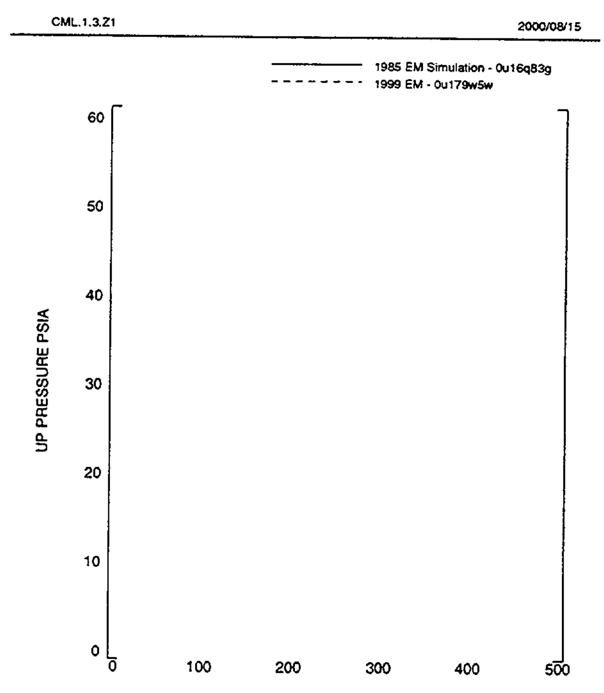


Figure 3.5-9
1985 EM VERSUS 1999 EM COMPERC-II
REFLOOD LIQ. MASS

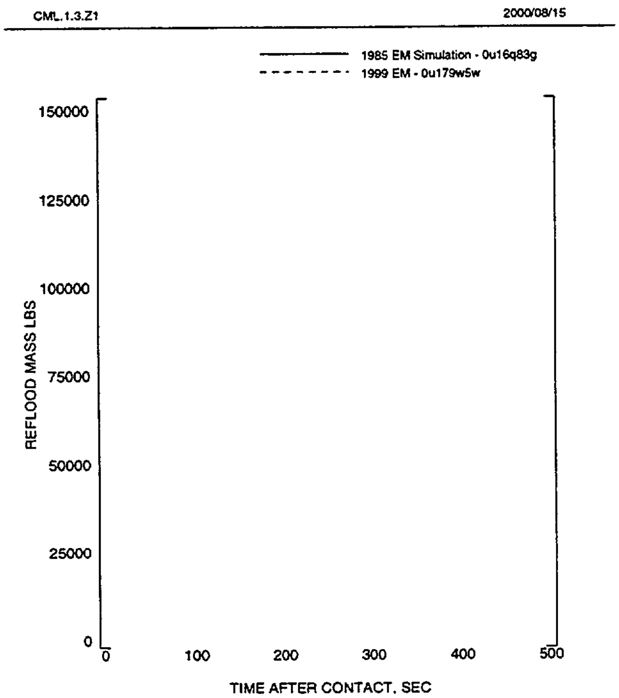


Figure 3.5-10
1985 EM VERSUS 1999 EM COMPERC-II
STEAM FLOW IN HOT LEG

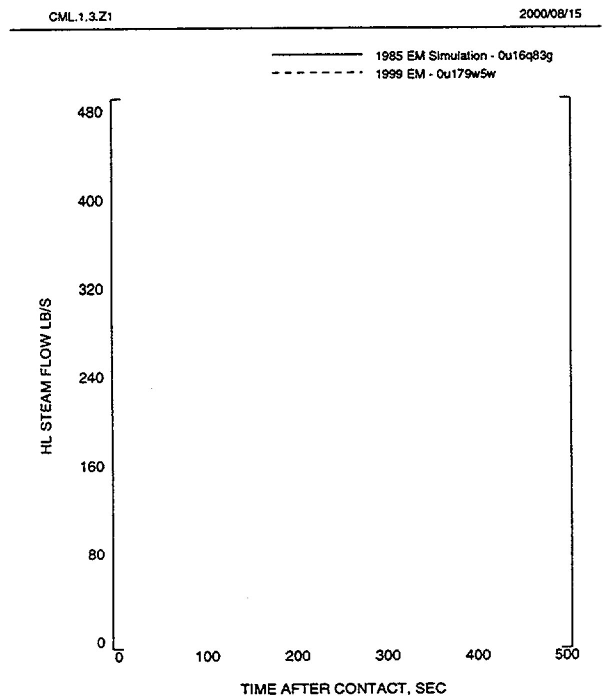


Figure 3.5-11
1985 EM VERSUS 1999 EM COMPERC-II
EQUIVALENT K-FACTOR

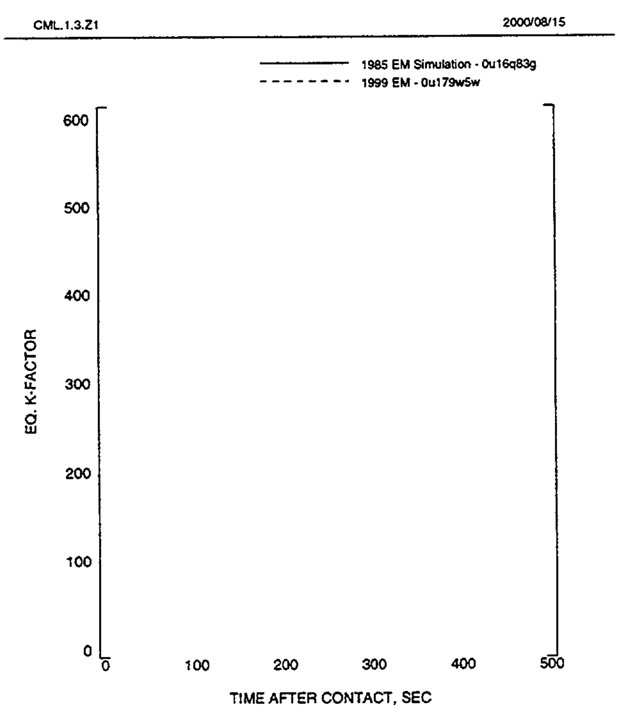


Figure 3.5-12
1985 EM VERSUS 1999 EM HTCOF
HEAT TRANS. COEFF. NODE 14
ELEV= 8.12

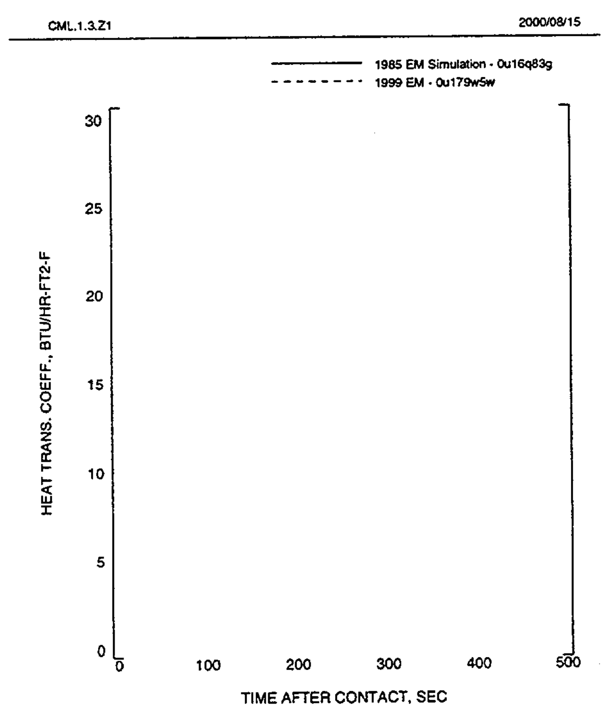


Figure 3.5-13
1985 EM VERSUS 1999 EM STRIKIN-II
CLADDING TEMP NODE 14
HOT CHANNEL

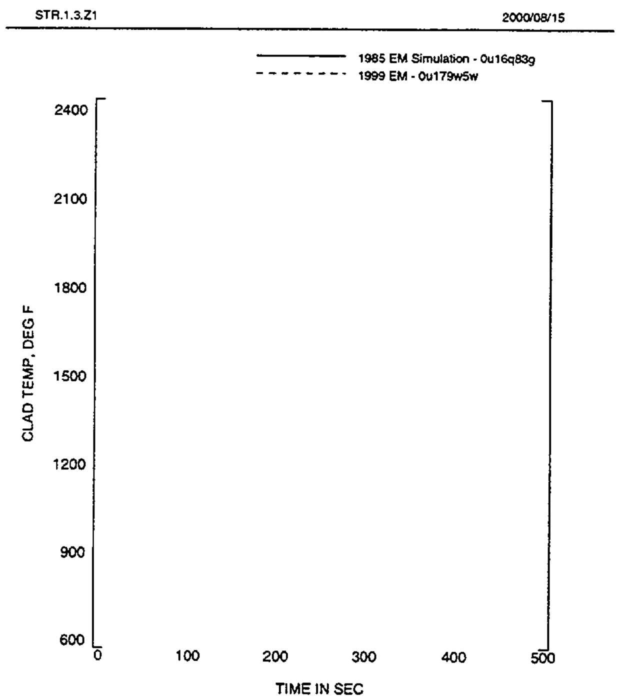


Figure 3.5-14
1985 EM VERSUS 1999 EM STRIKIN-II
CLADDING TEMP NODE 16
HOT CHANNEL

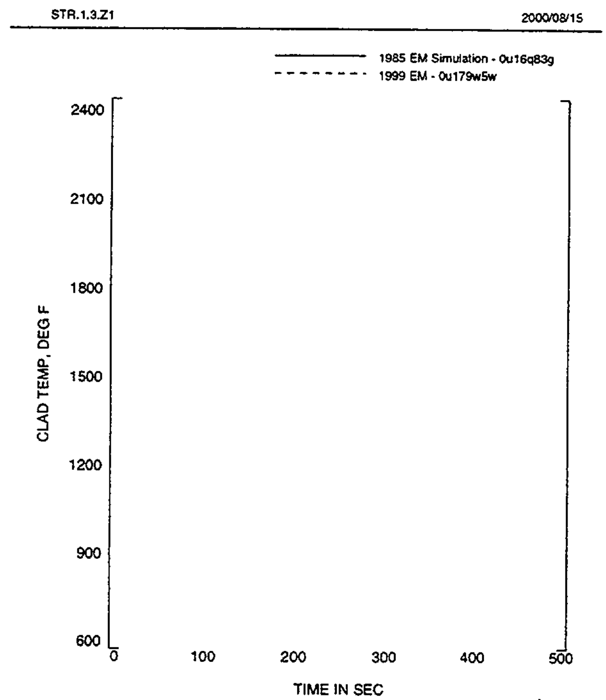


Figure 3.5-15
1985 EM VERSUS 1999 EM STRIKIN-II
PRCT ZIRC-WATER NODE 14
HOT CHANNEL

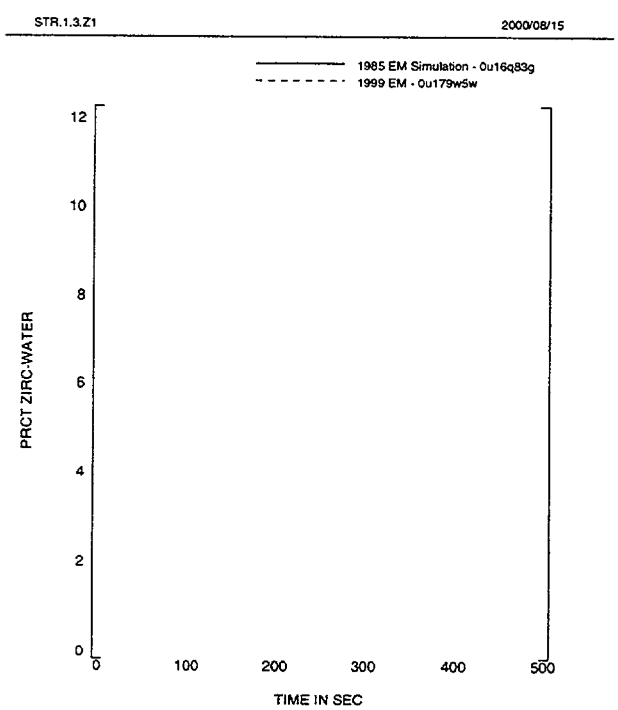
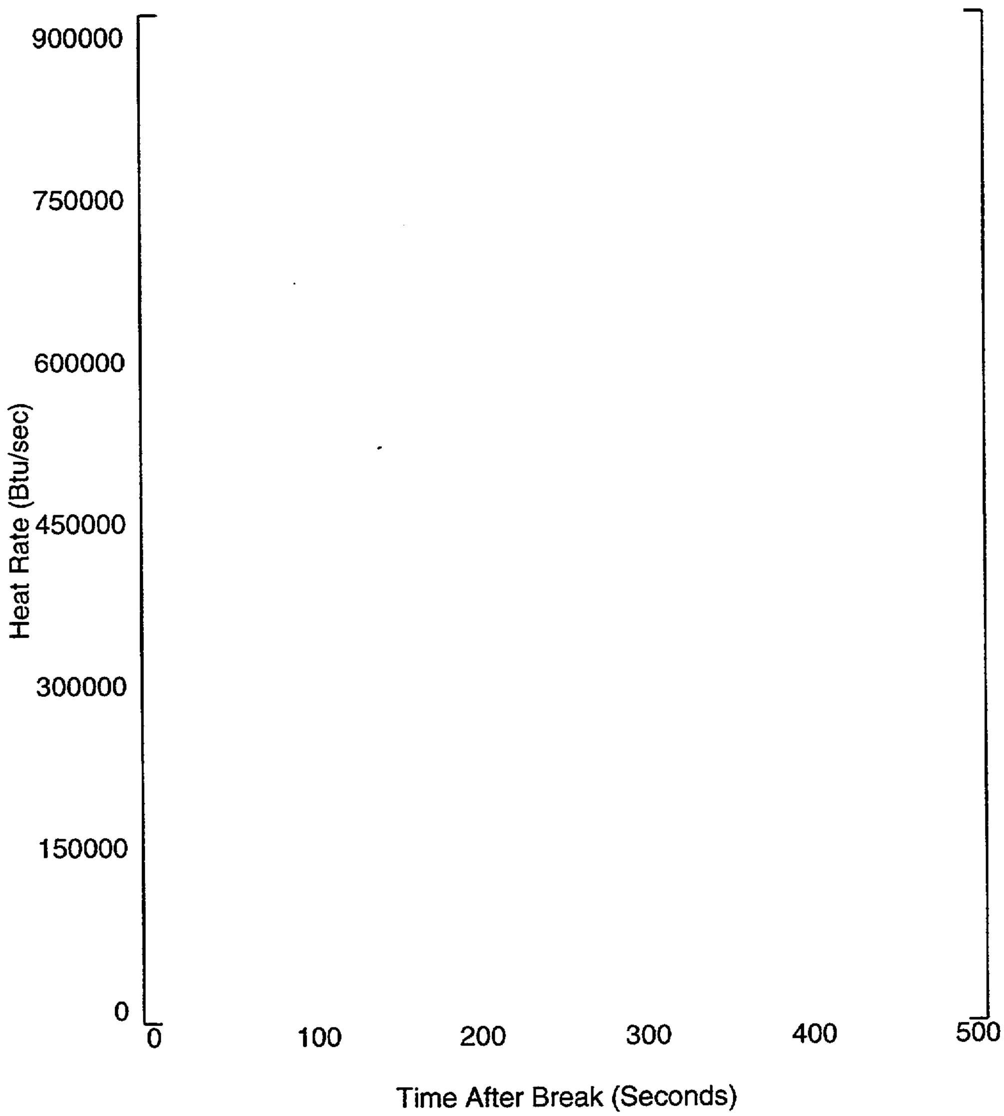


Figure 3.5-16

1985 EM VERSUS 1999 EM COMPERC-II

Condensation Heat Removal in Containment



3.6 Application Analysis Results Using the 1999 EM

This section presents the results of a break spectrum analysis performed with the 1999 EM for LBLOCA fully implemented. Appendix A provides a description of the required code inputs for this mode of analysis. The results are compared to the results of the reference analysis described in Section 3.4. The reference case is a 1985 EM simulation, where the PCT was calculated to be 2199°F using a specially selected PLHGR to establish a reference point very near the ECCS performance acceptance criterion limit of 2200°F. Table 3.6-1 lists the four breaks that are discussed in Section 3.4 and that are analyzed with the models described in Section 2.0.

Table 3.6-2 lists the values of important plant parameters and initial conditions that were used in the LBLOCA analysis. The fuel rod conditions listed are for the hot rod burnup that produces the highest PCT. The RCS initial conditions were selected in accordance with the LBLOCA EM and as described earlier in Table 3.3-1.

Table 3.6-3 lists important results for the spectrum of break sizes. The transient behavior of the parameters listed in Table 3.6-4 is shown in Figures 3.6-1 through 3.6-4. The PCT versus break size for the spectrum of cases is given in Figure 3.6-5.

The limiting break size is defined on the basis of the highest PCT. For the plant design data and characteristics given in Section 3.3, and for the spectrum of breaks described in Section 3.4, Table 3.6-3 shows that the limiting size is the 0.6xDEDLG. The calculated PCT for the limiting break is[ ]which represents a[ ]relative to the reference case PCT of 2199°F. The larger break sizes analyzed were roughly[ ]relative to the limiting break, and the smaller break size analyzed was[ ] This break size variation in PCT is typical of LBLOCA analyses for Combustion Engineering designed PWRs. The[ ]calculated for the limiting break size using the 1999 EM is comparable to the reductions calculated in Section 3.4 for the worst single failure[ ]and in Section 3.5 for the comparisons to the 1985 EM simulation[ ]

The results of the break spectrum analysis demonstrate the transient behavior that is expected from the improvements made for the 1999 EM. The magnitude of the improvement in PCT is attributable to the improved reflood thermal-hydraulics obtained by the 1999 EM improvements as evidenced by the [ ] listed in Table 3.6-3 compared to Table 3.4-1. The comparisons for the PARCH/HCROSS steam cooling analyses in Tables 3.6-3 and Table 3.4-1 also show [ ]

Table 3.6-1

1999 EM LBLOCA ECCS Performance Break Spectrum Analysis

Break Size, Type, and Location	Abbreviation	Figure No.
1.0 Double-Ended Guillotine Break in Pump Discharge Leg	1.0 DEDLG	3.6-1
0.8 Double-Ended Guillotine Break in Pump Discharge Leg	0.8 DEDLG	3.6-2
0.6 Double-Ended Guillotine Break in Pump Discharge Leg	0.6 DEDLG	3.6-3
0.4 Double-Ended Guillotine Break in Pump Discharge Leg	0.4 DEDLG	3.6-4

Table 3.6-2

Key System Parameters and Initial Conditions Used in the 1999 EM LBLOCA ECCS Performance Break Spectrum Analysis

Parameter	Units	Value
Reactor Power Level (102% of nominal)	MWt	3458
Peak Linear Heat Generation Rate (PLHGR) of the Hot Rod	kw/ft	12.43
PLHGR of the Average Rod in the Assembly with the Hot Rod	kw/ft	11.62
Fuel Centerline Temperature at the PLHGR	°F	3156
Fuel Average Temperature at the PLHGR	°F	2026
Gap Conductance at the PLHGR	Btu/hr-ft^2-°F	1548
Hot Rod Gas Pressure	psia	994
Hot Rod Burnup	MWD/MTU	1000
Radiation Enclosure X-factor (1)	--	1.825
Moderator Temperature Coefficient	$\Delta\rho$ / °F	0.0×10^{-4}
RCS Pressure	psia	2250
RCS Flow Rate	lbm/hr	144.9×10^{6}
Core Flow Rate	lbm/hr	140.5×10^{6}
Cold Leg Temperature	°F	530
Hot Leg Temperature	°F	593
Number of Plugged Tubes per Steam Generator	--	2200
Safety Injection Tank Gas Pressure (min / max)	psia	595 / 675
Safety Injection Tank Water Volume (min / max)	ft^3	1650 / 1825
Initial Containment Pressure	psia	13.3
Initial Containment Temperature	°F	50

(1) The radiation enclosure X-factor is a figure of merit describing the potential for rod-to-rod thermal radiation heat transfer. A low X-factor indicates a flat power distribution in the vicinity of the hot rod.

Table 3.6-3

Summary of Results

1999 EM LBLOCA ECCS Performance Break Spectrum Analysis

Result	1999 EM 1.0 DEDLG	1999 EM 0.8 DEDLG	1999 EM 0.6 DEDLG	1999 EM 0.4 DEDLG

Table 3.6-4

Parameters Plotted As A Function Of Time

1999 EM LBLOCA ECCS Performance Break Spectrum Analysis

Parameter	Units	Figure Number
Normalized Core Power	Fraction	A
Pressure in Center Hot Assembly Node	psia	B
Pressure in Steam Generator Secondary	psia	C
Break Flow Rate (Pump Side and Reactor Vessel Side)	lb/sec	D
Core Bulk Channel Flow Rate (Core Inlet and Core Outlet)	lb/sec	E
Hot Assembly Flow Rate (Below and Above Hot Spot)	lb/sec	F
Hot Assembly Quality (Below, At, and Above Hot Spot)	Fraction	G
Hot Assembly Fuel Average Temperature (Hot Spot)	°F	H
Hot Assembly Cladding Temperature (Hot Spot)	°F	I
Containment Pressure	psia	J
Mass Added to Core During Reflood	lbs	K
Peak Cladding Temperature (Hot Spot and Rupture Node)	°F	L
Gap Conductance (Hot Spot)	Btu/hr-ft^2-°F	M
Peak Local Cladding Oxidation Percentage	%	N
Fuel Centerline, Fuel Average, Cladding, Coolant Temperatures (Hot Spot)	°F	O
Heat Transfer Coefficient (Hot Spot)	Btu/hr-ft^2-°F	P

Figure 3.6-1 A
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
NORMALIZED CORE POWER

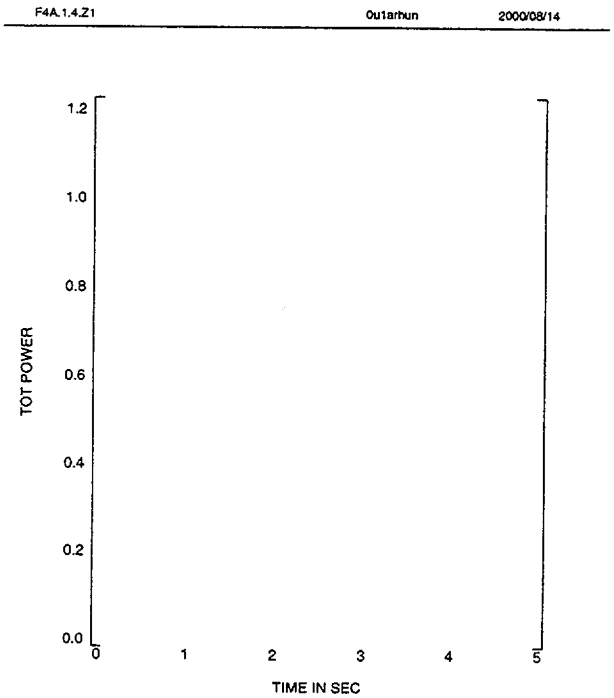


Figure 3.6-1 B
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
PRESSURE IN CENTER OF HOT ASSEMBLY
NODE 13

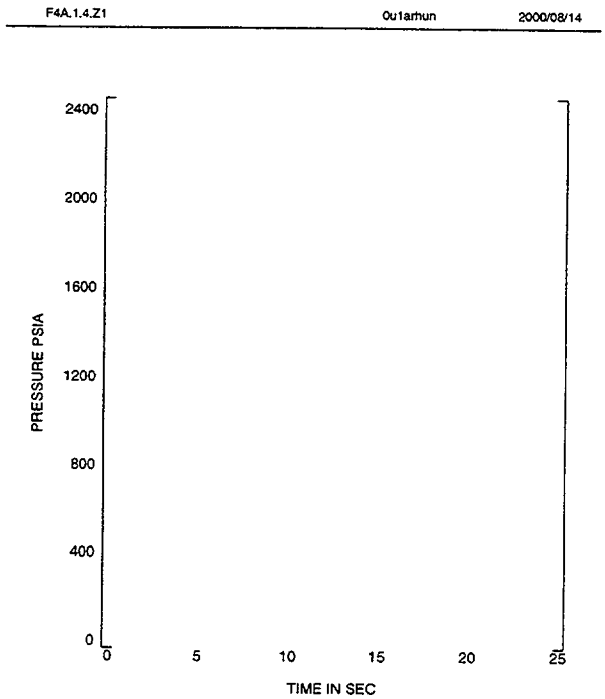


Figure 3.6-1 C
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
PRESSURE IN STEAM GENERATOR SECONDARY
NODE 53

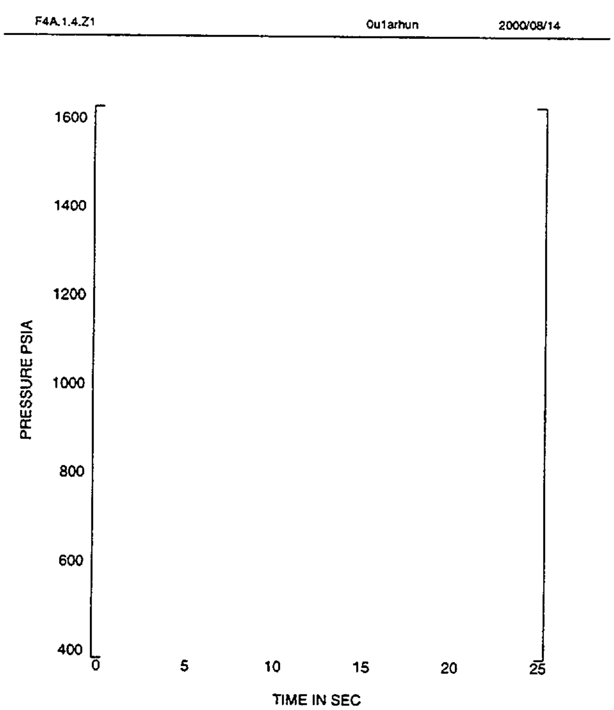


Figure 3.6-1 D
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
BREAK FLOW RATE
PATHS 72 AND 74

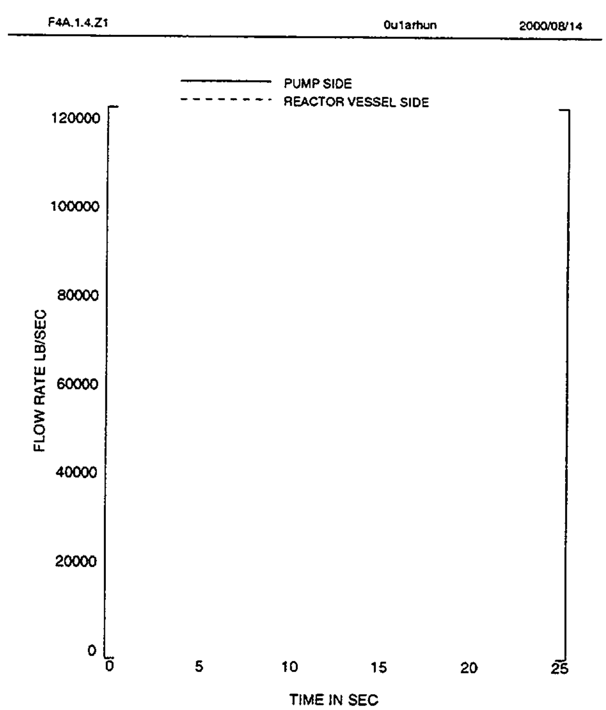


Figure 3.6-1 E
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
BULK CHANNEL CORE FLOW RATE
PATHS 1 AND 6

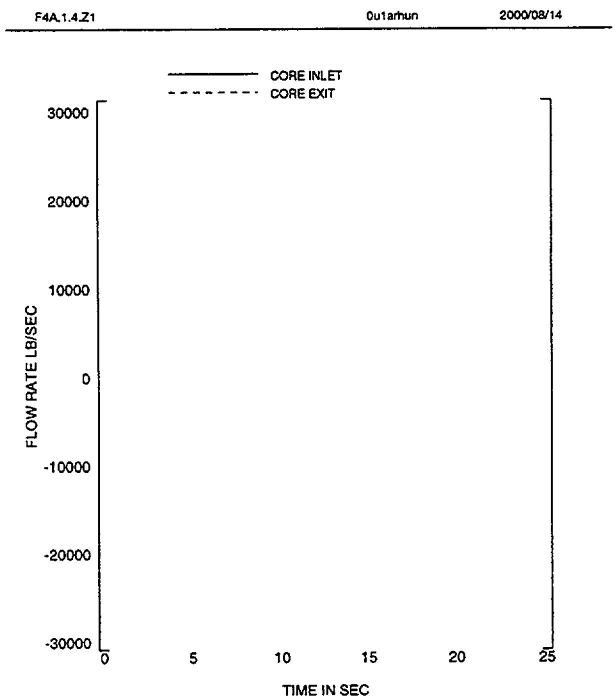


Figure 3.6-1 F
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT ASSEMBLY FLOW RATE
PATHS 16 AND 17

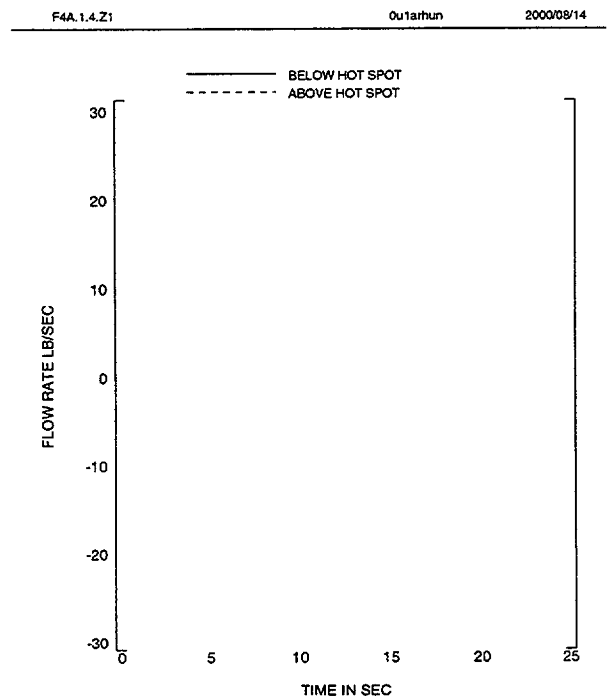


Figure 3.6-1 G
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT ASSEMBLY QUALITY
NODES 13, 14, AND 15

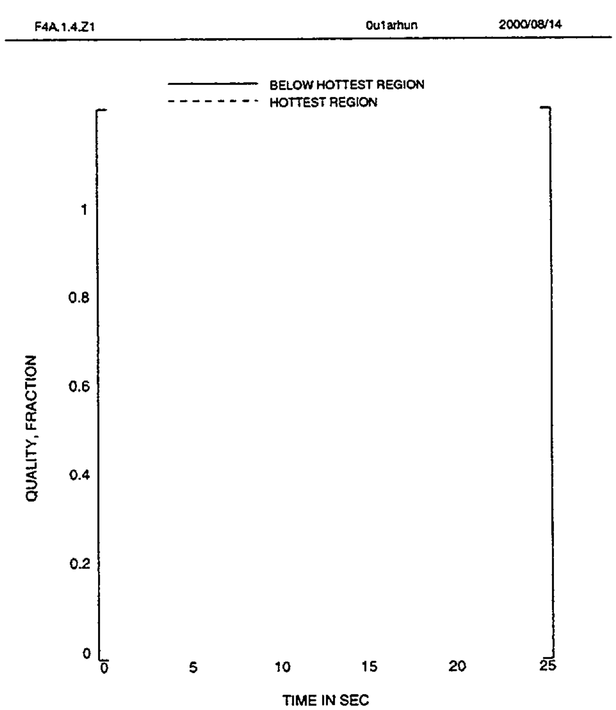


Figure 3.6-1 H
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
PCT NODE 14

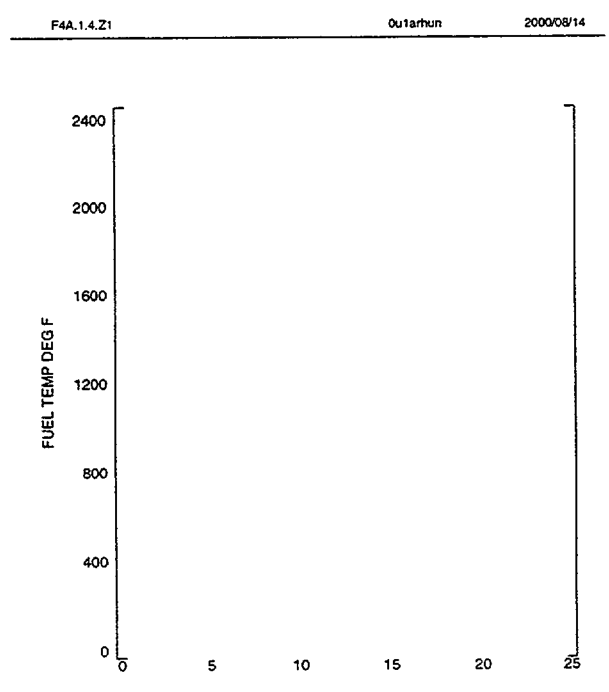


Figure 3.6-1 I
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT ASSEMBLY CLADDING TEMP HOT SPOT
PCT NODE 14

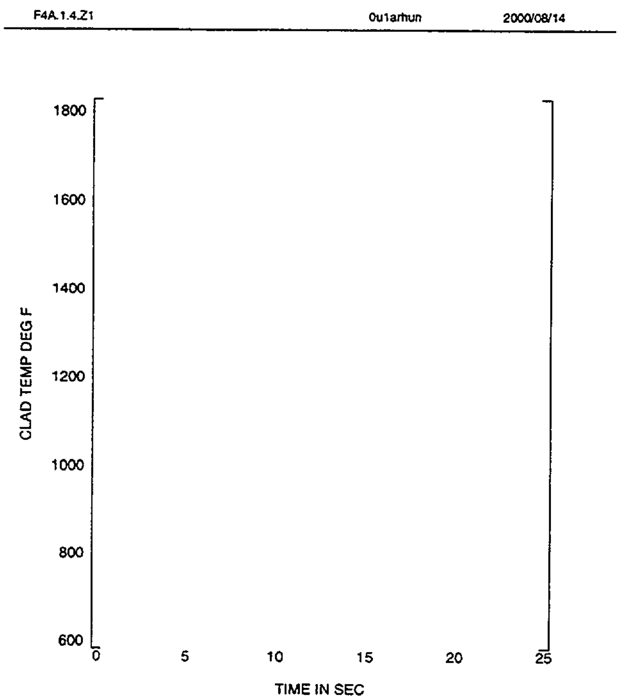


Figure 3.6-1 J
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
CONTAINMENT PRESSURE

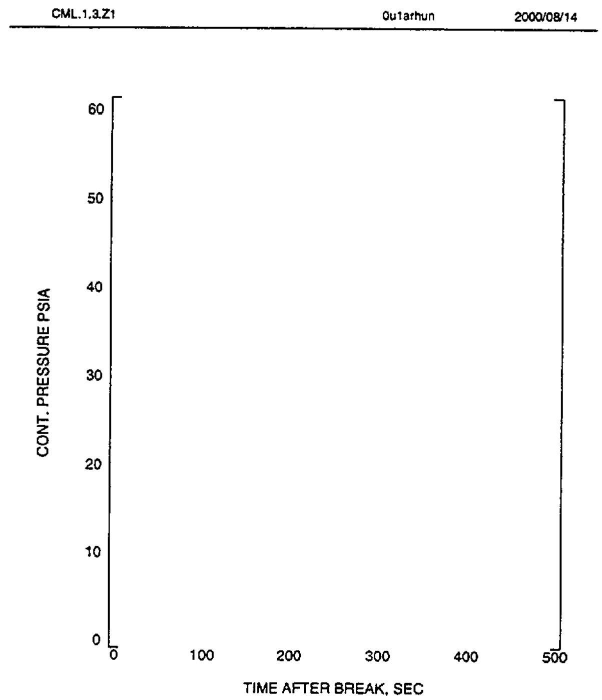


Figure 3.6-1 K
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
REFLOOD LIQ. MASS

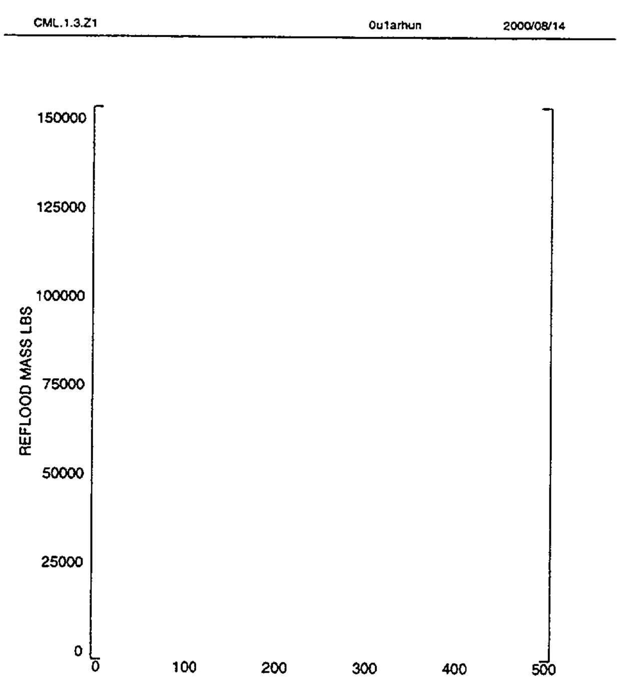


Figure 3.6-1 L
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
CLADDING TEMP
HOT CHANNEL

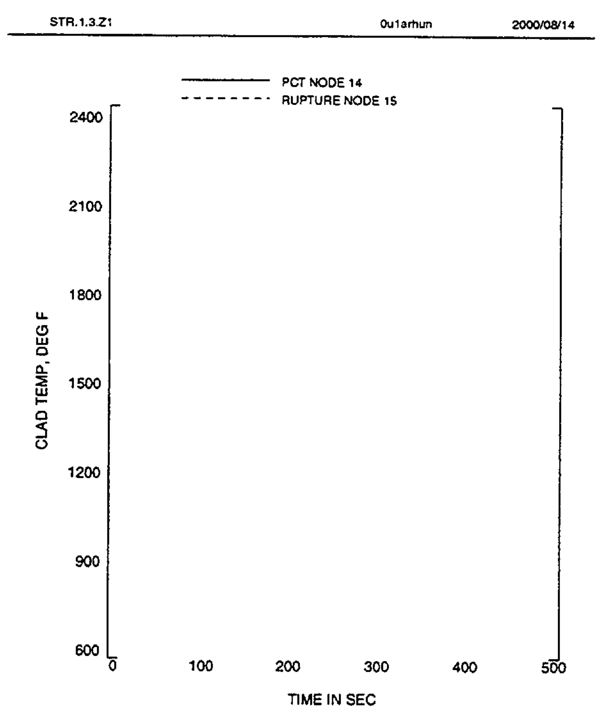


Figure 3.6-1 M
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT SPOT GAP CONDUCTANCE
HOT CHANNEL

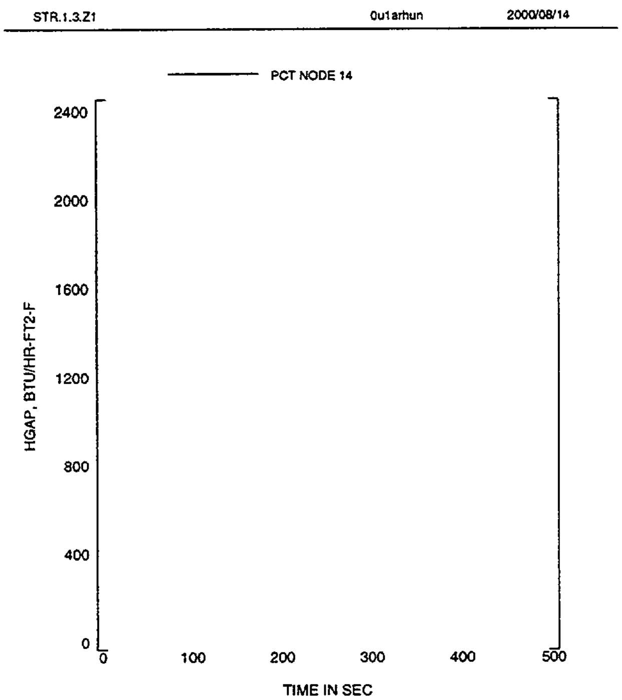


Figure 3.6-1 N
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
PEAK LOCAL CLAD OXIDATION PRCT
HOT CHANNEL

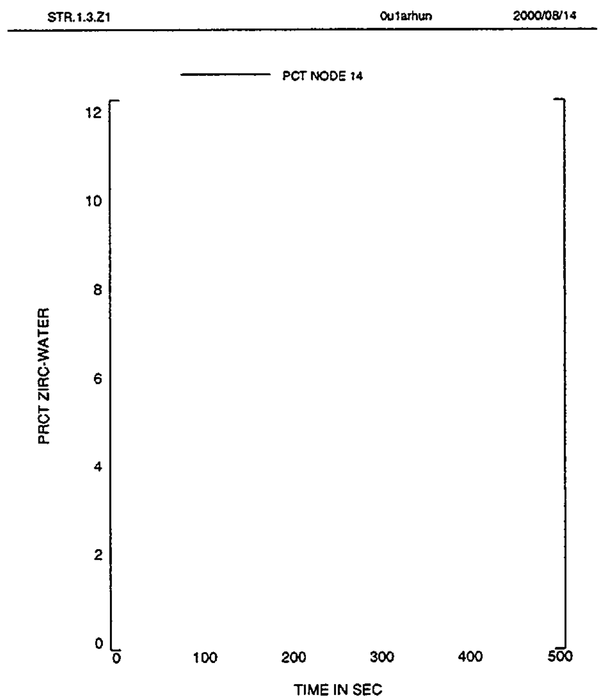


Figure 3.6-1 O
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HOT SPOT RADIAL TEMPERATURES
HOT CHANNEL

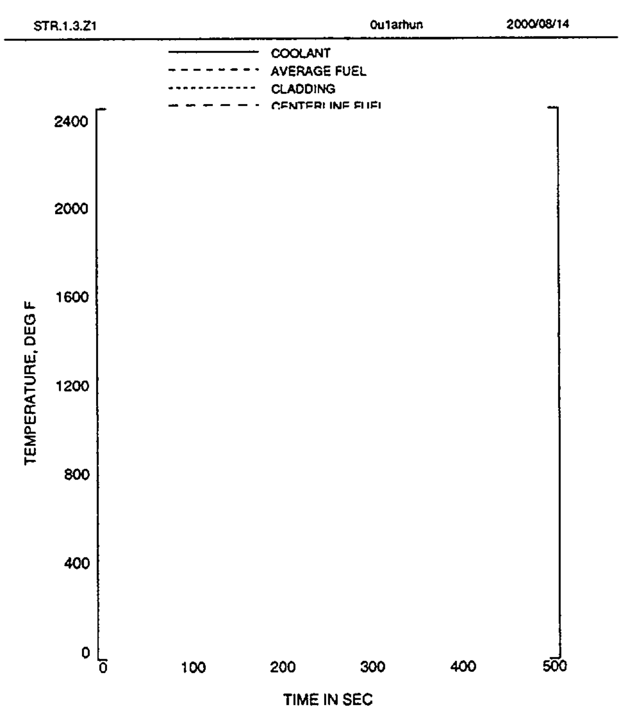


Figure 3.6-1 P
1999 EM SPECTRUM ANALYSIS 1.0 DEDLG
HEAT TRANSFER COEFF AT HOT SPOT
HOT CHANNEL

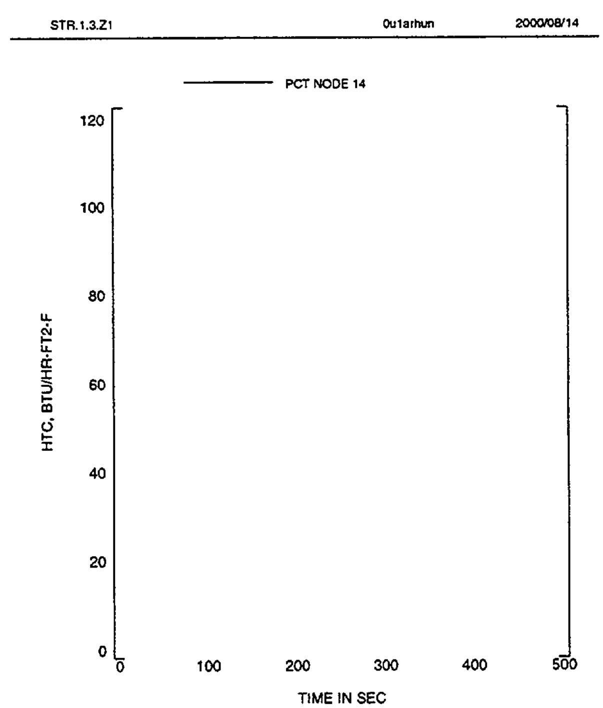


Figure 3.6-2 A
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
NORMALIZED CORE POWER

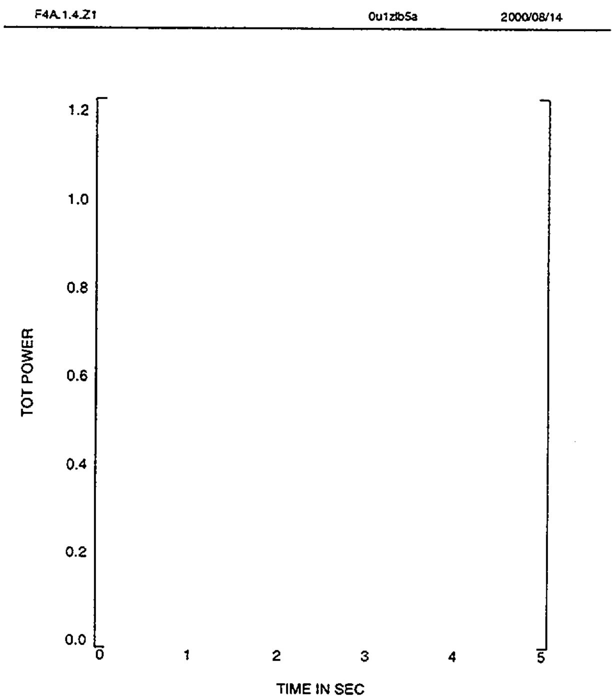


Figure 3.6-2 B
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
PRESSURE IN CENTER OF HOT ASSEMBLY
NODE 13

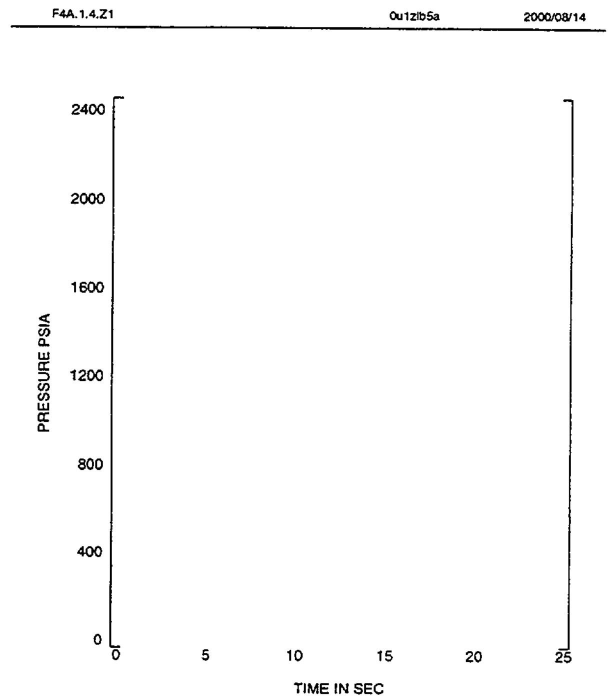


Figure 3.6-2 C
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
PRESSURE IN STEAM GENERATOR SECONDARY
NODE 53

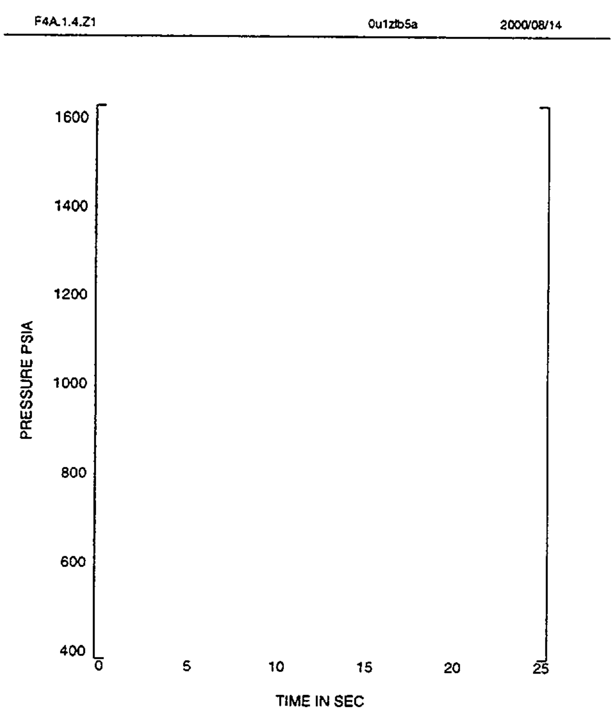


Figure 3.6-2 D
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
BREAK FLOW RATE
PATHS 72 AND 74

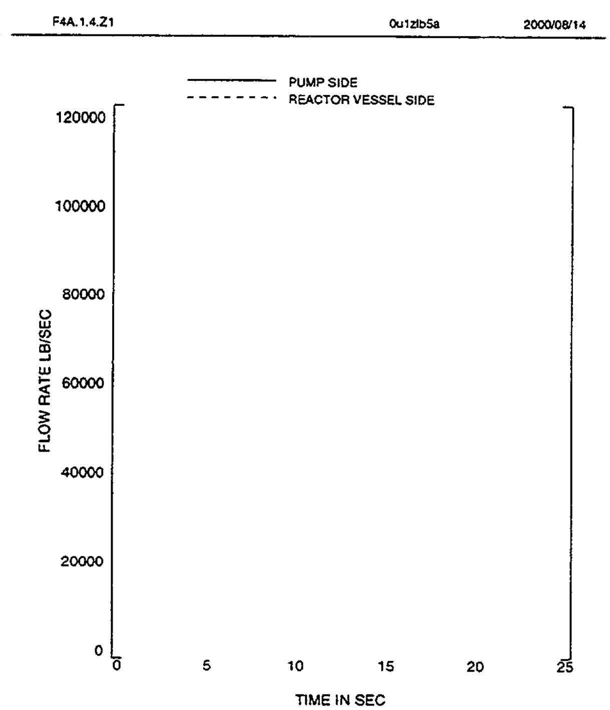


Figure 3.6-2 E
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
BULK CHANNEL CORE FLOW RATE
PATHS 1 AND 6

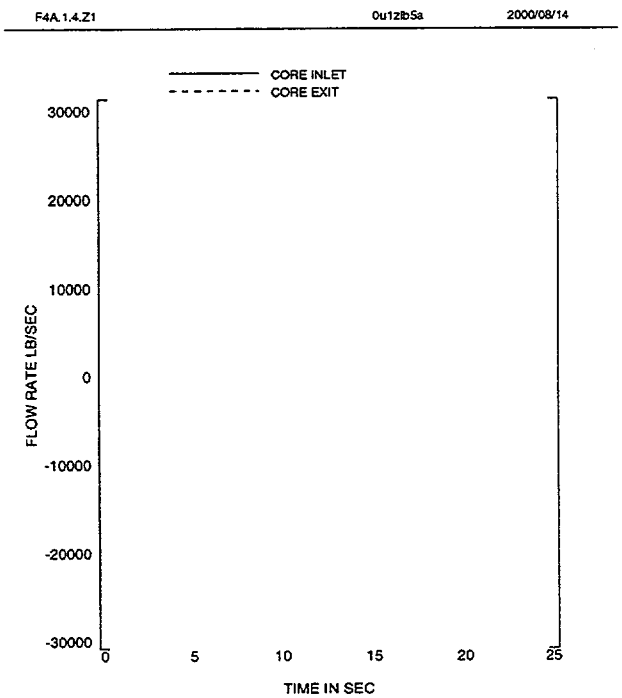


Figure 3.6-2 F
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT ASSEMBLY FLOW RATE
PATHS 16 AND 17

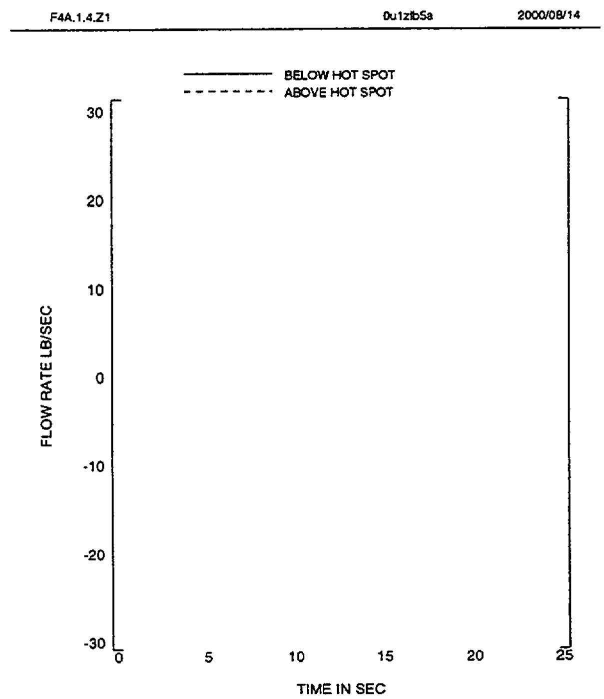


Figure 3.6-2 G
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT ASSEMBLY QUALITY
NODES 13, 14, AND 15

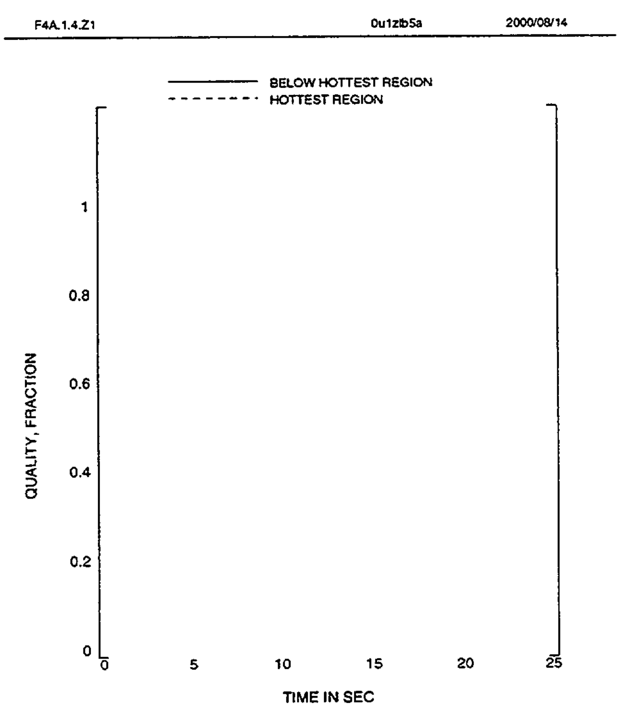


Figure 3.6-2 H
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
PCT NODE 14

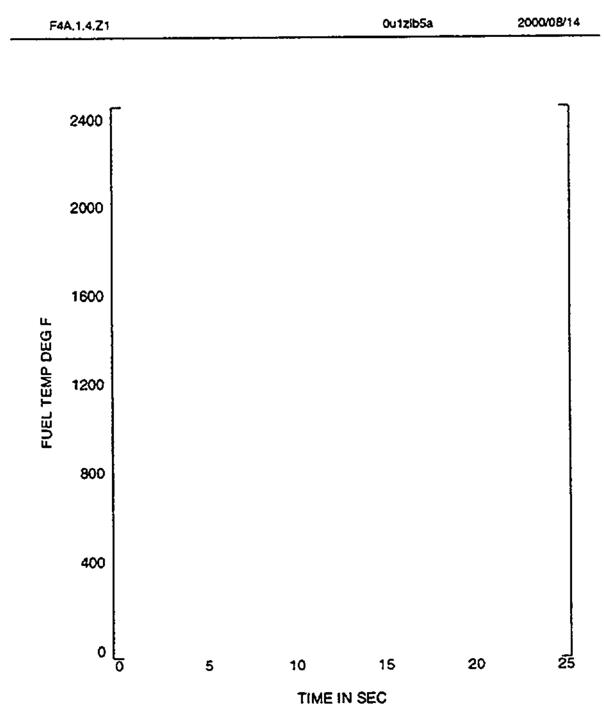


Figure 3.6-2 I
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT ASSEMBLY CLADDING TEMP HOT SPOT
PCT NODE 14

F4A.1.4.Z1 0u1zb5a 2000/08/14

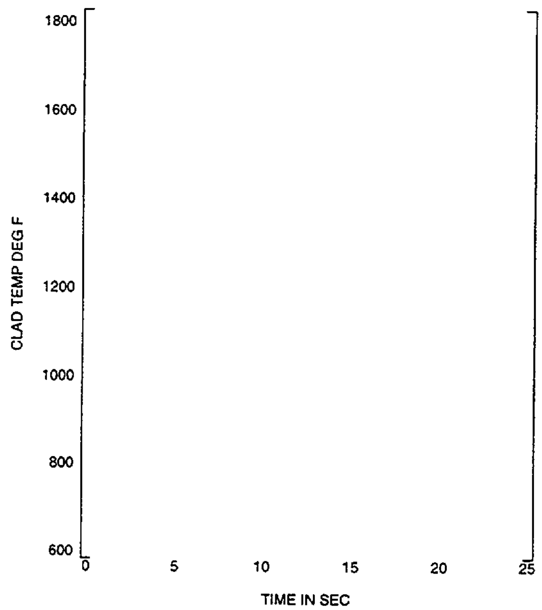


Figure 3.6-2 J
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
CONTAINMENT PRESSURE

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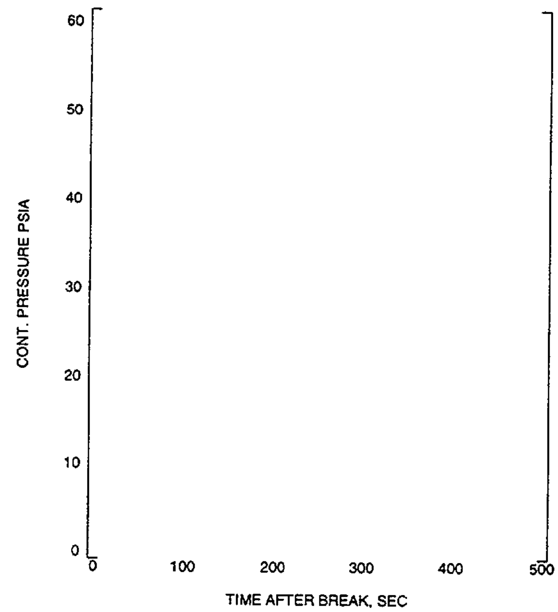


Figure 3.6-2 K
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
REFLOOD LIQ. MASS

CML.1.3.Z1 0u1zb5a 2000/08/14

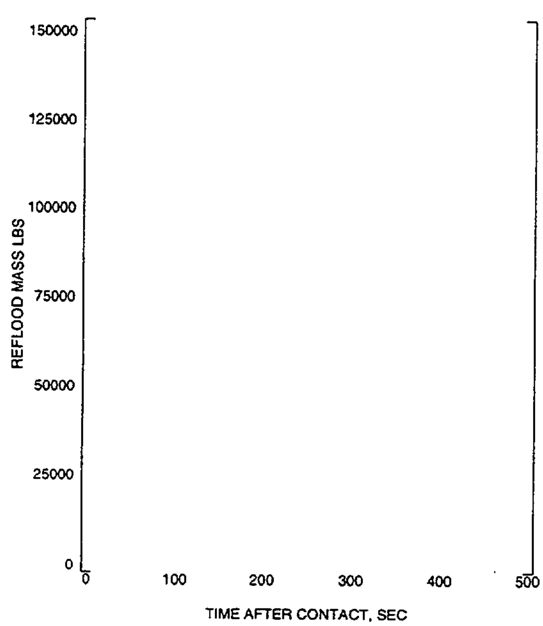


Figure 3.6-2 L
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
CLADDING TEMP
HOT CHANNEL

STR.1.3.Z1 0u1zb5a 2000/08/14

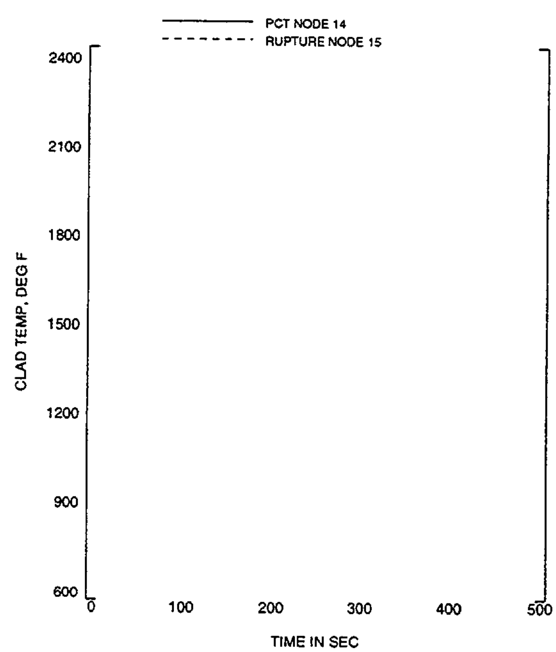


Figure 3.6-2 M
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT SPOT GAP CONDUCTANCE
HOT CHANNEL

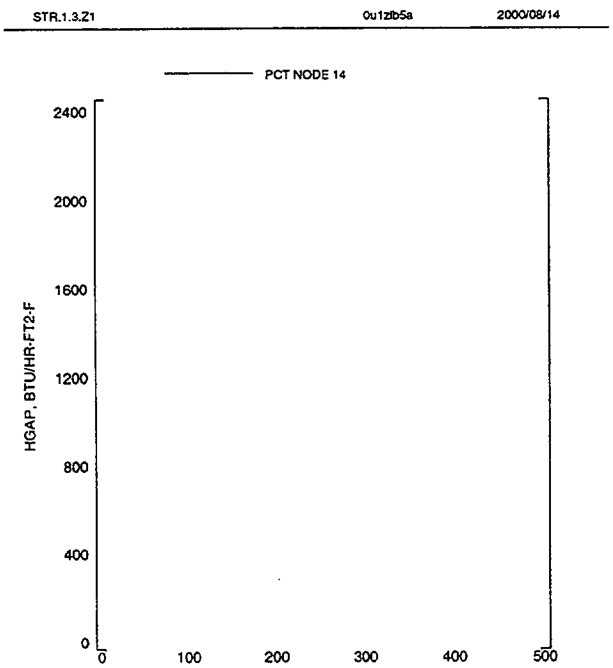


Figure 3.6-2 N
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
PEAK LOCAL CLAD OXIDATION PRCT
HOT CHANNEL

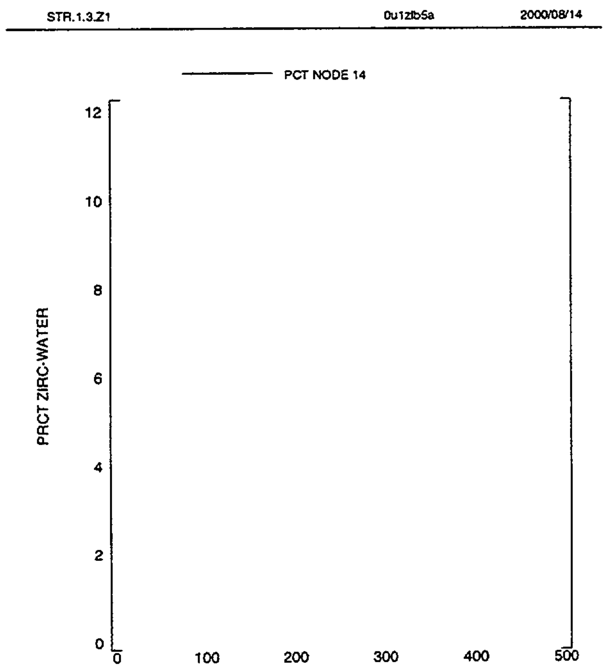


Figure 3.6-2 O
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HOT SPOT RADIAL TEMPERATURES
HOT CHANNEL

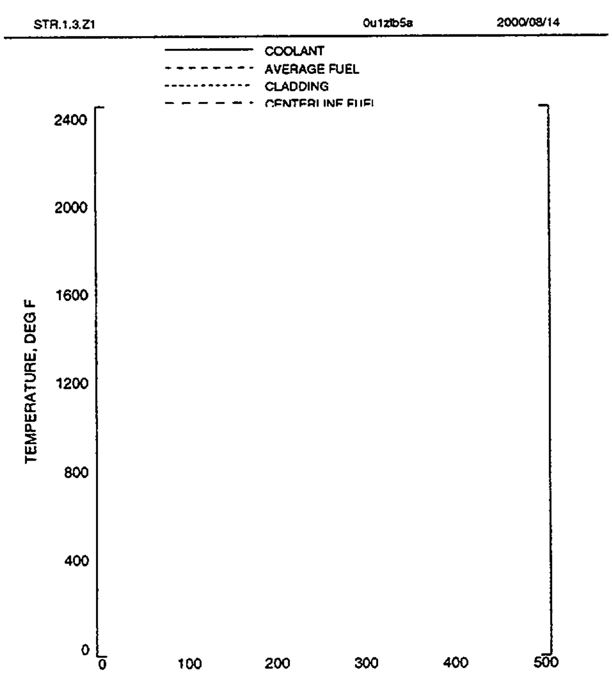


Figure 3.6-2 P
1999 EM SPECTRUM ANALYSIS 0.8 DEDLG
HEAT TRANSFER COEFF AT HOT SPOT
HOT CHANNEL

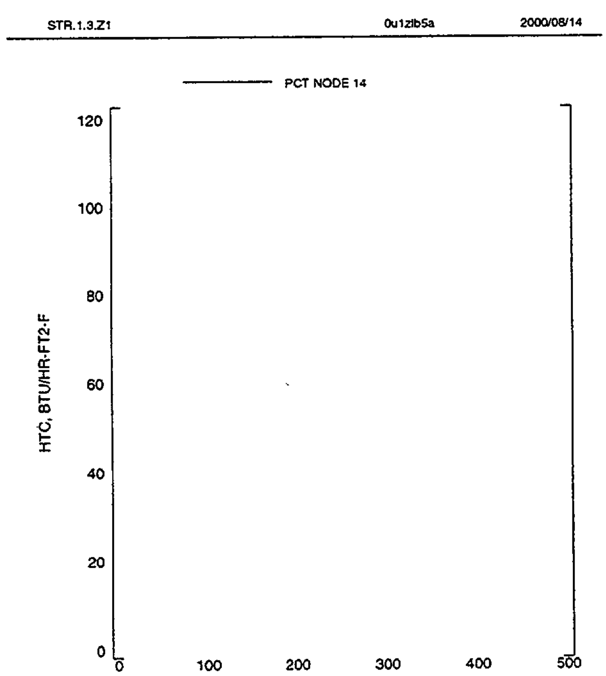


Figure 3.6-3 A
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
NORMALIZED CORE POWER

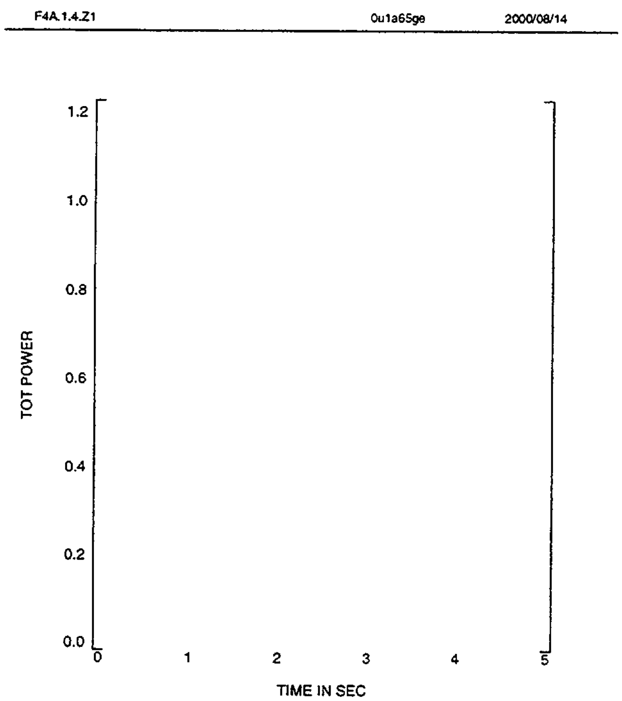


Figure 3.6-3 B
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
PRESSURE IN CENTER OF HOT ASSEMBLY
NODE 13

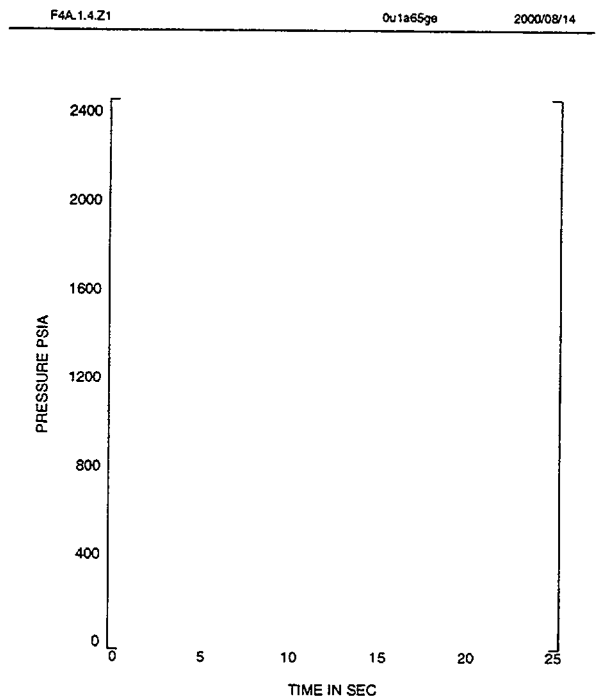


Figure 3.6-3 C
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
PRESSURE IN STEAM GENERATOR SECONDARY
NODE 53

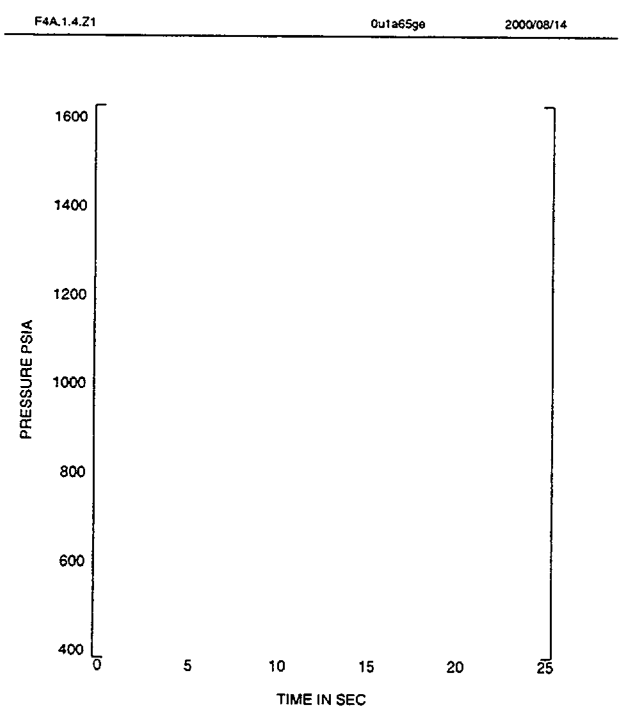


Figure 3.6-3 D
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
BREAK FLOW RATE
PATHS 72 AND 74

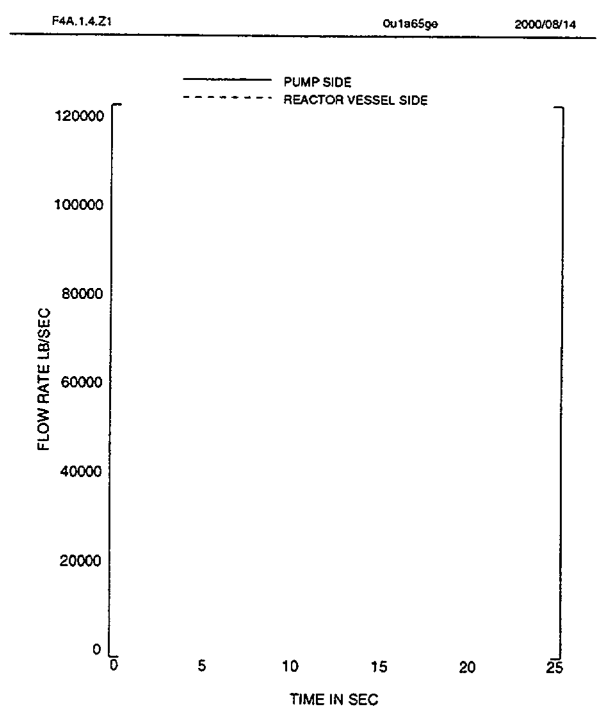


Figure 3.6-3 E
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
BULK CHANNEL CORE FLOW RATE
PATHS 1 AND 6

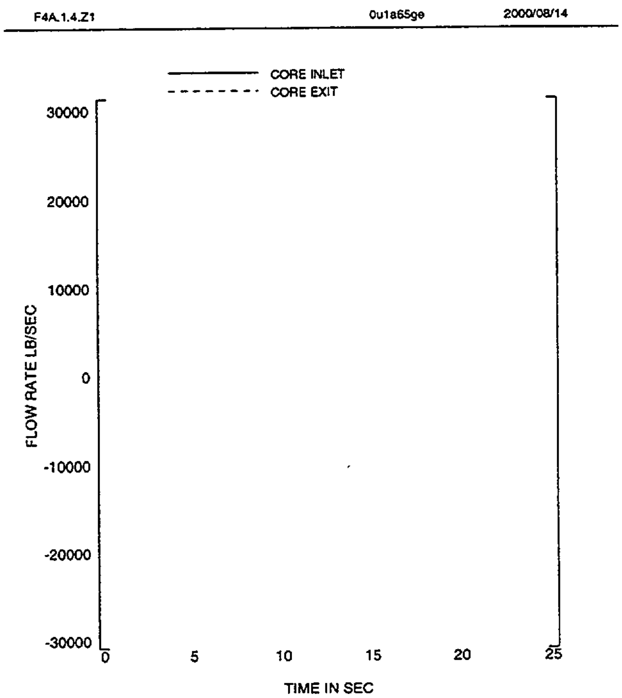


Figure 3.6-3 F
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT ASSEMBLY FLOW RATE
PATHS 16 AND 17

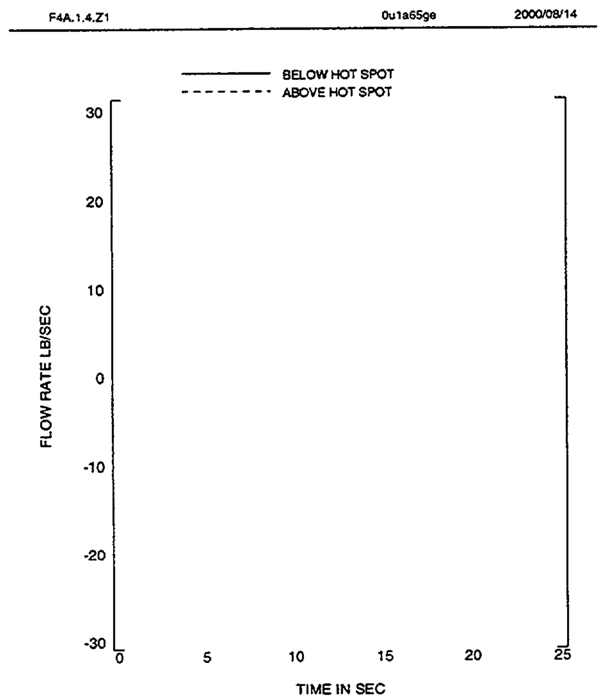


Figure 3.6-3 G
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT ASSEMBLY QUALITY
NODES 13, 14, AND 15

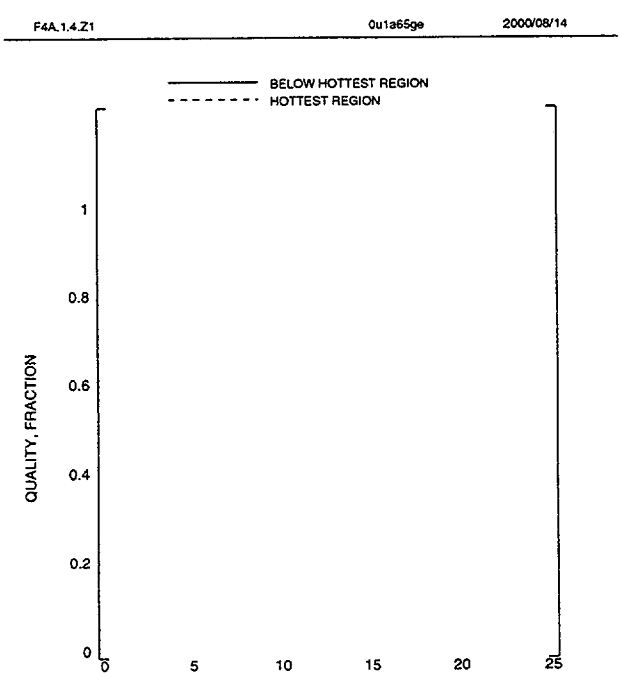


Figure 3.6-3 H
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
PCT NODE 14

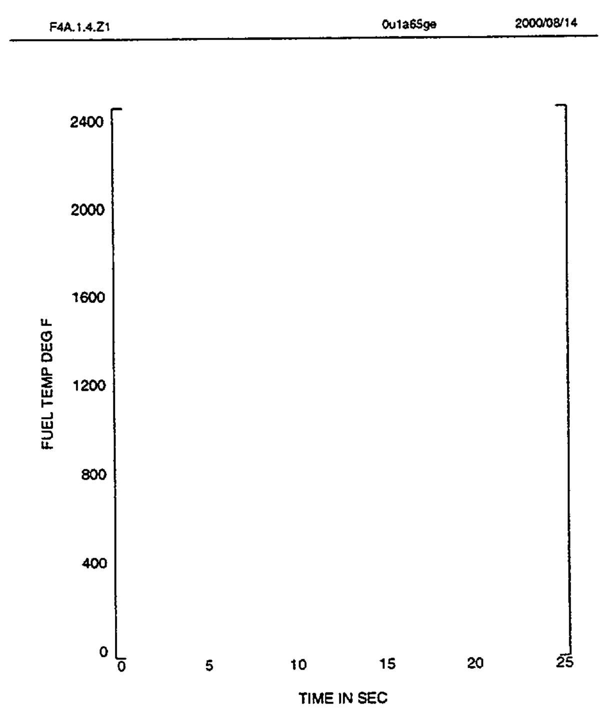


Figure 3.6-3 I
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT ASSEMBLY CLADDING TEMP HOT SPOT
PCT NODE 14

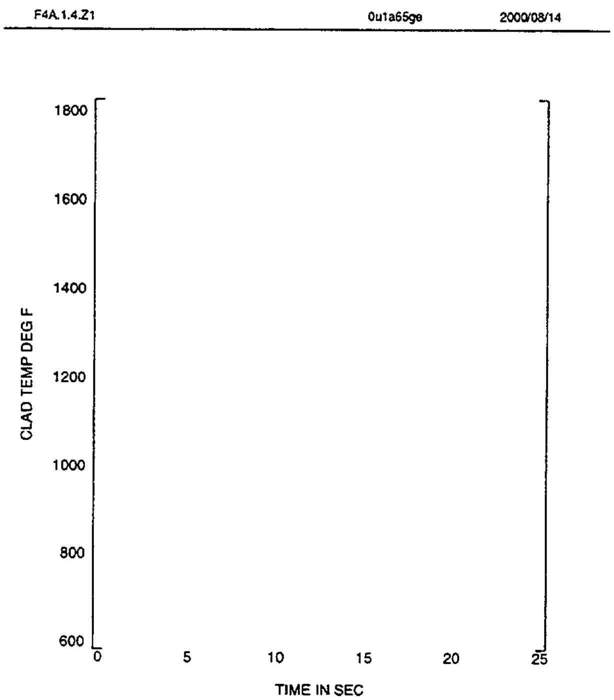


Figure 3.6-3 J
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
CONTAINMENT PRESSURE

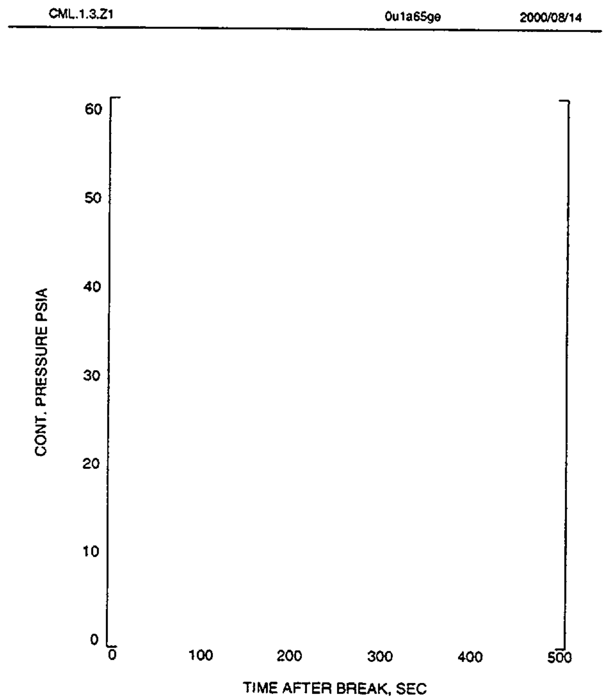


Figure 3.6-3 K
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
REFLOOD LIQ. MASS

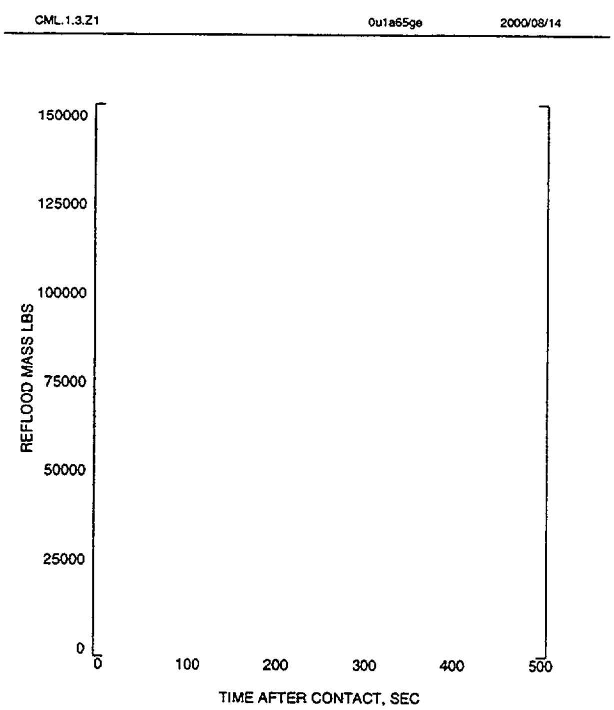


Figure 3.6-3 L
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
CLADDING TEMP
HOT CHANNEL

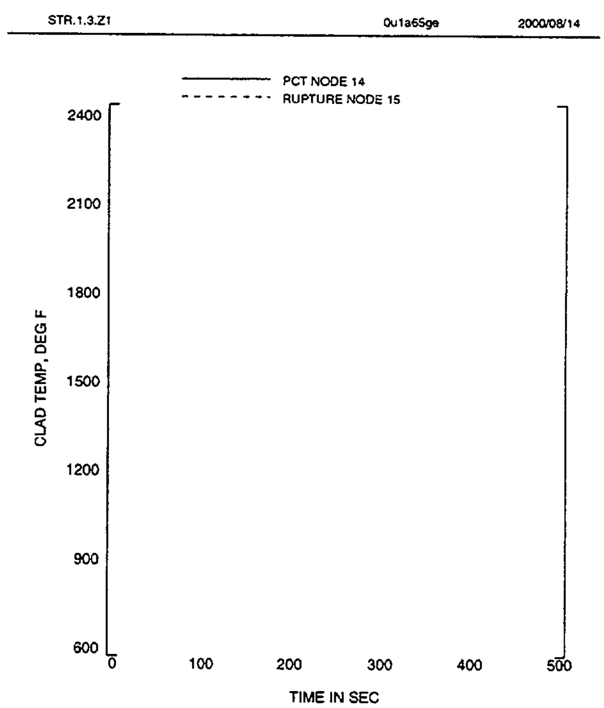


Figure 3.6-3 M
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT SPOT GAP CONDUCTANCE
HOT CHANNEL

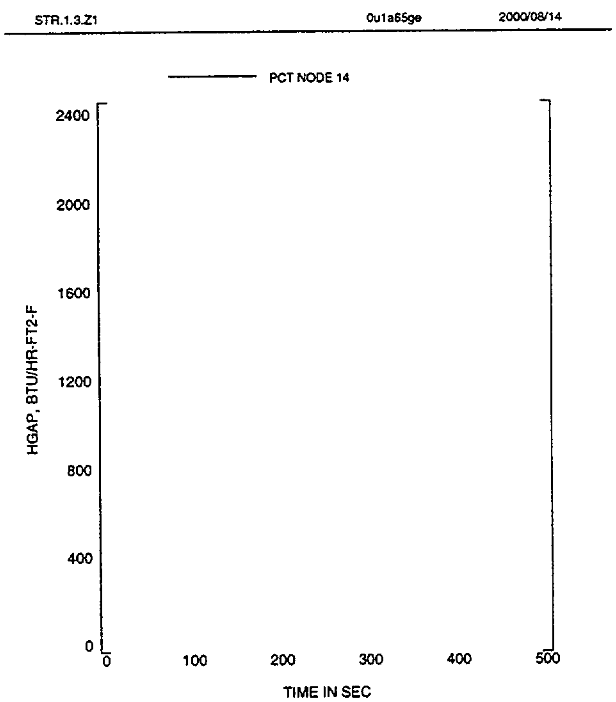


Figure 3.6-3 N
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
PEAK LOCAL CLAD OXIDATION PRCT
HOT CHANNEL

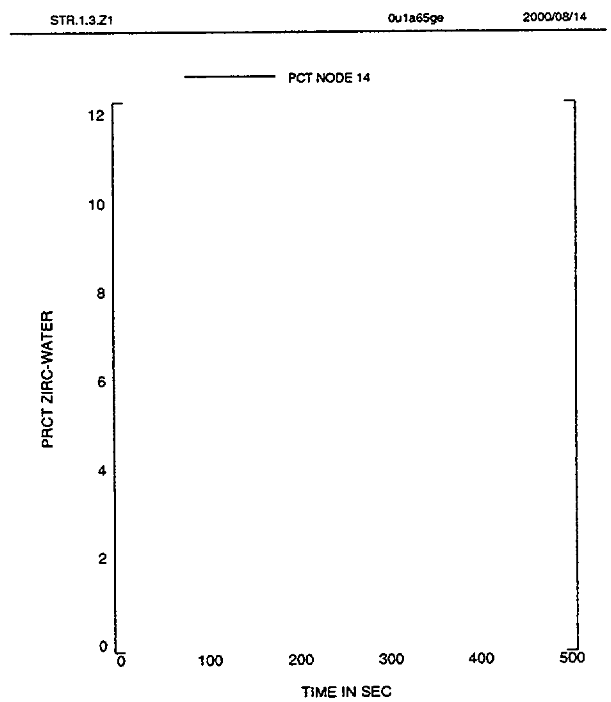


Figure 3.6-3 O
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HOT SPOT RADIAL TEMPERATURES
HOT CHANNEL

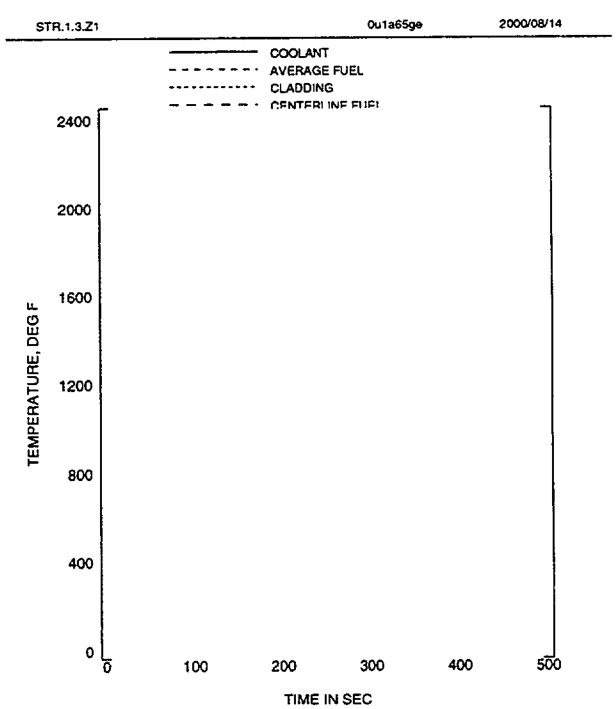


Figure 3.6-3 P
1999 EM SPECTRUM ANALYSIS 0.6 DEDLG
HEAT TRANSFER COEFF AT HOT SPOT
HOT CHANNEL

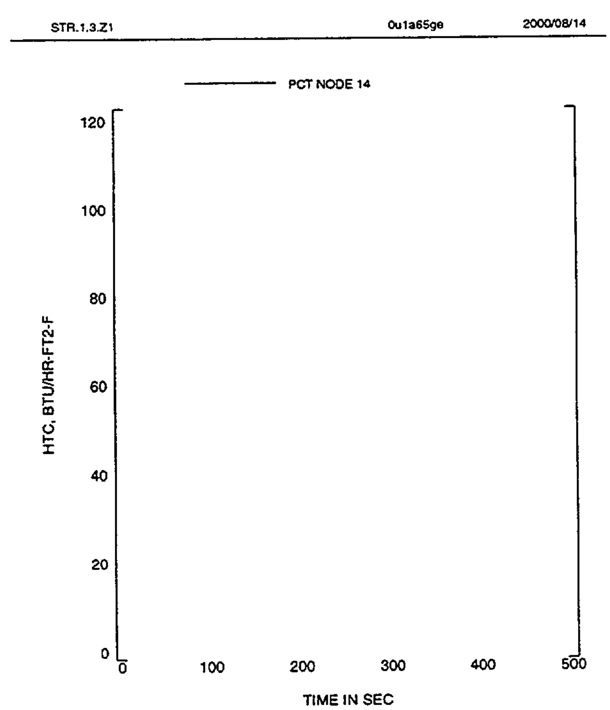


Figure 3.6-4 A
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
NORMALIZED CORE POWER

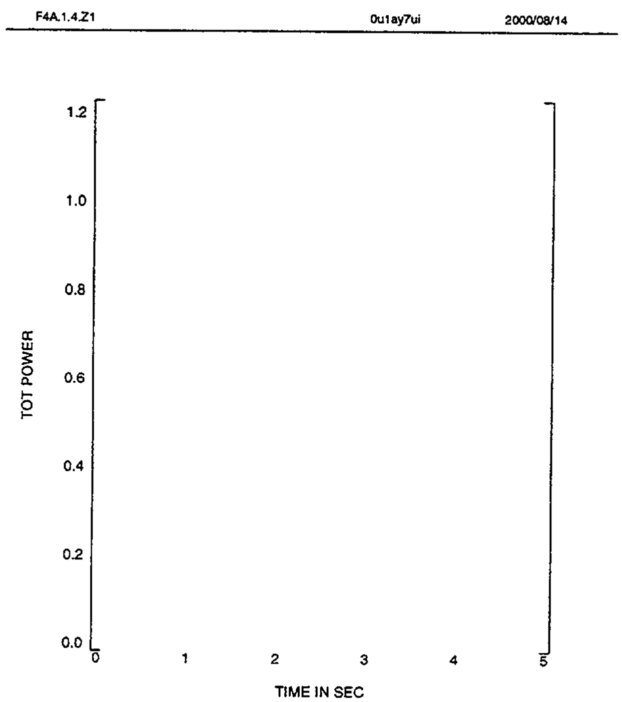


Figure 3.6-4 B
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
PRESSURE IN CENTER OF HOT ASSEMBLY
PCT NODE 13

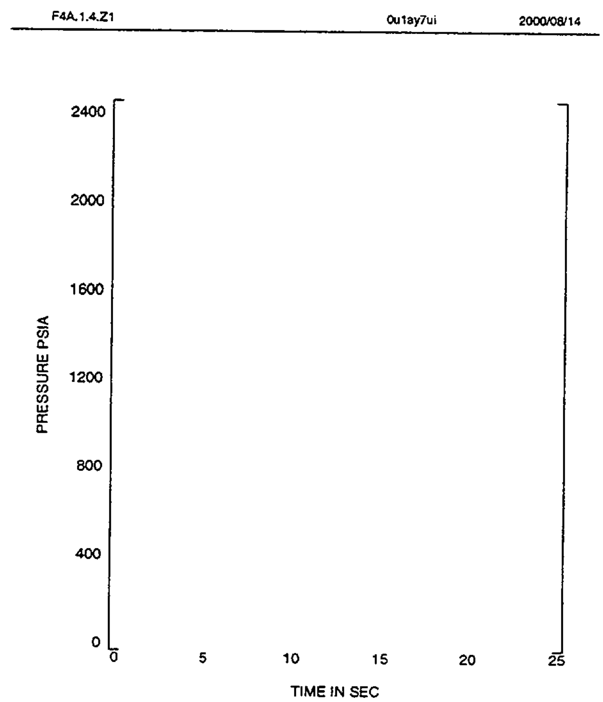


Figure 3.6-4 C
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
PRESSURE IN STEAM GENERATOR SECONDARY
NODE 53

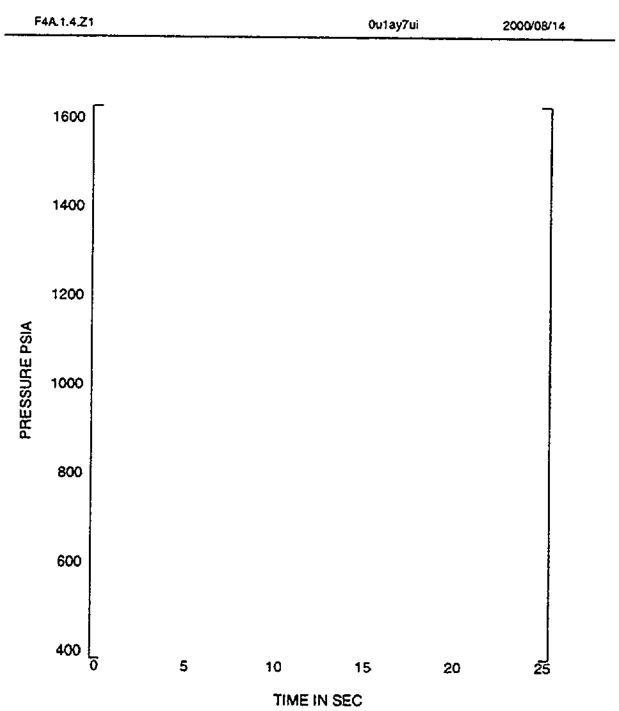


Figure 3.6-4 D
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
BREAK FLOW RATE
PATHS 72 AND 74

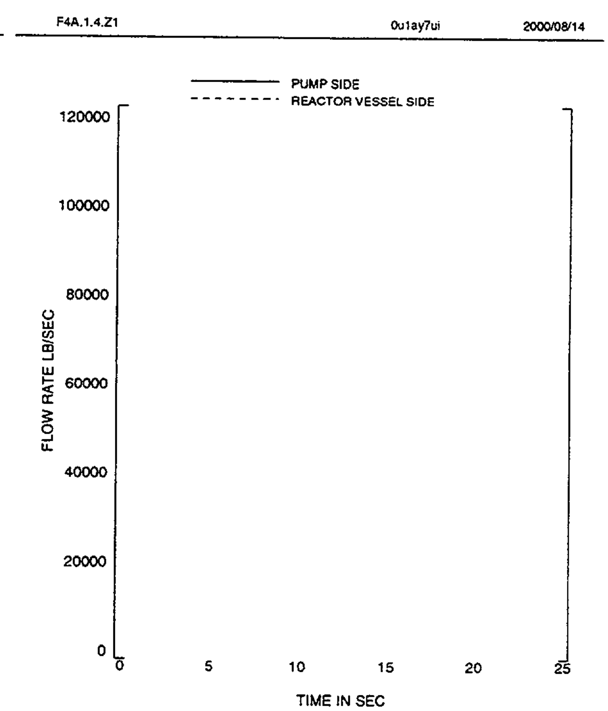


Figure 3.6-4 E
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
BULK CHANNEL CORE FLOW RATE
PATHS 1 AND 6

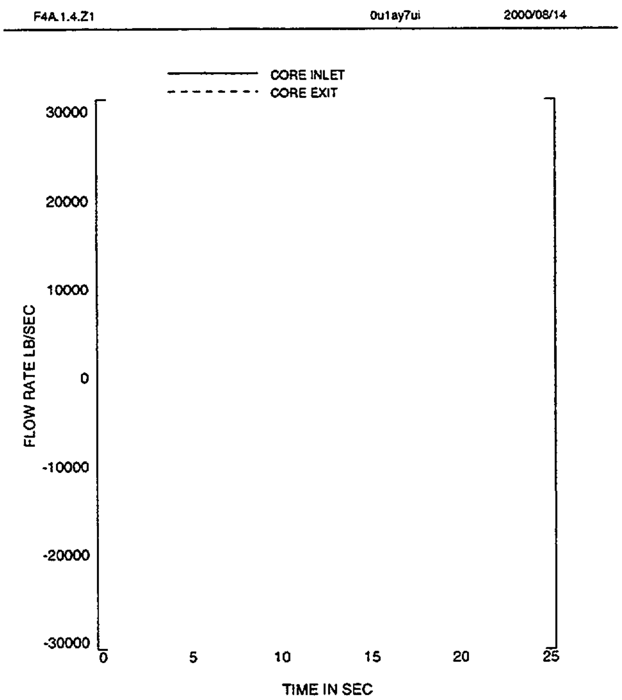


Figure 3.6-4 F
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT ASSEMBLY FLOW RATE
PATHS 16 AND 17

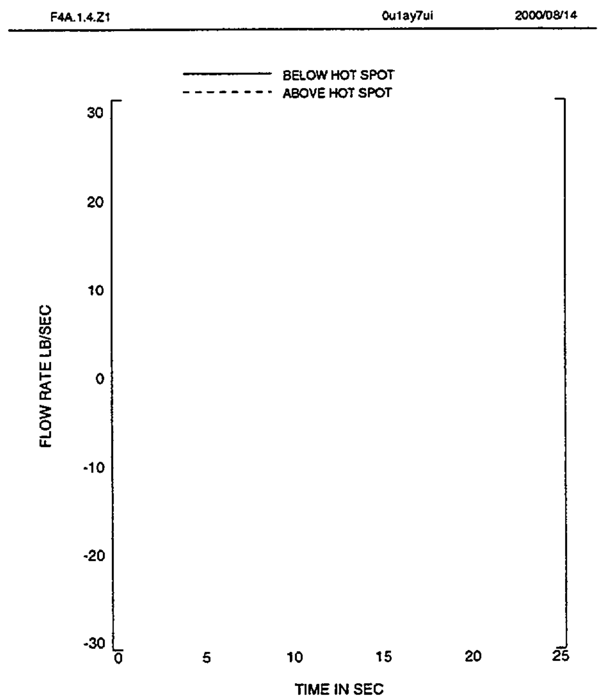


Figure 3.6-4 G
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT ASSEMBLY QUALITY
NODES 13, 14, AND 15

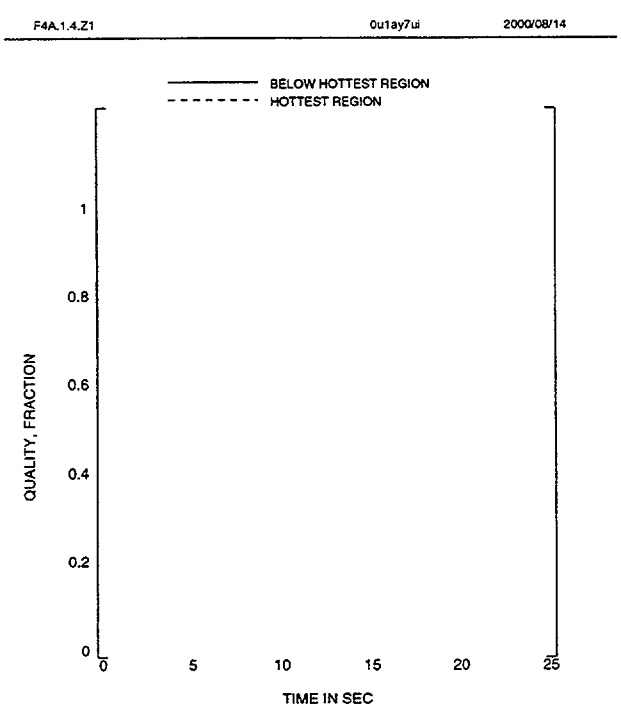


Figure 3.6-4 H
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
NODE 14

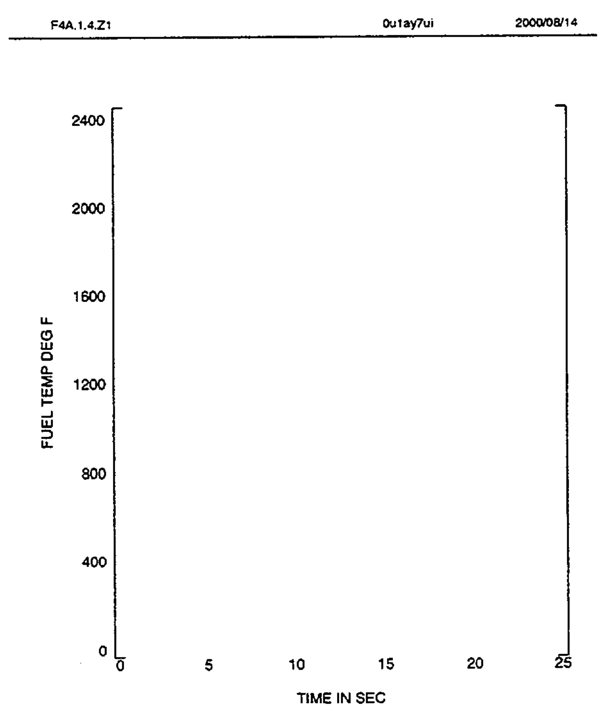


Figure 3.6-4 I
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT ASSEMBLY CLADDING TEMP HOT SPOT
NODE 14

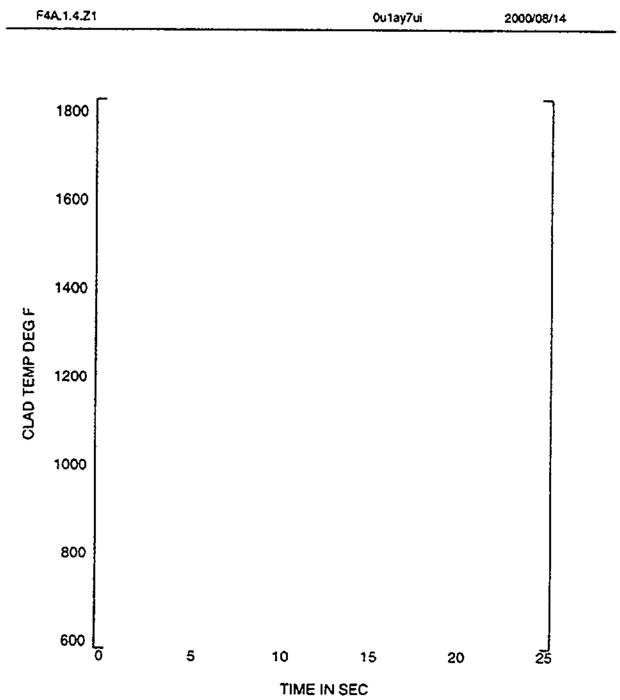


Figure 3.6-4 J
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
CONTAINMENT PRESSURE

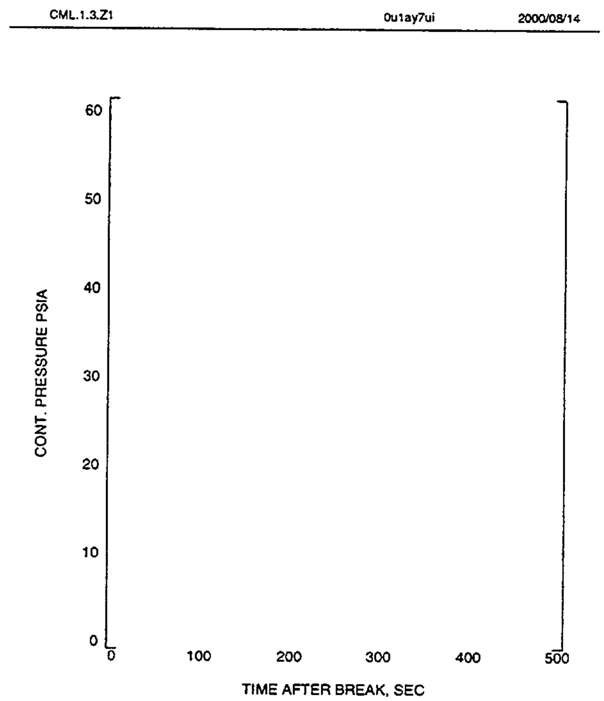


Figure 3.6-4 K
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
REFLOOD LIQ. MASS

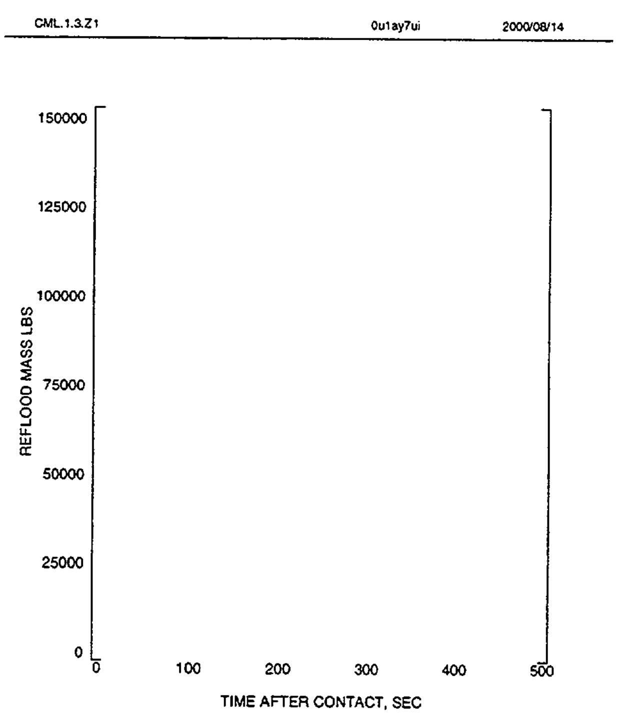


Figure 3.6-4 L
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
CLADDING TEMP
HOT CHANNEL

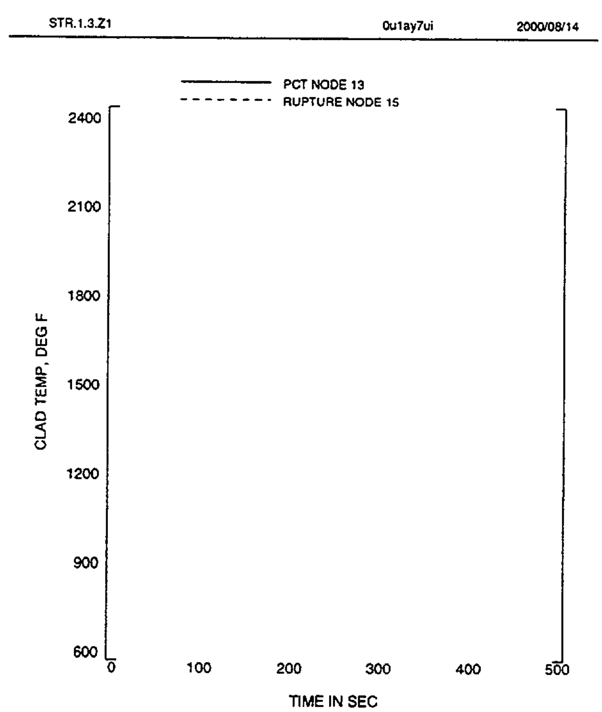


Figure 3.6-4 M
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT SPOT GAP CONDUCTANCE
HOT CHANNEL

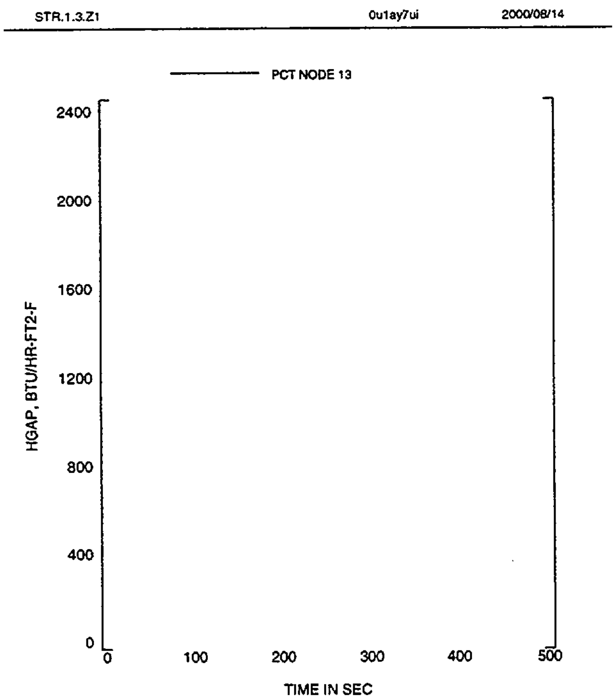


Figure 3.6-4 N
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
PEAK LOCAL CLAD OXIDATION PRCT
HOT CHANNEL

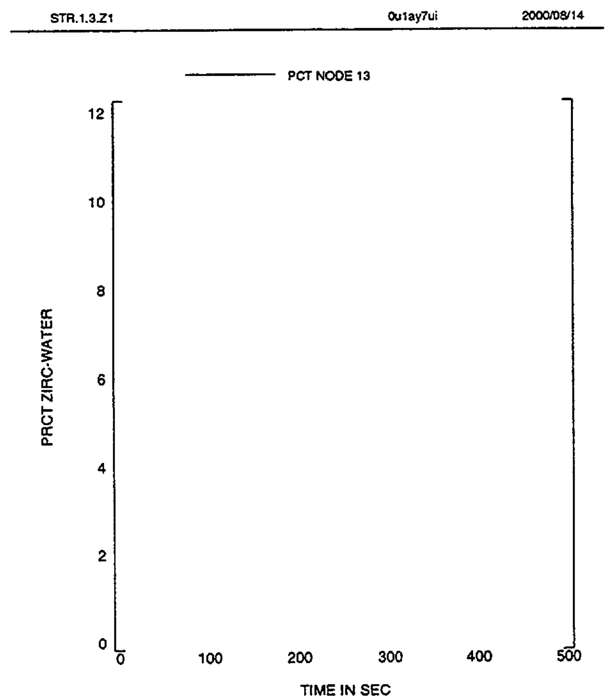


Figure 3.6-4 O
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HOT SPOT RADIAL TEMPERATURES
HOT CHANNEL

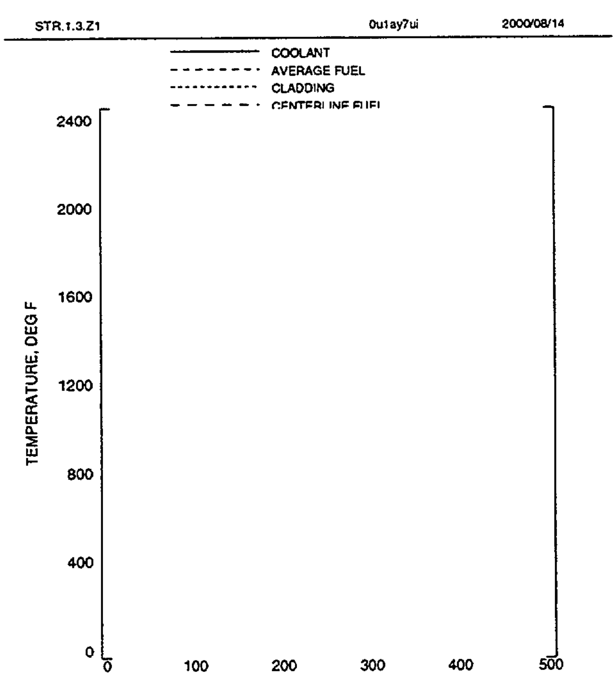


Figure 3.6-4 P
1999 EM SPECTRUM ANALYSIS 0.4 DEDLG
HEAT TRANSFER COEFF AT HOT SPOT
HOT CHANNEL

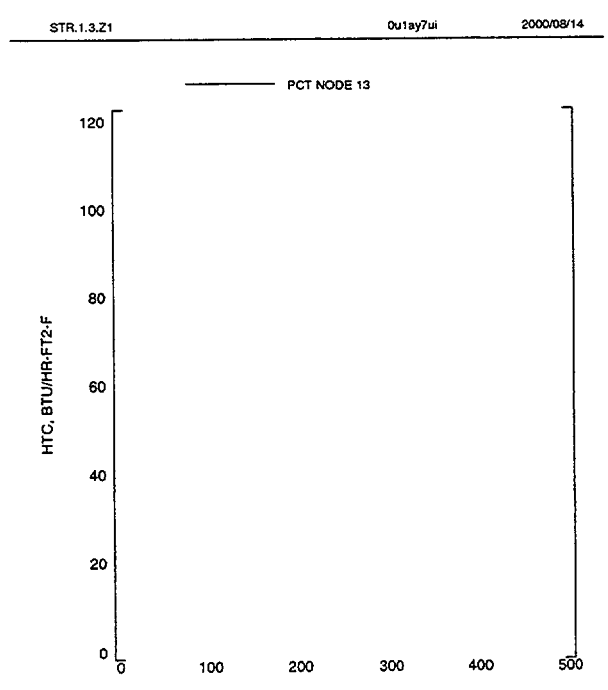
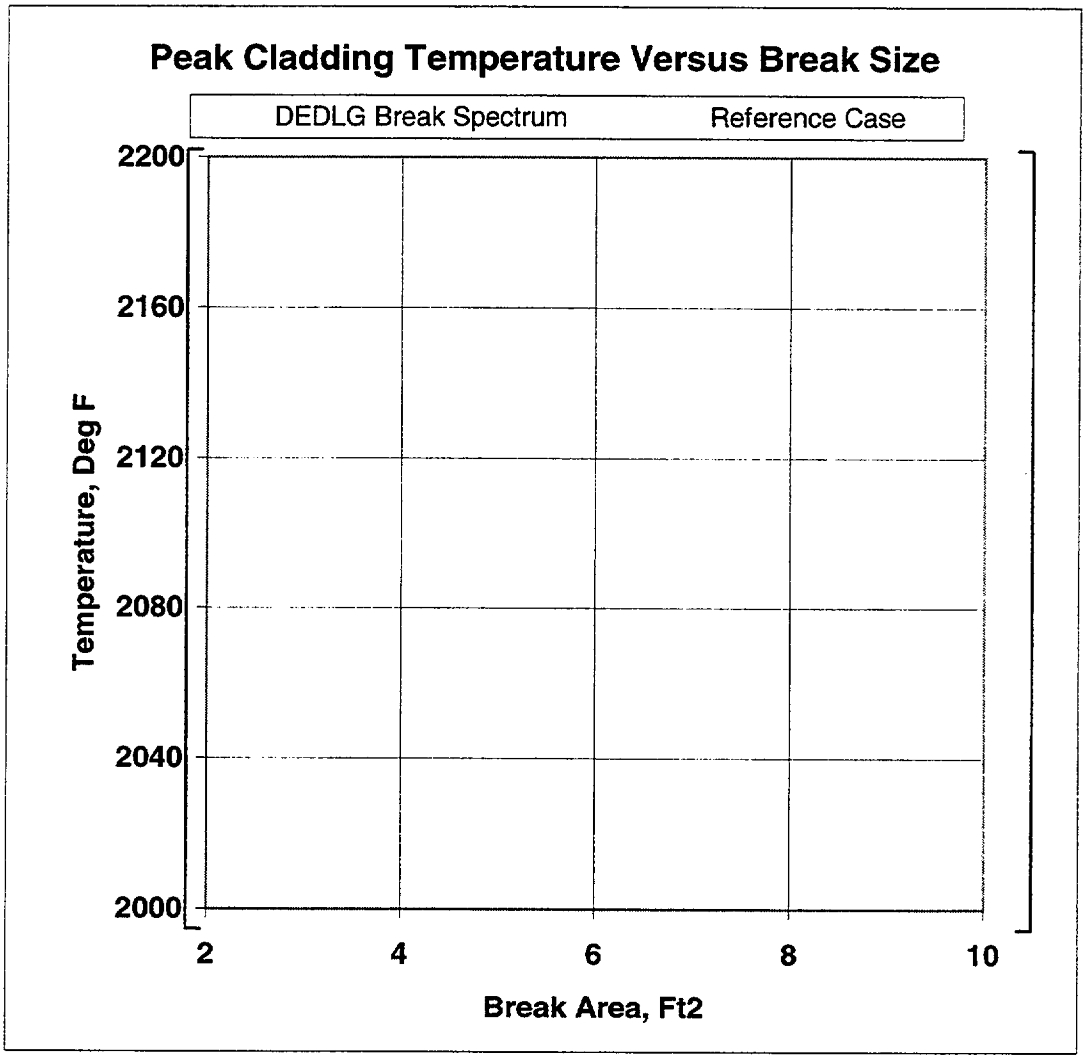


Figure 3.6-5

1999 EM Application Analysis Break Spectrum Results



3.7 Assessment of Overall Conservatism in the 1999 EM

The implementation of the entire set of model improvements in the 1999 EM to gain LBLOCA ECCS analysis margin is documented in Section 3.6, above. The PCT reduction for a typical Combustion Engineering designed PWR application is[ ] This reduction is achieved by removing conservatisms that exist in the current 1985 EM. In the following discussion, this reduction in conservatism is compared to the overall conservatism that can be demonstrated to exist in the Appendix K approach to licensing analyses.

In general, an Appendix K EM contains two types of conservatisms, namely, those that are required by Appendix K to 10CFR50 and those that are not. (The 1999 EM improvements are examples of conservatisms that are not required by Appendix K.) Together, the conservatisms contained within these two categories define the overall conservatism of the EM. In this context, an indication of the overall conservatism is the difference between the PCT calculated by the Appendix K model and that calculated by a realistic evaluation model (REM). Without a REM to provide the basis for the comparison, there is no simple, quantitative means to identify the overall conservatism in an Appendix K EM. Accordingly, this section assesses the overall conservatism of the 1985 Appendix K EM by comparing the decrease in PCT that it produces, relative to the current 1985 EM, when selected required features of Appendix K are removed. This case is referred to as a Non-EM case.

It is generally recognized that three key Appendix K requirements embody the majority of conservatism in an Appendix K EM for LBLOCA. Therefore, a comparison where these key requirements are replaced by realistic elements (a Non-EM case) offers a means to judge the impact on PCT of these changes relative to the important conservatisms in the Appendix K EM. Also, this comparison indicates that these key requirements still introduce significant conservative margin in the revised EM PCT even with the 1999 EM model improvements incorporated. The key Appendix K requirements considered in this assessment are the following:

1) Use of a multiplier of 1.2 times the 1971 ANS standard for decay heat
2) Use of a locked impeller for determining the primary system coolant pump resistance to steam venting during the reflood period
3) Use of steam heat transfer only on the hot rod at the rupture elevation and above when reflood rates are less than one inch per second

The 1985 EM utilizes the Dougall-Rohsenow film boiling correlation, which contains known non-conservatisms. Therefore, for this assessment of overall conservatism, this model is replaced with the 1999 EM correlation substitute described in Section 2.2.

It can be argued that conservatism in an Appendix K licensing calculation is not intuitively obvious. Due to competing effects of models and assumptions, what can be shown to be conservative for one analysis scenario may not be conservative under slightly different conditions. In a complex analytical system, the existence of interconnected model elements creates the need to carefully justify most assumptions or analysis conditions. However, by selecting Appendix K required elements for the Non-EM case and not selecting model-specific conservative features, this complexity in interrelated model issues is significantly reduced. An example of a model-specific feature with interrelated model complexities is the [

] This exercise to show overall conservatism in the 1999 EM would be more complex if this realistic model feature were selected for the study.

For this analysis, the 1999 EM automated/integrated code system is used in 1985 EM simulation mode, except that the Dougall-Rohsenow film boiling correlation is removed. The 1999 EM improvements and the other 1985 EM process changes are not incorporated into this calculation. The elements of this assessment are designed to demonstrate the overall conservatism inherent in the three Appendix K requirements identified above. The justification for the change to the three selected Appendix K requirements is as follows:

1. Decay Heat Multiplier. The Appendix K required decay heat model is 1.2 times the 1971 ANS standard for infinite operating time, including the heat from the decay of actinides. Figure 3.7-13 shows that the 1.0 1971 ANS standard decay heat curve is [ ] to the 1979 ANS standard decay heat curve at the $+2\sigma$ uncertainty level (Reference 3.0-3). Therefore, for this assessment, the reduction from a 1.2 multiplier to 1.0 for the 1971 ANS standard is [ ]

2. Steam Venting Resistance. Appendix K requires that the resistance to steam venting during the reflood period be based on assuming that the primary system coolant pumps have locked impellers if this assumption leads to the maximum PCT, otherwise the pump rotor should be assumed to be running free. Calculations have shown that the locked rotor hydraulic resistance to steam venting leads to the highest PCT. Therefore, for this assessment of overall conservatism, the pump rotor is assumed to be running free and that the resistance remaining in this segment of the overall resistance is that of the cold leg piping and reactor vessel nozzle. For cold leg breaks, a freely turning reactor coolant pump is not unexpected in at least one of the loops, providing a low resistance path for steam venting. In Appendix K licensing calculations, [ ] For this assessment, the margin of conservatism is explicitly calculated for the two extreme states of (1) a locked rotor condition for the EM (high pressure drop across the RCP) and (2) a free running pump for the non-EM case (low pressure drop across the RCP).

3. Steam Cooling on Hot Rod. Appendix K requires that the hot rod be cooled only by steam on the rupture node and above for reflood rates less than one inch per second.

To experimentally study this aspect of reflood heat transfer, the FLECHT-SEASET test program utilized various types and configurations of blockage to simulate cladding deformation and rupture. These tests essentially show that the Appendix K requirement for reflood rates less than one inch/sec, where only heat transfer to steam is allowed, is extremely conservative. The tests show that the liquid entrained by the reflood process passes through the blockage while the steam is redistributed around the blockage and back into the subchannels above the blockage. Droplet to steam interactions are enhanced by the blockage leading to improved heat transfer coefficients. That is, the observed effect of blockage was a heat transfer enhancement rather than a penalty as required by Appendix K. Therefore, these blocked tests support the conservatism inherent in the Appendix K requirement. For this assessment, the steam cooling model restrictions imposed on the 1985 EM are removed and the hot rod heat transfer methodology is calculated on the basis of the FLECHT correlation alone at all reflood rates and all nodes.

Table 3.7-1 shows the results of this Non-EM assessment analysis. The reference case for this assessment was described for analyses using the 1985 EM in Section 3.5. The PCT of the 1985 EM simulation case is 2189°F. The results of the three cases are as follows:

Case #1, 1.0 Multiplier on the ANS Decay Heat: Reducing the decay heat multiplier on the 1971 ANS decay heat from 1.2 to 1.0 significantly reduces the core heat source during the LBLOCA transient. Table 3.7-1 shows that the PCT for this case is [ ] which represents a reduction of [ ] relative to the reference Appendix K case. Figures 3.7-1 through 3.7-4 show the impact of the reduced decay heat multiplier on the blowdown thermal-hydraulics calculations. [

]

- Case #2, Reduced Steam Venting Resistance: Modeling the freely turning reactor coolant pump, significantly reduces the hydraulic resistance to steam venting during the LBLOCA transient. For this assessment, [ ] in the flow resistance (k-factor) was represented for the reactor coolant pump segment of the steam venting path. (This path includes the upper plenum, hot leg, steam generator U-tubes, suction leg, reactor coolant pump, and discharge leg.) That is, the flow resistance resulting from this reduction was roughly equivalent to the k-factor for an abrupt area contraction and expansion from the suction leg piping, through the pump volute, and into the discharge leg piping. The overall steam venting resistance was [ ] Table 3.7-1 shows that the PCT for this case is [ ] which is a reduction of [ ] relative to Case #1. Figures 3.7-5 through 3.7-8 show the significance of reducing the steam venting resistance. [

]

- Case #3, Steam Cooling Requirement: As a consequence of the previously analyzed change in the Appendix K requirement for steam venting resistance, the hot rod PCT became limited by the steam cooling requirement on the node above the cladding rupture location for core reflooding rates less than one inch per second. Therefore, to further determine the overall level of conservatism in the Appendix K EM, the steam cooling requirement was removed, and the hot rod PCT was determined by the FLECHT heat transfer methodology at all elevations of the hot rod and for all core reflood rates. Table 3.7-1 shows that the PCT for this case is [ ] which is a reduction of [ ] relative to Case #2. Figure 3.7-12 shows the reduction in PCT for this case.

In summary, the PCT, when combining the effects of removal of all three Appendix requirements, was [ ] Therefore, these comparisons show that replacing the three Appendix K required conservative models with less conservative but justified models produces a margin of conservatism associated with the three Appendix K requirements of [ ] As described in Sections 3.4, 3.5, and 3.6, the 1999 EM model improvements will reduce PCT by roughly [ ] Even if this reduction is double for other LBLOCA analyses, this leaves over [ ] of conservative margin in the 1999 EM calculation relative to the Non-EM case. Based on these comparisons, it is concluded that the 1999 EM maintains an appropriate amount of overall model conservatism.

Table 3.7-1

Non-EM Parameter Study

[

Result	1985 EM Simulation	Case #1 1.0xANS Decay Heat	Case #2 Unlocked RCP Rotor	Case #3 No Steam Cooling

]

Figure 3.7-1
NON-EM PARAMETER STUDY
NORMALIZED CORE POWER

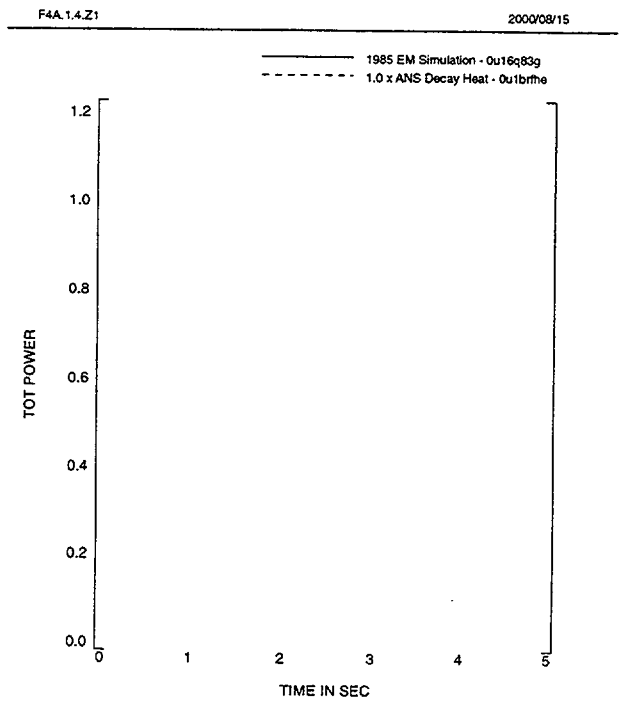


Figure 3.7-2
NON-EM PARAMETER STUDY
PRESSURE IN CENTER OF HOT ASSEMBLY
NODE 13

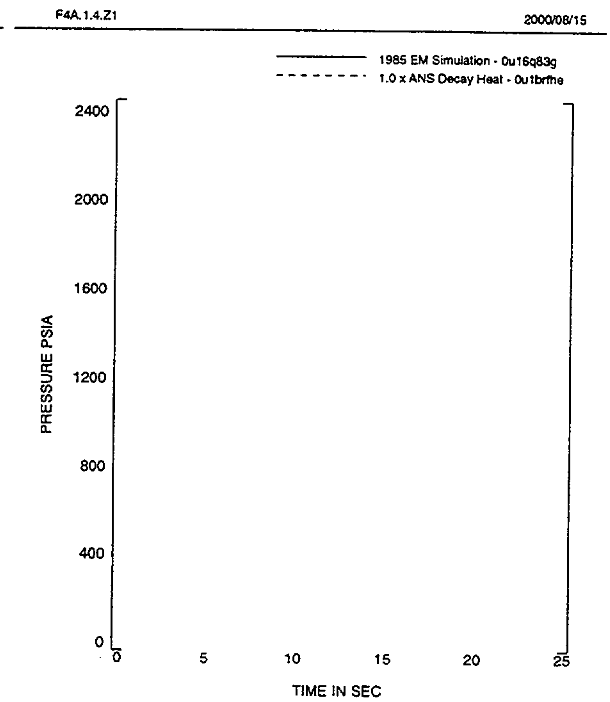


Figure 3.7-3
NON-EM PARAMETER STUDY
HOT ASSEMBLY FUEL AVE TEMP HOT SPOT
NODE 14

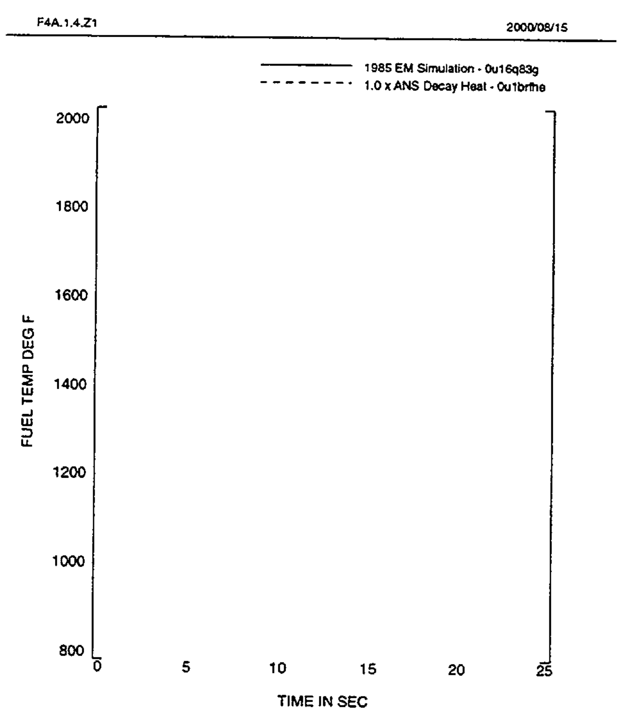


Figure 3.7-4
NON-EM PARAMETER STUDY
HOT ASSEMBLY CLADDING TEMP HOT SPOT
NODE 14

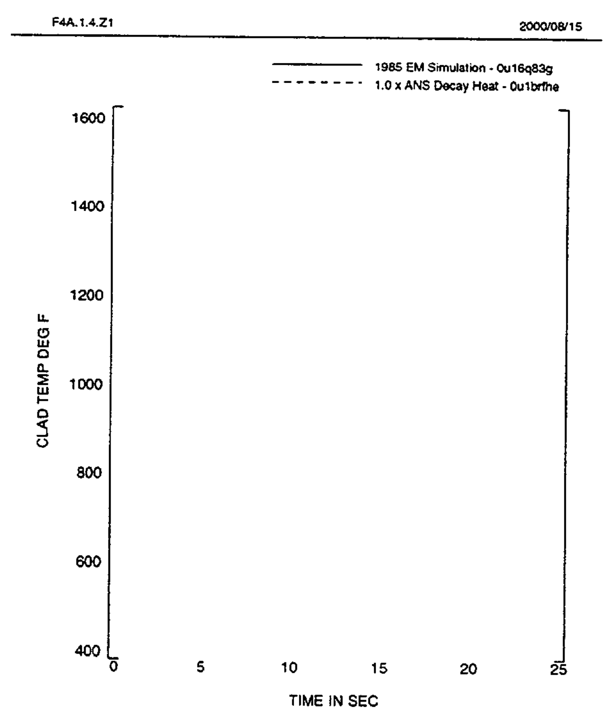


Figure 3.7-5
NON-EM PARAMETER STUDY
CONTAINMENT PRESSURE

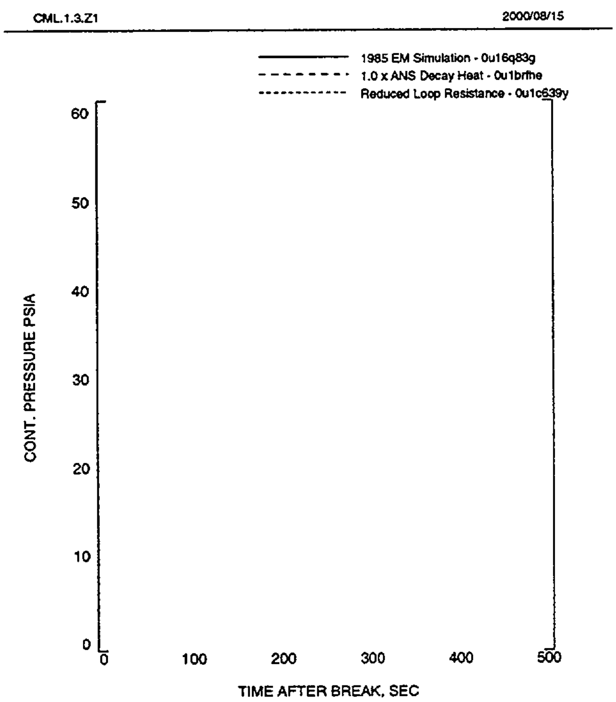


Figure 3.7-6
NON-EM PARAMETER STUDY
SUBCOOLED LIQ. LEVEL

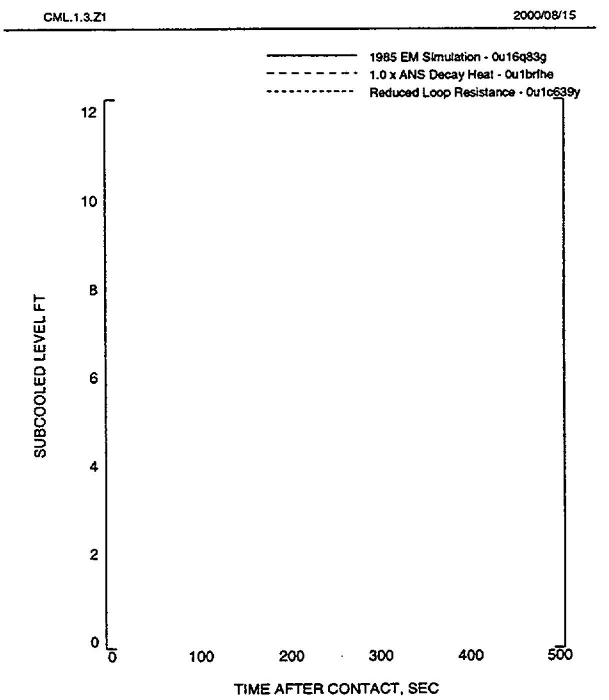


Figure 3.7-7
NON-EM PARAMETER STUDY
REFLOOD LIQ. MASS

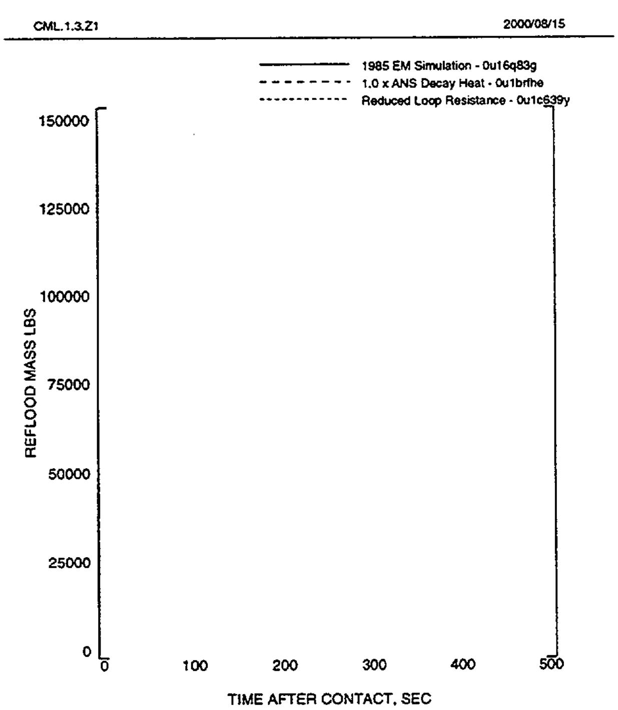


Figure 3.7-8
NON-EM PARAMETER STUDY
EQUIVALENT K-FACTOR

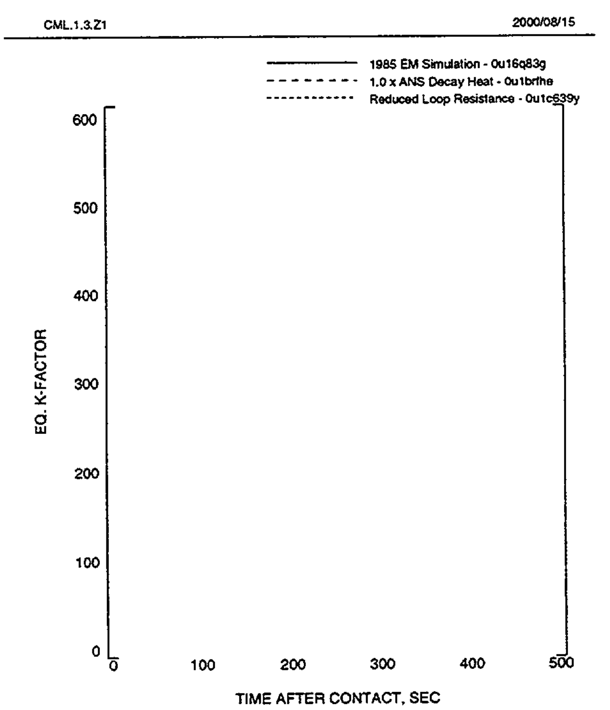


Figure 3.7-9
NON-EM PARAMETER STUDY
CLADDING TEMP NODE 14
HOT CHANNEL

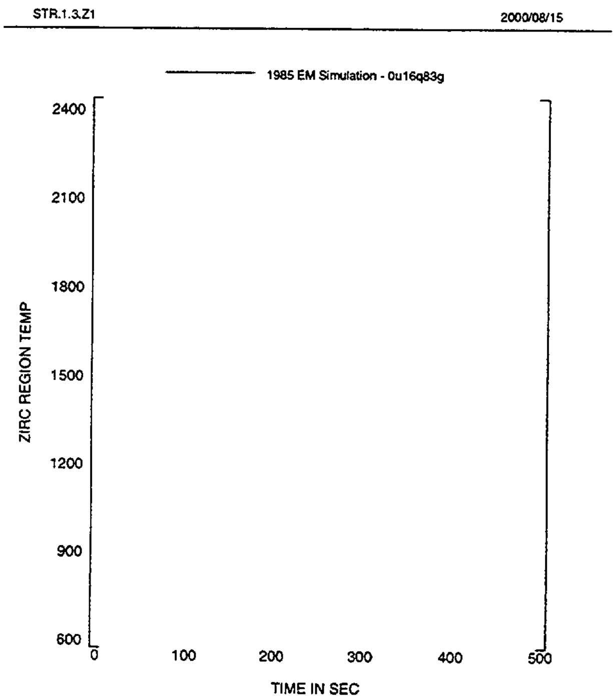


Figure 3.7-10
NON-EM PARAMETER STUDY
CLADDING TEMP NODE 14
HOT CHANNEL

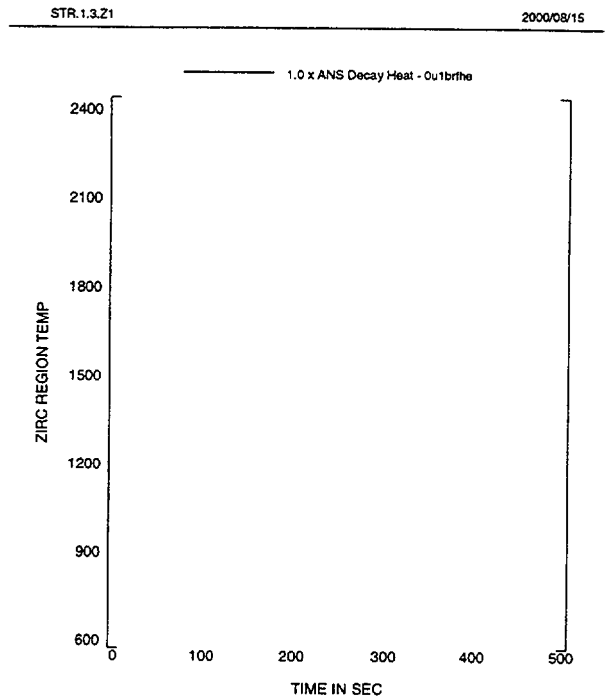


Figure 3.7-11
NON-EM PARAMETER STUDY
CLADDING TEMP NODE 14
HOT CHANNEL

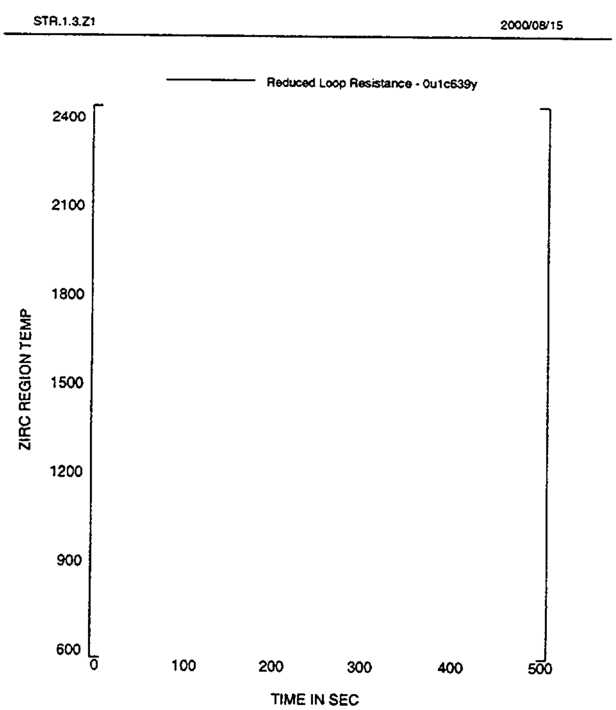


Figure 3.7-12
NON-EM PARAMETER STUDY
CLADDING TEMP NODE 14
HOT CHANNEL

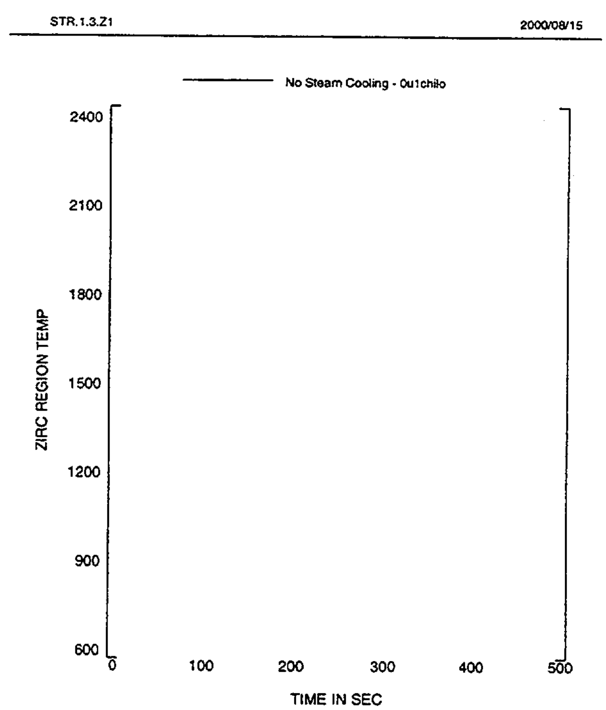
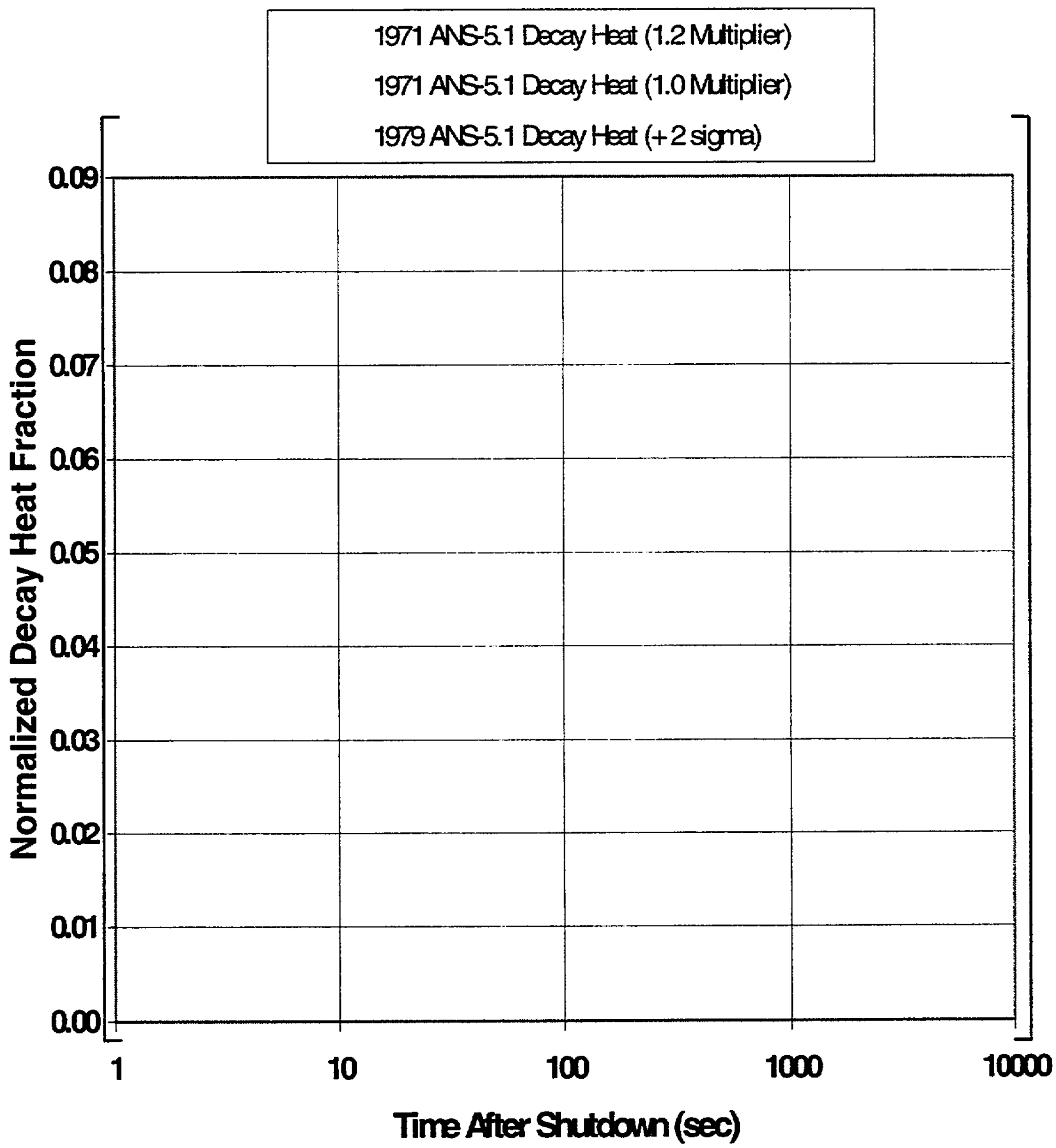


Figure 3.7-13
Decay Heat Curves



3.8 References

3.0-1 D. M. Crutchfield (NRC) to A. E. Scherer (CE), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Topical Reports," July 31, 1986.

3.0-2 CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985, Section 4.3.

3.0-3 ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," American National Standards Institute, Inc., August 1979.

Appendix A

Input and Output Descriptions for the 1999 EM Automated/Integrated Code System

A.1 The User Control Interface Input File

The User Control Interface (UCI) file is a single file containing input variables that control model options in CEFLASH-4A (Reference A-1), COMPERC-II (Reference A-2), STRIKIN-II (Reference A-3), PARCH (Reference A-4), and HCROSS (Reference A-5). The UCI file consists of line entries in free format. The first entry of each line describes the model option identifier (an alphanumeric variable without blanks). The second entry is a numeric variable identifying the option. Trailing comments are optional. The list of UCI variables is given in Table A-1. If a UCI input variable is not entered in the file, then its value will default to either zero or to the default value indicated in Table A-1.

Table A-1 also lists required and optional input options for executing the LBLOCA computer codes in one of the following three modes:

- 1985 EM consistent with currently approved EM
- 1985 EM with process improvements documented in this topical report consistent with currently approved EM
- 1999 EM as described by this topical report.

Table A-1
User Control Interface (UCI) Input Variables

[

Model ID	Model Used by Code	Standard Input 1985 EM	Optional Input 1985 EM Process Improved	Required Input 1999 EM	Description

A.2 The CEFLASH-4A and COMPERC-II Input File for the 1999 EM Code System

A new input file is defined for the CEFLASH-4A and COMPERC-II codes (f4acml.inp.mod1.jobid). This input file contains input data to execute CEFLASH-4A and COMPERC-II as part of the 1999 EM automated/integrated code system. The new input file consists of line entries in free format and in any order. The first entry of each line describes the model option (an alphanumeric variable without blanks). The second entry is a numeric variable identifying the option. Trailing comments are optional. The content of this input file consists of the following three parts:

1. Containment Spray Flow Delivery Data
2. Broken Loop SIT Data
3. Steam Generator Secondary Side Physical Characteristics and Model Options (Including Safety Injection Pump Actuation Setpoint Data)

Containment Spray Flow Delivery Data

The containment spray inputs describe the pressure (psia) vs. flow (gpm) for the containment spray pumps. These inputs are required for the automatic spray and spillage option.

Line 1	CSP NCSP	CSP is the containment spray alphanumeric ID NCSP is the number of Pressure vs. Flow pairs, an integer less than or equal to 50
Lines 2 (NCSP lines)	Pressure(i) Flow(i)	Pressure vs. Flow pairs Pressure (psia) Flow rate (gpm) One line per pair of pressure vs. flow values
Line 3	TSPRSTR TSPREND	Spray flow delivery start time (sec) Spray flow delivery end time (sec) (both times relative to start of LOCA)

Broken Loop SIT Data

For the automatic spray and spillage option, these inputs define the physical characteristics for the safety injection tank connected to the broken loop for discharge leg break configurations, i.e., for Rel. Add. 11, NTYPE = 3. These inputs are not required for hot leg or suction leg breaks, i.e., for Rel. Add. 11, NTYPE $\neq$ 3.

Line 1	SIT NTYPEX	SIT is the SIT inputs alphanumeric ID NTYPEX: Integer = 3. Do SIT calculations for discharge leg break NTYPEX: Integer $\neq$ 3. Do not model broken loop SIT for the automatic spray and spillage option. If NTYPEX < 1 then Line 2 must be omitted.
Line 2	AAAX	Area of tank injection line (ft^2)
	AKAKX	Flow coefficient (dimensionless)
	SPECVX	Liquid specific volume (ft^3/lbm)
	PGSITX	Gas pressure in tank (psia)
	PESITX	Elevation head (psia)
	VLSITX	Liquid volume (ft^3)
	VGSITX	Gas volume (ft^3)

Steam Generator Secondary Side Physical Characteristics and Model Options

Table A.2 provides the steam generator secondary side physical characteristics for the 1999 EM model for COMPERC-II steam venting reflood thermal-hydraulics. There are several model-related options that must be selected for the correct operation of the model.

This input file provides a means to introduce wall heat parameters into the secondary side nodes analyzed by CEFLASH-4A during the blowdown portion of the transient without modifying the CEFLASH-4A plant specific base deck. Entering the input once in this file, insures consistency between the wall heat modeling in CEFLASH-4A and COMPERC-II for the steam generator secondary side nodes.

This input file provides the plant data necessary to model safety injection pump actuation based on SIAS (low pressurizer pressure trip setpoint) with an appropriate delay time.

Table A-2
The CEFLASH-4A and COMPERC-II New Input File

Variable	Code Using Variable	Description	Notes

A.3 The PARCH Code Spacer Grid Input File

An improved model for steam cooling heat transfer has been implemented into the PARCH code in order to calculate the effects of spacer grids on steam heat transfer coefficients. [

]

The spacer grid input is optional. The input should be specified if parch_spacer_grid is equal to 1 in the UCI. The spacer grid inputs are the following:

NGRID	Number of spacer grids
BGRID	Area reduction or blockage fraction
DGRID	Hydraulic diameter of spacer grid, ft
HGRID	Height of spacer grid, ft
EGRID(I), I = 1, NGRID	Elevation of the top edge of the spacer grids, relative to the bottom of the core, ft

The first three variables (NGRID, BGRID, HGRID) are read with a format

I5, 2F10.0

Each of the spacer grid elevation heights (EGRID(I)) is read with the format

F10.0

A.4 The 1999 EM Automated/Integrated Code Interface Files

Execution of the 1999 EM automated /integrated code system creates a set of interface files that transfer data among the LOCA codes. This section lists the outputs for each of the new interface files. Figure 2.1-2 shows the LBLOCA computer codes and interface files that comprise the 1999 EM code system. This section does not include the many input and output files that are part of the code system execution.

The interface files are listed by file name. Each execution of the automated/integrated code system generates a unique case identifier called the "jobid." This designator is appended to each of the file names to provide uniqueness and traceability for each case that is run. As described in Section 2.1.1, it is the existence of the interface file that activates the automatic linkage of the codes.

The interface files that are already part of the 1985 EM code system are as follows:

File Name.jobid	Upstream Code	Downstream Code
4ap.cml.leak1	4APUNCH	COMPERC-II
4ap.str.bhyd	4APUNCH	STRIKIN-II
cml.hpu.fhtc	HTCOF	HPUNCH
hpu.str.fhtc	HPUNCH	STRIKIN-II
fat.str.[h/a]pin	FATES3B	STRIKIN-II

The interface files developed for the 1999 EM automated/integrated code system are as follows and are described in the following tables:

File Name.jobid	Upstream Code	Downstream Code	See Table
uci.inp	none	All	A-1
f4a.cml	CEFLASH-4A	COMPERC-II, COMZIRC	A-3
cml.htcof	COMPERC-II	HTCOF, FRELAPC, STRIKIN-II, COMZIRC	A-4
cml.prm	COMPERC-II	PARCH	A-5
fre.[htcof/cmz]	FRELAPC	HTCOF, COMZIRC	A-6
cml.hpu.inp	HTCOF	HPUNCH & STRIKIN-II	A-7
f4a.cmz	CEFLASH-4A	COMZIRC	A-8
cml.cmz	COMPERC-II	COMZIRC	A-9
str.cmz	STRIKIN-II	COMZIRC	A-10
str.f4a	STRIKIN-II	CEFLASH-4A	A-11

Table A-3

Interface File Content for

f4a.cml.jobid

[

Line ID	Typical Value	Description	Units

]

Table A-4
Interface File Content for
cml.htcof.jobid

[

Line ID	Typical Value	Code	Description	Units

]

Table A-5

Interface File Content for

cml.prm.jobid[1]

[

]

1 Typical values, first 31 entries of file (Continues to the end of the transient)

Table A-6

Interface File Content for

fre.htcof.jobid

[

]

Interface File Content for

fre.cmz.jobid

[

]

Table A-7
Interface File Content for
cml.hpu.inp.jobid

[

Line ID	Typical Value	Description	Units

]

Table A-8

Interface File Content for

f4a.cmz.jobid

[

]

Table A-9

Interface File Content for

cml.cmz.jobid

[

]

Table A-10
Interface File Content for
str.cmz.jobid

[

Line ID and/or Node Number	Typical Value	Description & Units

]

Table A-11
Interface File Content for
str.f4a.jobid

[

Line ID	Typical Value	Typical Value	Description & Units

]

A.5 Description of Extra Diagnostic Output Files for the 1999 EM

Several diagnostic output files were added to the CEFLASH-4A (Reference A-1), COMPERC-II (Reference A-2), STRIKIN-II (Reference A-3), and PARCH (Reference A-4) codes during the development work. Generation of the output files is controlled by the UCI control variable extra_output (= 0 Do not generate extra output, = 1 Generate extra output). Generally, the output files are written in tabular form with time as the first entry.

The list of diagnostic output files for the 1999 EM is given in Table A-12. The output variables of diagnostic output files related to the new models in COMPERC-II that present information not directly available in the COMPERC-II main output files are described in Tables A-13 through A-19. The contents of the other diagnostic output files listed in Table A-12 are not listed in this appendix. These files contain information used during the development phase of work to verify the consistency of the automated/integrated process models.

Table A-12

List of Diagnostic Output Files

File Name	Description	Reference
f4a.fiie.jobid	CEFLASH-4A time step control output	----
f4a.slip.jobid	CEFLASH-4A slip model output	----
f4a.t32.jobid	CEFLASH-4A clad rupture variables	----
f4a.t28.jobid	CEFLASH-4A plastic strain variables (node 14)	----
f4a.t29.jobid	CEFLASH-4A gap pressure variables (hot assembly)	----
cml.t18.jobid	COMPERC-II Spray and Spillage variables	Table A-13
cml.t19.jobid	COMPERC-II consistency of ECCS injection variables	Table A-14
cml.t22.jobid	COMPERC-II vessel variables	Table A-15
cml.t23.jobid	COMPERC-II SG Secondary Side Variables	Table A-16
cml.t24.jobid	COMPERC-II Containment Variables	Table A-17
cml.t25.jobid	COMPERC-II Reflood Rate Variables	----
cml.t26.jobid	COMPERC-II Spray and Spillage Variables	Table A-18
cml.t27.jobid	COMPERC-II New Nitrogen Injection Model Variables	Table A-19
str.t22.jobid	STRIKIN-II rupture node variables	----
str.t23.jobid	STRIKIN-II rupture and Plastic Strain Variables	----
str.t24.jobid	STRIKIN-II steam Cooling Heat Transfer Coefficient Variables	----
str.t25.jobid	STRIKIN-II film boiling correlation variables	----
str.t26.jobid	STRIKIN-II comparison of PARCH and STRIKIN-II variables	----
str.t27.jobid	STRIKIN-II and PARCH heat balance comparison variables	----
str.t28.jobid	STRIKIN-II plastic strain variables (node 14, ave. rod)	----
str.t49.jobid	STRIKIN-II gap pressure variables (average rod)	----
prm.t76.jobid	PARCH heat balance in the fuel rod	----
prm.t77.jobid	PARCH steam channel variables	----
prm.t78.jobid	PARCH droplet model	----
prm.t79.jobid	PARCH heat balance in the fuel rod	----

Table A-13

File cml.t18.jobid, COMPERC-II Spray and Spillage Variables

Variable Name	Description	Units
TIME	Time measured from time equal to zero	sec
W_SUMP(1)	Spillage flow from vessel	lbm/sec
W_SUMP(2)	SIT mass flow rate (broken loop, PERC)	lbm/sec
W_SUMP(3)	Steam condensed rate broken loop	lbm/sec
W_SUMP(4)	SI pump flow broken loop	lbm/sec
W_SUMP(5)	Total spillage mass flow to containment	lbm/sec
W_SUMP(6)	Spillage mass flow to sump	lbm/sec
W_SUMP(7)	Containment spray mass flow rate	lbm/sec
W_SUMP(8)	Spillage spray flow	lbm/sec
W_SUMP(9)	SIT broken loop spray flow rate	lbm/sec
W_SUMP(10)	Spray flow SIT broken loop (PERC)	lbm/sec
W_SUMP(11)	SIT flow for spray calculation (PRESS)	lbm/sec
W_SUMP(12)	Total flow to sump	lbm/sec
W_SUMP(13)	Blowdown break flow rate	lbm/sec

Table A-14

File cml.t19.jobid, COMPERC-II Consistency of ECCS Injection Variables

Variable Name	Description	Units
TIME	Time measured from TAD time (zero = TAD)	sec
WSI	Total liquid flow to vessel (SIT plus pump plus condensation)	lbm
HSI	Enthalpy of total liquid flow to vessel	Btu/lbm
W_SITLO	Total SIT flow to vessel	lbm/sec
W_PUMPLO	Total pump flow to vessel	lbm/sec
W_SILOOP	Total SI flow to vessel	lbm/sec
H_SITLO	SIT enthalpy to vessel	Btu/lbm
H_PUMPLO	Enthalpy of SI pump flow to vessel	Btu/lbm
H_SILOOP	Enthalpy of SIT plus SI pump flow	Btu/lbm
WSPIL	Spillage from vessel flow rate	lbm/sec
WSICW	Liquid total spillage flow (vessel plus SIT plus pump plus condensation broken loop)	lbm/sec
HSICW	Energy of liquid total spillage flow (flow times enthalpy)	Btu/sec
WSICS	Steam flow from loops	lbm/sec
HSICS	Steam energy from loops (flow times enthalpy)	Btu/sec

Table A-15

File cml.t22.jobid, COMPERC-II Vessel Variables

Variable Name	Description	Units
TIME	Time measured from TAD time (zero = TAD)	sec
PUP	Upper plenum pressure	psia
PANN	Annulus pressure	psia
ZB1T	Core subcooled liquid level	ft
ZB2T	Core two phase mixture level	ft
ZAT	Annulus liquid level	ft
WAB	Instantaneous core reflood rate	lbm/sec
WSX	Steam release rate exiting core	lbm/sec
WSCEX	Total steam flow exiting core	lbm/sec
WSON	Steam flow exiting upper plenum	lbm/sec
WSOP	Total steam flow in loops	lbm/sec
WSI	Liquid flow entering vessel	lbm/sec
WSPIL	Spillage flow from vessel	lbm/sec
PUP	Upper plenum pressure	psia
HUP	Upper plenum enthalpy	Btu/lbm
DENUP	Upper plenum density	lbm/ft^3

In addition the following variables are written once (when the condition occurs)

Variable Name	Description	Units
TIME	Contact time (measured from TAD)	sec
TFST	Time of end of first reflood rate (measured from contact)	sec
TIME	Time of end of first reflood rate (measured from TAD)	sec
ZAT	Annulus level	ft
ZMAX	Elevation of bottom of cold leg	ft
W_FIRST	Average first reflood rate (point calculation)	inch/sec
WI_FIRST	Integrated mass at end of first reflood rate	lbm

Table A-16

File cml.t23.jobid, COMPERC-II Steam Generator Secondary Side Variables

Variable Name	Description	Units
TIME	Time measured from TAD time (zero = TAD)	sec
SG_P	SG secondary side pressure	psia
SG_TSEC	SG secondary side average temperature	°F
SG_TPRII	SG tube inlet temperature	°F
SG_TPRIE	SG tube outlet temperature	°F
SG_TSS(1)	Temperature of SG secondary side axial region 1	°F
SG_TSS(2)	Temperature of SG secondary side axial region 2	°F
SG_TSS(3)	Temperature of SG secondary side axial region 3	°F
SG_TPS(1)	Temperature of SG primary side axial region 1	°F
SG_TPS(2)	Temperature of SG primary side axial region 2	°F
WSOP	Total steam flow	lbm/sec
SG_HI	SG tubes inlet enthalpy	Btu/lbm
HSEC	SG tubes exit enthalpy	Btu/lbm
SG_Q	Heat rate exiting SG primary side	Btu/sec
SG_QCH	Heat rate added to SG tubes steam flow	Btu/sec
SG_QSEC	Heat rate added to SG secondary side	Btu/sec
SG_WCON	SG secondary side total condensation (negative means boiling)	lbm/sec

Table A-16 (Cont)

File cml.t23.jobid, COMPERC-II Steam Generator Secondary Side Variables [1]

Variable Name	Description	Units
TIME	Contact time (measured from TAD)	sec
SG_P	SG secondary side pressure	psia
SG_VOID	SG secondary side void fraction	---
SG_HLIQ	SG secondary side liquid enthalpy (average)	Btu/lbm
SG_HF	SG secondary side saturation liquid enthalpy	Btu/lbm
SG_HG	SG secondary side saturated steam enthalpy	Btu/lbm
SG_TLIQ	SG secondary side liquid temperature (average)	°F
SG_TSAT	SG secondary side saturation temperature	°F
SG_TWALL	SG secondary side wall temperature	°F
SG_MLIQ	SG secondary side liquid mass	lbm
SG_MTOT	SG secondary side total mass	lbm
SG_MSTM	SG secondary side steam mass	lbm
SG_TPSB(1:30)	Sectionalized SG primary side temperature	°F
SG_TPS(1:30)	Sectionalized SG primary side temperature	°F
SG_TTS(1:30)	Sectionalized SG tube temperature	°F
SG_TSS(1:15)	Sectionalized SG secondary side temperature	°F
SG_QPS(1:30)	Sectionalized SG heat exiting primary side	Btu/sec
SG_QSS(1:30)	Sectionalized SG heat entering secondary side	Btu/sec
SG_QDROP(1:30)	Sectionalized SG heat rate from steam to droplets	Btu/sec
SG_QLIQS(1:30)	Sectionalized SG heat rate quenched region	Btu/sec
SG_Q	Heat rate exiting SG primary side	Btu/sec
SG_QCH	Heat rate added to SG tubes steam flow	Btu/sec
SG_QSEC	Heat rate added to SG secondary side	Btu/sec
SG_WSP(1:30)	Sectionalized SG steam flow in primary side	lbm/sec
SG_WLP(1:30)	Sectionalized SG liquid droplet flow in primary side	lbm/sec
SG_WEVAP(1:30)	Sectionalized SG droplet evaporation rate	lbm/sec
SG_VOP(1:30)	Sectionalized SG tube void fraction	---
SG_HPRIS(1:30)	Sectionalized SG primary side heat transfer coefficient	Btu/ft^2-hr-°F
SG_HSECS(1:30)	Sectionalized SG secondary side heat transfer coefficient	Btu/ft^2-hr-°F

1 Additional variables

Table A-17

File cml.t24.jobid, COMPERC-II Containment Variables

Variable Name	Description	Units
TIME	Time measured from time zero	sec
STMIN	Steam flow rate to containment gas space	lbm/sec
WTRIN	Liquid flow to containment sump	lbm/sec
STENIN	Steam energy rate to containment gas space	Btu/sec
WTENIN	Liquid energy rate to containment sump	Btu/sec
MAIR	Containment air mass	lbm
MSTM	Containment steam mass	lbm
MWTR	Containment sump liquid mass	lbm
PAIR	Containment air partial pressure	psia
PSTM	Containment steam partial pressure	psia
PWTR	Containment total pressure	psia
QSPRYY	Containment condensation heat rate	Btu/sec

Table A-18

File cml.t26.jobid, COMPERC-II Spray and Spillage Variables

Variable Name	Description	Units
TIME	Time measured from time zero	sec
QSPRAY(1)	Containment spray flow (automatic spray and spillage) or spray flow table 1	lbm/sec
QSPRAY(2)	Spillage spray flow (automatic spray and spillage) or spray flow table 2	lbm/sec
QSPRAY(3)	Broke loop SIT spray flow (automatic spray and spillage) or spray flow table 3	lbm/sec
QSPRAY(4)	zero (automatic spray and spillage) or spray flow table 4	lbm/sec
HHSPRAY(1)	Enthalpy for QSPRAY(1)	Btu/lbm
HHSPRAY(2)	Enthalpy for QSPRAY(2)	Btu/lbm
HHSPRAY(3)	Enthalpy for QSPRAY(3)	Btu/lbm
HHSPRAY(4)	Enthalpy for QSPRAY(4)	Btu/lbm
HMAXSP	Saturation liquid enthalpy at steam partial pressure	Btu/lbm
SPRFLO	Total spray flow	lbm/sec
QSPRYY	Containment condensation heat rate	Btu/sec

Table A-19

File cml.t27.jobid, COMPERC-II New Nitrogen Injection Model Variables[1]

Variable Name	Description	Units
TIME	Time measured from time of TAD	sec
BLK	K-Factor	dimensionless
DP_TOT	Injection section delta pressure (total)	psid
DP_MOM	Injection section delta pressure (momentum)	psid
SV_MIX	Mixture specific volume (liquid, steam, nitrogen)	ft^3/lbm
DM_NIT	Delta momentum, nitrogen phase	lbm-ft/sec
VE_NITP	Nitrogen, SI pump velocity	ft/sec
VE_NIT	Nitrogen, SI pump velocity (projected)	ft/sec
VE_STM	Steam velocity (upstream SI line)	ft/sec
VE_MIX	Mixture velocity (liquid, steam, nitrogen)	ft/sec
WSLTG	Steam flow (one cold leg)	lbm/sec
WN2	Nitrogen mass flow rate (one tank)	lbm/sec
WPUMP1	SI pump mass flow rate (one pump)	lbm/sec
VSPNOZ	Nitrogen specific volume at the nozzle	ft^3/lbm
VSPSH	Steam flow specific volume	ft^3/lbm

1 Variables written during the period of nitrogen injection.

A.6 Description of the Impact of Interface Variables on Existing Base Decks

The automated/integrated code system functions without the need for base deck modifications because a hierarchy was established for the impact of the interface variables on the computer code base decks. The purpose of this section is to document the relationship between the new interface variables and the base deck variables in the downstream computer codes. These new interface variables exist from the two new input files (the UCI file and the f4acml.inp.mod1 file) and from the numerous linkages created for the automated/integrated code system, see Figure 2.1-2. The relationships between the new variables and the base deck variables also depend on the input option selected. Therefore, the following three tables describe the impact of the relationship and the hierarchy of the new interface variables to the base deck variables for each of the input options and for each of the computer codes.

- Table A-20 describes the impact of the UCI input variables and their options on all downstream computer code input variables (see Table A-1 for the description of the UCI input file).
- Table A-21 describes the impact of the f4acml.inp.mod1 input variables on the downstream computer codes (see Table A-2 for a description of the f4acml.inp.mod1 input file).
- Table A-22 describes the impact of all of the interface variables created by an upstream computer code on the base deck variables for the downstream computer code. The interface files are described in Section A.4 in Tables A-3 through A-11.

Table A-20
User Control Interface (UCI) Input Variables Impact on Computer Code Base Decks

New Variable Line ID	Downstream Code	Input Option(s)	Base Deck Variable ID	Impact on Base Deck Variable	Description
[					

New Variable Line ID	Downstream Code	Input Option(s)	Base Deck Variable ID	Impact on Base Deck Variable	Description

]

Table A-21
New Input File (f4acml.inp.mod1.jobid) Variables Impact on Computer Code Base Decks

[

New Variable Line ID	Downstream Code	Base Deck Variable ID	Impact on Base Deck Variable	Description

]

Table A-22
Interface File Variables Impact on Computer Code Base Decks

Interface File Name	Upstream Code	Interface File Line ID	Downstream Code	Base Deck Variable ID	Impact on Base Deck Variable	Description

Interface File Name	Upstream Code	Interface File Line ID	Downstream Code	Base Deck Variable ID	Impact on Base Deck Variable	Description

Interface File Name	Upstream Code	Interface File Line ID	Downstream Code	Base Deck Variable ID	Impact on Base Deck Variable	Description

Interface File Name	Upstream Code	Interface File Line ID	Downstream Code	Base Deck Variable ID	Impact on Base Deck Variable	Description

A.7 References

A-1 CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

CENPD-133, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

A-2 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

A-3 CENPD-135P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.

CENPD-135P, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)," February 1975.

CENPD-135, Supplement 4-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.

CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

A-4 CENPD-138P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.

CENPD-138P, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications)," February 1975.

CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

A-5 Enclosure 1-P-A to LD-81-095, "C-E ECCS Evaluation Model Flow Blockage Analysis," December 1981.

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Appendix B

NRC ACCEPTANCE REVIEW

Reference: J. Cushing (NRC) to I. C. Rickard (ABB CE), "Acceptance of CENPD-132, Supplement 4, 'Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model,' for Review," October 4, 1999

Appendix B contains the results of the NRC staff's acceptance review for this topical report submittal for the 1999 EM and the scheduled completion date.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 4, 1999

Mr. Ian C. Rickard
Director, Nuclear Licensing
ABB Combustion Engineering Nuclear Power
P.O. Box 500
2000 Day Hill Road
Windsor, CT 06095-0500

SUBJECT: ACCEPTANCE OF CENPD-132, SUPPLEMENT 4, "CALCULATIVE METHODS FOR THE ABB CENP LARGE BREAK LOCA EVALUATION MODEL," FOR REVIEW

Dear Mr. Rickard:

Pursuant to the NRC staff's policy, an acceptance review of the material provided in your April 30, 1999, letter has been performed. We have found that the material presented is complete enough to begin a review. The scheduled completion date for this review is December 31, 2000. We will forward any requests for additional information as the review progresses.

Sincerely,

J Cushing

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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Appendix C

NRC REQUEST FOR ADDITIONAL INFORMATION

Reference: J. Cushing (NRC) to I. C. Rickard (ABB CE), "Request for Additional Information (RAI) Regarding CENPD-132-P, Supplement 4-P (TAC NO. MA5660)," December 14, 1999

Appendix C contains the NRC staff's request for additional information regarding the licensing review of this topical report submittal for the 1999 EM.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 14, 1999

Mr. Ian C. Rickard, Director
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2000 Day Hill Road
Windsor, Connecticut 06095-0500

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING CENPD-132-P, SUPPLEMENT 4-P (TAC NO. MA5660)

Dear Mr. Rickard:

Topical Report CENPD-132-P, Supplement 4-P, "Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model," was submitted for staff review by ABB Combustion Engineering Nuclear Power Company letter dated April 30, 1999. As a result of the review, the staff has determined that additional information is needed to complete the review. The information needed is detailed in the enclosure.

The enclosed request was discussed with Mr. Jagelar of your staff on December 6, 1999. A mutually agreeable target date of March 3, 2000, was established for responding to the request for additional information. If circumstances result in the need to revise the target date, please call me at your earliest opportunity at (301) 415-1424.

Sincerely,

J Cushing

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 692

Enclosure: Request for Additional Information

cc w/encl: Mr. Charles B. Brinkman, Manager
Washington Operations
ABB Combustion Engineering Nuclear Power
12300 Twinbrook Parkway, Suite 330
Rockville, MD 20852

REQUEST FOR ADDITIONAL INFORMATION

ON TOPICAL REPORT CENPD-132-P, SUPPLEMENT 4-P

"CALCULATIVE METHODS FOR THE ABB CENP LARGE BREAK

LOCA EVALUATION MODEL"

1. Section 2.4.1.2 of the topical report described two cases for the calculation of the steam generator secondary side pressure in the COMPERC-II code for the revised steam venting reflood thermal-hydraulics model of the 1999 evaluation model (EM). Describe selection criteria or logic between the two cases in the COMPERC-II code.

2. In the 1999 EM revised steam venting reflood thermal-hydraulics calculation, the steam generator (SG) secondary side heat transfer coefficient is calculated with Equation 2.4.1.3-1. The basic component of Equation 2.4.1.3-1, which was derived from a correlation for natural convection for vertical plates, appears to omit the definition of the characteristic length (L). Also, the FLECHT-SEASET report NUREG/CR-1534 indicated that among many natural convection correlations examined, the Eckert-Jackson correlation gives the most consistent results in the evaluation of the FLECHT-SEASET data.

   (a) Confirm that equation 2.4.1.3-1 is correct, or make correction if necessary.

   (b) Explain the merit of using Equation 2.4.1.3-1 instead of the Eckert-Jackson correlation for the SG secondary side heat transfer calculation.

3. In the assessment of the effect of steam generator inlet quality on LBLOCA analysis, Section 2.4.2.3.3 of the topical report indicated a specific value of the liquid entrainment fraction for the 1999 EM COMPERC-II entrainment model. Provide the origin and the basis of this value of the entrainment fraction in the 1999 EM COMPERC-II.

4. For the evaluation of the effects of de-entrainment in the upper plenum and hot legs on the reflood rate and peak cladding temperature (PCT), Section 2.4.2.3.4 of the report described the implementation of a special model in COMPERC-II in which the downcomer and lower plenum mass and energy equations are modified with the de-entrainment flow in the upper plenum and hot legs. Explain why the de-entrainment liquid is not added to the upper plenum, but to the downcomer and lower plenum; and discuss the effect of this modeling (adding the de-entrainment liquid to the downcomer) on the reflood rate and PCT.

Enclosure

Appendix D

ABB CENP RESPONSE TO THE NRC REQUEST FOR ADDITIONAL INFORMATION

Reference: LD-2000-0011, I. C. Rickard (ABB CE) to U. S. Nuclear Regulatory Commission (Document Control Desk), "ABB CENP Response to NRC Request for Additional Information Regarding CENPD-132-P, Supplement 4-P," February 22, 2000

Appendix D contains the ABB CENP response to the NRC staff's request for additional information regarding the licensing review of this topical report submittal for the 1999 EM.

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ASEA BROWN BOVERI

22 February, 2000
LD-2000-0011

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

SUBJECT: ABB CENP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING CENPD-132-P, SUPPLEMENT 4-P
{ENCLOSURE 1-P CONTAINS PROPRIETARY INFORMATION}

References: 1) Letter, I. C. Rickard (ABB CENP) to USNRC Document Control Desk, "Revision to ABB CENP ECCS Performance Appendix K Evaluation Model", LD-99-026, April 30, 1999

2) Letter, J. S. Cushing (NRC) to I. C. Rickard (ABB CENP), "Request for Additional Information (RAI) Regarding CENPD-132-P, Supplement 4-P (TAC No. MA5660)", December 14, 1999

By letter dated April 30, 1999 (Reference 1), ABB C-E Nuclear Power, Inc. (ABB CENP) submitted and requested Nuclear Regulatory Commission (NRC) review and approval of CENPD-132-P, Supplement 4-P – "Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model". On December 14, 1999 (Reference 2), the NRC issued a Request for Additional Information (RAI) necessary for completion of their CENPD-132-P, Supplement 4-P review effort. Enclosure 1-P to this letter (PROPRIETARY) provides ABB CENP's response to the NRC RAIs. These responses will be incorporated into the Topical Report as Appendix D, and therefore, the pages are so numbered.

ABB CENP has determined that the material provided in Enclosure 1-P is PROPRIETARY in nature. Consequently, it is requested that Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that these copies be appropriately safeguarded. The reasons for the classification of this information as PROPRIETARY are delineated in the affidavit provided in Enclosure 2.

ABB C-E Nuclear Power, Inc.

P.O. Box 500
2000 Day Hill Rd.
Windsor, CT 06095-0500

Telephone (860) 285-9678
Fax (860) 285-3253

If you have any questions concerning this matter, please do not hesitate to call me or Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
ABB CE NUCLEAR POWER, INC.

Ian C. Rickard, Director
Nuclear Licensing

Enclosures: As stated

xc: w/o Enclosures
J. S. Cushing (NRC/NRR/DLPM/LPD4)
Y.H. Hsii (NRC/NRR/DSSA/SRXB)

ABB Combustion Engineering Nuclear Power, Inc.

ABB CENP Response to NRC Request for Additional Information Regarding CENPD-132-P, Supplement 4-P

ABB CENP Response to NRC Request for Additional Information (RAI) Regarding CENPD-132-P, Supplement 4-P

(TAC NO. MA5660)

Introduction

ABB C-E Nuclear Power (ABB CENP) submitted topical report CENPD-132, Supplement 4-P (1999 EM) to the Nuclear Regulatory Commission (NRC) in April 1999, Reference D-1. This topical report submittal describes modifications to the Emergency Core Cooling System (ECCS) Evaluation Model (EM) that is used for the analysis of the large break loss-of-coolant accident (LBLOCA). The modifications include implementation of process changes within the currently NRC-accepted evaluation model (i.e., the 1985 EM), the replacement of the Dougall-Rohsenow film boiling correlation, as well as improved models designed to reduce conservatism. Model improvements are made in the areas of (1) cladding swelling and rupture, (2) steam venting reflood thermal hydraulics, (3) steam/water interaction during nitrogen discharge from the safety injection tanks, (4) reflood heat transfer, and (5) hot rod heat transfer to steam. The 1999 EM submittal presents sensitivity studies and comparisons with experimental data along with the results of a break spectrum analysis for a typical ABB CENP designed Pressurized Water Reactor (PWR).

The NRC notified ABB CENP of the Acceptance for Review of the topical report supplement on October 4, 1999, Reference D-2. As a result of the NRC review, the staff determined that additional information is needed to complete the review. The NRC issued this Request for Additional Information (RAI) on December 14, 1999, Reference D-3. The RAI identified four (4) items requiring additional information. The following material is provided in response to the RAI. The Acceptance for Review notification, the RAI, and the ABB CENP response to the RAI contained herein will be inserted into the topical report as Appendices B, C, and D, respectively.

Note that the RAI is repeated in the following material and is denoted by a **bold** font. An *Italics* font denotes the ABB CENP responses.

References

D-1 LD-99-026, "Revision to ABB CENP ECCS Performance Appendix K Evaluation Model," letter from I. C. Rickard (ABB CENP) to U. S. Nuclear Regulatory Commission Document Control Desk, April 30, 1999.

D-2 Letter from J. Cushing (NRC) to I. C. Rickard (ABB CENP), "Acceptance of CENPD-132, Supplement 4, 'Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model,' for Review," October 4, 1999.

D-3 Letter from J. Cushing (NRC) to I. C. Rickard (ABB CENP), "Request for Additional Information (RAI) Regarding CENPD-132-P, Supplement 4-P (TAC NO. MA5660)," December 14, 1999.

RAI 1. Section 2.4.1.2 of the topical report described two cases for the calculation of the secondary side pressure in the COMPERC-II code for the revised steam venting reflood thermal-hydraulics model in the 1999 evaluation model (EM). Describe selection criteria or logic between the two cases in the COMPERC-II Code.

<u>ABB CENP Response:</u>

This response describes the selection criteria and logic for the secondary side pressure model. Section 2.4.1.2 of the topical report will be revised to include this information.

The 1999 EM version of the COMPERC-II code dynamically determines the thermal state of the steam generator (SG) secondary side during the transient. As described in Section 2.4.1.2, a single control volume is used to represent the liquid and steam phases in the steam generators. The liquid and steam are assumed to be fully separated (liquid at the bottom, steam at the top) in thermal equilibrium or non-equilibrium. The two cases described in Section 2.4.1.2 are used to determine if the SG secondary side is in a state of thermal equilibrium or non-equilibrium. Upon initialization at the Time of Annulus Downflow (TAD), COMPERC-II assumes that the secondary side is in thermal equilibrium at the saturation temperature at TAD. The initial saturation conditions are calculated by the code from the SG secondary side pressure at TAD.

For each time step during the transient, COMPERC-II integrates the mass and energy conservation equations for the control volume, and calculates the secondary side thermal conditions (pressure and thermal state of the liquid and steam phases) assuming the thermal non-equilibrium conditions described in Case 1 in Section 2.4.1.2 (iv). If a consistent solution is found, i.e., the iterative process converges and calculates [ ] then the solution is accepted. If the iterative process converges [ ] then the Case 1 calculations are rejected, [ ] are assumed to occur in the secondary side, and the pressure is recalculated from the mass and energy using the method described in Case 2 in Section 2.4.1.2 (iv).

If the iterative process fails to converge for Cases 1 or 2, then the COMPERC-II execution is terminated with an error message.

RAI 2. In the 1999 EM revised steam venting reflood thermal-hydraulics calculation, the steam generator (SG) secondary side heat transfer coefficient is calculated with Equation 2.4.1.3-1. The basic component of Equation 2.4.1.3-1, which was derived from a correlation for natural convection for vertical plates, appears to omit the definition of the characteristic length (L). Also, the FLECHT-SEASET report NUREG/CR-1534 indicated that among many natural convection correlations examined, the Eckert-Jackson correlation gives the most consistent results in the evaluation of the FLECHT-SEASET data.

(a) Confirm that equation 2.4.1.3-1 is correct, or make correction if necessary.

(b) Explain the merit of using Equation 2.4.1.3-1 instead of the Eckert-Jackson correlation for the SG secondary side heat transfer calculation.

<u>*ABB CENP Response:*</u>

(a) *Equation (2.4.1.3-1) was mistyped and, therefore, will be revised in the topical report. The COMPERC-II coding is not affected. The definition of the characteristic length will be inserted in the definition of terms.*

The correct entry of Equation (2.4.1.3-1) is as follows:

[ ] *(2.4.1.3-1)*

The value of the characteristic length (L) used in Equation (2.4.1.3-1) is [

.]

(b) *Before proceeding with the evaluation of the Eckert-Jackson correlation it is important to note that the Eckert-Jackson correlation given in NUREG/CR-1534 (Reference D.2-1, Page 6-5) is typed incorrectly. (ABB CENP recommends that NRC advise anyone that may be using Reference D.2-1 for this correlation of a potential problem). For this RAI response, the Eckert-Jackson correlation is taken from source documents and is as follows (written as a heat transfer coefficient expression instead of in the Nusselt form, Reference D.2-2, Page 6):*

$$h_{sec} = 0.021(k_L/L)\,(Gr_L\,Pr_L)^{0.4}$$

where

h_{sec} = *Secondary side heat transfer coefficient (Btu/hr ft^2 °F)*

k_L = *Liquid conductivity (Btu/hr ft °F)*

Gr_L = *Grashof number evaluated at the film temperature*

$= \rho^2 \beta g L^3 \Delta T / \mu^2$

L = *Characteristic length (ft)*

Pr_L = *Prandtl number*

ρ = *Density* (lbm/ft^3)

β = *Thermal expansion coefficient* ($1/°F$)

g = 32.17 (ft/sec^2)

ΔT = $T_{tube} - T_{sec}$ ($°F$)

μ = *Viscosity (lbm/sec ft)*

D = *Hydraulic diameter (riser region, ft)*

This part of the RAI response will explain the merit of using Equation (2.4.1.3-1) instead of the Eckert-Jackson correlation in the 1999 EM.

Equation (2.4.1.3-1) is the model used in several instances for modeling free convection in the ABB CENP large break and small break evaluation models. It was used in the 1999 EM for consistency and convenience. Its choice for the 1999 EM was justified on Page 2.4-35 in Section 2.4.4.iii.a.

In order to evaluate the merit of using Equation (2.4.1.3-1) vs. the Eckert-Jackson correlation, the correlations were compared over a wide range of conditions that included approximately 900 points, with pressure ranging from 500 to 1000 psia, liquid temperatures from saturation to 100 °F subcooling and wall-to-liquid delta temperatures ranging from 0 °F to 50 °F. The results of this comparison are shown in Figure D.2-1. The Eckert-Jackson correlation was derived from heat transfer data that was in good agreement with experimental data in the range of Grashof numbers from 10^{10} *to* 10^{12}*. The formula may be used for higher Grashof numbers (Reference D.2-2). Grashof numbers used for the above comparison range from* 10^{12} *to* 10^{14}*.*

The Eckert-Jackson correlation and Equation (2.4.1.3-1) are of [

]

The COMPERC-II FLECHT-SEASET simulation described in Sections 2.4.2.2 and 2.4.2.3.2 used Equation (2.4.1.3-1) to calculate the secondary side free-convection heat transfer coefficients. The COMPERC-II results for FLECHT-SEASET Test 22920 (Section 2.4.2.2 (iv)) demonstrate the adequacy of Equation (2.4.1.3-1) used for the secondary side. This test is a pure steam test at the SG tube inlet and shows that the COMPERC-II SG model realistically calculates the results of the test including the secondary side temperature distribution and heat transfer (see Figures 2.4.2.2-11 and 2.4.2.2-12).

(The COMPERC-II simulation of the other FLECHT-SEASET tests (Sections 2.4.2.2 (ii) and (iii)) are [

]

Implementation of the Eckert-Jackson correlation in a special version of the COMPERC-II code and simulation of the FLECHT-SEASET Test 22920 yielded the following results:

- *The results of Figures 2.4.2.2-9, 2.4.2.2-10, 2.4.2.2-11, and 2.4.2.2-12 are essentially unchanged when calculated with the Eckert-Jackson correlation (that is, there are no visually observable differences).*
- *Figure D.2-2 shows a comparison of the secondary side heat transfer coefficients calculated by the Eckert-Jackson correlation and Equation (2.4.1.3-1) axially along the tube at 1500 seconds in the test. This figure shows that Eckert-Jackson [ ] Equation (2.4.1.3-1) along the tube with the exception of a small upper region of the tube with small wall-to-coolant delta temperatures.*
- *The simulation using Equation (2.4.1.3-1) calculates [ ] secondary side SG tube exit temperatures than the simulation using Eckert-Jackson. For example, at 1500 seconds in Figure 2.4.2.2-12 (the time where the difference is the largest), the simulation with Eckert-Jackson calculates a [ ] in primary side SG fluid exit temperature compared to the base case with Equation (2.4.1.3-1). Since the resistance to steam venting is determined by the SG tube exit temperature, Equation (2.4.1.3-1) is [ ] than Eckert-Jackson.*

Evaluation of the impact of using the Eckert-Jackson correlation vs. Equation (2.4.1.3-1) was tested on a limiting break size/location PWR test case using the special version of COMPERC-II. The result was a [ ] when using Eckert-Jackson. The [

]

Therefore the following conclusions are stated:

- *Equation (2.4.1.3-1) calculates heat transfer coefficients which are consistent with those calculated by Eckert-Jackson.*
- *Use of Equation (2.4.1.3-1) or the Eckert-Jackson correlation produces comparable results for COMPERC-II FLECHT-SEASET simulations.*
- *Use of Equation (2.4.1.3-1) or Eckert-Jackson produces comparable PCTs that [ ] for PWR limiting case LOCA calculations.*

References:

D.2-1 *NUREG/CR-1534, "PWR FLECHT SEASET Steam generator Separate-Effects Task, Data Analysis and Evaluation Report, February, 1982.*

D.2-2 *Eckert, E. R. G., and Jackson, T. W., "Analysis of Turbulent Free-Convection Boundary Layer on Flat Plate", NACA-1015, 1951.*

Figure D.2-1

COMPARISON OF Eq (2.4.1.3-1) AND ECKERT-JACKSON

FREE CONVECTION

HEAT TRANSFER COEFFICIENTS

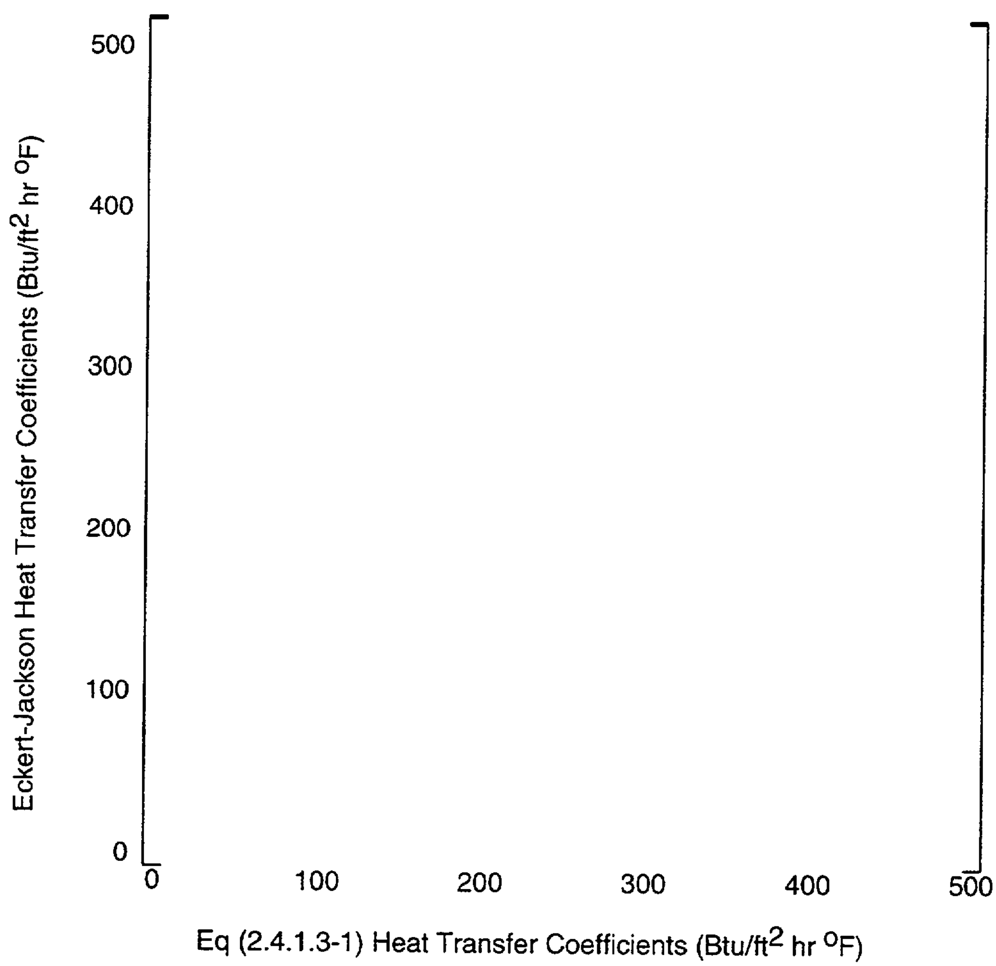
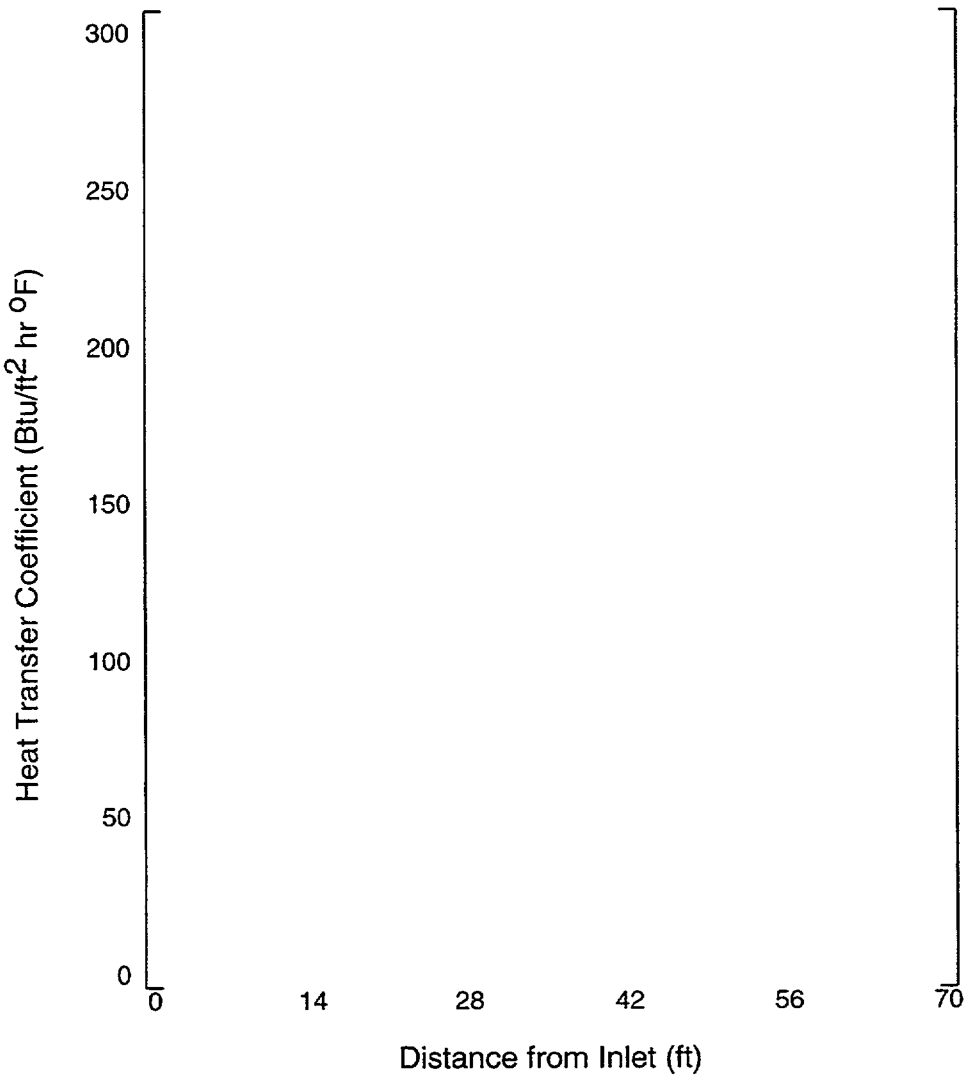


Figure D.2-2

FLECHT-SEASET Test 22920

Secondary Side SG Tube Axial H. T. Coeff

Comparison of Eq (2.4.1.3-1) and Eckert-Jackson H. T. Coeff.



RAI 3. In the assessment of the effect of steam generator inlet quality on LBLOCA analysis, Section 2.4.2.3.3 of the topical report indicated a specific value of the liquid entrainment fraction for the 1999 EM COMPERC-II entrainment model. Provide the origin and the basis of this value of the entrainment fraction in the 1999 EM COMPERC-II.

ABB CENP Response:

The origin and the basis of the value of the entrainment fraction in the 1999 EM COMPERC-II methodology described in Section 2.4.2.3.3 is unchanged from the currently approved methodology documented in Reference D.3-1 and accepted for use in the SER, Reference D.3-2. This currently approved methodology remains acceptable for the 1999 EM for three reasons:

(1) *None of the methodology improvements for the 1999 EM specifically relate to the COMPERC-II core flow hydraulics modeling elements that include the liquid entrainment fraction.*
(2) *The PWR performance of the 1999 EM remains consistent with the variation of key parameters (nominal containment pressure, reflood rate, initial peak cladding temperature, and inlet water subcooling) represented by the original COMPERC-II core flow hydraulics model justification.*
(3) *The currently approved core flow hydraulics methodology (even in combination with the methodology improvements of the 1999 EM) remains a significant source of conservatism for PWR calculations compared to more realistic analysis approaches.*

Each of these reasons is discussed in the following review of the material from Reference D.3-1.

In accordance with 10 CFR 50, Appendix K (Reference D.3-3), the ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction or entrainment fraction) is used to determine the core exit flow and is determined in accordance with applicable experimental data (for example, from the PWR FLECHT test program).

It is shown in Reference D.3-1 that proper selection of the void fraction and entrainment fraction in the core region above the mixture level results in a COMPERC-II prediction of the core reflood rate that is conservative relative to the FLECHT runs. In the COMPERC-II model for a PWR, the void fraction and entrainment fraction are proprietary constants input by the user. The selection of these constants is described and justified in Reference D.3-1 using comparisons to FLECHT runs. These COMPERC-II comparisons of the FLECHT runs used measured test boundary conditions at the inlet of the core, core ΔP data, and the location of the quench front in the core for the test simulation. These comparisons did not utilize the COMPERC-II steam venting reflood thermal-hydraulics methodology. Therefore, the 1999 EM improvements for steam venting described in Section 2.4 have no impact on the basis for the selection or justification of the void fraction and entrainment fraction in the core.

Furthermore, the FLECHT runs contained variations in pressure (19-61 psia), reflood rate (1.0-2.0 in/sec), initial peak cladding temperature (1200 °F-2145 °F), inlet water subcooling (22 °F-148 °F), and powers like those of hot rod assemblies that are representative of conditions expected during the reflood phase of a LBLOCA. The PWR performance of the reflood portion of the 1999 EM is consistent with the performance of the currently approved methodology with tendencies to

produce slightly higher (improved) reflood rates for the same boundary conditions. Therefore, the parameter variations used in the original justification of the liquid entrainment model remain applicable for the 1999 EM.

Finally, this core reflood hydraulics aspect of the PWR performance of the 1999 EM is very conservative relative to more realistic modeling, as would be expected of a 10 CFR 50, Appendix K methodology. This was demonstrated in Sections 2.4.2.3.3 through 2.4.2.3.5, where more realistic modeling was used to show the effects of SG[ ] compared to 10 CFR 50, Appendix K modeling. TRAC-P calculations cited in Section 2.4.2.3.4 establish 300°F conservatism in peak cladding temperature between Appendix K modeling and best estimate modeling. Analyses presented for the 1999 EM further establish a comparable degree of conservatism in the new COMPERC-II methodology.

References:

D.3-1 CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974, (Section III.D.4.c).

D.3-2 Letter from Olan D. Parr (NRC) to F. M. Stern (C-E), June 13, 1975; Enclosure: "Status Report by the Directorate of Licensing in the Matter of Combustion Engineering, Inc. ECCS Evaluation Model Conformance to 10CFR50 Appendix K," pages 4-57, 4-58, 4-61(J.a), 4-62, 4-63, and 4-65.

D.3-3 Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

Code of Federal Regulations, Title 10, Part 50, Appendix K, "ECCS Evaluation Models."

RAI 4. For the evaluation of the effects of de-entrainment in the upper plenum and hot legs on the reflood rate and peak cladding temperature (PCT), Section 2.4.2.3.4 of the report described the implementation of a special model in COMPERC-II in which the downcomer and lower plenum mass and energy equations are modified with the de-entrainment flow in the upper plenum and hot legs. Explain why the de-entrainment liquid is not added to the upper plenum, but to the downcomer and lower plenum; and discuss the effect of this modeling (adding the de-entrainment liquid to the downcomer) on the reflood rate and PCT.

ABB CENP Response:

The de-entrained liquid was added to the lower plenum and not to the upper plenum and hot legs in the special version of COMPERC-II used for the de-entrainment studies (quantification of the margin of conservatism), because this implementation provided a reasonable simulation of the effect of de-entrainment on the reflood phenomena while maintaining consistency with the COMPERC-II formulation. That is, the implementation of the de-entrainment of liquid model was made consistent with the following two main constraints: (i) COMPERC-II conserves liquid mass only in the downcomer/lower plenum and core nodes, (ii) Any liquid that is added to the core in COMPERC-II is by reflood from the bottom of the core.

Since the core reflood rate in COMPERC-II is calculated by solving a manometer type equation between the downcomer and the core and upper plenum, (Reference D.4-1, Section III.A), adding the de-entrained liquid to the lower plenum, lets the manometer equation calculate the impact of de-entrainment on the rate of liquid addition to the core. Note that addition of the de-entrained liquid to the lower plenum does not necessarily mean that the liquid is added to the core. That is, after the initial refill period of the downcomer, the downcomer is usually full to the level of the break, and thus, any addition of the liquid to the lower plenum effectively represents an equivalent increase of the spillage rate out the break into the containment.

The most important parameter affecting the core reflood rate is the resistance to steam venting around the loops. De-entrainment of the liquid in the upper plenum and hot legs reduces the steam flow around the loops, thus reducing the resistance to steam venting, and therefore results in an increased reflood rate. As demonstrated by the results in Section 2.4.2.3.4, this response calculated by the special version of COMPERC-II is consistent with observations of the 2D/3D Test Program and the best estimate calculations with TRAC-P (Reference D.4-2, Section 4.8).

In order to evaluate the effect of leaving the de-entrained liquid in the upper plenum and hot legs instead of adding it to the lower plenum, a separate special version of COMPERC-II was created where all the de-entrained liquid was kept in the upper plenum and hot legs. For this version, the upper plenum pressure calculations were not modified to account for the redistribution of system mass in the upper plenum and hot legs. Also, thermal-hydraulic interactions between the liquid in the upper plenum and the two-phase mixture exiting the core were not addressed. The results of this analysis[

]

Thus, the results showing the effect of the implementation of the de-entrainment model on PCT given in Section 2.4.2.3.4 (iii) and in Figure 2.4.2.3.4-2, are not significantly impacted by the approach of adding the de-entrained liquid to the lower plenum in the special version of COMPERC-II.

References:

D.4-1 CENPD-134P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974.

CENPD-134P, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," February 1975.

CENPD-134, Supplement 2-A, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," June 1985.

D.4-2 "Reactor Safety Issues Resolved by the 2D/3D Program," NUREG/IA-0127, GRS-101, MPR-1346, July 1993.

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Appendix E

TRANSMITTAL TO NRC OF ADDITIONAL SUPPORTING INFORMATION

Reference: LD-1999-0064, I. C. Rickard (ABB CE) to U. S. Nuclear Regulatory Commission (Attn: Yi-Hsiung Hsii), "COMPERC-II Topical Report Set – Information Copy," December 9, 1999

Appendix E contains the transmittal of a set of the COMPERC-II Computer Code Topical Reports for use by the NRC staff during the review of this topical report submittal for the 1999 EM.

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ASEA BROWN BOVERI

9 December, 1999
LD-1999-0064

U. S. Nuclear Regulatory Commission
Attn: Yi-Hsiung Hsii (10 B3)
Washington, D.C. 20555

SUBJECT: COMPERC-II TOPICAL REPORT SET - INFORMATION COPY

References:
1) CENPD-134P, "COMPERC-II – A Program for Emergency – Refill – Reflood of the Core", August 1974 {PROPRIETARY COPY NO. 000143}
2) CENPD-134P, Supplement 1, "COMPERC-II – A Program for Emergency Refill – Reflood of the Core (Modifications)", February 1975 {PROPRIETARY COPY NO. 00091}
3) CENPD-134, Supplement 2, "COMPERC-II – A Program for Emergency Refill – Reflood of the Core", June 1985

In accordance with your November 1999 verbal request to Ernie Jageler of ABB Combustion Engineering Nuclear Power's (ABB CENP) staff, enclosed herewith is a copy of the COMPERC-II topical report set (References 1 through 3). Although this information was previously provided to the Nuclear Regulatory Commission (NRC), this set is being provided in support of your review of ABB CENP's 1999 Large Break Loss-of-Coolant Accident (LBLOCA) Evaluation Model review activities.

ABB CENP has determined that the information contained in References 1 and 2 is PROPRIETARY in nature. Consequently, it is requested that these documents be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that these documents be appropriately safeguarded. The reasons for the classification of this information as proprietary were delineated in the affidavits provided at the time of their original submittal to the NRC.

If you have any questions regarding this matter, please do not hesitate to call me or Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
ABB COMBUSTION ENGINEERING NUCLEAR POWER, INC.

for Ian C. Rickard, Director
Nuclear Licensing

Enclosure: As stated

ABB Combustion Engineering Nuclear Power, Inc.

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Appendix F

CENP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION DURING MEETING ON NOVEMBER 3, 2000

Reference: LD-2000-0057, P. W. Richardson (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Response to Questions Regarding CENPD-132, Supplement 4-P, Rev. 1," Enclosure 3, "CENP Non-Proprietary Response to NRC Questions Regarding CENPD-132, Supplement 4-P, Revision 1, Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," November 13, 2000

Appendix F contains the CENP response to the NRC staff's request for additional information regarding the licensing review of this topical report submittal for the 1999 EM, which occurred during the meeting with the staff on November 3, 2000.

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Westinghouse Electric Company
CE Nuclear Power LLC

2000 Day Hill Road
Windsor, CT 06095
USA

13 November, 2000
LD-2000-0057

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

SUBJECT: RESPONSE TO QUESTIONS REGARDING CENPD-132, SUPPLEMENT 4-P, REV. 1
{Enclosure 1-P Contains Proprietary Information}

Reference(s): 1) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", August 2000
2) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model", LD-2000-0046, August 30, 2000

Representatives from CE Nuclear Power LLC (CENP) and the Nuclear Regulatory Commission (NRC) participated in a meeting on Friday, November 3, 2000 to discuss the ongoing review of CENPD-132, Supplement 4-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (Reference 1). This topical report was submitted to the NRC on August 30, 2000 (Reference 2). A number of NRC questions arose during the meeting to which CENP responses are provided herein. Pursuant to prior agreement, CENP is furnishing one (1) proprietary and one (1) non-proprietary copy of this letter and enclosures to the NRC Document Control Desk and three (3) proprietary copies to Jack Cushing, NRC, CENP Project Manager.

As mentioned above, following submittal of Revision 1 to the Topical Report, a meeting was held on Nov. 3, 2000 for the purpose of addressing NRC questions on the revision. After detailed discussions that covered the entire topical report, seven items remained that required closure. Five of these items were CENP actions and two were NRC. It was also agreed that a conference call would be held to address the final resolution of these items. The phone call was held on Nov. 8, 2000. CENP and NRC verbally described responses to each of the seven items. The material contained in Enclosure 1-P provides the official CENP response to the five action items it was assigned. Also, the NRC's responses to its two action items are summarized along with the required CENP follow-up actions. The non-proprietary responses are provided in Enclosure 3.

CENP has determined that the information provided in Enclosure 1-P is proprietary in nature. Consequently, it is requested that Enclosure 1-P be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that this information be appropriately safeguarded. The reasons for the classification of this information as proprietary are delineated in the affidavit provided in Enclosure 2.

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
CE NUCLEAR POWER LLC

Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Enclosure(s): As stated

xc: J. S. Cushing (NRC, with 3 copies of Enclosure 1-P)

CE NUCLEAR POWER LLC

NON-PROPRIETARY RESPONSE TO NRC QUESTIONS REGARDING CENPD-132, SUPPLEMENT 4-P, REVISION 1 CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

November 2000

CENP Response to NRC Request for Additional Information (RAI) During Meeting on November 3, 2000 Regarding CENPD-132-P, Supplement 4-P, Revision 1

(TAC NO. MA5660)

Introduction

CE Nuclear Power LLC (CENP) submitted topical report CENPD-132, Supplement 4-P (1999 EM) to the Nuclear Regulatory Commission (NRC) in April 1999, Reference 1. This topical report submittal describes modifications to the Emergency Core Cooling System (ECCS) Evaluation Model (EM) that is used for the analysis of the large break loss-of-coolant accident (LBLOCA). The modifications include implementation of process changes within the currently NRC-accepted evaluation model (i.e., the 1985 EM), the replacement of the Dougall-Rohsenow film boiling correlation, as well as improved models designed to reduce conservatism. Model improvements are made in the areas of (1) cladding swelling and rupture, (2) steam venting reflood thermal hydraulics, (3) steam/water interaction during nitrogen discharge from the safety injection tanks, (4) reflood heat transfer, and (5) hot rod heat transfer to steam. The 1999 EM submittal presents sensitivity studies and comparisons with experimental data along with the results of a break spectrum analysis for a typical CENP designed Pressurized Water Reactor (PWR).

The NRC notified CENP of the Acceptance for Review of the topical report supplement on October 4, 1999, Reference 2. As a result of the NRC review, the staff determined that additional information is needed to complete the review. The NRC issued a Request for Additional Information (RAI) on December 14, 1999, Reference 3. The RAI identified four (4) items requiring additional information. CENP provided a response to the RAI on February 22, 2000, Reference 4.

After the topical report was originally issued, Westinghouse Electric Corporation acquired ABB Combustion Engineering Nuclear Power and the company name was changed to CE Nuclear Power LLC. Also, during the licensing review process, the NRC suggested that a number of modifications and additions to the topical report should be made to both correct and clarify the technical documentation. Also, with training and usage, the Automated/Integrated Code System was enhanced, which lead to additional documentation changes to incorporate the latest capabilities and user-guidance material. Therefore, prior to completion of the licensing process, Revision 1 of the topical report was prepared and submitted to NRC, Reference 5, for the following reasons:

- Change the company name
- Incorporate the NRC RAI and the responses to the RAI
- Incorporate corrections and clarifications identified by NRC during the licensing review process
- Provide updated user guidance material associated with the content and usage of the Automated/Integrated Code System.

The Acceptance for Review notification, the RAI, and the CENP response to the RAI were inserted into the topical report revision as Appendices B, C, and D, respectively. Also, NRC requested additional supporting information related to the COMPERC-II computer code. The record of that transmittal, Reference 6, was inserted into the topical report revision as Appendix E.

After submittal of the revised topical report, a meeting was held on November 3, 2000 at the NRC headquarters in Rockville, Maryland, for the purpose of addressing NRC questions and issues. After detailed discussions covering the entire topical report submittal, there remained seven issues requiring closure. Five of these issues were CENP action items and two were NRC. It was also agreed that a conference call would be held on or before November 13, 2000, for the resolution of these items. The phone call was held on November 8, 2000. CENP and NRC verbally described responses to each of the seven items. The material contained in the following sections officially responds to the five CENP action items. Also, the NRC's responses to its two action items are summarized along with the required CENP follow-up actions.

Note that the NRC request for information is denoted by a **bold** font in the following material. An *Italics* font denotes the CENP responses and a regular font denotes the NRC responses.

References

1. LD-99-026, "Revision to ABB CENP ECCS Performance Appendix K Evaluation Model," letter from I. C. Rickard (ABB CENP) to U. S. Nuclear Regulatory Commission Document Control Desk, April 30, 1999.
2. Letter from J. Cushing (NRC) to I. C. Rickard (ABB CENP), "Acceptance of CENPD-132, Supplement 4, 'Calculative Methods for the ABB CENP Large Break LOCA Evaluation Model,' for Review," October 4, 1999.
3. Letter from J. Cushing (NRC) to I. C. Rickard (ABB CENP), "Request for Additional Information (RAI) Regarding CENPD-132-P, Supplement 4-P (TAC NO. MA5660)," December 14, 1999.
4. LD-2000-0011, I. C. Rickard (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "ABB CENP Response to NRC Request for Additional Information Regarding CENPD-132-P, Supplement 4-P," February 22, 2000.
5. LD-2000-0046, P. W. Richardson (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model," August 30, 2000.
6. LD-1999-0064, I. C. Rickard (CENP) to U. S. Nuclear Regulatory Commission (Attn: Yi-Hsiung Hsii), "COMPERC-II Topical Report Set – Information Copy," December 9, 1999.

1. CENP will evaluate the effects of possible U-tube uncovery and provide justification for its assumption in the steam venting steam generator model that the U-tubes are covered by steam generator secondary side liquid.

<u>CENP Response:</u>

In order to evaluate the effect of steam generator (SG) U-tube uncovery on the application of the 1999 EM to the LBLOCA scenario, a special version of the COMPERC-II code was created which explicitly represents the effect of U-tube uncovery on the system. As described in Section 2.4.1.3 (v) of the Topical Report, the heat transfer rate for each axial section on the steam generator tubes secondary side (i ranging from 1 to 2N, where N is the number of axial layers in the evaporator region) is calculated using the equation

$$Q_{sec,i} = H_{sec,i} A_{sec,i} (T_{tube,i} - T_{sec,j})$$

where

$Q_{sec,i}$	*Steam generator tube heat transfer rate (secondary side) (Btu/sec) (section i)*
$H_{sec,i}$	*Overall secondary side heat transfer coefficient (Btu/sec ft^2 °F) (section i)*
$A_{sec,i}$	*Secondary side tube heat transfer area (ft^2) (section i)*
$T_{tube,i}$	*Steam generator tube temperature (°F) (section i)*
$T_{sec,j}$	*Secondary side layer temperature (°F) (layer j)*
j	*Secondary side layer in contact with section i of the tubes*

The changes made to this special version of COMPERC-II are as follows:

- *The heat transfer coefficient $H_{sec,i}$ is calculated as follows:*

  If axial layer j is liquid, then the heat transfer coefficients are calculated with the [ ] correlation for single phase liquid flow as described in Section 2.4.1.3 (v), and the liquid temperature in contact with the tube is the corresponding secondary side layer temperature.

  If the U-tube is uncovered, then a heat transfer coefficient equal to [ ] is used in the above equation, and the SG secondary side steam temperature is used in place of the secondary side layer temperature. This surface heat transfer coefficient is appropriate for [ ] at the pressures and temperatures typical of this time period of the transient.

- *The heat transfer from the steam phase to the tubes is included in the conservation of energy equation described in Section 2.4.1.2 (iii). That is, Q_{tube} in that equation includes the heat transfer from the steam phase described above.*

- *The location of the mixture level in the SG secondary side for this special version of the code is specified through input.*

The computer case in the Topical Report used to determine the effect of activating the 1999 EM steam generator model in COMPERC-II (Table 2.4.3-1) was used for the evaluation of the effect of U-tube uncovery, by reducing the initial liquid inventory in the SGs and by specifying the location of the liquid level in the secondary side through input. The comparison shows the results for the following cases:

Case 1: *Case with U-tubes covered based on nominal initial liquid inventory in the SG (this is the case for the 1999 EM SG Model in Table 2.4.3-1).*

Case 2: *Case with reduced liquid inventory in the SGs leading to U-tube uncovery. This case was run by reducing the initial liquid inventory in the SG by 60% and by reducing the U-tube coverage by 40%. The reduction in the SG inventory was implemented in both the CEFLASH-4A and COMPERC-II codes.*

Case 3: *Case with reduced liquid inventory (like Case 2) but forcing the code to calculate heat transfer coefficients to the liquid phase (consistent with the calculations of the 1999 EM version of the COMPERC-II code). This case isolates the effect of using heat transfer coefficients to liquid instead of heat transfer coefficients to steam in the uncovered region of the U-tubes.*

The comparison of key parameters for these three cases is shown in the following table:

Effect of Steam Generator U-Tube Uncovery

[

Result	Case 1 Nominal Inventory U-Tubes Covered	Case 2 Reduced Inventory U-Tubes Uncovered	Case 3 Reduced Inventory U-Tubes Covered

]

These cases show the following:

- *The effect of U-tube uncovery during the large break LOCA event is a[ ] Thus, the 1999 EM methodology assuming no U-tube uncovery is [ ]*
- *The effect on PCT of calculating heat transfer to liquid in the upper regions of the U-tubes when U-tube uncovery occurs is[ ]*

The following figures show various comparisons for the cases described above.

- *Figure 1-1 shows that the cladding temperature transients on the limiting node (i.e., the PCT node) are[ ]between Cases 1 and 2.*
- *Figure 1-2 shows a comparison of the SG pressure response for Cases 1 and 2. Note that since Case 2 is run with a reduced liquid inventory, the SG pressurization during blowdown is[ ]liquid temperature at TAD. Also, the fuel average temperature of the hot rod at TAD is[ ]change in calculated blowdown hydraulics.*
- *Figure 1-3 shows a comparison of the SG steam temperature in the U-tubes for Cases 2 (U-tubes uncovered) and 3 (U-tubes covered) at 130 seconds into the transient. This comparison shows that the difference in temperature in the upper region of the U-tubes is[.] The difference at the U-tube exit is[ ]between these two cases.*

In summary, these results show that heat transfer from the secondary to primary is[ ]for the 1999 EM Appendix K model.

Figure 1-1
Effect of Uncovery of the SG U-Tubes
Peak Clad Temperature Curve

STR.1.3.Z1 2000/11/08

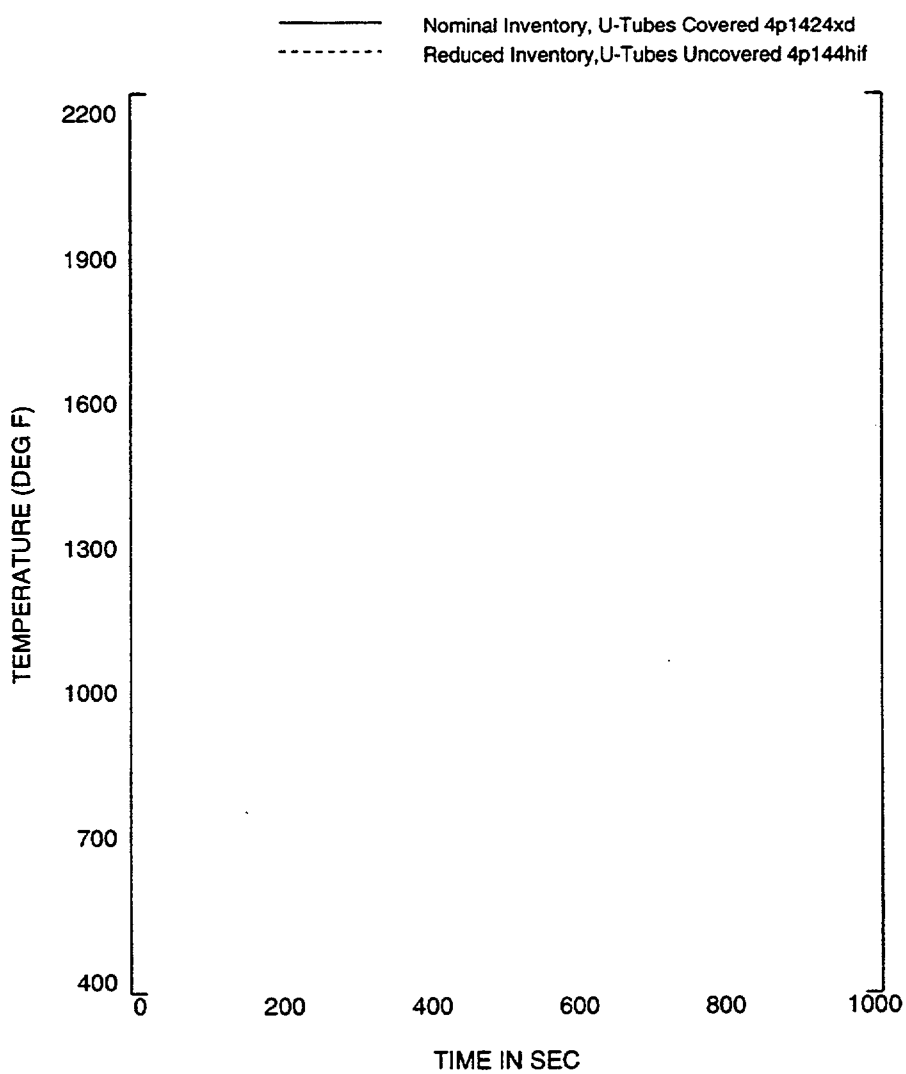


Figure 1-2

SG Secondary Side Pressure

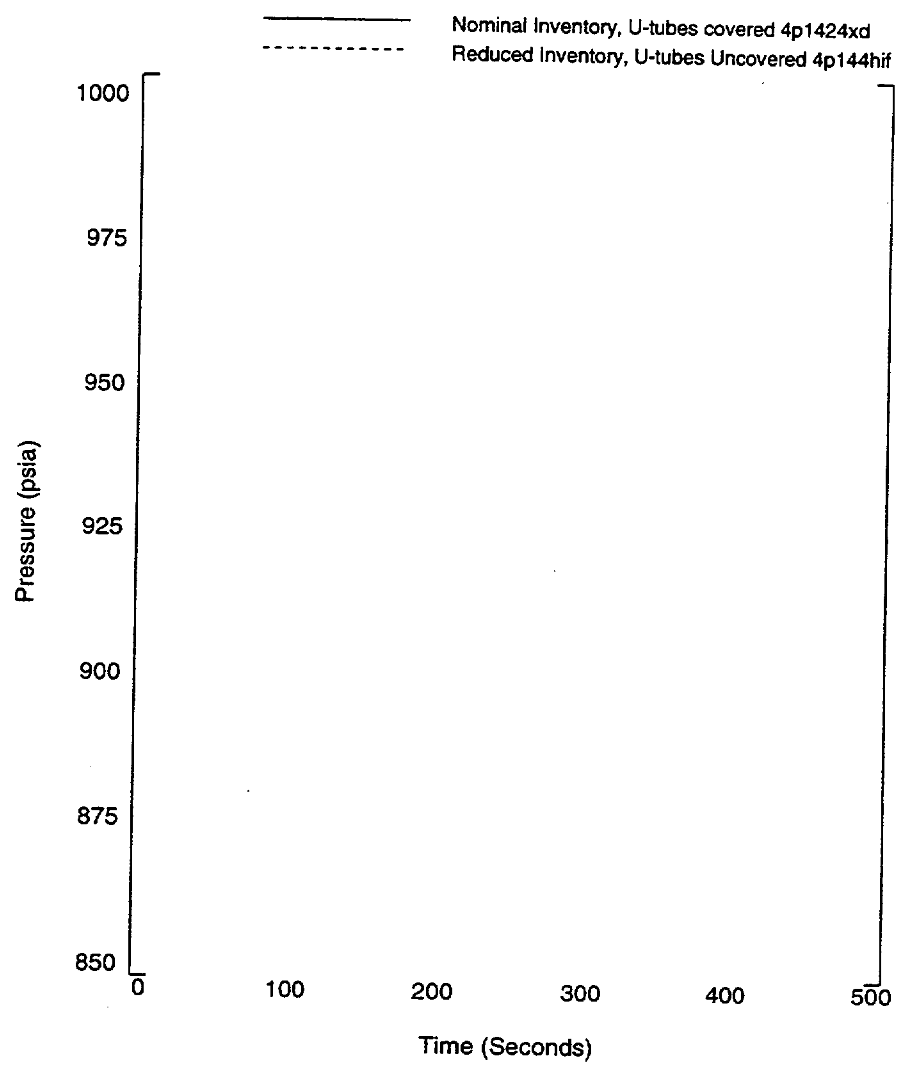
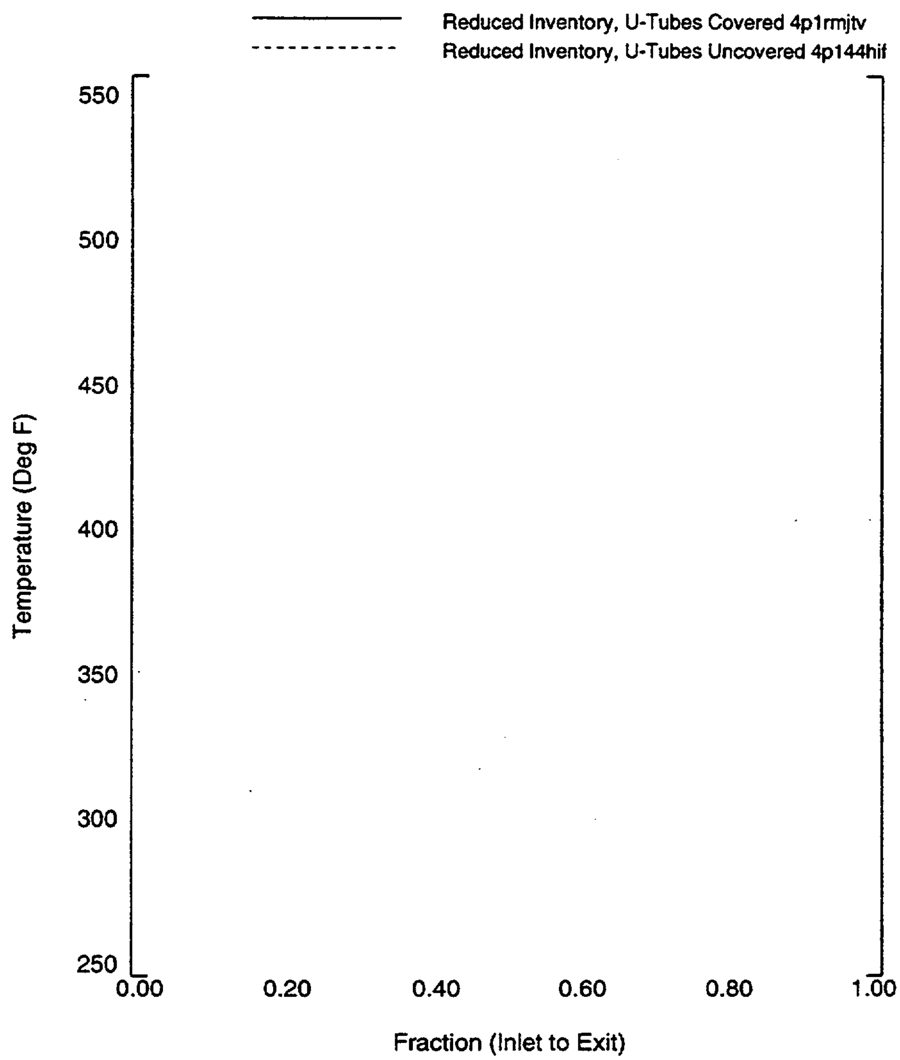


Figure 1-3

SG Tube Steam Temperature (at 130 Seconds)



2. CENP will evaluate the effects of neglecting the wall heat transfer to the SG secondary side steam region in the SG modeling.

<u>CENP Response:</u>

In order to evaluate the effect of wall heat transfer to steam, special versions of the CEFLASH-4A and COMPERC-II codes were created, which model the SG walls in contact with steam. The changes that were made to the codes are as follows:

- *Two separate types of walls were implemented in each SG to represent thick walls in contact with the steam, like the SG vessel and dome, and thin walls in contact with the steam representing the internal structures in the steam generators, like separators and dryers.*
- *The wall heat rate for each of these walls was calculated as follows:*

$$Q_{wall} = H\,A\,(T_{wall} - T_{stm})$$

where

Q_{wall}	= *Heat rate from the wall to the steam (Btu/sec)*
H	= *Wall surface heat transfer coefficient (Btu/sec ft^2 °F)*
A	= *Wall surface heat transfer area (ft^2)*
T_{wall}	= *Wall average temperature (°F)*
T_{stm}	= *Steam region temperature (°F)*

For the analysis, a wall surface heat transfer coefficient equal to [ ] was used. This surface heat transfer coefficient is appropriate for [ ] at the pressures and temperatures typical of this time period of the transient. The wall temperature for each wall was calculated by integrating the energy equation

$$MC_p\, dT_{wall}/dt = -Q_{wall}$$

where

MCp	= *Wall heat capacity (Btu/°F)*
t	= *Time (sec)*

- *The wall temperatures in contact with the steam were initialized in CEFLASH-4A using the SG saturation temperature. The values of the wall temperatures at TAD were transferred from CEFLASH-4A to COMPERC-II to initialize the wall temperatures in COMPERC-II. Since COMPERC-II combines the two SGs into one model while CEFLASH-4A explicitly represents two SGs, the value used for the transfer was the [ ] temperature of each of the walls in the two SGs.*
- *The heat transfer from the wall heat to the steam was added as an extra term in the conservation of energy equation for COMPERC-II described in Section 2.4.1.2 (iii) and in the one for CEFLASH-4A described in Reference 2-1, Page 9.*

The computer case in the Topical Report used to determine the effect of activating the 1999 EM steam generator model in COMPERC-II (Table 2.4.3-1) was used for this evaluation by implementing representative input for the heat capacities and overall heat transfer coefficients for

the thin and thick walls described above in both the CEFLASH-4A and COMPERC-II codes. The comparison shows the results for the following cases:

Case 1: *Case without wall heat to steam (this is the case for the 1999 EM SG Model given in Table 2.4.3-1).*

Case 2: *Case 1 modified to include wall heat to steam.*

Case 3: *Case 1 modified to include wall heat to steam and reduced liquid inventory in the SGs leading to U-tube uncovery (combining Cases 2 of Questions 1 and 2).*

The comparison of key parameters for these three cases is shown in the following table:

Effect of Wall Heat to Steam

[

Result	Case 1 Nominal Inventory No Wall Heat to Steam	Case 2 Nominal Inventory with Wall Heat to Steam	Case 3 Reduced Inventory with U-Tube uncovery and Wall Heat to Steam

]

These cases show the following:

- *The effect of wall heat to steam on PCT is [.] Since the SGs are isolated at time zero, causing subsequent SG pressurization, the walls in contact with the steam act as heat sinks during the transient, absorbing heat from the secondary side and causing a [.] The SG pressure transient comparison between the reference case (Case 1) and the case with wall heat to steam (Case 2) is shown in Figure 2-1. The secondary side wall temperature responses for the walls in contact with steam and the SG secondary side steam temperature response for Case 2 are shown in Figure 2-2.*
- *The combined effect of wall heat to steam and SG U-tube uncovery [ ] the conclusions of the responses to Question 1. That is, the effect of the use of wall heat to steam for the case with U-tube uncovery (Case 3) compared to Case 2 from Question 1 is [ ]*

Section 3.3.2 of the Topical Report provides a sensitivity study showing the impact of steam generator secondary initial pressure and steam generator physical parameters, in particular, the wall heat capacity for storing and releasing energy to the liquid region is increased by 10% for the study. The results confirm that variations in wall heat parameters, even of this magnitude, have a [ ] on the calculated PCT.

In summary, these results show that wall heat transfer to the secondary side steam region has [ ] on the calculation of PCT using the 1999 EM steam venting model.

Reference:

2-1 *CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.*

Figure 2-1

SG Secondary Side Pressure

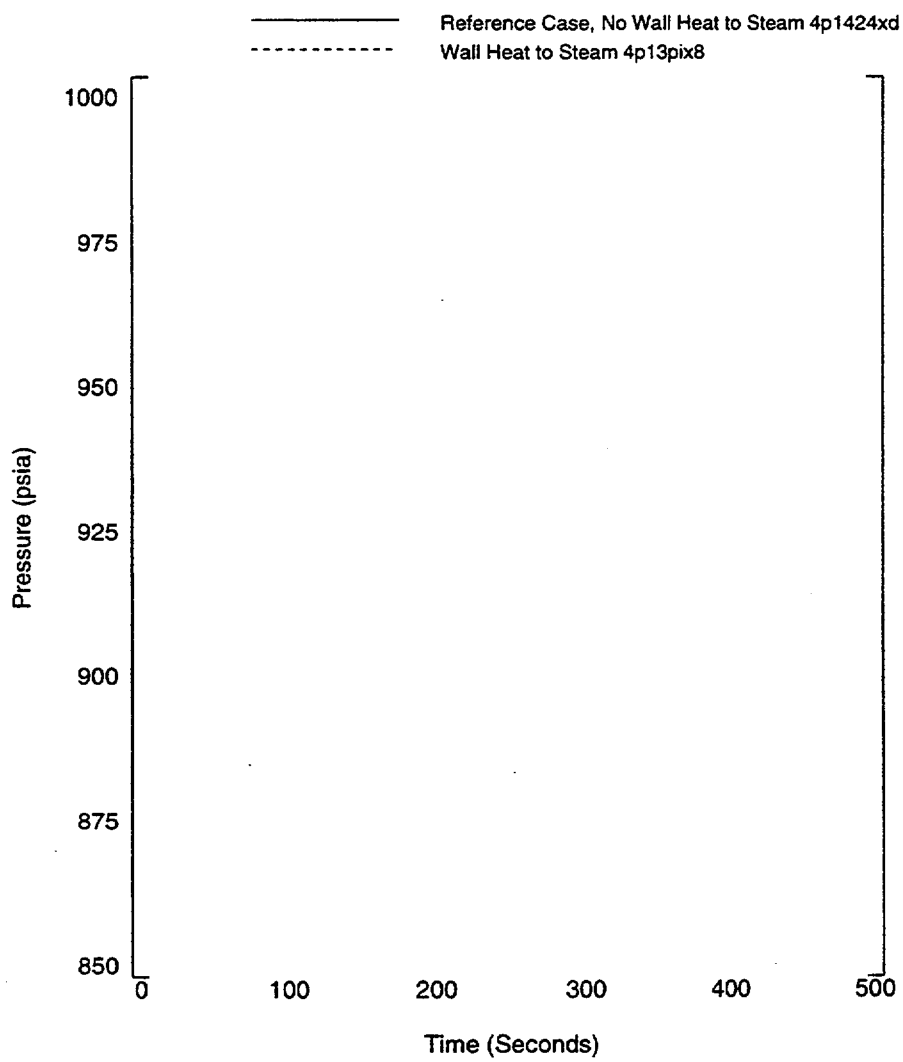
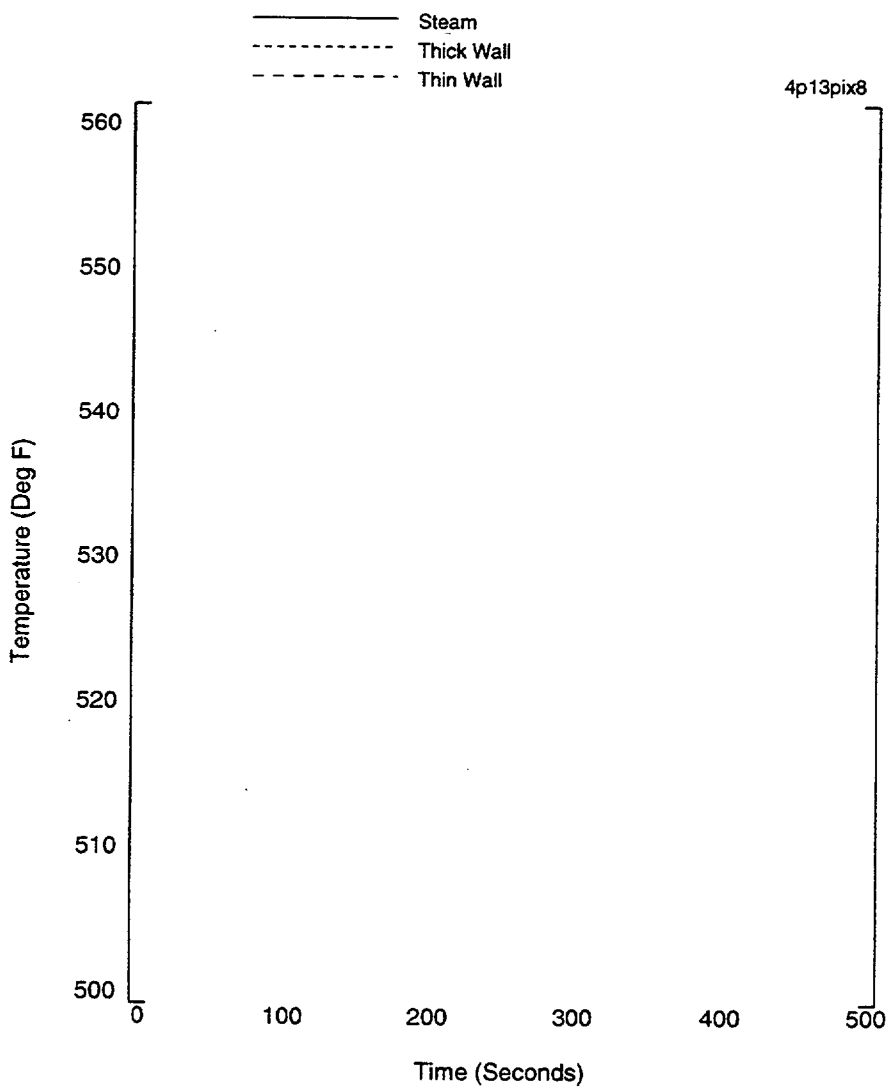


Figure 2-2

SG Secondary Side Temperatures



3. CENP will provide comparisons of two additional FLECHT-SEASET separate effects test cases (Nos. 22010 and 22503) for the validation of the steam venting SG model. CENP will also provide comparisons of the primary side steam temperatures for those cases analyzed.

<u>CENP Response:</u>

As requested, two additional FLECHT-SEASET tests have been added to the verification database, Tests 22010 and 22503, References 3-1 and 3-2. These tests are shaded in the table below for comparison to the tests already documented in Topical Report Section 2.4.2.2.

	Primary Side Test Conditions (1)			Secondary Side Initial Conditions (2)	
Test ID	Primary Side Pressure (psia)	Total U-Tube Mass Flow (lbm/sec)	U-Tubes Inlet Quality	Secondary Side Level (ft)	Secondary Side Temperature (°F)
22010	40.0	0.503	0.801	33.6	525
22503	19.9	0.494	0.798	33.6	525
20904	60.4	0.494	0.798	32.4	525
22213	40.0	0.991	0.797	33.7	524
22920	39.8	0.493	1.0	32.5	523

Notes:

(1) U-tube boundary conditions are held constant for the duration of the test
(2) Secondary side values are the initial test conditions

Topical Report Section 2.4.2.2.i. defines the FLECHT-SEASET test conditions with the following categories:

- *High pressure (~60 psia), mid pressure (~40 psia), low pressure (~20 psia)*
- *High flow (~0.99 lbm/sec), mid flow (~0.75 lbm/sec), low flow (~0.49 lbm/sec)*
- *High quality (1.), mid quality (~0.8), low quality (~0.7 and below)*

Using these categories, the verification test basis is characterized as follows including the two additional tests:

- *Test ID 22010: mid pressure, low flow, mid quality*
- *Test ID 22503: low pressure, low flow, mid quality*
- *Test ID 20904: high pressure, low flow, mid quality*
- *Test ID 22213: mid pressure, high flow, mid quality*
- *Test ID 22920: mid pressure, low flow, high quality*

The results of the additional test comparisons are shown in Figures 3.1-1 through 3.1-8 for Test 22010 and Figures 3.2-1 through 3.2-8 for Test 22503. These figures include comparisons to data on both the secondary and primary sides.

Similar to the original three tests reported in the Topical Report, the comparisons for Tests 22010 and 22503 provide results that show that the 1999 EM COMPERC-II model[ ] the measured values of the secondary side temperature both at the end of the test, see

Figures 3.1-3 and 3.2-3, and at the tube exit elevation throughout the test, see Figures 3.1-4, and 3.2-4.

The Topical Report figures for the original three tests do not include comparisons to primary side data. Therefore, Figures 3.3-1 through 3.5-4 have been added here to provide these comparisons. For each of the five test comparisons, the primary side temperatures are shown at four axial positions along the 70 ft of U-tube length, that is, at 4, 10, 60, and 69 ft. These test facility locations correspond to the available data. Unlike the inherently uniform conditions measured on the secondary side, the primary side temperature measurements represent the local conditions inside one particular U-tube and variations on the order of ±10 °F are observed. The COMPERC calculated values of average U-tube primary temperature are linearly interpolated from the COMPERC 30 axial node model to correspond to the measurement location.

At the outlet location of 69 ft, the results of the comparisons show that the 1999 EM COMPERC-II model[ ] the measured values of the primary side temperature at the steam generator U-tube outlet position by as much as [ ] at the end of the test for all cases except for Test 22920. Test 22920 is a saturated steam test (no liquid at the inlet), where the agreement between test measurement and COMPERC calculation is excellent, which validates the 1999 EM formulation. These comparisons at the 69 ft location are shown in Figures 3.1-8, 3.2-8, 3.3-4, 3.4-4, and 3.5-4.

The comparisons at the other axial locations are not as important as the outlet condition mentioned above because it is the outlet condition that has the greatest impact on steam venting from the steam generator to the break through the cold leg piping and reactor coolant pump. The comparisons within the axial extent of the steam generator are complicated due to the influence of local conditions within the particular U-tube holding the steam probe on the measurements. For example, a 15 °F difference between the measured temperature and the calculated value can be attributed to just a 10% variation in the local U-tube flow rate from the average flow being used in the COMPERC calculation. The results of the comparisons at the other axial positions are given as follows:

- *At the inlet location of 4 ft, the COMPERC calculations[ ] the measured temperatures by as much as[ ] at the end of the test for all cases except the saturated steam test 22920, which shows excellent agreement. This overall conservatism of the inlet comparisons is related to the outlet comparisons at 69 ft discussed above, because the stratified secondary side fluid temperature strongly influences both inlet and outlet calculations. These comparisons at the 4 ft location are shown in Figures 3.1-5, 3.2-5, 3.3-1, 3.4-1, and 3.5-1.*
- *At the 10 ft location, the COMPERC calculations are an[ ] representation of the measured values for all cases except Test 22213, which is the high flow rate test where the comparison is[ ] at the end of the test), and Test 22920, which is the saturated steam test where the calculation [ ] which is of the same order as the observed range of variation in the measurements. In fact, the comparisons for the other tests at this same location are similarly biased during the early portion of the test by roughly [ ]. After 600 seconds, the calculated values at this location[ ] at the end of the test. These comparisons at the 10 ft location are shown in Figures 3.1-6, 3.2-6, 3.3-2, 3.4-2, and 3.5-2.*

- *At the 60 ft location, the COMPERC calculations are in[ ]agreement with the measured values (within[ ]for the majority of the time of the test) for all tests except Test 22213, which is the high flow test where the calculations[ ]at the end of the test. This measurement location is on the downflow side of the steam generator, where the temperature differential between the primary and secondary sides is small and where the steam temperature is therefore less sensitive to the local U-tube flow rate. Also, the secondary side stratification is not as large at 60 ft as 69 ft, therefore, the temperature comparisons between the calculated and measured values are in[ ]agreement, again showing the adequacy of the 1999 EM steam generator model. These comparisons at the 60 ft location are shown in Figures 3.1-7, 3.2-7, 3.3-3, 3.4-3, and 3.5-3.*

References:

3-1 "PWR FLECHT-SEASET Steam Generator Separate-Effects Task, Data Analysis and Evaluation Report," EPRI NP-1461, NUREG/CR-1534, WCAP-9724, February 1982.

3-2 "PWR FLECHT-SEASET Steam Generator Separate-Effects Task, Data Report," NRC/EPRI/Westinghouse Report No. 4.

Figure 3.1-1

FLECHT-SEASET Test 22010

Primary Side SG Tube Axial Temperature

COMPERC-II Calculation (Every 150 seconds)

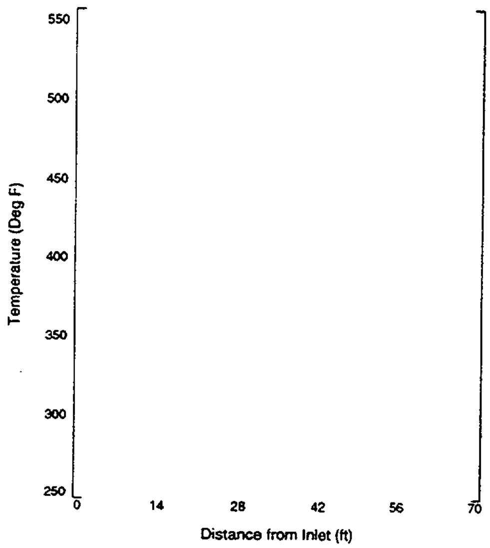


Figure 3.1-2

FLECHT-SEASET Test 22010

Secondary Side Fluid Temperature along SG Tube

COMPERC-II Calculation (Every 150 seconds)

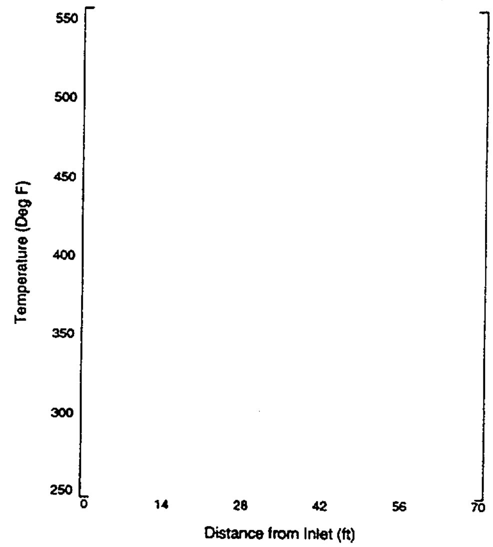


Figure 3.1-3

FLECHT-SEASET Test 22010

Secondary Side Fluid Temperature along SG Tube

Comparison to COMPERC-II Calculation (1500 Seconds)

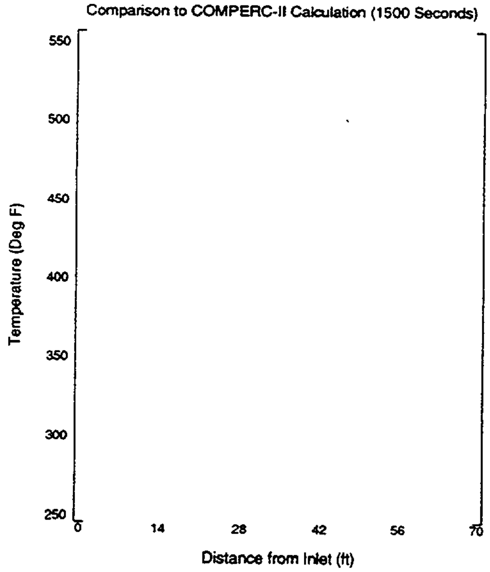


Figure 3.1-4

FLECHT-SEASET Test 22010

Secondary Side Fluid Temperature (at tube exit)

COMPERC-II Calculation

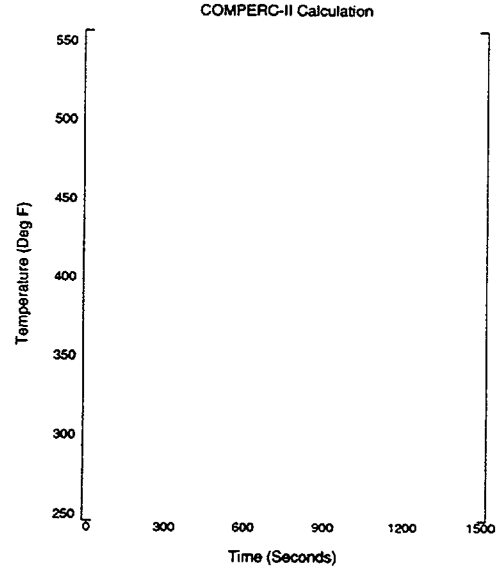


Figure 3.1-5
FLECHT-SEASET Test 22010
Primary Side Steam Temperature (at 4 ft)
COMPERC-II Calculation

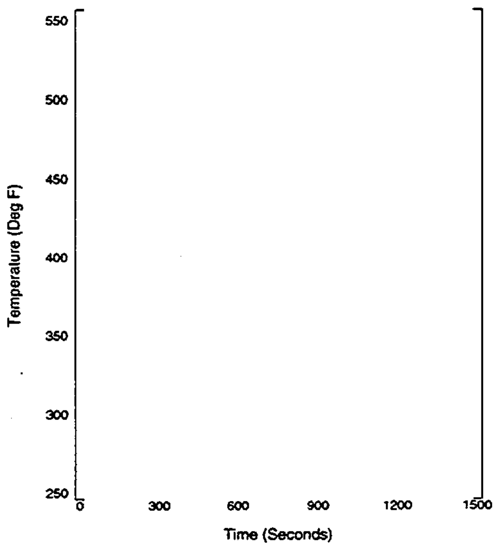


Figure 3.1-6
FLECHT-SEASET Test 22010
Primary Side Steam Temperature (at 10 ft)
COMPERC-II Calculation

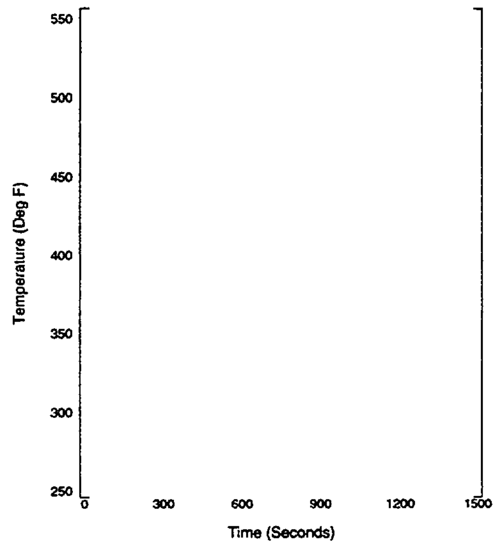


Figure 3.1-7
FLECHT-SEASET Test 22010
Primary Side Steam Temperature (at 60 ft)
COMPERC-II Calculation

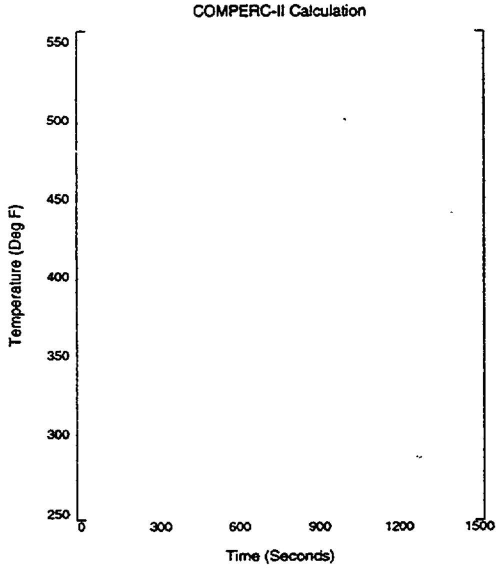


Figure 3.1-8
FLECHT-SEASET Test 22010
Primary Side Steam Temperature (at 69 ft)
COMPERC-II Calculation

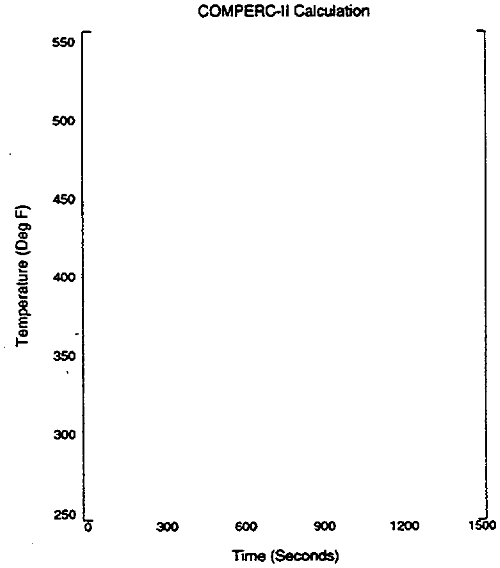


Figure 3.2-1
FLECHT-SEASET Test 22503
Primary Side SG Tube Axial Temperature
COMPERC-II Calculation (Every 150 seconds)

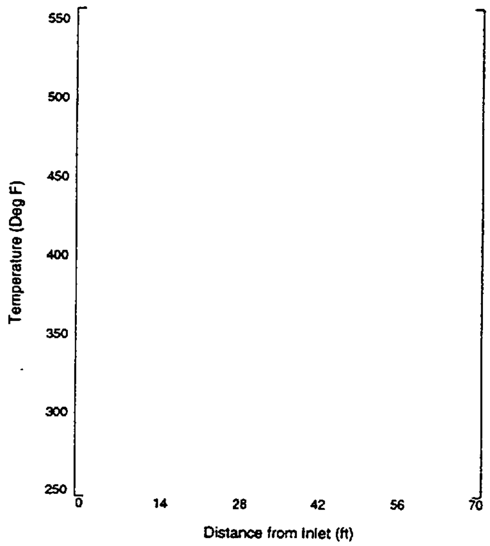


Figure 3.2-2
FLECHT-SEASET Test 22503
Secondary Side Fluid Temperature along SG Tube
COMPERC-II Calculation (Every 150 seconds)

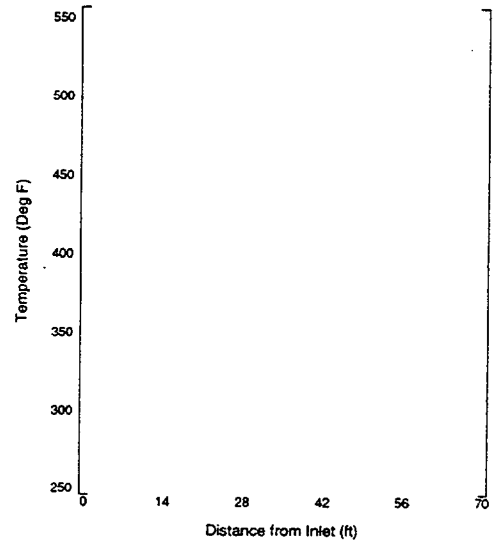


Figure 3.2-3
FLECHT-SEASET Test 22503
Secondary Side Fluid Temperature along SG Tube
Comparison to COMPERC-II Calculation (1500 Seconds)

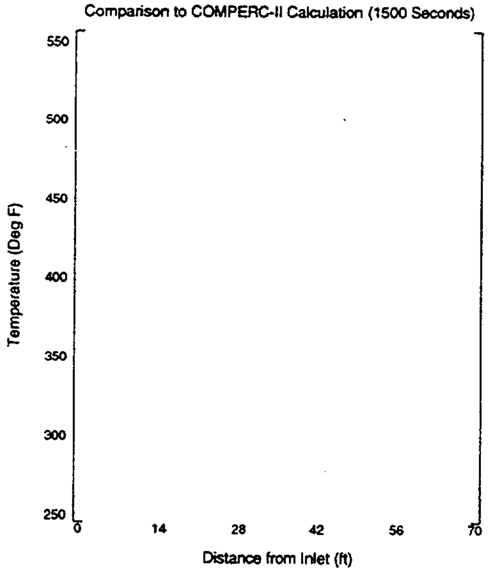


Figure 3.2-4
FLECHT-SEASET Test 22503
Secondary Side Fluid Temperature (at tube exit)
COMPERC-II Calculation

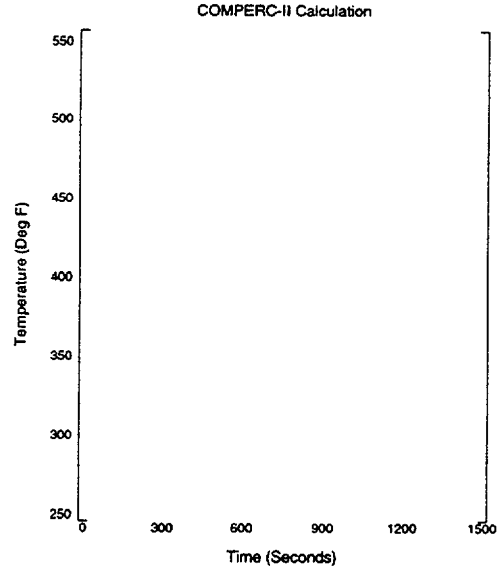


Figure 3.2-5
FLECHT-SEASET Test 22503
Primary Side Steam Temperature (at 4 ft)
COMPERC-II Calculation

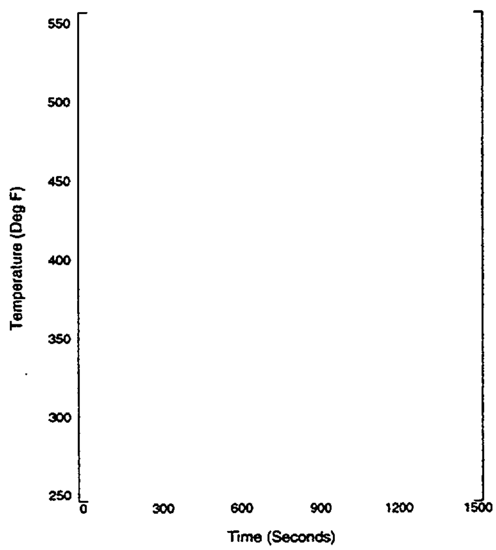


Figure 3.2-6
FLECHT-SEASET Test 22503
Primary Side Steam Temperature (at 10 ft)
COMPERC-II Calculation

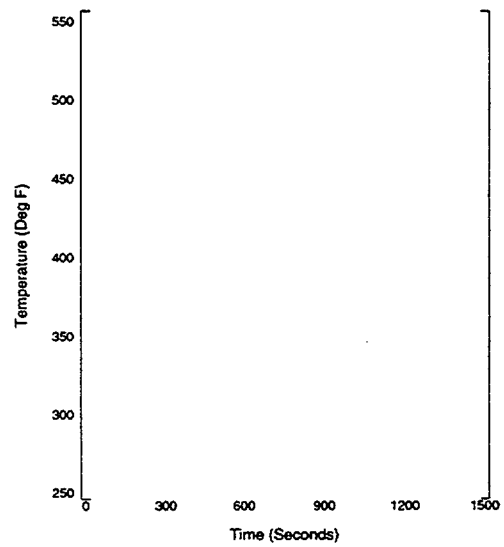


Figure 3.2-7
FLECHT-SEASET Test 22503
Primary Side Steam Temperature (at 60 ft)
COMPERC-II Calculation

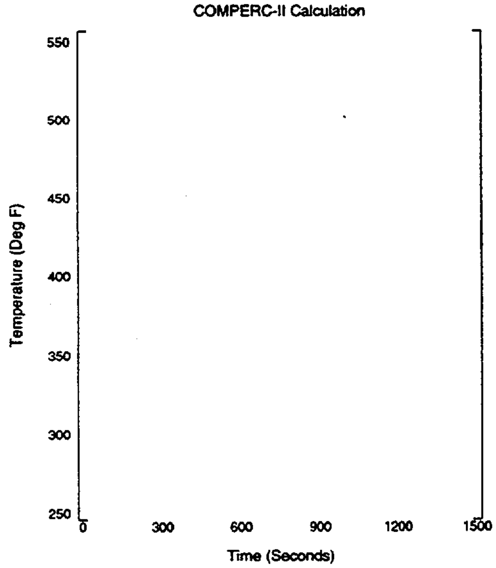


Figure 3.2-8
FLECHT-SEASET Test 22503
Primary Side Steam Temperature (at 69 ft)
COMPERC-II Calculation

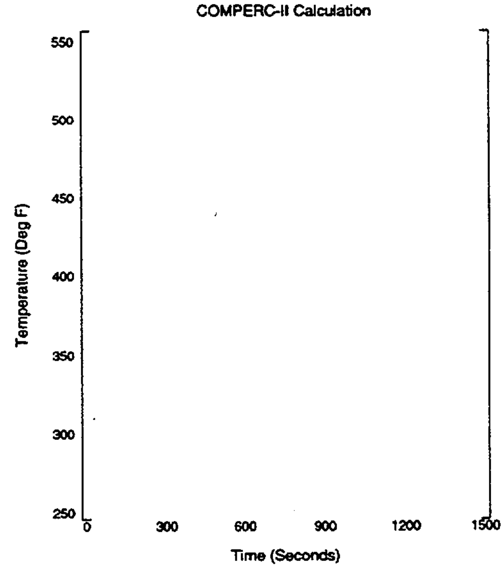


Figure 3.3-1
FLECHT-SEASET Test 20904
Primary Side Steam Temperature (at 4 ft)
COMPERC-II Calculation

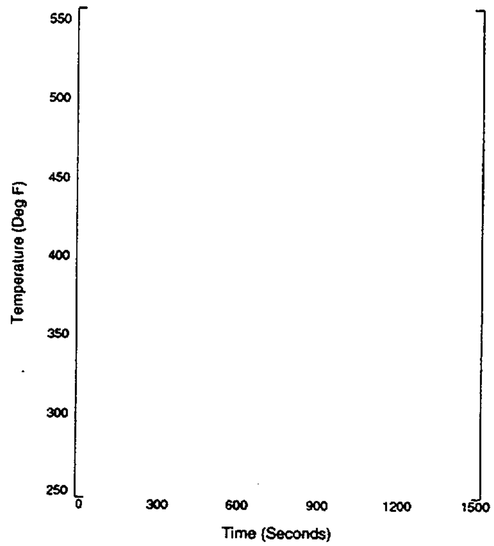


Figure 3.3-2
FLECHT-SEASET Test 20904
Primary Side Steam Temperature (at 10 ft)
COMPERC-II Calculation

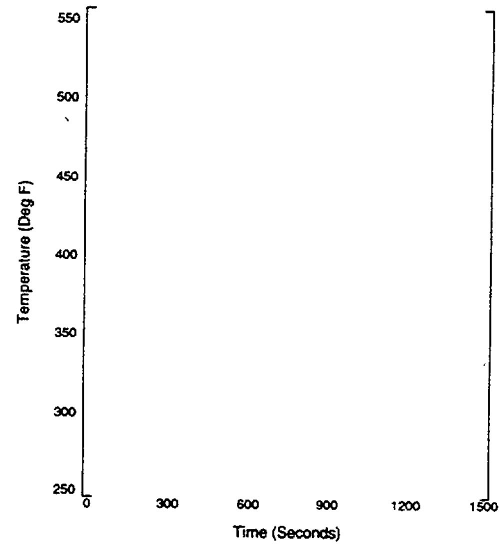


Figure 3.3-3
FLECHT-SEASET Test 20904
Primary Side Steam Temperature (at 60 ft)
COMPERC-II Calculation

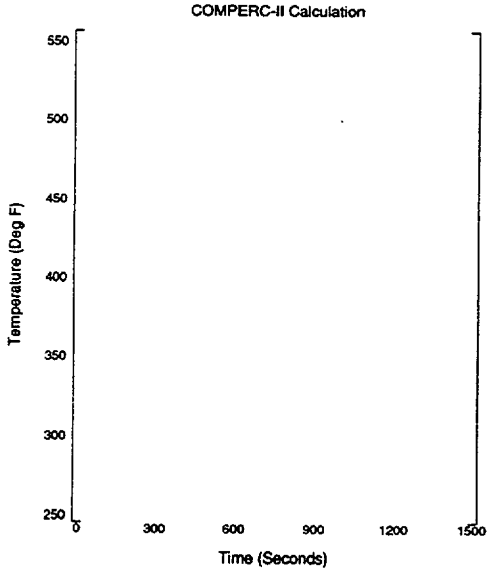


Figure 3.3-4
FLECHT-SEASET Test 20904
Primary Side Steam Temperature (at 69 ft)
COMPERC-II Calculation

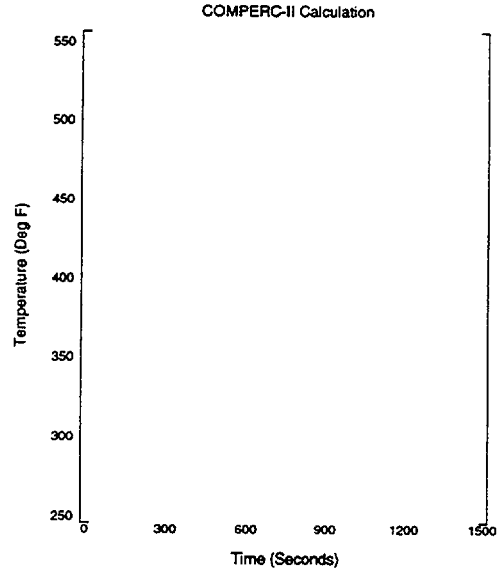


Figure 3.4-1
FLECHT-SEASET Test 22213
Primary Side Steam Temperature (at 4 ft)
COMPERC-II Calculation

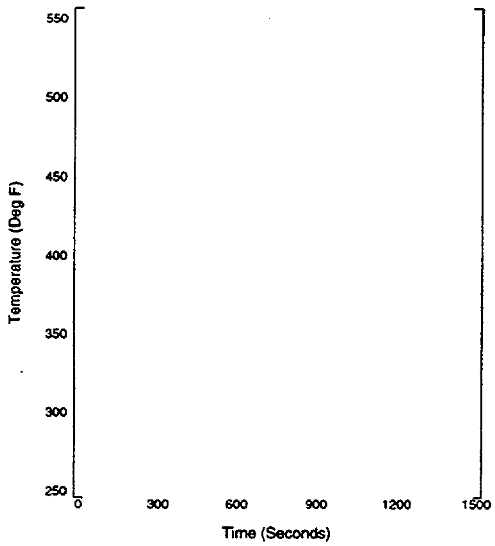


Figure 3.4-2
FLECHT-SEASET Test 22213
Primary Side Steam Temperature (at 10 ft)
COMPERC-II Calculation

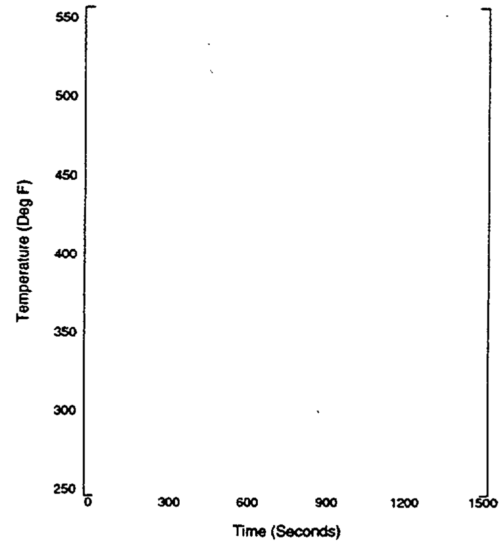


Figure 3.4-3
FLECHT-SEASET Test 22213
Primary Side Steam Temperature (at 60 ft)
COMPERC-II Calculation

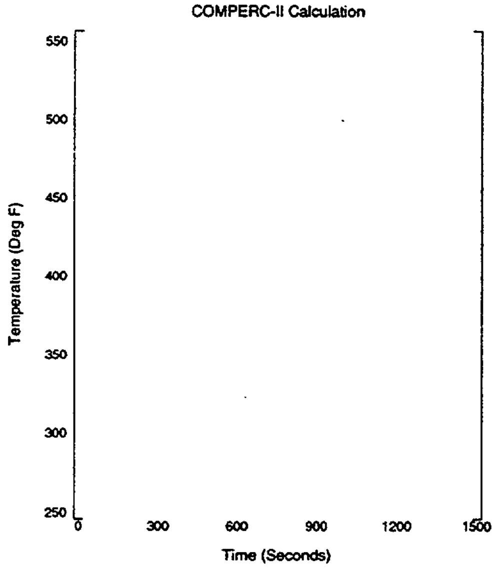


Figure 3.4-4
FLECHT-SEASET Test 22213
Primary Side Steam Temperature (at 69 ft)
COMPERC-II Calculation

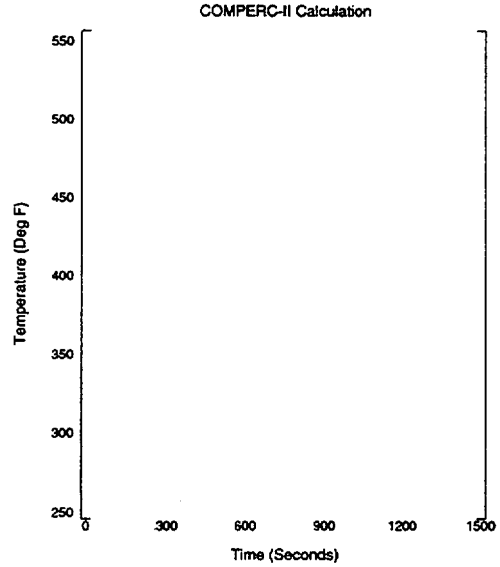


Figure 3.5-1
FLECHT-SEASET Test 22920
Primary Side Steam Temperature (at 4 ft)
COMPERC-II Calculation

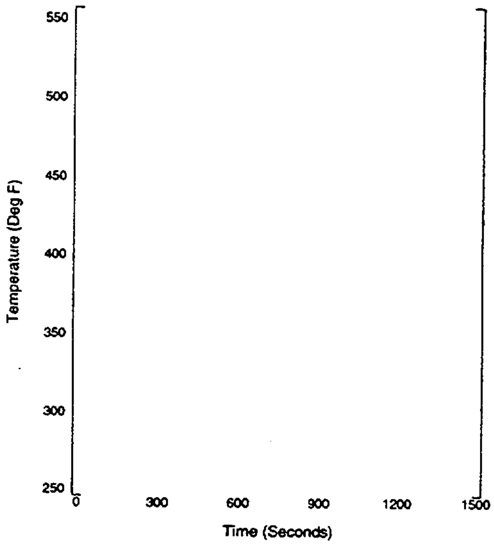


Figure 3.5-2
FLECHT-SEASET Test 22920
Primary Side Steam Temperature (at 10 ft)
COMPERC-II Calculation

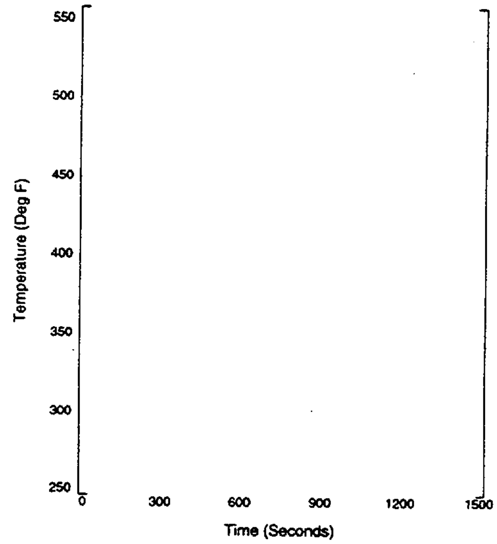


Figure 3.5-3
FLECHT-SEASET Test 22920
Primary Side Steam Temperature (at 60 ft)
COMPERC-II Calculation

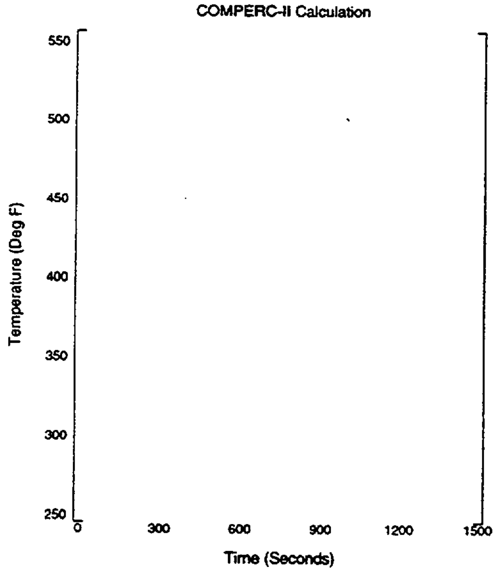
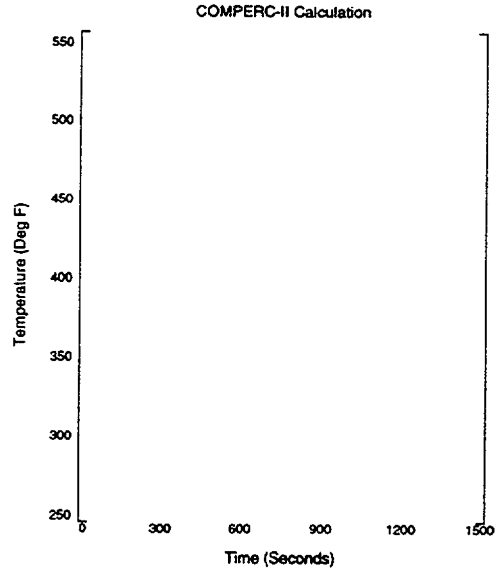


Figure 3.5-4
FLECHT-SEASET Test 22920
Primary Side Steam Temperature (at 69 ft)
COMPERC-II Calculation



4. **CENP will provide a brief description of the control logic for the calculation of the injection section differential pressure from nitrogen injection to ensure compliance with the existing safety evaluation report limits on the differential pressures of the safety injection tank injection and pump injection.**

CENP Response:

Section 2.5.5 of the Topical Report will be modified to include the following information, which explains that the code logic implemented in the 1999 EM ensures that the regulatory limits on injection section differential pressure are not violated by the new model for SIT nitrogen discharge:

> *The differential pressure across the injection line is calculated using Equation (2.5-1). To ensure conformance to regulatory limits on the injection section differential pressure, the updated code logic subjects the results of Equation (2.5-1) to the following limitations:*
>
> - [
>
> -]

5. CENP will provide a summary statement that all of the methodology changes described in CENPD-133 Supplement 4-P, have been included in the 1985 EM.

<u>*CENP Response:*</u>

In Section 3.2 of the Topical Report, Item #2, CENP requested closure on legacy topical report CENPD-133 Supplement 4-P submitted to NRC in 1977, which was inadvertently not cited in the reference list of the SER for the 1985 EM. This is one of the references that comprise the 1985 EM and which will also comprise the 1999 EM. The paragraph in Section 3.2 of the Topical Report will be modified to include the following statement:

> *All of the methodology changes described in CENPD-133 Supplement 4-P, including the change made to conform to the Appendix K requirement on "no return to nucleate boiling," were incorporated into the 1985 EM.*

6. NRC will evaluate CENP's position that the result of a sensitivity study of the refueling water storage tank water temperature used for safety injection and containment spray is generically applicable to all CENP-designed plants.

NRC Response:

In the phone call on November 8, 2000, NRC stated that it could not extend the use of minimum refueling water storage tank (RWST) water temperature as bounding for all applications of the 1999 EM. The NRC indicated that the SER will say that future plant submittals using the 1999 EM for its ECCS performance evaluation must confirm that the use of minimum temperature as described in the Topical Report is bounding.

CENP Follow-up Actions:

As described in Section 2.1.3 of the Topical Report, the use of the automated/integrated code system for the 1999 EM includes consistent modeling of spray and spillage from the break into the containment during the reflood thermal hydraulics portion of the analysis. In this regard, Section 3.3.1 of the Topical Report describes the change from the old approach of using [ ] *to the approach of using* [ ]

To demonstrate the impact of this change, Topical Report Section 3.3.1 presents a sensitivity study, representative of a typical CENP designed plant, demonstrating that use of minimum temperature for the RWST is roughly [ ] *more conservative in the calculation of peak cladding temperature than using maximum temperature for the RWST.*

Section 3.4.4 of the Topical Report states that a worst single failure analysis will be performed for each application of the 1999 EM. This section will be modified to include the statement that, the worst single failure of an ECCS component must include consideration of the most limiting value of the RWST temperature as described in Section 3.3.1.

7. NRC will evaluate the acceptability of using the 1999 EM automated/integrated code system as a vehicle for performing the 1985 EM licensing calculations.

NRC Response:

In the phone call on November 8, 2000, NRC stated that use of the automated/integrated code system as a vehicle for performing 1985 EM ECCS licensing analyses is a change to the evaluation model that requires NRC review and approval. The NRC stated that because the automated/integrated code system changes the way code inputs are prepared that this alters the analysis process and is therefore a change to the evaluation model. Also, because this represents a change to the evaluation model, the Dougall-Rohsenow film boiling correlation must be removed.

CENP Follow-up Actions:

CENP submitted the Topical Report assuming that since the automated/integrated code system was essentially replicating the manual processes and was performing calculations in conformance with already approved methodologies that this was not a change to the evaluation model and therefore did not require NRC review and approval nor removal of Dougall-Rohsenow. However, in the interest of concluding the review and having the SER issued for the 1999 EM, CENP withdraws its application to use the automated/integrated code system in conjunction with the 1985 EM.

To this end, the following modifications to the Topical Report will be made:

> *In the first paragraph of Section 1.2.1, text will be removed that says the 1985 EM process improvements may be implemented separate from the 1999 EM.*
>
> *The second paragraph of Section 2.1 is removed. This removes text saying that the 1985 EM process improvements may be implemented separate from the 1999 EM.*
>
> *In Section A.1, the words "fully compliant" used to describe the options for user input for the 1985 EM will be replaced with the word "consistent."*

Sections 3.4.1, 3.4.2, and 3.5.1, contain analytical results simulating the 1985 EM using the automated/integrated code system. These results remain in the text of the Topical Report in support of the 1999 EM. These results provide useful comparisons between ECCS performance evaluations using the currently accepted 1985 EM methodology and the 1999 EM.

Appendix G

CHANGE PAGES FROM THE SUBMITTED CENPD-132, SUPPLEMENT 4-P, REVISION 1

Reference: LD-2000-0060, P. W. Richardson (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Replacement Pages for CENPD-132, Supplement 4-P, Revision 1," December 1, 2000

Appendix G contains the CENP correspondence to the NRC staff identifying various replacement pages for the licensing review of this topical report submittal for the 1999 EM. These replacement pages are contained within the "-A" version of the topical report. The original Revision 1 pages are contained in Appendix H.

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Westinghouse Electric Company
CE Nuclear Power LLC

2000 Day Hill Road
Windsor, CT 06095
USA

1 December, 2000
LD-2000-0060

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

SUBJECT: REPLACEMENT PAGES FOR CENPD-132, SUPPLEMENTR4-P, REV. 1
{ENCLOSURE 1-P CONTAINS PROPRIETARY INFORMATION}

Reference(s): 1) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", August 2000

2) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model", LD-2000-0046, August 30, 2000

3) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Response to Questions Regarding CENPD-132, Supplement 4-P, Rev. 1", LD-2000-0057, November 13, 2000

Representatives from CE Nuclear Power LLC (CENP) and the Nuclear Regulatory Commission (NRC) participated in a meeting on Friday, November 3, 2000 to discuss the ongoing review of CENPD-132, Supplement 4-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (Reference 1). This topical report was submitted to the NRC on August 30, 2000 (Reference 2). A number of NRC questions arose during the meeting to which CENP responses were provided in Reference 3. At the November 3, 2000 meeting, CENP also agreed to provide the NRC with change pages for the topical report that incorporate the responses to the questions and make other changes requested by the NRC reviewer. Pursuant to prior agreement, CENP is furnishing one (1) proprietary and one (1) non-proprietary copy of this letter and enclosures to the NRC Document Control Desk and three (3) proprietary copies to Jack Cushing, NRC, CENP Project Manager.

Enclosure 1-P contains a copy of the change pages for CENPD-132, Supplement 4-P, Rev. 1. This submittal completes CENP commitments to the NRC for updating the latest version of the topical report. A copy of the non-proprietary change pages are provided in Enclosure 3. The actual change pages in Enclosures 1-P and 3 are preceded by a tabular list of change pages for CENPD-132 Supplement 4-P, Revision 1 with a brief descriptive comment regarding the nature of the change to the page.

CENP has determined that the change pages provided in Enclosure 1-P are proprietary in nature. Consequently, it is requested that Enclosure 1-P be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that this information be appropriately safeguarded. The reasons for the classification of this information as proprietary are covered by

and were delineated in the affidavits provided in the transmittals of References 2 and 3. Copies of these affidavits are provided in Enclosure 2 for your information.

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
CE NUCLEAR POWER LLC

Chuck Molnar

for Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Enclosure(s): As stated

xc: J. S. Cushing (NRC, 3 copies)

CE Nuclear Power LLC

Non-Proprietary Change Pages for CENPD-132, Supplement 4-P, Revision 1 Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model

December 2000

List of Change Pages for CENPD-132 Supplement 4-P, Revision 1

Page Number	Description of Change
v	Addition of Appendices F and G to table of contents
1.0-4 and 1.0-5	Removal of statements about 1985 EM process changes
1.0-12	Removal of proprietary brackets on Table 1.0-1
2.1-1	Removal of statements about 1985 EM process changes
2.2-4	Delta PCT correction
2.2-10	Delta PCT corrections in Tables 2.2-4 and 2.2-5
2.4-8 through 2.4-12	Model description nomenclature corrections and clarifications Removal of proprietary brackets on model description in Section 2.4.2.1
2.4-37	Added definition of average specific volume used in model as coded
2.4-40	Correction to Table 2.4.3-1
2.4-52	Addition of proprietary brackets on Figure 2.4.2.3.4-2
2.5-11	Added description of model as coded to define imposed regulatory limits
2.5-13	Consistent representation of times relative to beginning of transient
2.7-6 and 2.7-7	Typographical correction of reference numbers
3.0-4	Added text for use of CENPD-133 Supplement 4-P
3.0-12	Typographical correction of JOBID in Table 3.3-4
3.0-19	Added text on RWST temperature
A-1	Replacement of words for the 1985 EM process improvements
A-9	Added reference to COMZIRC code for interface file usage
A-11	Correction to Table A-6 heading
D-4	Correction to nomenclature definitions

Appendix H

ORIGINAL REPORT PAGES THAT WERE REPLACED IN THE "-A" VERSION

Reference: LD-2000-0060, P. W. Richardson (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Replacement Pages for CENPD-132, Supplement 4-P, Revision 1," December 1, 2000

Appendix H contains the original report pages that were replaced by the reference correspondence to the NRC and by final errata.

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TABLE OF CONTENTS

Appendices

LIST OF FIGURES

In addition, plant design data, used to provide analysis inputs, are conservatively biased in accordance with the LBLOCA EM to produce conservatism in the calculated limiting PCT and limiting peak cladding local oxidation percentage (PLO). In some cases, conservative biases have been selected that in reality cannot mutually co-exist.

In the 1985 EM, the CEFLASH-4A computer code (Reference 1.0-3) is used to perform the blowdown hydraulic analysis of the reactor coolant system (RCS) and the COMPERC-II computer code (Reference 1.0-4) is used to perform the refill/reflood hydraulic analysis and to calculate FLECHT-based reflood heat transfer coefficients. The HCROSS (Reference 1.0-5) and PARCH (Reference 1.0-6) computer codes are used to calculate steam cooling heat transfer coefficients. The PCT and PLO are calculated by the STRIKIN-II computer code (Reference 1.0-7). Core-wide cladding oxidation is calculated using the COMZIRC computer code (Appendix C of Supplement 1 of Reference 1.0-4). The initial steady state fuel rod conditions used in the analysis are determined using an NRC-approved fuel performance computer code, FATES3B (Reference 1.0-8).

1.2 Summary Description of Modifications to the LBLOCA EM

The following is a brief summary description of the modifications to the LBLOCA Evaluation Model for the 1999 EM. The modifications are organized into the three categories of changes described earlier.

1.2.1 1985 EM Process Changes Within the Current EM

There are three process changes to the LBLOCA ECCS performance analysis methodology that maintain consistency with the currently NRC-accepted EM. These 1985 EM process changes are included in this topical report in the context of the 1999 EM, but they do not require NRC review since they do not represent any change to the NRC-accepted EM. These process changes may be implemented into the LBLOCA analysis process upon completion of the required quality assurance activities prior to or in combination with the 1999 EM model

improvements. If these 1985 EM process changes are needed prior to completing the licensing process for the 1999 EM improvements, then the NRC will be informed by separate informational correspondence regarding the timing and manner of implementation. These process changes are as follows:

- Automated/Integrated Code System

  The automated/integrated code system for the 1985 EM and the 1999 EM is a process change that combines the various computer codes of the 1985 EM into an integrated code system. With the automated/integrated code system, the analysis process can be executed from start to finish without analyst intervention. The automated/integrated code system can be executed with selected computer codes, models, options, and features that fully represent the methodologies that comprise the currently NRC-accepted 1985 EM. This is referred to as "1985 EM Simulation." The modifications to the 1985 EM that are included in the 1999 EM are activated through options in the User Control Interface (UCI) file, which is part of the automated/integrated code system. The benefit of this 1985 EM process change is to reduce the introduction of discretionary conservatism that is commonly used to avoid repetitive case running and to eliminate interface hand calculations involved in manual transfer of data from one code to the next.

- Explicit NUREG-0630 Cladding Swelling/Rupture

  In the 1985 EM, the NUREG-0630 cladding swelling and rupture models are implemented through user controlled inputs. This process often leads to repetitive calculations while the analyst iterates on heating rate dependent inputs. The 1999 EM automated/integrated code system explicitly calculates the NUREG-0630 model components without analyst intervention or iteration in a manner fully consistent with the approach described in the NRC-approved documentation for the 1985 EM. Therefore, this 1985 EM process change improves numerical precision, eliminates iterative case running, and remains fully compliant with currently approved methodology.

Table 1.0-1
1985 EM Major Sources of Conservatism

[

Category of Conservatism	List of Sources of Conservatism

]

2.1 Process Change within the Currently NRC-Accepted EM

This section describes three process changes to the LBLOCA ECCS performance analysis methodology that remain consistent with the currently NRC-accepted 1985 EM. These process changes are presented here in the context of the 1999 EM but they do not require NRC review since they do not represent any change to the NRC-accepted 1985 EM. These changes are provided herein for completeness of the 1999 EM description. These three process changes are the following:

1. Automated/Integrated Code System
2. Explicit NUREG-0630 Cladding Swelling/Rupture
3. Consistent Modeling of Spray and Spillage into the Containment

The implementation of each of these three process changes into the LBLOCA 1985 EM is described in the following subsections. These process changes may be implemented into the LBLOCA analysis process upon completion of the required quality assurance activities either prior to or in combination with the 1999 EM model improvements. If these 1985 EM process changes are needed prior to completing the licensing process for the 1999 EM model improvements, then the NRC will be informed by separate correspondence regarding the timing and manner of implementation.

In the 1985 EM, the analyst may introduce conservatism in certain parameters in order to eliminate repetitive case running and excessive interface hand calculations required to transfer data from one code to the next. That is, the analyst may control the manner in which interface data is transferred from one code to the next by deliberately selecting values to conservatively bias the data transfer process. The purpose of these three 1985 EM process changes is to reduce this type of discretionary conservatism and bring more consistency to the analysis process and its results. These 1985 EM process changes represent no change to the NRC-accepted methodology. Conservatism is maintained through the many conservative aspects of the 1985 EM and through the other discretionary conservatisms listed in Section 2.1.1.1.

] These results are shown in Tables 2.2-4 and 2.2-5. Table 2.2-5 shows that the hot rod PCT during the late reflood [

]

2.2.4 Applicability to LBLOCA Analysis

The [ ] film boiling correlation is [

]

2.2.5 Model as Coded

2.2.5.1 CEFLASH-4A Code

The [ ] correlation is implemented in the CEFLASH-4A code in a heat transfer calculations subroutine. The correlation is implemented in the form given in Equation (2.2.1-1) with the following [

]

Table 2.2-4

Effect of Removing the Dougall-Rohsenow Correlation from CEFLASH-4A which Analyzes the Hot Assembly Average Rod during the Blowdown Portion of the LBLOCA Transient

[

]

Table 2.2-5

Effect of Removing the Dougall-Rohsenow Correlation from STRIKIN-II which Analyzes the Hot Rod During the Entire LBLOCA Transient

[

]

]

2.4.1.3 Sectionalized Steam Generator Model

[

.]

i. Sectionalized Secondary Side Temperature Model

[

]

ii. Steam Generator Tube Temperature Model

[

]

iii. Steam Generator Tube Primary Side Temperature Model

[

]

iv. Primary Side Heat Transfer Coefficients

The primary side heat transfer coefficients for each axial section in the steam generator tubes are calculated with the [

]

All properties are evaluated at the bulk temperature for section i.

The overall heat transfer coefficient on the primary side of the tubes for section i is calculated as follows:

[

]

v. Secondary Side Heat Transfer Coefficients

The heat transfer coefficients for each axial section on the steam generator tubes secondary side are calculated with the[

]

The overall heat transfer coefficient on the secondary side of the tubes for each section is calculated as follows:

[

]

2.4.2 Model Assessment

2.4.2.1 Steam Generator Model Performance for LBLOCA Analysis

Implementation of the 1999 EM steam generator model to[

] An evaluation of the system response to the implementation of the secondary side model on a COMPERC-II LBLOCA calculation is shown in Figures 2.4.2.1-1 through 2.4.2.1-16. In these figures, the performance of the 1985 EM is compared to the 1999 EM. These comparisons are intended to illustrate the effect and

]

i. FLECHT-SEASET Test No. 20904 (Simulation with Inlet Quality equal to 0.798)

Test Case 20904 is a high pressure, low flow, mid quality test. The test comparison is run with the following boundary conditions (Reference 2.4-3, Page 20904-1 or Reference 2.4-2, Table 4-2A):

Boundary Condition	Value
SG Tubes Pressure (psia)	60.4
SG tubes total flow (lbm/sec)	0.494
SG tubes inlet quality	0.798
SG pressure (secondary side, psia)	850.
Secondary side void fraction	0.874

Test comparisons using the special COMPERC-II calculations are shown in Figures 2.4.2.3.2-1 through 2.4.2.3.2-4.

Figure 2.4.2.3.2-1 shows the primary side steam generator tube temperature response as a function of distance from the inlet every 150 seconds.

Figure 2.4.2.3.2-2 shows the secondary side fluid temperature along the steam generator tubes as a function of distance from the inlet every 150 seconds. The results are [ ] with test data (Reference 2.4-3, Page 20904-17). The steam generator secondary side fluid temperature at the tube exit [ ] Measurements during the FLECHT-SEASET experiments showed that the steam exits the steam generator tubes wet.

Figure 2.4.2.3.2-3 shows the comparison of the calculated secondary side fluid temperature along the steam generator tubes at 1500 seconds against the test results from Reference 2.4-3,

Page 20904-17. Figure 2.4.2.3.2-3 shows that [

]

Figure 2.4.2.3.2-4 shows the transient response of the secondary side fluid temperature at the steam generator tube exit, and shows that the [

] The reference data is again plotted from the data in the figures of Reference 2.4-3, Page 20904-17.

Comparison of the COMPERC-II results [

]

ii. *FLECHT-SEASET Test No. 22213 (Simulation with Inlet Quality equal to 0.797)*

Test Case 22213 is a mid pressure, high flow, mid quality test. The test comparison is run with the following boundary conditions (Reference 2.4-3, Page 22213-1 or Reference 2.4-2, Table 4-2A):

Boundary Condition	Value
SG Tubes Pressure (psia)	40.0
SG tubes total flow (lbm/sec)	0.991
SG tubes inlet quality	0.797
SG pressure (secondary side, psia)	841.
Secondary side void fraction	0.837

Test comparisons using the special COMPERC-II calculations are shown in Figures 2.4.2.3.2-5 through 2.4.2.3.2-8.

]

Minkowycs and Sparrow (Reference 2.4-6) obtained analytical solutions for flow over vertical cylinders when the above criteria is not met. They have shown that for[

]

b. The [ ] Correlation

The [ ] correlation is used to calculate heat transfer to the steam phase in the steam generator tubes. The correlation was developed from data for heating and cooling in tubes in Reference 2.4-4. Its applicability to the turbulent region (Re > 6000) has been confirmed by experiments to within $\pm$25% (Reference 2.4-11)

2.4.5 Model as Coded

The steam generator tube model coding follows the description of the model in Sections 2.4.1.3 (iii) and 2.4.2.3.1.

The steam generator secondary side model coding follows the description of the model in Sections 2.4.1.2 and 2.4.1.3 ((i) and (ii)).

The [ ] correlation (Section 2.1.4.3 (v)) and the [ ] correlation (Section 2.1.4.3 (iv)) are implemented as described.

Table 2.4.3-1

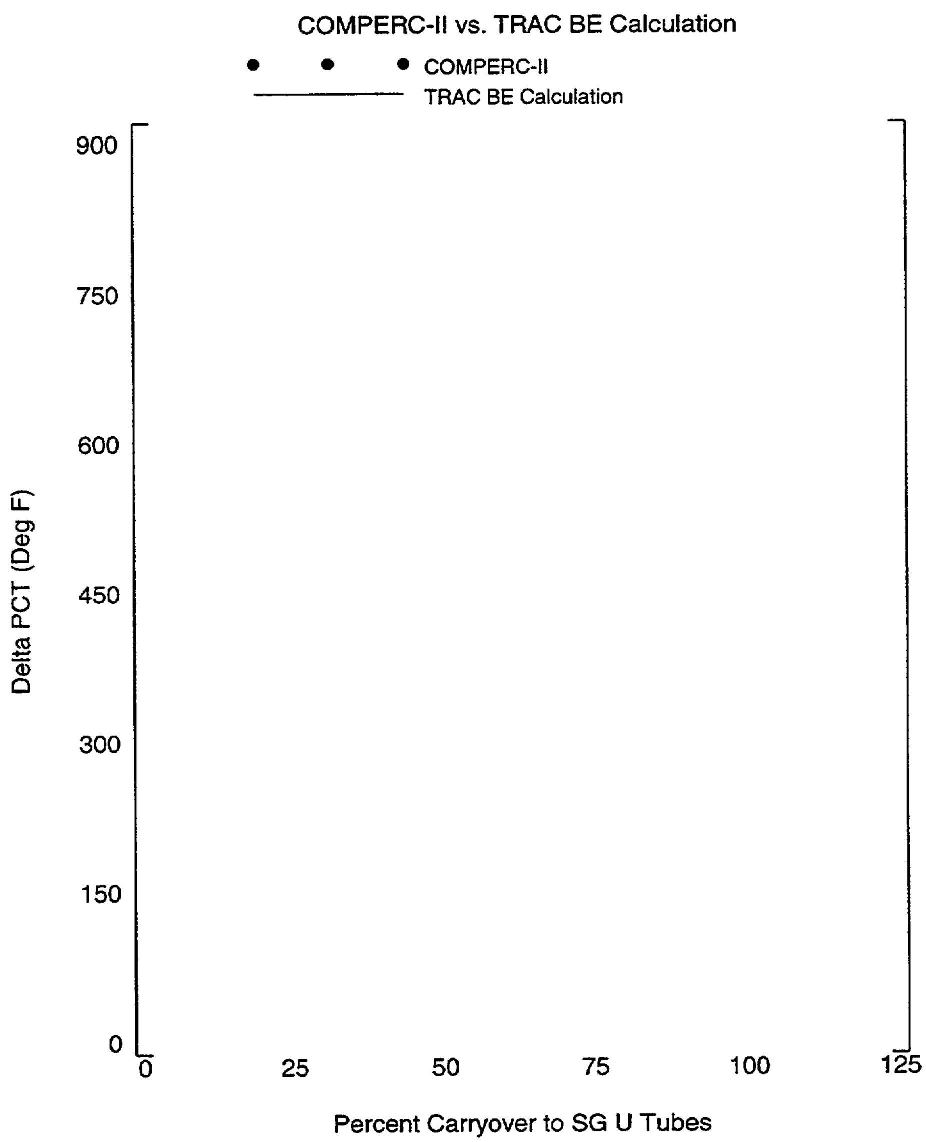
Effect of Activating the 1999 EM Steam Generator Model in COMPERC-II

[

]

Figure 2.4.2.3.4-2

Effect of Entrainment on PCT



2.5.4 Applicability to LBLOCA Analysis

- The discussion in CENPD-132 Volume 1, Section III.D.5 (Reference 2.5-2) includes results from one-fifth and one-third scale models of the cold leg piping, and showed that ECCS injection can be described[

  ]

- The 2D/3D Test Program (Reference 2.5-6), and the discussion in Section 2.5.2, show that the realistic observed effect of the nitrogen discharge process is a net addition of water into the core. Since the 1999 EM injection section delta pressure model reduces the amount of water entering the core during the time of nitrogen injection, the model [ ]

2.5.5 Model as Coded

The COMPERC-II nitrogen blowdown delta pressure is calculated as described. The nitrogen and liquid mixture velocity in the safety injection line is calculated using Equations (2.5-5), (2.5-6) and (2.5-7).

The liquid, steam and nitrogen mixture velocity in the cold legs is calculated using Equations (2.5-2), (2.5-3) and (2.5-4).

The differential pressure across the injection line is calculated using Equation (2.5-1).

Table 2.5-1

Effect of the 1999 EM Nitrogen Release Flow Resistance Model on PCT

Case	Reduction in PCT °F	Comparison of time of one inch/sec (1985 EM vs. 1999 EM) Seconds	Time of end of nitrogen blowdown (1985 EM and 1999 EM) Seconds
[			
			]

2.7.4 Applicability to LBLOCA Analysis

- The STRIKIN-II rod-to-rod radiation model is an approved NRC model for LBLOCA analysis. The STRIKIN-II calculated heat flux is implemented into PARCH and is applied fully consistent with the STRIKIN-II application.
- The use of the minimum of the FLECHT and the steam cooling heat transfer coefficients in the PARCH code for the rupture node is consistent with the SER requirement on the STRIKIN-II calculation.
- The HCROSS steam cross-flow model for the calculation of flow blockage and flow redistribution is also a NRC approved model for LBLOCA calculations.

2.7.5 Model as Coded

2.7.5.1 Implementation of the STRIKIN-II Rod-to-Rod Radiation Heat Flux into PARCH

The rod-to-rod radiation heat flux is calculated in STRIKIN-II. The radiation heat flux is transferred to PARCH at each time step and each STRIKIN-II axial node through the heat flux interface variable in units of Btu/hr-ft^2.

The PARCH program transposes the radiation heat flux array from the STRIKIN-II nodalization into an array for the PARCH nodalization by[

]

To implement the radiation heat flux into the integration of the fuel rod temperature equations in PARCH, the term b3 in Equation (3.2.1-21) in CENPD-138, Reference 2, is modified as follows:

[

]

2.7.5.2 Transfer of FLECHT Heat Transfer Coefficients for the Rupture Node to PARCH

STRIKIN-II calculates the heat transfer coefficients for all nodes during the reflood period. For the hot rod nodes equal to or above the rupture node elevation, the required heat transfer coefficient is equal to [

] In PARCH, the interface heat transfer coefficient is temperature corrected using the steam temperature, and is used to bound the new steam cooling heat transfer coefficient.

2.7.5.3 Transfer of all HCROSS Steam Cross-Flows to the PARCH Steam Channel

The steam cross-flows are calculated in HCROSS as axial flow fractions. The PARCH code reads the HCROSS flow fractions and defines the PARCH code flow fraction variables. The steam flow at each elevation is defined at each HCROSS axial node. This flow rate is [

.] The PARCH coolant energy balance is calculated using the same methodology as the 1985 EM with a finer nodalization.

that no other SER for the 1985 EM or earlier versions of the EM implied any such similar limitation. Moreover, CE Nuclear Power LLC believes that the 1985 EM and the 1999 EM are in fact applicable to analysis of mixed core configurations.

Resolution of this issue is also important because of the acquisition of CE Nuclear Power LLC by Westinghouse. This means that the fuel for Combustion Engineering PWRs will be manufactured at a Westinghouse facility and could therefore be considered Westinghouse manufactured fuel, not CE manufactured fuel.

2. **Referencing CENPD-133 Supplement 4-P**

The SER for the 1985 EM, Reference 3.0-1, inadvertently failed to cite in its reference list one of the topical report supplements that comprise the 1985 EM; and which will also comprise the 1999 EM. During our research for the 1999 EM model changes documented in this submittal, no evidence was found regarding NRC disposition of CEFLASH-4A Supplement 4. It is possible that the supplement did not require an SER since it impacted PCT by less than the 20°F limit imposed in the 1970's. Furthermore, the exact model change had been made earlier to the STRIKIN-II code and was approved by the staff. Nevertheless, there should be closure to this issue since the submitted change is made in order to conform to the Appendix K requirement on "no return to nucleate boiling." The 1999 EM automated/integrated code system licensing basis includes the referenced supplement since it is part of the 1985 EM.

Table 3.3-4

1999 EM Parameter Study

Safety Injection Pump Actuation Time

[

Result	Case (1) 1999 EM SI on Early Reflood	Case (2) 1999 EM SI on at TAD	Case (3) 1999 EM SI on after SITs Empty

]

iii. 1999 EM with Failure of a Diesel Generator

This analysis utilizes all of the proposed 1999 EM improvements including the automated/integrated code system with the automatic spray and spillage model. ECCS delivery is represented by the minimum injection to the cold legs from one LPSI pump and one HPSI pump. Containment spray delivery is represented with one spray pump, and for conservatism, maximum delivery is assumed. Table 3.4-2 shows that the PCT for this case is [ ]which is a[ ]relative to the case with no ECCS component failure.

Figures 3.4-5 through 3.4-8 show the impact of the failure of a diesel to the previous cases with no failure and loss of a LPSI. [

]

3.4.4 Summary of Worst Single Failure Study

In summary, this worst single failure analysis shows that no failure of an ECCS component produces the highest PCT for the 1999 EM. Consistent representation of ECCS injection to the cold legs and spillage of ECCS inventory to containment using the automatic spray and spillage model produces a worse single failure response that is slightly different than that calculated in the 1985 EM reference case, which is characterized by a conservative representation of these effects. The PCTs for the 1999 EM worst single failure cases were [ ] with the largest difference being [ ] Other plant configuration combinations of containment size and ECCS delivery rates, may lead to a different conclusion, therefore, the worst single failure analysis will be performed for each application of the 1999 EM.

Appendix A

Input and Output Descriptions for the 1999 EM Automated/Integrated Code System

A.1 The User Control Interface Input File

The User Control Interface (UCI) file is a single file containing input variables that control model options in CEFLASH-4A (Reference A-1), COMPERC-II (Reference A-2), STRIKIN-II (Reference A-3), PARCH (Reference A-4), and HCROSS (Reference A-5). The UCI file consists of line entries in free format. The first entry of each line describes the model option identifier (an alphanumeric variable without blanks). The second entry is a numeric variable identifying the option. Trailing comments are optional. The list of UCI variables is given in Table A-1. If a UCI input variable is not entered in the file, then its value will default to either zero or to the default value indicated in Table A-1.

Table A-1 also lists required and optional input options for executing the LBLOCA computer codes in one of the following three modes:

- 1985 EM fully compliant with currently approved EM
- 1985 EM with process improvements documented in this topical report fully compliant with currently approved EM
- 1999 EM as described by this topical report.

Table A-1
User Control Interface (UCI) Input Variables

Model ID	Model Used by Code	Standard Input 1985 EM	Optional Input 1985 EM Process Improved	Required Input 1999 EM	Description

Table A-3
Interface File Content for
f4a.cml.jobid

[

Line ID	Typical Value	Description	Units

]

Table A-4
Interface File Content for
cml.htcof.jobid

Line ID	Typical Value	Code	Description	Units

Table A-6

Interface File Content for

fre.htcof.jobid

[

]

Interface File Content for

fre.cmz.jobid

[

]

Table A-8

Interface File Content for

f4a.cmz.inp.jobid

[

]

$= \rho \beta g L^3 \Delta T / \mu^2$

L	= *Characteristic length (ft)*
Pr_L	= *Prandtl number*
ρ	= *Density* (lbm/ft^3)
β	= *Thermal expansion coefficient*
g	= $32.17\ (lbm\ ft / lbf\ sec^2)$
ΔT	= $T_{wall} - T_{liq}$ (°F)
μ	= *Viscosity (lbm/sec ft)*
D	= *Hydraulic diameter (riser region, ft)*

This part of the RAI response will explain the merit of using Equation (2.4.1.3-1) instead of the Eckert-Jackson correlation in the 1999 EM.

Equation (2.4.1.3-1) is the model used in several instances for modeling free convection in the ABB CENP large break and small break evaluation models. It was used in the 1999 EM for consistency and convenience. Its choice for the 1999 EM was justified on Page 2.4-35 in Section 2.4.4.iii.a.

In order to evaluate the merit of using Equation (2.4.1.3-1) vs. the Eckert-Jackson correlation, the correlations were compared over a wide range of conditions that included approximately 900 points, with pressure ranging from 500 to 1000 psia, liquid temperatures from saturation to 100 °F subcooling and wall-to-liquid delta temperatures ranging from 0 °F to 50 °F. The results of this comparison are shown in Figure D.2-1. The Eckert-Jackson correlation was derived from heat transfer data that was in good agreement with experimental data in the range of Grashof numbers from 10^{10} *to* 10^{12}. *The formula may be used for higher Grashof numbers (Reference D.2-2). Grashof numbers used for the above comparison range from* 10^{12} *to* 10^{14}.

The Eckert-Jackson correlation and Equation (2.4.1.3-1) are of [

]

The COMPERC-II FLECHT-SEASET simulation described in Sections 2.4.2.2 and 2.4.2.3.2 used Equation (2.4.1.3-1) to calculate the secondary side free-convection heat transfer coefficients. The COMPERC-II results for FLECHT-SEASET Test 22920 (Section 2.4.2.2 (iv)) demonstrate the adequacy of Equation (2.4.1.3-1) used for the secondary side. This test is a pure steam test at the SG tube inlet and shows that the COMPERC-II SG model realistically calculates the results of the test including the secondary side temperature distribution and heat transfer (see Figures 2.4.2.2-11 and 2.4.2.2-12).

(The COMPERC-II simulation of the other FLECHT-SEASET tests (Sections 2.4.2.2 (ii) and (iii)) are [

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Appendix I

NOTES FOR NOVEMBER 3, 2000
NRC MEETING REGARDING THE 1999 EM

Reference: LD-2000-0058, P. W. Richardson (CENP) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Notes for November 3, 2000 NRC Meeting Regarding CE Nuclear Power Large Break LOCA 1999 EM," November 20, 2000

Appendix I contains the CENP meeting notes prepared to be used as talking points during the meeting, which were submitted to NRC for the record and for NRC information and use in preparing the SER for the 1999 EM.

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Westinghouse Electric Company
CE Nuclear Power LLC

2000 Day Hill Road
Windsor, CT 06095
USA

20 November, 2000
LD-2000-0058

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

SUBJECT: NOTES FOR NOVEMBER 3, 2000 NRC MEETING REGARDING CE NUCLEAR POWER LARGE BREAK LOCA 1999 EM
(Enclosure 1-P Contains Proprietary Information)

Reference(s):
1) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Response to Questions Regarding CENPD-132, Supplement 4-P, Rev. 1", LD-2000-0057, November 13, 2000
2) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", August 2000
3) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model", LD-2000-0046, August 30, 2000

On November 13, 2000, CE Nuclear Power LLC (CENP) submitted responses to NRC questions that resulted from a meeting held on November 3, 2000 (Reference 1). The purpose of the November 3, 2000 meeting was to discuss the ongoing review of CENPD-132, Supplement 4-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (Reference 2) and NRC questions on the Revision 1 of the topical report. This topical report was submitted to the NRC on August 30, 2000 (Reference 3). Pursuant to prior agreement, CENP is furnishing one (1) proprietary and one (1) non-proprietary copy of this letter and enclosures to the NRC Document Control Desk and three (3) proprietary copies to Jack Cushing, NRC, CENP Project Manager.

Enclosure 1-P contains a copy of the meeting notes prepared by the CENP representatives to be used as talking points during the meeting. CENP is submitting a copy of these notes for the record and for NRC information and use, as needed. The non-proprietary responses are provided in Enclosure 3.

CENP has determined that the information provided in Enclosure 1-P is proprietary in nature. Consequently, it is requested that Enclosure 1-P be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that this information be appropriately safeguarded. The reasons for the classification of this information as proprietary are delineated in the affidavit provided in Enclosure 2.

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
CE NUCLEAR POWER LLC

Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Enclosure(s): As stated

xc: J. S. Cushing (NRC, with 3 copies of Enclosure 1-P)

Enclosure 3 to LD-2000-0058

CE NUCLEAR POWER LLC

NON-PROPRIETARY NOTES FOR NOVEMBER 3, 2000 NRC MEETING REGARDING CE NUCLEAR POWER LARGE BREAK LOCA 1999 EM

November 2000

CE Nuclear Power LLC
NRC Technical Review Meeting
November 3, 2000

Licensing Review of CENPD-132 Supplement 4-P Revision 1

AGENDA

8:00	Opening Statement by NRC
8:05	Opening Statement by CENP
8:10	Steam Venting Model
9:30	Nitrogen Discharge Model
10:00	Reflood Heat Transfer Model
10:30	Break
10:45	Discuss the Following Items as Needed
	Automated/Integrated Code System
	Replacement of Dougall-Rohsenow
	Hot Assembly Gap Internal Gas Pressure
	Hot Rod Steam Cooling Heat Transfer
	Section 3 - Application Analyses and Other Items
12:00	Lunch
1:00	Clarification and Review of Issues
2:30	Break - CENP Independent Discussion
3:00	Meeting Summary and Finalize Action Items
4:30	Adjourn

CE Nuclear Power LLC
NRC Technical Review Meeting
November 3, 2000

Licensing Review of CENPD-132 Supplement 4-P Revision 1

List of Attendees

Name	Representing	Phone Number
Ernie Jageler	Westinghouse Electric Co.	860-285-2289
Ruben Espinosa	Westinghouse Electric Co.	860-285-4990

Opening Statement

- We are prepared to address and close out any and all questions or issues if possible
- We are prepared to take action items to resolve any remaining items
- We are prepared to meet your requests for supporting information or references
- We have no presentation slides but will speak extemporaneously
- We have one-page bullet summaries of each of the major model elements of our submittal, which Ruben and I will refer to if necessary to cover our points. These pages are essentially the outline of an SER that we would write highlighting the major changes are why they are acceptable.
- We have prepared material to answer Gene Hsii's questions (LN from 10/30) with supporting documentation and reference material
- How best to capture the results of this meeting
    - Copy of materials that we show transmitted later with the meeting summary
    - Add results of any new cases that are needed to the meeting summary
    - Add the meeting summary as an Appendix to the topical report
- Have also brought reference materials with us
    - User documentation topical reports
    - SER's
    - Code listings
    - Selected case results
- Recommend that NRC speak first in each topic area to express the question or concern that needs to be addressed
- Recommend that CENP speak last to summarize closeout or action item if needed

Update Large Break LOCA Appendix K Evaluation Model – 1999 EM

- Reduce Conservatism in Selected Models to Obtain PLHGR Margin Improvement
  1) Process Changes Within the Currently NRC-Approved EM
     - ❖ Automated/Integrated Code System
     - ❖ Explicit NUREG-0630 Cladding Swelling/Rupture
     - ❖ Consistent Modeling of Spray and Spillage into the Containment
  2) NRC-Required Changes
     - ❖ Replacement of Dougall-Rohsenow
  3) Changes Requiring Licensing Review
     - ❖ Hot Assembly Fuel Rod Internal Gas Pressure (thermal-hydraulic blowdown code)
     - ❖ Steam Venting Reflood Thermal-Hydraulics (reduced steam superheat at SG primary exit)
     - ❖ Steam/Water Interaction During Nitrogen Discharge (reduced injection section pressure drop)
     - ❖ Reflood Heat Transfer (improved time-shift term for multiple reflood rates)
     - ❖ Hot Rod Steam Cooling Heat Transfer (improved spatially-dependent data transfer)

Licensing Process Schedule

- Topical Report Submittal
  - April 30, 1999
- NRC Acceptance Review
  - October 4, 1999
  - Scheduled Completion Date: December 31, 2000
- NRC Request for Additional Information
  - December 14, 1999
  - Response to RAI Transmitted to NRC: February 22, 2000
- NRC Suspends the Licensing Review Process
  - March 15, 2000
  - Restart September, 2000
  - Brief meeting on October 17, 2000
  - Scheduled Completion Date in Jeopardy
    - Agreed to review SER for proprietary material
    - Agreed to provide reference material at NRC request
    - Agreed to face-to-face technical review meeting
- NRC Approval of Topical Report Needed to Support Plant Analyses
  - December 31, 2000

NRC Guidance from November 17, 1998 Planning Meeting

- No objections to technical aspects of planned EM revisions
- Meeting with CENP customers in January was unnecessary
- Licensing review would be done in-house
- Follow-up conference call on November 18, 1998; CENP received the following SEVEN ITEM NRC POLICY GUIDANCE
  1. NRC acceptance review based on accomplishing licensing with one round of RAIs
  2. NRC requires transmittal of the EM computer codes. CENP provided technical description of workstation requirements
  3. NRC policy allows changes within the context of already approved methods without additional NRC approval, but documentation and configuration control must be in place such that the changes could be audited
  4. Plans to revise the use of discretionary conservatism were acceptable to NRC provided documentation was in place for audit
  5. NRC's policy takes strong exception to non-physical, non-conservative changes that are a consequence of Appendix K modeling
  6. NRC accepts demonstration of retained conservatism as a valid approach to licensing a change, but usually in combination with other justification arguments.
  7. NRC strongly suggested that the model changes be submitted as soon as possible to meet time line discussed at the 11/17/98 planning meeting

Utilities That Will Take Advantage of Improved Model

- Entergy – Arkansas Nuclear One – Unit 2
- Arizona Public Service – Palo Verde Units 1, 2, & 3

Technical Justification of Model Change

COMPERC-II Steam Venting Thermal-Hydraulics

What was changed?

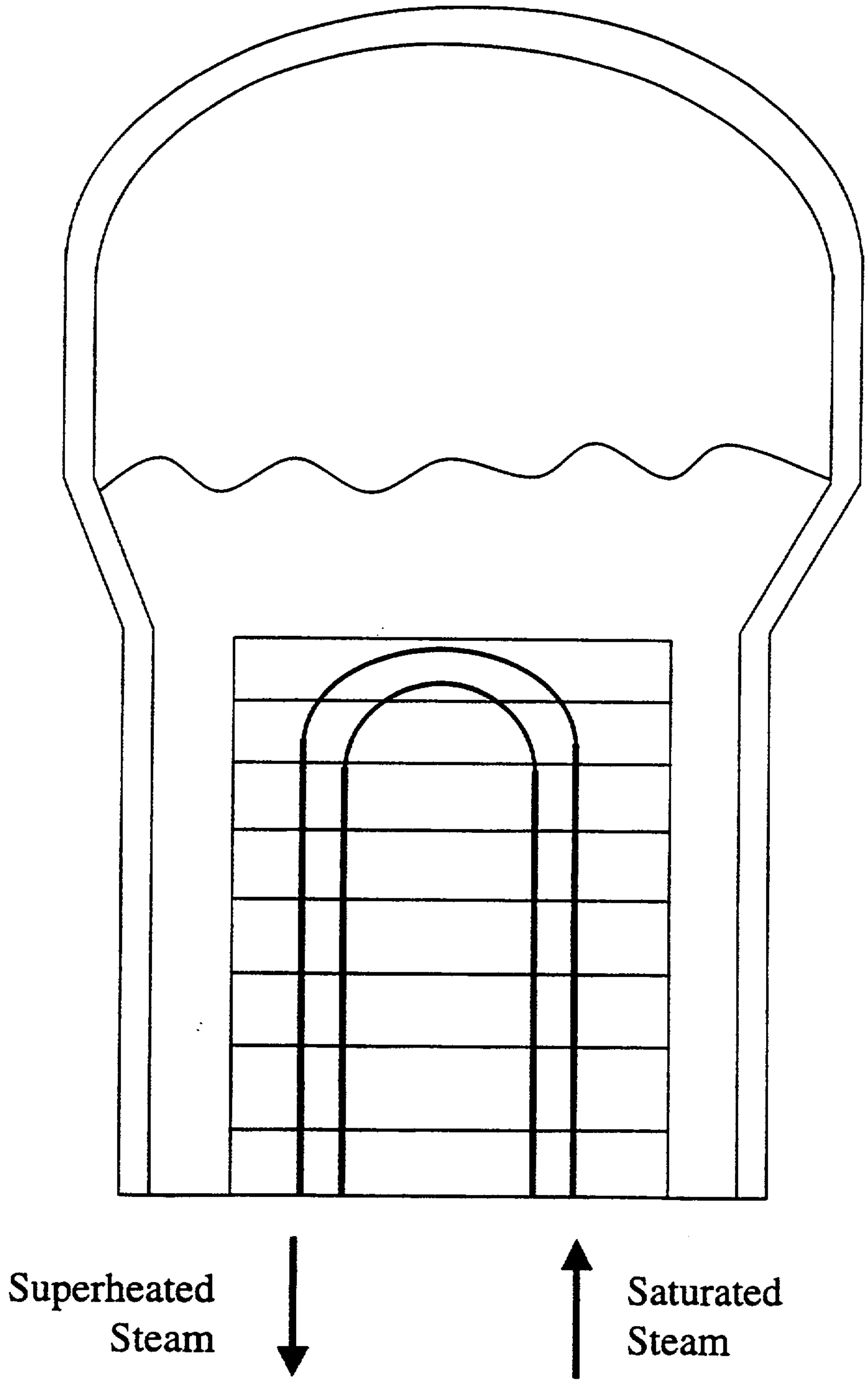
- SG secondary side model was added to COMPERC-II. Model replaces conservative infinite heat source with a finite heat source during reflood period.
- Model represents thermal stratification of SG secondary side consistent with experimental observations.
- Heat transfer implemented using representative correlations.
- Model implemented using conservative assumptions.
- Model reduces the SG tube exit temperature which then reduces the resistance to steam flow, and increases the reflood rate.

Why is the change reasonable?

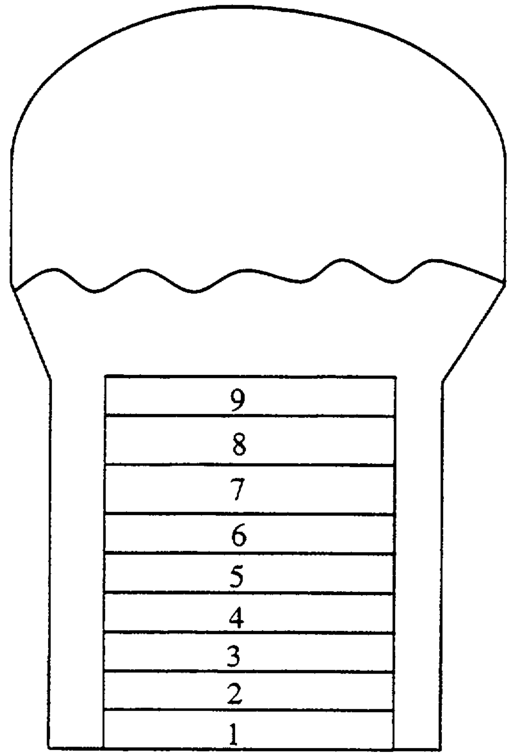
- Calculated system response consistent with experimental observations in FLECHT-SEASET tests.
- Model validated by comparison to experimental FLECHT-SEASET test data.
- Model implemented using conservative assumptions. Verified that removal of overly conservative assumptions would significantly improve comparison to experimental data and reduce PCT.
- Appendix K conservative assumptions were not removed or modified.
- TRAC BE calculations show that Appendix K assumptions have at least a 300 °F margin of conservatism on PCT.
- Changes yield a[ ]
- Demonstrated in RAI that results are not significantly affected by choice of heat transfer correlations.

Effect of the Change

- A[ ]
- Calculation of secondary side[ ]consistent with experimental data.
- Conservative calculation of FLECHT-SEASET test data[ ]

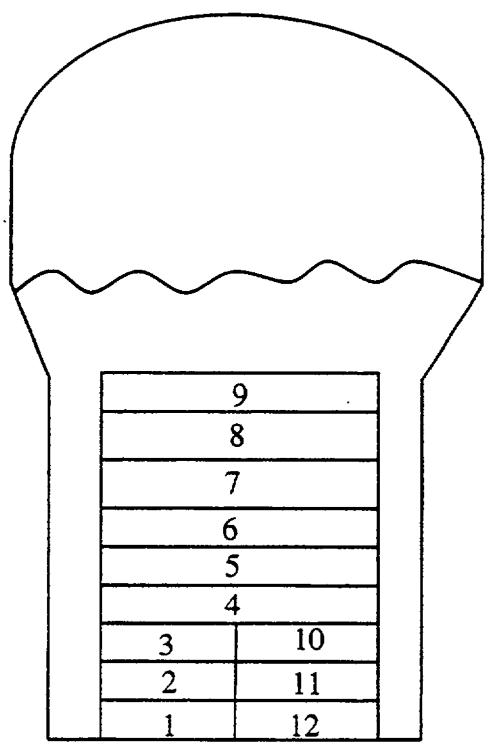


Superheated
Steam
Saturated
Steam



SG Secondary Side Without Economizer

SG_NSEC=9, SG_IECON=0



SG Secondary Side With Economizer

SG_NSEC=9, SG_IECON=3

Explanation of the [ ] Assumption at the SG Inlet

- The [ ] assumption at the SG inlet was not modified. It is the same assumption used in the previous version of the code.
- Comparison of loop fluid conditions between old version and new version of code

Model	Previous Version	New Version
Loop Flow	[	
Hot legs		
SG tubes		
Suction and Discharge legs		]

- Large amount of entrained liquid supports assumption that [
  -
  -
  -

  ]

- Hand calculations to illustrate that [ ] at the SG inlet.

  For a LBLOCA reflood calculation, code calculates approximately [

  ]

- Above hand calculation does [

]

The Effect of the New Steam Venting Model on PCT

- The new steam venting model results in a [ ]
- Reasons for the [ ] in PCT
  - [
  -
  -
  ]
- Quantification of the effect
  - [
  -
  -
  ]
- Observations on the Effect of the individual models on PCT
  - [
    -
    -
  -
  -
    -
    -
  ]

Technical Justification of Model Change

Modification of COMPERC-II Steam/Water Interaction during Nitrogen Discharge

Summary

- The scope of the modifications for the injection section delta pressure calculations is limited to the time of nitrogen blowdown. It does not affect the time of SIT discharge or the time of SI pump injection.
- The modified model remains consistent with current SER constraints. The modifications remove some of the conservatism that was left in the model relative to the SER requirements.

History of the Model for all three reflood conditions (SIT, Nitrogen discharge, SI pump)

- [
-
-
-
-]

What was changed?

- COMPERC-II calculation of the injection section delta pressure during nitrogen discharge was replaced with a[ ]
- The injection section delta pressure is the[ ]

Why is the change reasonable?

- There is no experimental evidence that injection section delta pressure increases during period of nitrogen discharge. 2D/3D Program shows the opposite. Nitrogen addition results in a net addition of water into the core. TRAC simulations of PWR and UPTF tests suggest that nitrogen discharge and the resulting surge in core water level are beneficial to core cooling. In the Achilles test, the surge of water into the core also enhanced core cooling.
- On the other hand the injection section delta pressure calculated during nitrogen discharge with the previous model is [ ]
-
- The applicable SER does not explicitly require use of the [ ]
- The SER required delta pressure limit values during SIT discharge and SI pump discharge bound the injection section experimental delta pressure data given in CENPD-132 Volume I, and Supplement 2 to CENPD-132. No data points greater than [ ] were observed in the tests, including cases with non-condensibles injection.
- The injection section delta pressure calculated by the new model during nitrogen injection is larger than the required delta pressure at the end-points.

Effect of the Change

- PCT can be [ ] for plants where one inch/sec time coincides with initiation of nitrogen blowdown. The [ ]
- The injection section delta pressure is calculated consistent with the end-point delta pressure values used by the code.

Reflood Heat Transfer

- Improve the MOD-1C reflood heat transfer model
  - The basic FLECHT correlation for constant flooding rates that is contained within the MOD-1C methodology is unchanged
  - The MOD-2C procedure is changed by improving the "time-shift" equation for variable flooding rates utilizing FLECHT-SEASET and CCTF test data
- The time-shift equation is both empirical and mechanistic
  - The equation contains an empirical elevation dependent correlation adjustment multiplier
  - A reflood mass integral represents the long term flooding rate impact inherent in the constant flooding rate heat transfer correlation
  - Only the empirical elevation dependent correlation adjustment multiplier is improved using the newer FLECHT-SEASET and CCTF test data
  - The new model is developed from one test condition then benchmarked against all of the other test data for multiple reflood rates
- The model change is assessed against all available multiple reflood rate test results from the FLECHT-SEASET and CCTF test programs
  - The range of applicability for the new reflood heat transfer model is expanded compared to the current NRC-accepted model
    - The range of applicability of the basic FLECHT heat transfer correlation is unchanged
    - The original MOD-2C procedure was benchmarked against 3 FLECHT tests with multiple flooding rates
    - These 3 FLECHT tests were forced flooding with only two reflood intervals
    - The second interval was 1 in/sec flooding rate only, only one power level was covered, with very limited ranges in pressure (55-64 psia), peak temperature and inlet subcooling
    - Comparisons were primarily at only one elevation (8 ft) with only selected comparisons at one other elevation (6ft)
  - FLECHT-SEASET data is available for both forced flooding and gravity feed
    - Five of six multiple reflood tests in the unblocked test series were used
    - Two forced reflood tests with variable flooding rates and three gravity reflood tests, one of which simulated a different initial axial temperature distribution
    - In some tests, 3 flooding intervals were simulated with flooding rates as low as 0.62 in/sec, and pressure was extended to as low as 20 psia
    - One test had radial power variation, another had low coolant temperature
    - Test data from these tests were available at eight elevations between 6.49 ft and 11.61 ft elevation, covering 36 separate transient profiles at different elevations and rod locations

- CCTF data was available for full-height core and RCS reflood simulation
    - Four tests were selected based on availability of data and adequacy of test conditions for the range of LBLOCA conditions
    - Test data from these tests are available at four elevations between 3 ft and 10 ft elevation, covering 16 separate transient profiles
- For both FLECHT-SEASET and CCTF conservatism is assured
    - Test measurements and the heat transfer coefficients calculated from the measurements represent the heat transfer coefficients from the highest measured temperatures
    - Fitting of the empirical adjustment multiplier leads to overall conservative comparison to the data
    - That is, the average result of the revised model is the prediction of a lower heat transfer coefficient than measured

- The revised reflood heat transfer model [ ]
    - Depending on location of rupture, the FLECHT heat transfer cooled node PCT is [ ]
    - Depending on location of rupture, the steam heat transfer cooled node PCT is [ ]
- Conservatism of the revised reflood heat transfer model is assured
    - The basic FLECHT heat transfer correlation for constant flooding rates is unchanged and is known to be conservative relative to data and covers the range of parameter variation required for LBLOCA
    - Test data comparisons from multi-rod and multi-assembly test facilities are made only on the test rods with the highest measured temperatures
    - The correlation approach is conservatively biased to underpredict the measured heat transfer coefficients

FRELAPC Computer Code

- Analytical FLECHT Rod Elevation and Power Correction Program
  - Methodology documented in CENPD-213, "Reflood Heat Transfer: Application of FLECHT Reflood Heat Transfer Coefficients to C-E's 16x16 Fuel Bundles," January 1976.
  - Methodology NRC-accepted in SER: K. Kniel (NRC) to A. E. Scherer (C-E), "Application of FLECHT Reflood Heat Transfer Coefficients to CE's 16x16 Fuel Bundle," August 2, 1976
  - FRELAPC was developed to address NRC concerns regarding the application of the FLECHT heat transfer correlation to axially skewed power shapes and to fuel assembly designs different from the FLECHT facility
- Analytical approach is mechanistic
  - Procedure based on an approach developed by L. J. Ybarrondo (Aerojet Nuclear Company), "An Empirical Flooding Heat Transfer Coefficient Including Quench Time Prediction Applicable to a 12-foot Long PWR Core and Including a Modification for Variable Flooding Rates, Short Cores, and Non-symmetrical Power Profiles," November 24, 1971
  - The method identifies the particular FLECHT rod elevation and power level at which the local coolant conditions (integral heatup) and surface heat flux of the STRIKIN-II (or COMZIRC) node of interest are precisely duplicated
  - FRELAPC generates a set of corrected FLECHT elevations and power levels for any fuel assembly design and axial power shape

Process Change within the Currently NRC-Accepted EM

- Three process changes that remain consistent with currently NRC-accepted EM and do not need NRC review
  - Automated/Integrated Code System
  - Explicit NUREG-0630 Cladding Swelling/Rupture
  - Consistent Modeling of Spray and Spillage into the Containment
- These changes will be implemented in both the 1985 EM and the 1999 EM
- Purpose of these changes is to reduce the introduction of conservatism by the analyst
  - Control the interface data transfer process between computer codes, precisely and consistently
  - Eliminate conservative bias in the use of NUREG-0630 cladding rupture and swelling models
  - Eliminate conservative manual estimation of spray and spillage into the containment during reflood by utilizing existing approved methodology with consistent modeling of sources of dispersed water

Technical Justification of Model Change

Replacement of Dougall-Rohsenow Film Boiling Correlation

What was changed?

- Dougall-Rohsenow film boiling correlation was replaced with [ ] correlation in CEFLASH-4A and STRIKIN-II

Why is the change reasonable?

- [ ] correlation developed using large set of data which covers the range of interest in LBLOCA analyses.

- Correlation evaluated at ORNL using THTF test data. [ ] correlation underpredicts most THTF data. Dougall-Rohsenow overpredicts most THTF data.

- Correlation endorsed by NRC in Compendium of ECCS Research for Realistic LOCA Analysis.

Effect of the Change

- Typical effect: PCT [ ]

Hot Assembly Fuel Rod Internal Gas Pressure

- Update the CEFLASH-4A model for the hot assembly fuel rod internal gas pressure
  - Replace constant fuel rod internal pressure with dynamic model
  - The purpose of the change is to achieve consistency between the hot rod heatup calculation of cladding rupture in STRIKIN-II with that in CEFLASH-4A
  - Inconsistent calculation of assembly blockage flow redistribution in CEFLASH-4A when fuel rod cladding rupture is not calculated during blowdown in STRIKIN-II is overly conservative
- Model implemented in CEFLASH-4A is the same model used in STRIKIN-II and previously accepted by NRC for the hot rod heatup calculation
  - Model based on first principles, ideal gas law, including effects of cladding and fuel dimensional changes due to thermal and mechanical expansion and contraction
  - For consistency between CEFLASH-4A and STRIKIN-II, the internal pressure is updated for each radial core region at each time step in response to pre-rupture cladding plastic strain driven by cladding heatup
  - Each fuel node gas gap has an average transient temperature and volume for each time step
- The CEFLASH-4A dynamic model is benchmarked against STRIKIN-II
  - A special case is run where the CEFLASH-4A and STRIKIN-II average rods are initialized with the same stored energy and internal gas pressure
  - The dynamic comparison is very similar for the two codes with CEFLASH-4A having[ ]over the majority of the blowdown transient time period
  - The transient pressures are[ ]
- The dynamic hot assembly fuel rod internal gas pressure model achieves consistent calculation of the time of rupture between CEFLASH-4A and STRIKIN-II
  - The STRIKIN-II hot rod PCT is[ ]compared to a case with inconsistent calculation of blowdown rupture in CEFLASH-4A but reflood rupture in STRIKIN-II
- Conservatism is assured through the conservative nature of the NUREG-0630 and Coffman swelling and rupture models, as required by Appendix K

Technical Justification of Model Change

Hot Rod Steam Cooling Heat Transfer Model Changes

What was changed?

- [
-

-

]

Why is the change reasonable?

- The STRIKIN-II [ ] which is a NRC approved model, was implemented into the PARCH. This implementation improves the consistency of the hot rod temperature calculations in PARCH and STRIKIN-II.
- The transfer of the [ ] improves the consistency between the STRIKIN-II and PARCH calculations.
- The transfer of [ ]

Effect of the Change

- Implementation of the [ ] produces consistent temperatures between STRIKIN-II and PARCH.
- Transfer of [ ] in STRIKIN-II PCT.
- The overall effect of the three changes described above is a [ ]

Section 3 ECCS Performance Analyses

- Consistent with all previous evaluation model submittals to NRC, the application analyses contained in the topical report are representative of the typical CE designed PWR
    - Every CE designed PWR is different
    - Classification is not the standard approach
    - Consistent with the past, this submittal for the LBLOCA evaluation model change is meant to be generic. The results are meant to be representative (not necessarily bounding).
    - Plant specific evaluation model submittals for LBLOCA ECCS performance have not been used previously by CENP
- The 1999 EM Automated/Integrated Code System remains in compliance with all SER's relevant to the 1985 EM and is fully compliant with all current SER constraints and limitations for the 1985 EM
    - Request removal of constraint on non-CENP manufactured fuel
    - Request closure on referencing CENPD-133 Supplement 4-P
- Request approval for using the 1999 EM AICS for 1985 EM analyses
    - Produces nearly the same PCT result as the stand-alone operation of the previous code versions
    - AICS process improvements for the 1985 EM simulation improve PCT results
- Application of the 1999 EM AICS in design analyses requires consideration of the impact of uncertainties in the input parameters compared to standard practice
    - [
    -
    -

  ]
- Worse single failure of ECCS component analyses for the 1985 EM and the 1999 EM are shown
    - For the 1999 EM, the worse single failure is[ ]
- Break spectrum analysis results for the 1999 EM show margin gain
    - [
    -
    -]

- An assessment of overall conservatism in the 1999 EM is given
    - Relaxed three Appendix K requirements
        - Reduce the decay heat multiplier from 1.2 to 1.0
        - Do not use the locked-rotor K-factor for steam venting through the RCS, but assume the pumps are free spinning
        - Do not assume that cooling must be only from steam when the reflood rate is less than 1 in/sec, but assume that FLECHT cooling still applies
    - PCT is [ ] for the non-EM cases in total
    - Since the 1999 EM [ ] it is concluded that the 1999 EM maintains an appropriate amount of overall model conservatism

Current Licensing Issues - 1985 EM

- **Applicability to non-CENP manufactured fuel**
  - One SER constraint (July 31, 1986, Crutchfield to Scherer) limits applicability to only CENP manufactured Zircaloy clad fuel
  - Need approval to apply LBLOCA EM to non-CENP manufactured fuel assemblies so that transition mixed core analyses can be performed in the future
  - CENP does not understand why our implementation of the NRC's required model, NUREG-0630, should be restricted to only CENP manufactured Zircaloy clad fuel
  - Request removing this constraint

Current Licensing Issues - 1985 EM

- **Referencing CENPD-133 Supplement 4-P (Submitted to NRC in April 1977)**
  - CENP did not implement the methodology change between April 1977 and July 1986 because we had not received the SER. Subsequently, the methodology change was made part of the 1985 EM.
  - The SER for the 1985 EM failed to cite in its reference list the supplement for this model change
  - A meeting between Brinkman (CENP) and Collins (NRC) was held on 11/4/98. CENP presented the topical report timeline, a copy of the relevant page from the NRC's status report showing the TAC still open, and a brief discussion of the subject of the topical report supplement.
  - NRC agreed to research how the TAC could have been closed without an SER or rejection letter. NRC has not yet responded.
  - CENP suspects that the reference was either inadvertently omitted from the SER reference list, which contained several other topical reports being accepted as part of the 1985 EM, or an SER was not required since the model change impacted PCT less than the 20°F limit imposed back in the 70's.
  - Both CENP and NRC agree that this issue should be rectified. Request final closure on the issue.

SER Constraints and Limitations

- CENPD-132 Supplement 4-P Revision 1 is acceptable for referencing.
- For completeness, this acceptability includes reference to CENPD-133 Supplement 4-P, submitted in April 1977, but previously omitted in the list of acceptable references.
- The 1999 EM Automated/Integrated Code System (AICS) is approved for use to perform an ECCS performance analysis consistent with the features described in the topical report.
- Also, the 1999 EM AICS is approved for use to perform an analysis consistent with the required and acceptable options and features of the previously accepted evaluation model, i.e., the 1985 EM, which is referred to as the 1985 EM simulation mode of operation including the process improvements available from the AICS at the discretion of CENP
- All current SER constraints and limitations for the 1985 EM will continue to apply for the 1999 EM with one exception:
    - Both the 1985 EM and the 1999 EM can be applied to ECCS performance design analyses with a reactor core containing non-CENP manufactured Zircaloy fuel
- As with all previous CENP ECCS performance evaluation models, the 1999 EM is acceptable for CE four loop plants, dry containments, only a bottom flooding ECCS, and only for ECCS performance analyses
- Also, as with all previous CENP ECCS performance evaluation models, the 1999 EM is acceptable provided a break size spectrum, a time-in-life sensitivity study, and a worst single failure sensitivity study are performed for each complete analysis

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Appendix J

CENP RESPONSE TO
NRC REQUEST FOR PROPRIETARY CONTENT REVIEW OF THE SER

Reference: LD-2000-0061, P. W. Richardson (CE) to U. S. Nuclear Regulatory Commission (Document Control Desk), "Response to NRC Request for Proprietary Content Review of CENPD-132, Supplement 4-P, Revision 1, Safety Evaluation Report," Enclosure 1, "Identification of Unprotected Proprietary Information in NRC Safety Evaluation Report for CENPD-132, Supplement 4-P, Revision 1, Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," December 19, 2000

Appendix J contains the identification of unprotected proprietary information in the NRC SER for this topical report submittal for the 1999 EM.

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Westinghouse Electric Company
CE Nuclear Power LLC

2000 Day Hill Road
Windsor, CT 06095
USA

19 December, 2000
LD-2000-0061

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

SUBJECT: RESPONSE TO NRC REQUEST FOR PROPRIETARY CONTENT REVIEW OF CENPD-132, SUPPLEMENT 4-P, REVISION 1, SAFETY EVALUATION REPORT [CONTAINS PROPRIETARY INFORMATION]

Reference(s):
1) Letter, S. A. Richards (NRC) to P. W. Richardson (CENP), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model' (TAC No. MA5660)", December 15, 2000
2) CENPD-132, Supplement 4-P, Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model", August 2000
3) Letter, P. W. Richardson (CENP) to U.S. Nuclear Regulatory Commission Document Control Desk, "Revision to CE Nuclear Power LLC ECCS Performance Appendix K Evaluation Model", LD-2000-0046, August 30, 2000
4) Letter, J. Cushing (NRC) to P. W. Richardson (CENP), "Request for Withholding Information from Public Disclosure – Westinghouse Electric Company CE Nuclear Power, LLC (TAC No. MA5660)", November 16, 2000

On December 15, 2000, the Nuclear Regulatory Commission (NRC) issued (Reference 1) its Safety Evaluation Report (SER) for the CE Nuclear Power LLC (CENP) Large Break LOCA 1999 Evaluation Model. This evaluation model is documented in topical report CENPD-132, Supplement 4-P, Revision 1 (Reference 2), which was submitted to the NRC on August 30, 2000 (Reference 3). The SER requested that CENP review the non-proprietary Enclosure 1 to assure that no proprietary information had been inadvertently included. CENP has conducted the requested review and has determined that the non-proprietary Enclosure 1 does indeed contain some proprietary information that was not appropriately blanked out.

Enclosure 1 to this letter provides a tabular listing of the affected pages, as well as the pages themselves, annotated to indicate the additional information that should be considered proprietary. CENP compared the SER information with the topical report and determined that the topical did indeed indicate, via brackets [ ], that the information was proprietary in nature and that it should be withheld from public disclosure. Therefore, CENP requests that the NRC not release Reference 1, Enclosure 1 to the public without first protecting the additional information identified herein. The proprietary affidavit filed with the original submittal of the topical report (Reference 3) delineated the reasons for declaring the information to be proprietary in nature. In a letter dated November 16, 2000 (Reference 4), the NRC concurred

with CENP's proprietary information declaration without exception and indicated that the delineated information would be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 285-5205.

Very truly yours,
CE NUCLEAR POWER LLC

Philip W. Richardson
Licensing Project Manager
Windsor Nuclear Licensing

Enclosure(s): As stated

xc: J. S. Cushing (NRC)

CE NUCLEAR POWER LLC

IDENTIFICATION OF UNPROTECTED PROPRIETARY INFORMATION IN NRC SAFETY EVALUATION REPORT FOR CENPD-132, SUPPLEMENT 4-P, REVISION 1 CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL

December 2000

The following additional text should be bracketed in the non-proprietary Enclosure 1 of the NRC's December 15, 2000 Safety Evaluation transmittal letter.

Item	Additional Text to be Bracketed	Corresponding Reference Text Previously Bracketed
1.	Page 4, 6th Paragraph The word " " on the 4th Line	TR Section 2.1.3 Page 2.1-12, 2nd Paragraph
2.	Page 8, 3rd Paragraph Beginning with and including the word " " on the 2nd Line and extending through and including the word " " on the 5th Line.	TR Section 2.4.1 Pages 2.4-3 through 2.4-4
3.	Page 8, 4th Paragraph The word " " on the 9th Line	LD-2000-0057 Page 5, 1st Bullet, 3rd Line
4.	Page 8, 4th Paragraph The word " " on the 10th Line	LD-2000-0057 Page 11, 1st Bullet, 1st Line
5.	Page 10, 4th Paragraph The word " " on the 11th Line	TR Section 2.4.2.3.4 Page 2.4-32, 4th Paragraph

CE NUCLEAR POWER LLC