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,

/RA/

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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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 **[**

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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 **[**

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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 <u>LBLOCA ECCS PERFORMANCE ANALYSIS</u>

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.

<u>Applicability of Non-ABB CENP Manufactured Fuel</u>

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."

<u>Referencing CENPD-133, Supplement 4</u>

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.

<u>SG Secondary Initial Pressure and Physical Parameters</u>

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.

<u>Safety Injection Pump Actuation Time</u>

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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(2) Letter from Ian C. Rickard, ABB Combustion Engineering Nuclear Power (LD-2000-LD-2000–0011) to U.S. Nuclear Regulatory Commission, "ABB CENP Response to NRC Request for Additional Information Regarding CENPD-132-P, Supplement 4-P," February 22, 2000.

(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.

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

(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.

(7) NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," April 1980.

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

(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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