ENCLOSURE

EVALUATION OF WESTINGHOUSE NUCLEAR DESIGN AND ANALYSIS PROGRAMS FOR BOILING WATER REACTORS-WCAP 10106 (TACS 48566)

1. Summary of Report

This report describes the calculational models used by Westinghouse for the nuclear design and analysis of Boiling Water Reactor (BWR) cores. The computer codes used in these calculational models were originally developed and used by ASEA-ATOM, Vasteras, Sweden and were obtained by Westinghouse under a licensee agreement with ASEA-ATOM. A detailed description of the two major codes in the nuclear design system, PHOENIX and POLCA, is presented in the report as well as descriptive overviews of the auxiliary codes. The design system is composed of the following computer codes:

PHOENIX- A two-dimensional, multi-group transport theory code used to calculate the lattice physics constants of fuel assemblies.

POLCA- A modified one-group nodal code used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions of BWR cores.

FOBUS- A Monte Carlo transport theory code used for the generation of burnable absorber neutron cross sections.

PHOEBE- A library processing code used to construct a nuclear cross section data library.

PHIPO- A linking code used to generate input for POLCA from PHOENIX output.

ANALOAD- A fuel shuffling optimization code used to generate core fuel loading patterns.

POREF- A fuel shuffling code which performs manual refueling and lays out the geometry of the refueled core.

STROD- A code used for preliminary analysis of shutdown margin.

BAREST- A code used to determine the residual burnable absorber reactivity in each fuel assembly at the end of cycle.

2. Summary of Evaluation

We have reviewed the information presented with regard to calculational methods and assumptions. The comparisons of calculations and experiment, which is a major justification made by Westinghouse for the adequacy and accuracy of the BWR analysis methodology, is being reviewed separately. A companion report, WCAP-10841, "Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Programs for Boiling Water Reactors", dated June 1985 provides the relevant benchmark results.

The PHOENIX lattice physics code is used by Westinghouse for the nuclear design and analysis of BWRs. The code was developed by ASEA-ATOM for the calculation of lattice physics constants for fuel assemblies typical of those encountered in both BWR and PWR designs. Constants generated by PHOENIX are used in the three-dimensional POLCA code which simulates the neutronic and thermal-hydraulic conditions in typical BWR cores.

The code computes neutron spectra and two-group assembly averaged cross sections via a two-step procedure: 1) A 25-group nodal solution for the assembly which retains the spatial heterogeneities (based on transmission probabilities and response fluxes), followed by 2) a discrete ordinates (S_4) transport solution in up to 12-groups for the partially homogenized assembly. Burnable absorbers are treated by a perturbation technique which makes use of a Monte Carlo calculation for the absorber and its surrounding lattice environment. Branch calculations are automatically performed as functions of burnup for void,

Doppler and control rod insertions. POLCA is Westinghouse's coupled 3-D neutronics/thermal-hydraulic core simulator code intended for nuclear design and analysis of BWRs. The neutronics in POLCA utilize a modified, coarse mesh, one-group diffusion theory model. Two-group macroscopic cross-sections determined with the PHOENIX code are used and the core-reflector interface is represented with an albedo boundary condition. Feedback reactivity effects due to voids and Doppler broadening, as well as due to control rod insertion and xenon buildup are accounted for explicitly. The thermal-hydraulic model in POLCA is based on the CONDOR program, which did not form a part of the present review. The capabilities of the POLCA code include: xenon dynamics, depletion calculations, critical power ratio (CPR) and linear heat generation rate (LHGR) determination and simulation of in-core detector signals.

Based on the descriptions of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup, and stability, we find the calculational models presented in WCAP-10106 acceptable.

3. Evaluation Procedure

We have reviewed the report within the guidelines provided by Section 4.3 of the Standard Review Plan. Included in our review was the description of the analytical methods used and the types of calculations which are to be performed for Westinghouse BWR licensing actions. The Core Performance Branch has been assisted by our consultants at Brookhaven National Laboratory under Technical Assistance contract FIN No. A-3407 in the review of this topical report.

4. Regulatory Position

We have reviewed the BWR methodology described in WCAP-10106 and find it to be acceptable for nuclear core design and analysis made by Westinghouse in

licensing actions including calculations of lattice physics constants and neutronic and thermal-hydraulic analyses in three dimensions. The qualification of the results obtained by the described procedures relative to measurement or other calculations is being reviewed separately and was not addressed in this review.