July 24, 2000

Mr. Ian C. Rickard, Director
Nuclear Licensing
Combustion Engineering Nuclear Power
2000 Day Hill Road
P.O. Box 500
Windsor, Connecticut 06095-0500

SUBJECT: ACCEPTANCE FOR REFERENCING OF CENPD-390-P, "THE ADVANCED PHOENIX AND POLCA CODES FOR NUCLEAR DESIGN OF BOILING WATER REACTORS" (TAC NO. MA5659)

Dear Mr. Rickard:

We have concluded our review of the subject topical report submitted by Combustion Engineering Nuclear Power, LLC by letter dated April 15, 1999. The report is acceptable for referencing in licensing applications and analysis, subject to the limitations specified in the report and in the associated NRC safety evaluation (SE), which is enclosed. The SE defines the basis of acceptance of the report.

The topical report describes the changes made to the PHOENIX and POLCA Codes since they were previously reviewed and approved. It also provides assessment against operational data and measurements to demonstrate that the codes are capable of predicting power distributions, thermal limits and critical conditions necessary to design a boiling water reactor.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

We do not intend to repeat our review of the matters described in the report, and found acceptable, when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to matters described in the report.

In accordance with procedures established in NUREG-0390, "Topical Report Review Status," we request that Combustion Engineering Nuclear Power publish accepted versions of this topical report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted versions shall include 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, Combustion Engineering 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 by Stephen Dembek for/

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

Project No. 692

Enclosure: Safety Evaluation

cc w/encl:
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 692

Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT CENPD-390-P,

"THE ADVANCED PHOENIX AND POLCA CODES FOR

NUCLEAR DESIGN OF BOILING WATER REACTORS,"

COMBUSTION ENGINEERING NUCLEAR POWER, LLC

1.0 INTRODUCTION

By letter dated April 15, 1999, Combustion Engineering Nuclear Power, LLC (CENP) requested review and approval for a modified version of their PHOENIX/POLCA reactor physics methodology for boiling water reactor (BWR) analysis. The staff received one erratum on March 2, 2000, and performed an onsite visit to run and evaluate the codes in Windsor, Connecticut. CENP proposes to use the approved topical report to support BWR licensing analysis. The topical report describes the changes made to the code since it was previously reviewed and approved (Reference 1). It also provides an assessment against operational data and measurements to demonstrate that the code is capable of predicting power distributions, thermal limits and critical conditions necessary to design a BWR.

2.0 DISCUSSION

PHOENIX solves the discrete ordinate’s form of the two-dimensional neutron transport equation in Cartesian coordinates. It uses a two-step method where a pin cell calculation is performed using an integral method to obtain a spectrum to collapse the fine group structure to a problem specific working group structure which is then used to complete the two-dimensional S_N calculation.

POLCA is a three-dimensional diffusion theory code used to predict the core and pin-wise power distribution. POLCA is a steady state code using the analytical nodal method (Reference 2) with assembly discontinuity factors and a thermal-hydraulic model based on the CONDOR code (Reference 3).

The review was structured in three parts: (1) review of changes to the codes; (2) review of validation information; and (3) an onsite visit to exercise the code system and evaluate code documentation. This format will be followed for the remainder of this safety evaluation report.

2.1 Changes to the Codes

PHOENIX itself is not being changed from what was previously approved, but its cross section library has been replaced. There are two ENDF/B-VI based libraries available; an 89-group

library and a 34-group library which was derived from the 89-group library. This change required that an assessment be performed.

The changes to the POLCA code have been more substantial than the changes to the PHOENIX code. The thermal-hydraulic model in POLCA has been extended to allow for more geometric flexibility, more flow path models, and additional pressure drop and void fraction models. Changes have also been made to the following models in the POLCA code:

a. Detector model
b. Reflector albedo
c. Xenon transient model
d. Pin-power reconstruction
e. Improved axial power distribution representation
f. Depletion
g. Thermal limits

The majority of the changes involve modifications or additions of neutronic models. The review, therefore, focused primarily on the neutronic capabilities of the code, but the thermal-hydraulic models were also reassessed.

2.2 Validation

PHOENIX was re-evaluated against a set of critical experiments and fission rate data to assess the new library. The data set includes the Strawbridge & Barry criticals, the BAPL criticals and KRITZ data. The critical pin cell experiments modeled geometries which cover the range of current light water reactor fuel assemblies. Both the 34- and 89-group ENDF/B-VI libraries were shown to have an eigenvalue bias of less than 600 pcm. Comparing PHOENIX to the KRITZ fission rate data shows that PHOENIX can accurately predict local pin power distributions.

POLCA validation includes code to code comparisons, comparisons to test loop data and gamma scan data. Three code to code comparisons were performed to assess the capability of the advanced nodal method (ANM) solver. POLCA results compare well with the other codes with assembly power differences well below 1 percent and eigenvalue differences below 20 pcm. POLCA was also assessed against the NEA International Standard problem NEACRP-L336 which is designed to assess the pin power reconstruction capability of computer codes. This is a code to code comparison and the POLCA predicted root mean square (RMS) pin power error was less than 1 percent. POLCA was also used to predict the reactor eigenvalue and traversing in-core probe (TIP) results for 43 operating cycles from 3 BWRs. Comparisons between TIP measurements and predictions show an overall RMS error of 4.3 percent. Pressure drop and assembly flow rate data were used to assess the thermal-hydraulic models in POLCA. Pressure drop data were obtained from the FRIGG loop and flow rate data were measured in eight ABB designed BWRs. POLCA predictions were in good agreement with the measured data. Finally, gamma scans were performed on both of the assembly power distributions and nodal powers. Data was obtained for pin power distributions from 3 assemblies (a total of 100 individual fuel rods) and nodal measurements were obtained from 72 assemblies. Once again, the POLCA predictions were in good agreement with the measured data.

2.3 Onsite Visit

During the week of March 6, 2000, NRC staff visited the Combustion Engineering offices to run sample problems, examine documentation and meet with CENP personnel involved with PHOENIX/POLCA. POLCA cases were run to assess its capabilities over a wide range of conditions including cases involving large radial power distribution gradients, large inlet subcooling, and symmetry effects. No apparent deficiencies were noted. A review of the POLCA users manual and theory manual found that the manuals provide enough information to allow the user to understand how to use the code and the models in the code.

3.0 EVALUATION

The information presented in the topical report describes the new models and assesses the new code against both data and the previously approved code. The data assessment provided shows that the revised PHOENIX/POLCA code system is capable of predicting power distributions and critical conditions. The assessment performed by CENP used modern fuel and is, therefore, representative of how the code system will perform for actual design applications. Furthermore, the new code system is shown to provide equivalent or, in some cases, better predictions when compared to results from the currently approved methods. The demonstrated errors are consistent with what the staff has observed for other methods and are acceptable as long as they are appropriately considered when operating limits are formulated or monitored.

The assessment performed was not intended to necessarily assess all of the new models individually, but, rather, it shows that they provide acceptable predictions of relevant parameters. For example, the pin-power reconstruction and axial homogenization models are significant improvements over the previously approved models and the assessment shows that they provide acceptable results. The detector and the depletion models are assessed by the reactor power distribution and eigenvalue comparisons, respectively. Thermal limit predictions were not provided for evaluation, but CENP will need to use an approved correlation where appropriate. For cases where correlations are not used (i.e., linear heat generation rate), the code’s ability to predict power distributions can be used to conclude that PHOENIX/POLCA is an adequate tool for these applications.

4.0 APPLICATION OF PHOENIX/POLCA TO NON-ABB/CE FUEL TYPES

When using PHOENIX/POLCA to analyze a transition core with non-ABB/CE co-resident fuel the following must be considered in the analysis. First, CENP will need to ensure that the thermal and hydraulic behavior is properly modeled in the code by using fuel specific data to determine the pressure loss coefficients and critical power behavior. Second, power distribution uncertainties derived in the report for ABB/CE fuel types should be confirmed by comparison to available operating data such as linear power range monitor (LRPM) output. The mixed core methodology from CENPD-300-PA, "Safety Report for Boiling Water Reactor Reload Fuel," should be used to perform thermal limits calculations for non-ABB/CE coresident fuel.

5.0 CONCLUSIONS

The staff has reviewed the application of PHOENIX/POLCA as described in CENPD-390-P to BWR code reload design. The staff finds this application acceptable subject to the following conditions which were agreed to in Reference 4:

a. As discussed above, when applying PHOENIX/POLCA to transition cores, CENP should use fuel specific data to model the thermal and hydraulic behavior of the non-ABB/CE fuel and confirm that the uncertainties derived for ABB fuel are applicable to the non-ABB/CE fuel.

b. PHOENIX/POLCA are approved for analysis of ABB/CE fuel types up to and including 10x10 lattices with a maximum enrichment of 5 w/o UO_2. Non-ABB/CE fuel types may be analyzed assuming that analyses are performed consistent with (a) above. The code is approved for application to fuel with burnable absorbers composed of a mixture of UO_2 and Gd_2O_3 with concentrations up to 9 w/o Gd_2O_3. Application of the code to non-UO_2 fuel or the fuel using burnable poisons other than Gadolina will need to be justified.

c. When applying the PHOENIX/POLCA code to fuel other than what is approved in this SE (see (b) above), the NRC should be informed by letter of this application and be provided an opportunity for review.

d. PHOENIX/POLCA contains several models for BWR analysis not used to generate the information contained in the topical report. If CENP determines that one of these models is needed for a licensing analysis, the staff should be informed of the application and be given an opportunity for review.

6.0 REFERENCES

1. Letter from C. O. Thomas (USNRC) to E. P. Rahe (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10106, “A Description of Nuclear Design and Analysis Programs for Boiling Water Reactors,” September 3, 1985.

2. K. S. Smith, "An Analytic Nodal Method for Solving the Two-Group Multidimensional, Static and Transient Neutron Diffusion Equation," Department of Nuclear Engineering Thesis, M.I.T., Cambridge, Mass., March 1979.

3. CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors, BR 91-255-P-A, May 1991.

4. Letter from I. C. Rickard (ABBCE) to USNRC, "Application of CENPD-390-P to Mixed BWR Cores," June 12, 2000.

Principal Contributor: A. Ulses

Date: July 24, 2000