WNP-2 Core Model Upgrade

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ABSTRACT

The paper describes the core model upgrade of the WNP-2 training simulator and the reasons for the upgrade. The core model as well as the interface with the rest of the simulator are briefly described . The paper also describes the procedure that will be used by WNP-2 to update the simulator core data after future core reloads. Results from the fully integrated simulator are presented.

DESCRIPTION OF THE OLD CORE MODEL

Exitech Corporation provided the previous core model as a part of the original simulator software delivered by Westinghouse in 1994. This model uses the Quasi-Static Method to calculate the neutron flux level and flux distribution within the reactor core and the thermal power output into the fuel and the coolant. In this method, the neutron density is factored into an amplitude function and a shape function. The model used 185 radial and eight axial nodes for the core.

DESCRIPTION OF OLD CORE UPDATE PROCESS AND PROBLEMS

After the initial delivery, the core data was updated once by Exitech Corporation. This process was lengthy and costly and included several steps – collection of data from the fuels group, conversion of data to the required format by a third party, update calculations, and installation and testing. This process took approximately 6-9 months. Based on INPO review, it was recommended that WNP-2 should update the core data prior to each fuel load, to provide operators the ability to train on the new core. This was not possible with the old process, and this was the major driving force behind the project to purchase a new core model.

THE WNP-2 NEW CORE MODEL

Scope of Simulation

The core is modeled using 185 neutronic channels corresponding in general to a control cell. There are 16 peripheral bundles that are not controlled and these are included in the bundle grouping. The model uses 12 axial layers. The cross-section library was generated based on lattice data for cycle 15.

The simulation includes reactivity effects of changes in density, fuel temperature, control rod position, boron, and fission products such as xenon and samarium. Exposure and density history of the cross sections, as well as burnup dependence of the delayed neutron fraction, are held constant in the dynamic calculation and are updated for each core life. Prompt and delayed power production are simulated and the power deposition model discriminates between power deposited in the fuel rod an power deposited directly in the coolant by gamma heating. In-core detectors are modeled.

Core conditions at the following points in the cycle were prepared: BOC, MOC, EOC, and EOC coastdown. A cycle update tool is provided as part of a THOR delivery to allow WNP-2 to upgrade the model to future cycles. Interfacing programs to CASMO-3 and MICROBURN-B were also provided.

Model Features

a) Full 3D model based on 2-group nodal method

b) Calculation done in 2 groups with a special treatment of the thermal flux leakage to achieve fast calculation.

c) Two-group cross-section model with functional dependence on the following nodal properties:

- Fuel burnup
- Void history
- Instantaneous density
- Fuel temperature
- Control rod presence
- Soluble boron
- Xenon
- Samarium

d) All control rods are treated explicitly and are interfaced with existing failures. The effects of these failures are dynamically calculated.

e) The code calculates concentrations of fission product poisons. Buildup and burnout of iodine, xenon, promethium and samarium are provided. Flags are provided to change the fission product poisons in fast time to create a steady-state initial condition.

f) The core model takes into account the fact that fission energy is deposited as thermal energy both inside the fuel pellet where the fission takes place and outside the pellet due to neutron and gamma attenuation. Also, the heat deposition location for prompt and delayed power may be different.

g) Delayed neutrons precursors in six groups

h) Decay-heat model treated in 23 groups similarly to the delayed neutron precursors and is based on ANSI/ANS-5.1-1979 standard.

i) The model includes in-core detectors.

CORE RELOAD PROCESS

As part of the delivery to WNP-2 a Cycle-Update Utility was developed. It consists of two auxiliary programs: THORGEN and THORSS. THORGEN is the cross-section-processing program and is used to read lattice data and generate the cross-section file for THOR. THORSS is essentially a standalone version of the THOR neutronics model and includes the tools for collapsing the distributions and cross-section data from the engineering nodalization to the THOR nodalization and an interface to the client's engineering steady state simulator output.

The cross-section data are input to THOR in the form of polynomials coefficients generated per "super" node, i.e. a 12"-height node in the case of this project . The coefficients are calculated based on

- polynomials per fuel type
- flux solution in a "normal" node, i.e., the node used engineering codes

The cross-section representation in THOR is identical in both following situations:

- in a normal node size with data given per fuel type
- in a super node size with the data given per node

The model takes into account the following effects:

- Exposure (E)
- Density history (ρ_x)
- Spacer grids (SP)
- Instantaneous density (ρ)
- Fuel temperature (T_F)
- Control rod insertion (CF)
- Soluble boron (N_B)
- Xenon (N_X)
- Samarium (N_S)

The data are given for each fuel type and control type and are split into two types:

- Base cross sections, which describe the lattice properties at nominal power density, fuel temperature, boron concentration, equilibrium xenon and samarium, and no control rod or spacers as a function of depletion for various moderator densities. At a given burnup and density history only the density may vary instantaneously.
- Differential cross sections describing deviations from base cross sections as the result of perturbations in the independent parameters.

The lattice calculational matrix includes depletion calculations at three void fractions and branchoff calculations at six density levels for conditions ranging from cold to highly voided.

The steady-state solution as well as the averaging process is done by means of the THORSS. THORSS produces the initialization data for the THOR neutronics in the nodalization of THOR. The functions of THORSS are:

- Read THORGEN file
- Fetch core loading, nodal distributions, control rod pattern from the steady-state simulator output file

- Process input data related to core shape, dimensions, steady-state simulator nodalization, THOR nodalization, control rod locations and pattern, nuclear instrumentation (type, location, …)
- Calculate steady-state solution in steady-state simulator nodalization (keff, power, fluxes, delayed neutron concentration,…).
- Perform transformation of nuclear parameters and independent parameters (i.e., history data, density, …) from steady-state simulator nodalization to THOR nodalization.
- Calculate steady-state solution in THOR nodalization (keff, power, fluxes, delayed neutron concentration,…). .
- Write THOR initialization data

The following steps are used in generating cross section and initialization data for THOR:

Step 1	Calculate polynomial coefficients per fuel type by means of THORGEN based on lattice files
Step 2	Calculate the polynomial coefficients per super node by means of the averaging procedure in THORSS
Step 3	Produce an initialization file by means of THORSS

INTERFACE BETWEEN THERMAL HYDRAULICS AND THE NEW REACTOR KINETICS

The ATHENA code performs the thermal hydraulics calculations for the WNP-2 simulator. ATHENA computes the voids, liquid, vapor and boron densities, and the coolant and fuel temperatures at eight elevations in the core. The new reactor kinetics requires the moderator densities, boron concentrations, and the fuel temperatures within all the new kinetics nodes, i.e. at 12 elevations and 185 lateral locations. The necessary “mapping” between the eight ATHENA core thermal hydraulics cells, and the 2220 kinetics nodes is performed by four modules: THIPOL, THNEUT, THTOKI and THFRKI.

THIPOL computes the voids, liquid, vapor and boron densities, and the coolant and fuel temperatures at twelve elevations in the core by linear interpolation of the ATHENA output.

THNEUT computes the moderator density, boron density and fuel temperature in each of the kinetics nodes. The starting point for the computation is the integration the coolant energy equation from the bottom to the top of each of the fuel assemblies. This integration takes into account the flow into each individual fuel assembly and the diffusion of energy to neighboring assemblies. The result is the coolant thermodynamic equilibrium qualities and the temperatures for all the neutronics nodes. The nodal thermodynamic equilibrium qualities are then converted to flow-qualities by using a relationship for subcooled boiling together with the detachment quality. The voids in the neutronics nodes follow by applying the void/quality relationship for the average channel. The nodal density is then computed using the nodal void and the phasic interpolated densities at the twelve core elevations. For nodes where the void is zero, the density is equal to the liquid density evaluated at the nodal enthalpy.

The interpolated fuel temperature at the twelve elevations is the starting point for the evaluation of the fuel temperatures in the neutronics nodes. They are computed from the nodal nuclear power delivered to the fuel by assuming the coefficient of heat transfer is constant across the core.

THTOKI converts the units for moderator density, boron density, and the average fuel temperature in the neutronics nodes.

THFRKI converts the units of nuclear power and sums the results to provide the total power deposited in the fuel and in the water in each of the eight ATHENA core cells.

RESULTS

Core performance testing took place before the plant underwent refueling with the longer cycle core. This forced us to compare the response of the neutronics model to the predicted core response obtained from MICROBURN-B calculations. Acceptance tests that were conducted on the replacement core model included:

- 15 Hour Steady State
- Fission Product Poison Tests (For Samarium and Xenon)
- Reverse Power Effects (During Control Rod Withdrawal)
- Doppler (Fuel Temperature) Reactivity
- Moderator Temperature Reactivity
- Void Reactivity Effects
- Boron Injection Shutdown
- Verification of Estimated Critical Rod Configuration for BOC, MOC and EOC Conditions
- All ANSI Transient Tests (involving the core or thermal-hydraulic models)

The results of the transient tests were measured against the existing simulator response data. All of our test results correlated closely with the MICROBURN-B predictions and previously observed simulator responses. Once the outage was completed in October, the neutronics model was adjusted to match the observed response of the plant core during startup. An additional adjustment is expected in February, after 4 months of operation, at which time the new core model will be placed in service for use with the certified training load.

SUMMARY

In anticipation of the new core configuration Energy Northwest decided that a more advanced core neutronics model was an integral part of the overall transition plan. The analysis and recommendations of INPO SOER 96-2, "Design and Operating Considerations for Reactor Cores" supported this decision. SOER 96-2 made the following recommendation as part of the overall strategy to improve operator competency:

"For significant changes to the reactor core design, such as changing from a negative to a positive moderator temperature coefficient, evaluate simulator response compared to expected plant response. Update the simulator to the extent possible to support operator training on the revised core performance and integrated plant response"

Energy Northwest's initiative to optimize fuel cycle economics has resulted in a longer fuel cycle and higher-energy output core. We feel confident that the improved core response provided by the upgraded neutronics model and the ability to keep current with plant fuel cycles, provides operators with the most effective training available outside the control room.

REFERENCES

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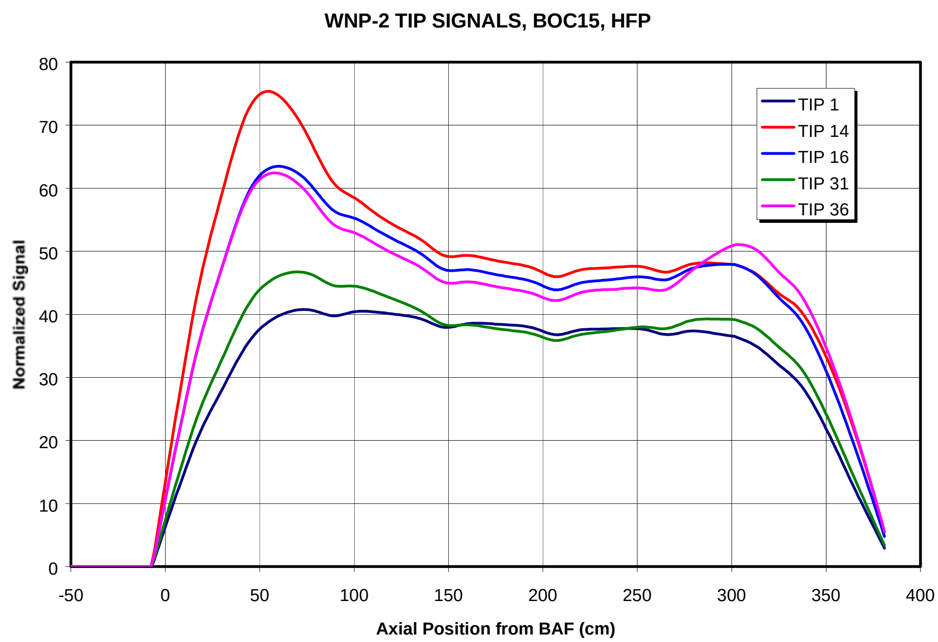

Figure 1. TIP traces at five different core locations.

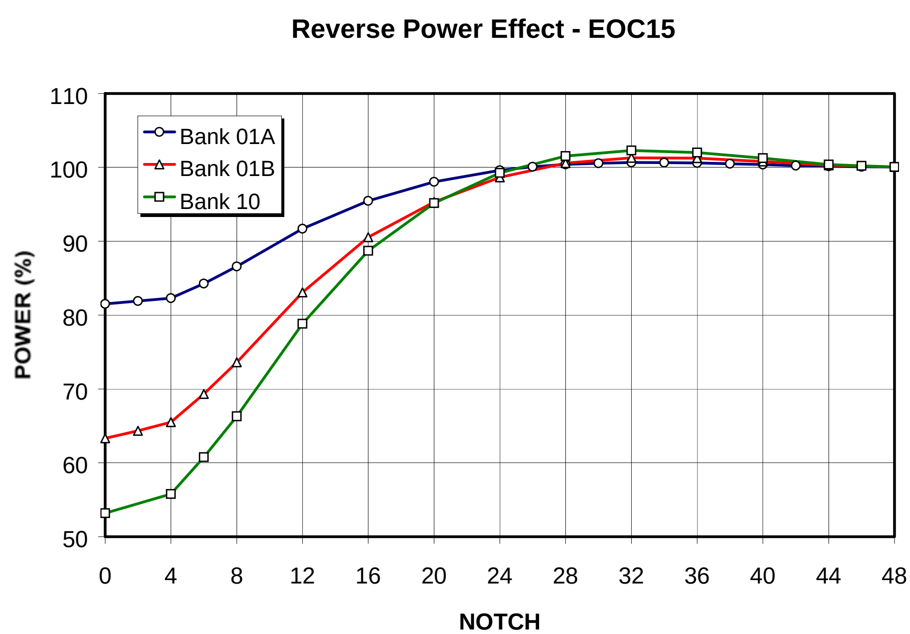

Figure 2. Reverse power effect at end of cycle for selected control banks.

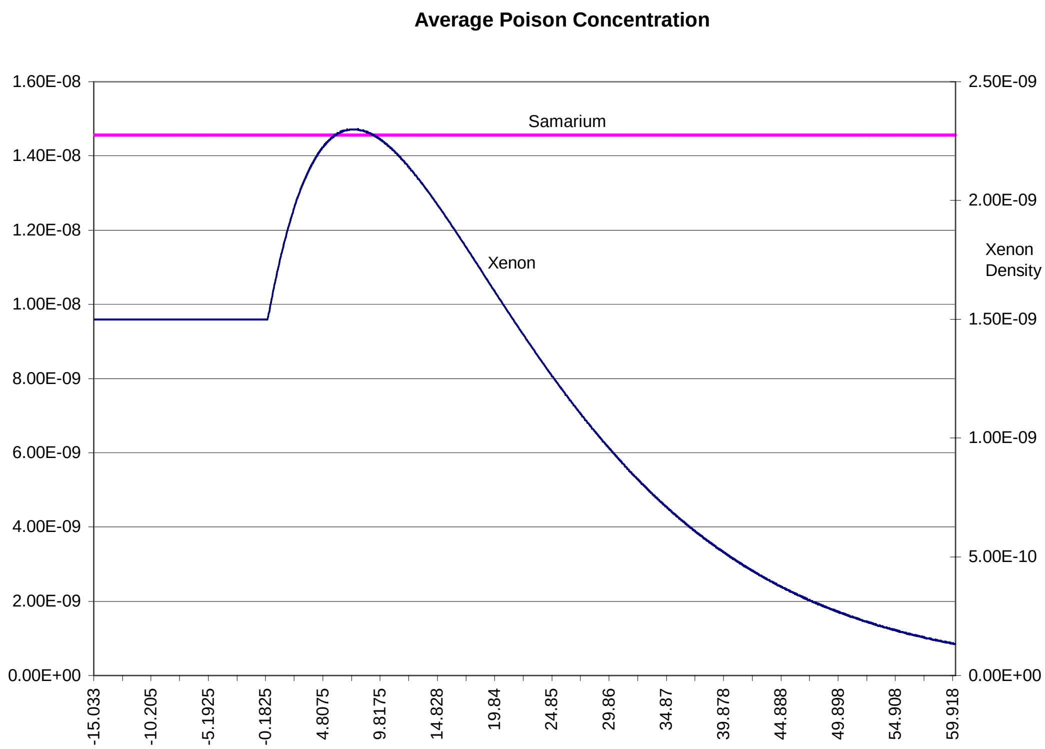

Figure 3, Average Poison Concentration Post Trip

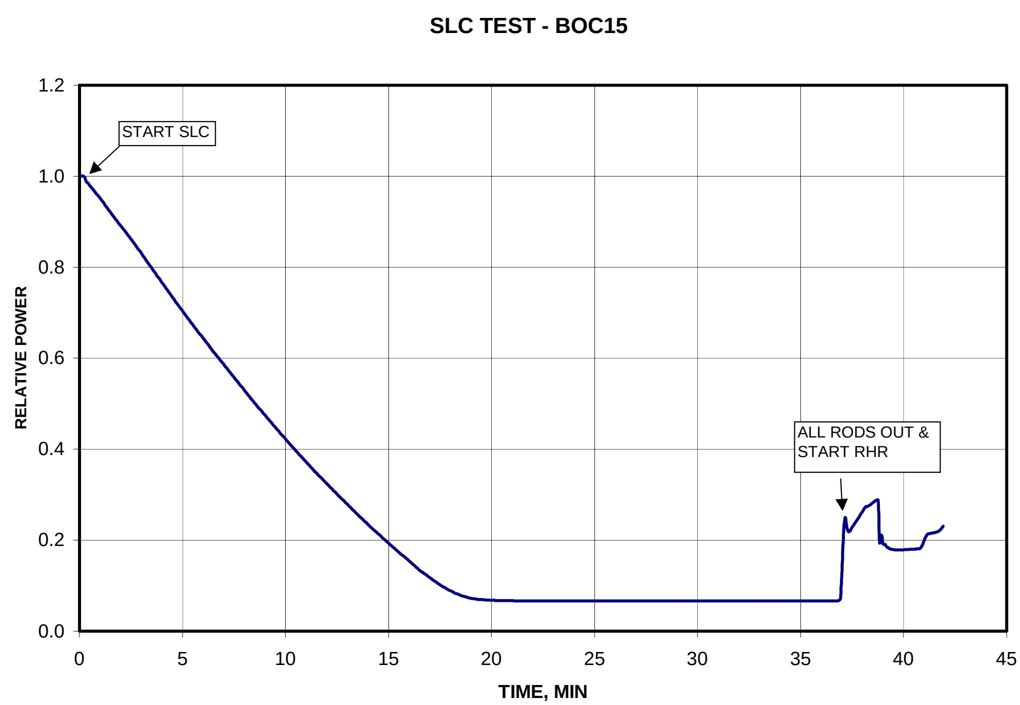

Figure 4, Boron Injection Shutdown Test