THE NEW CORE MONITORING SYSTEM FOR THE HOPE CREEK STATION

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ABSTRACT

A new Core Monitoring System (CMS) has been delivered to Public Service Electric and Gas (PSEG) Nuclear for installation at the Hope Creek Station. The CMS will replace the core monitoring system that was delivered by the Nuclear Steam Supply System (NSSS) supplier, as original equipment. The original equipment plant process computer is being removed and a new software package is being installed to provide the functionality required to perform the core supervision and surveillance tasks in accordance with the existing Technical Specifications.

Plant data is acquired by the new CMS process computer and is used to perform the thermal balance calculation for the core. Control rod positions and core operating parameters are used in the 3-D neutronics calculation to determine the detailed 3-D power distribution and determine the margin to the licensed thermal limits.

The Heat Balance program calculates the reactor power as used to verify compliance with the operating license. The thermal limits evaluations and the depletion calculations are carried out at specified time intervals and the results are available for display to the end users on various color graphics displays. On-demand calculations are triggered to evaluate the result of Traveling In-Core Probe (TIP) reading tasks, to produce reports and to calculate detailed distributions of isotopes important for safeguard requirements.

The coupled thermal hydraulic and neutronics equations in 3-dimensions and 2-energy groups are solved using cross sections calculated by the lattice code PHOENIX™. The power in each node, for each fuel pin, is calculated from the neutron flux solution. The small-scale solution is then the basis for the simulation of detector readings, linear heat generation rate calculation, dryout calculations and PCI evaluations. The CMS has the capability to use the Local Power Range Monitor (LPRM) adaptive method UPDAT for evaluation of the thermal margins.

The User Interface Module (UIM) provides access to the system for different types of users: Reactor Operators, Reactor Engineers, Fuel Engineers and System Administrators. The requirements of the users are accommodated by a graphical user interface. Different functions are accessed through drop-down menus that provide selections for the presentation of plant parameters in graphical, tabular or text form.

The combination of ENDFB-VI cross sections used in the PHOENIX™ code and the 3-D, 2-energy group calculations using the POLCA™ code provide state-of-the-art methodology and high accuracy for a core monitoring system in a commercial operating plant.

A measure of the accuracy of the CMS is provided by TIP comparisons that show rms errors in the radial power distribution are less than 2% and the overall (3-D nodal) TIP distributions below 5%. Other measurements to calculation comparisons on radial power distributions have been made on gamma scanning results. These evaluations are based on power distributions inferred from the activity of La-140. The nodal rms errors are 5.8% and the radial errors are 1.6%.

1.0 INTRODUCTION

Earlier generations of CMS have been developed and implemented at a number of utilities over the last 20 years. Significant improvements in computer hardware performance and lower computer costs have prompted the development of an integrated approach to core design activities and on-line supervision. The replacement of earlier versions of CMS has been achieved by making use of state-of-the art core neutronics, modern computer platforms, and Graphical User Interfaces (GUI) and database integration.

2.0 SYSTEM FUNCTIONALITY

The functionality of CMS for the Hope Creek Station was developed in cooperation with PSEG Nuclear, the owner and operator of the plant. The hardware configuration is shown in Figure 1. The functions that were required for implementation are briefly noted below.

One critical requirement of the new system was satisfied by an interface that allowed for some functions to be preserved in part of the existing Balance Of Plant (BOP) plant computer, as identified in Figure 1.

The Nuclear Stem Supply (NSS) Applications were completely replaced by the introduction of the POLCA7™ code to calculate the 3-D power distribution, which forms the basis for the on-line core monitoring and core supervision functions.

The purpose of the core supervision function is to calculate and update the values of the thermal margins and detailed neutronics information. This was accomplished by implementing a control module to scan the parameters that are made available by the Data Acquisition System. User defined triggering parameters start the calculations performed by the POLCA7™ code, using two neutron energy groups, a fine axial spatial division and pin power reconstitution.

The power distribution calculated with input plant parameters is used to update the distribution of iodine, xenon, promethium and samarium. The thermal margin calculations are performed using the power distribution and the results presented graphically and as hard copy, in a core performance report. A database utility program takes periodic copies of the most important measured or calculated data for later use as input for off-line calculations or history edits. The frequency and the amount of data to be stored are user defined.

The power distribution calculated by the POLCA7™ code is used to update a number of quantities which are tracked by the system. Such parameters are the exposure arrays for control rods, fuel bundles and the LPRMs.

The LPRMs are also monitored with respect to drift, by verifying that under stable operating conditions, the measured LPRM readings do not exceed a pre-defined noise band. If the band is exceeded, the system flags the drifting detector and issues an alarm.

Periodically, as required by Technical Specifications, the LPRMs are calibrated using data obtained by the TIP system. The data acquired by the TIP system is processed by the CMS and used to calculate LPRM correction factors which are needed to correct the depletion of the LPRMs, which are used for both the supervision and protection systems of the plant.

A graphical interface has been designed to allow access to calculated and measured data, to provide the means for requesting information and to ensure that the system is operable.

Reactor Operators, Reactor Engineers and Fuel Design Engineers have different roles in ensuring that the plant is operating in a manner consistent with the regulatory requirements. The CMS provides information that is needed by the three types of users to ensure safe operation.

The detailed calculation of the power distribution in each fuel rod makes it possible to evaluate pellet powers and assess the compliance with Pellet-Clad Interaction (PCI) supervision rules for all the fuel in the reactor. Detailed distributions of important isotopes in the reactor core are also required to conform to safeguard reporting requirements.

4.0 THE 3-D CORE SIMULATOR

4.1 OVERVIEW OF THE POLCA7™ CODE

The POLCA7™ code solves the two-group diffusion equation employing an analytic nodal method. The code calculates the three-dimensional power distribution in the reactor taking into account all-important phenomena that affect the neutronic, thermal and hydraulic behavior of the core. The reactor core is divided into computational nodes in which the neutronic characteristics of each node are described by homogenized equivalent two-group macroscopic cross sections. The three-dimensional power distribution calculated by the POLCA7™ code includes the thermal-hydraulic feedback effects of the coolant flow, the influence of control rods, the reactivity feedback effects due to Doppler feedback, xenon absorption, soluble boron and coolant density.

The POLCA7™ code is the main working tool for in-core fuel management activities. Therefore the CMS for the Hope Creek Station has the same accuracy and model fidelity as a state-of-the-art nuclear design code.

The structure of the POLCA7™ code is summarised in Figure 2. The POLCA7™ code solves the two-group diffusion equation employing a method similar to that referred to as the Analytic Nodal Method (ANM). The method takes the three-dimensional diffusion equation and converts it into three one-dimensional equations, with one equation for each spatial direction. The equations are coupled through the neutron leakage from one direction to another, referred to as transverse leakage. The analytical solution to the inhomogeneous one-dimensional diffusion equation is used to derive a relationship between node surface net currents and node average fluxes. This relationship is then substituted into the node balance equation to eliminate net currents as unknowns to yield an equation, which is similar to that resulting from the finite difference approximation.

In addition to the use of transverse leakage for spatial decoupling, another feature of the POLCA7™ code nodal method is a spectral analysis method used to compute the analytic matrix functions, which appear in the nodal coupling relations.

Modern homogenization principles are also accounted for in the nodal equations through the use of discontinuity factors to describe flux continuity conditions at nodal interfaces. In addition, smooth intra-nodal variations of cross sections are allowed to account for burnup induced heterogeneities.

The neutronic computational module produces node average fluxes and node interface average fluxes and net currents. The flux variation inside the node is calculated by the pin power reconstruction method.

The POLCA7™ code handles two types of response simulations for detectors: neutron in-core detectors and gamma in-core detectors.

The in-core neutron sensitive response calculation is based on computing the reaction rate induced in the detector by impinging neutrons. This is done by combining detector response functions generated by the lattice code with the point fluxes in the detector location computed by the POLCA7™ code and summing over the energy groups.

The in-core gamma response is correlated to a weighted sum of the powers of the fuel pins in the bundles surrounding the detector.

The real time performance of the CMS requires that the I-135 and X-135 distributions be calculated using the most recently calculated neutron flux distributions. The non-equilibrium I-135 and Xe-135 distributions are calculated assuming that other nuclide concentrations remain constant.

The pin power reconstruction process involves the superposition of heterogeneous information at the fuel from the lattice code with a smoothly varying homogeneous power distribution determined from the POLCA7™ code to obtain a composite power distribution on the pellet level.

Material variations within the axial nodes can lead to important reaction rate and flux variations, which would not be captured, by traditional node average fluxes. Spacers, control rod tips, burnable absorber variation, enrichment zoning, and reflector regions at the assembly ends are examples of those material variations.

The nodal cross sections used by the POLCA7™ code account for the presence of axial heterogeneities through axial homogenization corrections derived from an axial homogenization method included in the POLCA7™ code. This method also yields axial discontinuity factors, which provide neutron balance in the presence of axial material variations within the node and are utilized in the nodal coupling coefficients. The POLCA7™ code treatment of the axial variations also provides a smoothly varying axial flux, which can be used to correct the axial power distributions with the axial fine structure function to accommodate the heterogeneous variations within the node.

5.0 QUALIFICATION OF THE POLCA7™ CODE

The qualification of the POLCA7™ code has been carried out by testing individual models or combinations of models to verify that they perform as intended as well as comparisons of the POLCA7™ code predictions with measured data.

The POLCA7™ code comparisons have been performed with computational benchmarks generated by means of reference calculations as well as by comparison with experimental data suitable for evaluating the individual model being verified. Some of the comparisons are discussed below.

The neutronics model was verified by comparison with established two-dimensional analytical benchmarks. Some of the analytical benchmarks involve power calculations without depletion for both PWR and BWR cores. Another benchmark provides verification of the POLCA7™ code depletion models.

The POLCA7™ code pin power reconstruction model is verified by comparison with a pin power distribution benchmark.

The POLCA7™ code validation involves the evaluation of core follow predictions for the Hope Creek operating core. Specifically, the reactivity rundown and the detector responses calculated by the POLCA7™ code are compared with the similar measured parameters.

5.1 NEUTRONIC MODEL VERIFICATION

Comparisons of the POLCA7™ code neutronic model calculations with reference solution results for benchmark problems have been performed. One of the benchmarks is the International Atomic Energy Agency (IAEA) two-dimensional benchmark problem specified in Reference 1. The identifier in Reference 1 for this problem is 11-A2. This benchmark consists of two different fuel bundle types with reflector bundles on the edges of the core and a total of 177 assemblies. The configuration is one-eighth core symmetric. Each assembly has a width of 20 cm and a height of 340 cm. The fuel and reflector bundles have no internal pin structure, and are represented by uniform two-group cross sections. The large flux gradients in the vicinity of the reflector and near the control rods provide a very good test for a code such as the POLCA7™ code.

The relative assembly powers and $k_{effective}$ predicted by the POLCA7™ code were compared with a benchmark reference solution for this configuration. In the discussion below, the term "error" refers to the magnitude of the difference between the relative assembly power predicted by the POLCA7™ code (P) and the relative assembly power predicted by the reference solution (P_{ref}).

The POLCA7™ code solution for this benchmark was compared with Solution 3 (i.e., 11-A2-3) in Reference 1. This solution utilized a nodal method referred to as the nodal expansion method, which was run on a very refined spatial, mesh (36 meshes per assembly) and should provide a very accurate solution to this problem.

The results obtained with the POLCA7™ code are compared with those predicted by solution 11-A2-3 in Figure 3. Relative to solution 11-A2-3, the POLCA7™ code solution has a root mean square error in relative bundle average power of 0.002 and a maximum error of 0.004. The deviation in $k_{effective}$ is 8 pcm.

The conclusion from the two-dimensional IAEA benchmark is that the two computations (the POLCA7™ code and Solution 11-A2-3 in Reference 1) yield virtually identical results. The small differences observed are characteristic of expected numerical deviations.

5.1.2 BIBLIS Benchmark

The BIBLIS benchmark is a two-dimensional model of an operating Pressurized Water Reactor (PWR) core with a multi-zone, checkerboard loading. This configuration is one-eighth core symmetric with seven different homogenized fuel compositions and a homogenized reflector. Each assembly has a width of 23.1 cm. The realistic nature of this problem makes it a good test for the POLCA7™ code neutronics model, which will provide errors or uncertainties similar to those expected in operating plants.

The POLCA7™ code solution was compared with a reference solution generated with the code LABAN as reported in Reference 2. The results obtained with the POLCA7™ code are compared with those predicted by the LABAN code in Figure 3.

5.1.3 Operating Plant Data Validation

The POLCA7™ code validation effort has been on going for several years, as POLCA7™ was first introduced for core design calculations in PWRs. Recent validations have been reported in References 3, 4 and 5.

Extensive comparisons against operating data have been performed for BWR validation. Fourteen cycles of the Olkiluoto 2 (OL-2), ten cycles of the Leibstadt Plant (KKL) and eleven cycles of Forsmark 3 have been analyzed to validate the performance of the POLCA7™ code.

Power distribution comparisons have been performed for gamma-scan measurements in 26 fuel assemblies of OL-2 at the end of Cycle 8. The power distribution measurements were inferred from gamma scanning and the activity of ^{140}La. The comparisons show that the nodal rms errors are 5.8% and the radial rms errors are 1.6%. Some of this material has been reported in References 3, 4 and 5.

The comparisons for the Hope Creek Station cover 8 cycles of operation of the plant. The Hope Creek Station is a Boiling Water Reactor.

Comparisons of calculated and measured $k_{effective}$ values and TIP have been made and are summarized below.

Figure 5 shows the $k_{effective}$ values calculated at hot conditions for the cases at which the TIP measurements were performed. In addition, the figure also shows the $k_{effective}$ values at cold conditions when they were measured. These comparisons demonstrate consistent and accurate trends over many cycles of operating data.

Figure 6 shows a comparison between TIP measurements and the POLCA™ code calculated detector responses in the nodal values. Figure 7 shows the comparison for the calculated and measured radial values.

These types of comparisons have been carried out in order to establish the accuracy of the POLCA™ code for the core supervision and monitoring function at the Hope Creek Station.

CONCLUSIONS

The new CMS for the Hope Creek Station has been designed, implemented and tested to replace the original NSSS equipment plant process computer. Extensive validation of the ability of the POLCA™ code to calculate the operating core parameters needed to satisfy the Technical Specification requirements for core supervision and monitoring have been carried out.

The addition of Hope Creek specific operating data comparisons to the existing comparisons database insures that the new CMS will comply with current licensing requirements.

REFERENCES

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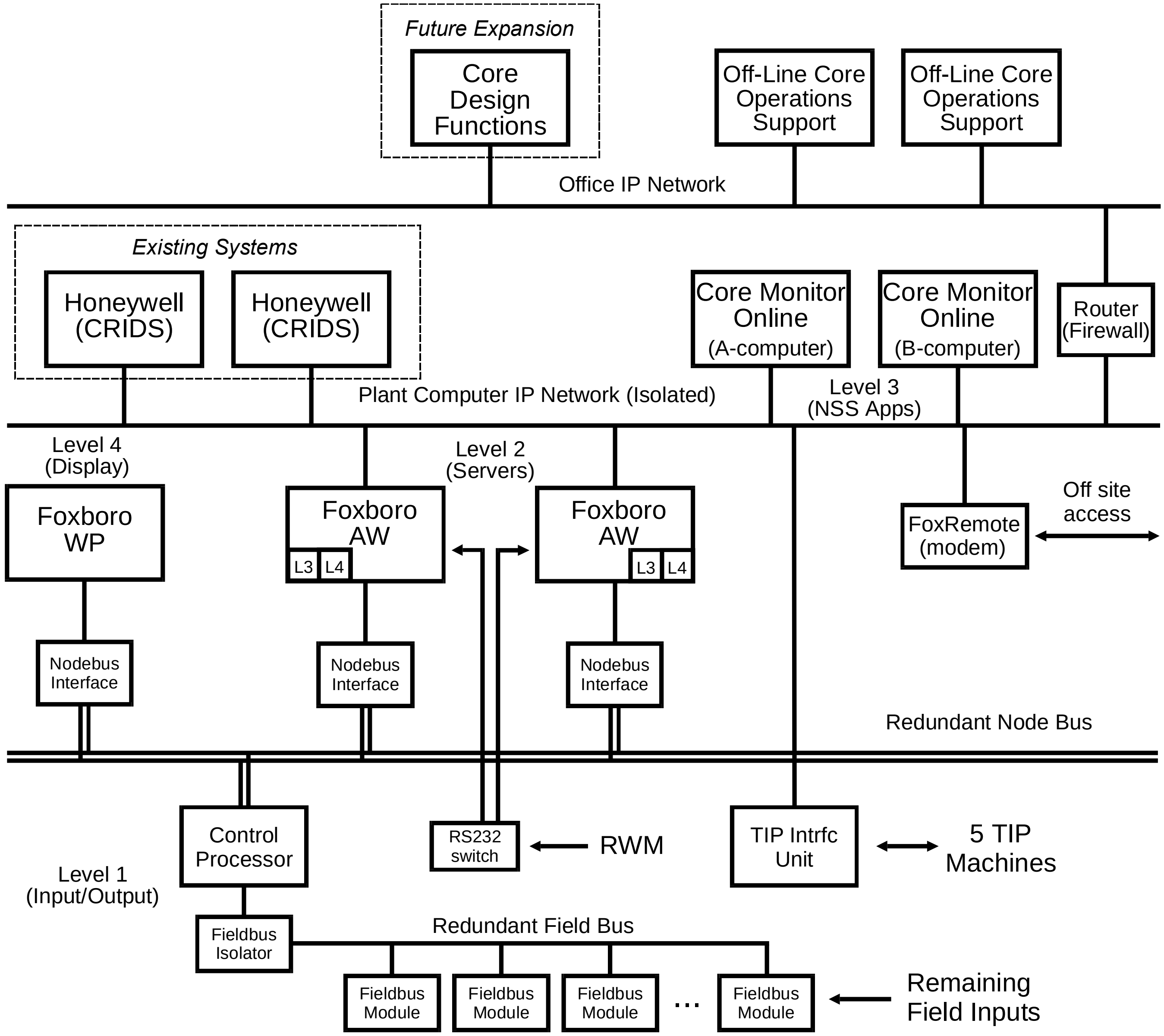

Figure 1: NSS Process Computer replacement architecture.

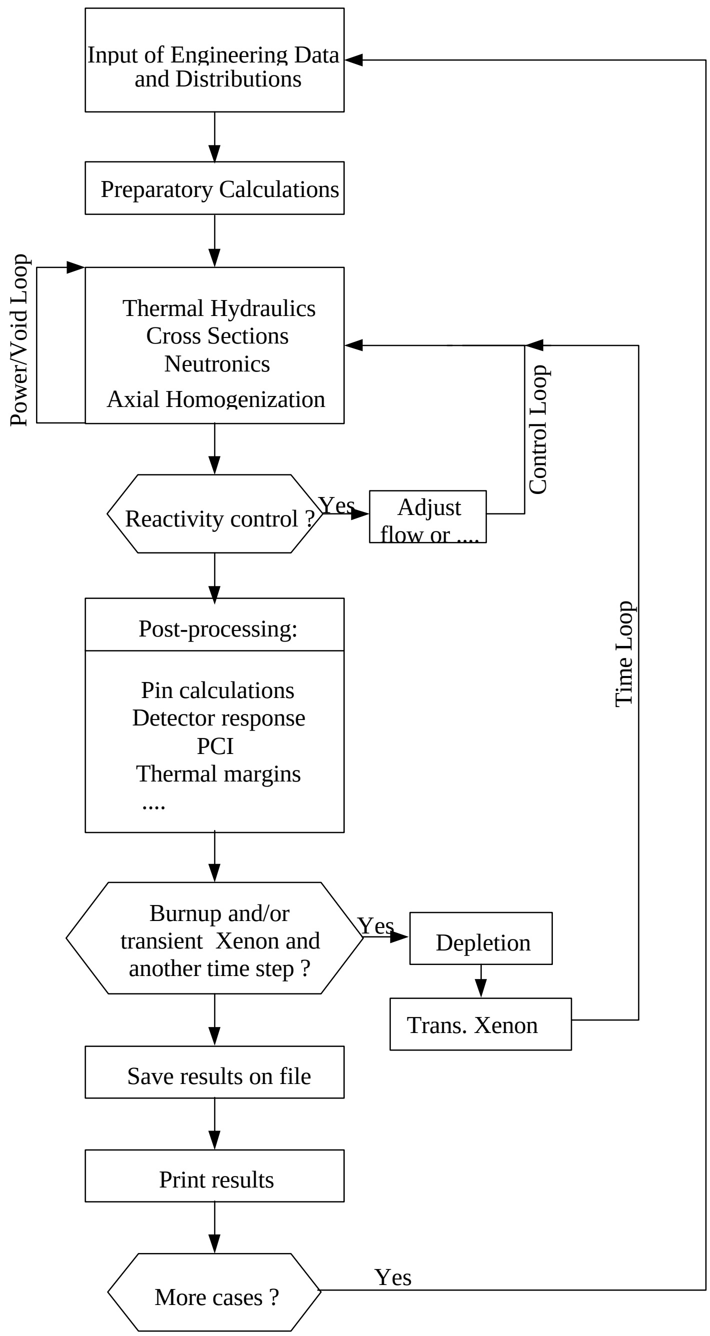

Figure 2: The POLCA7 Code Calculational Flow

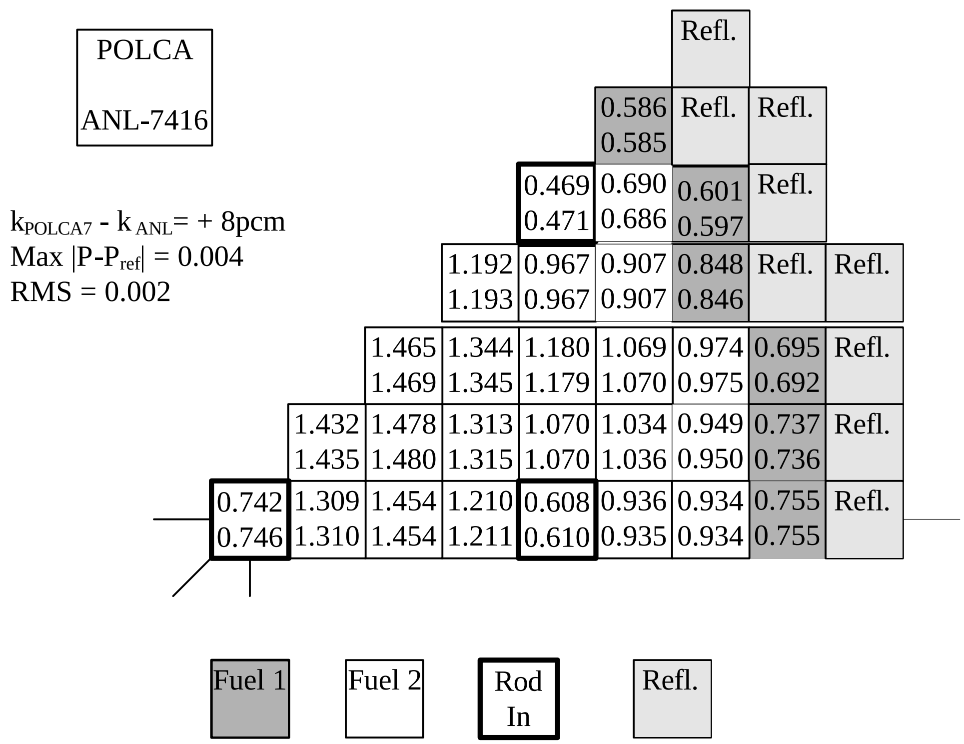

Figure 3: IAEA 2D Benchmark, the POLCA7 code results compared to ANL-7416

Assembly Average Power
BIBLIS Benchmark

Bundle Type
LABAN
POLCA

$k_{POLCA7} - k_{LABAN} = + 15\text{pcm}$
Max $|P-P_{ref}| = 0.022$
RMS = 0.007

3	3	3	3	3				
4 1.015 1.012	4 0.971 0.969	4 0.825 0.828	4 0.545 0.551	3	3	3		
1 1.095 1.093	1 1.072 1.070	1 0.931 0.934	4 0.765 0.769	4 0.874 0.887	4 0.684 0.687	3	3	
7 0.982 0.977	1 1.032 1.030	7 0.924 0.921	1 0.950 0.955	5 0.993 1.002	4 1.199 1.221	4 0.684 0.687	3	
1 1.089 1.085	8 1.068 1.062	2 1.120 1.120	8 1.039 1.038	2 1.123 1.128	5 0.993 1.002	4 0.874 0.887	3	3
6 1.221 1.212	2 1.224 1.219	8 1.105 1.099	2 1.161 1.161	8 1.039 1.038	1 0.950 0.955	4 0.765 0.769	4 0.545 0.551	3
2 1.243 1.237	8 1.134 1.126	1 1.122 1.118	8 1.105 1.099	2 1.120 1.120	7 0.924 0.921	1 0.931 0.934	4 0.825 0.828	3
8 1.102 1.092	1 1.118 1.112	8 1.134 1.126	2 1.224 1.219	8 1.068 1.062	1 1.032 1.030	1 1.072 1.070	4 0.971 0.969	3
1 1.091 1.084	8 1.102 1.092	2 1.243 1.237	6 1.221 1.212	1 1.089 1.085	7 0.982 0.977	1 1.095 1.093	4 1.015 1.012	3

Figure 4: BIBLIS 2D Benchmark, the POLCA7 code results compared to LABAN

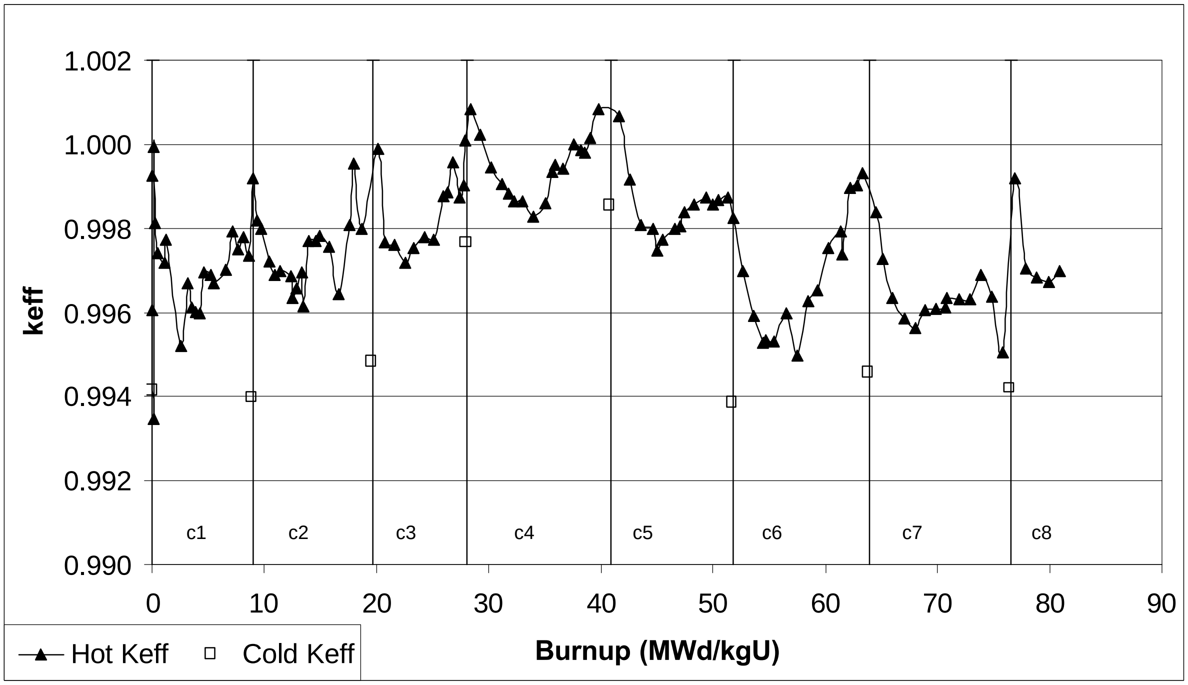

Figure 5: Calculated $k_{effective}$ by Cycle for Hope Creek

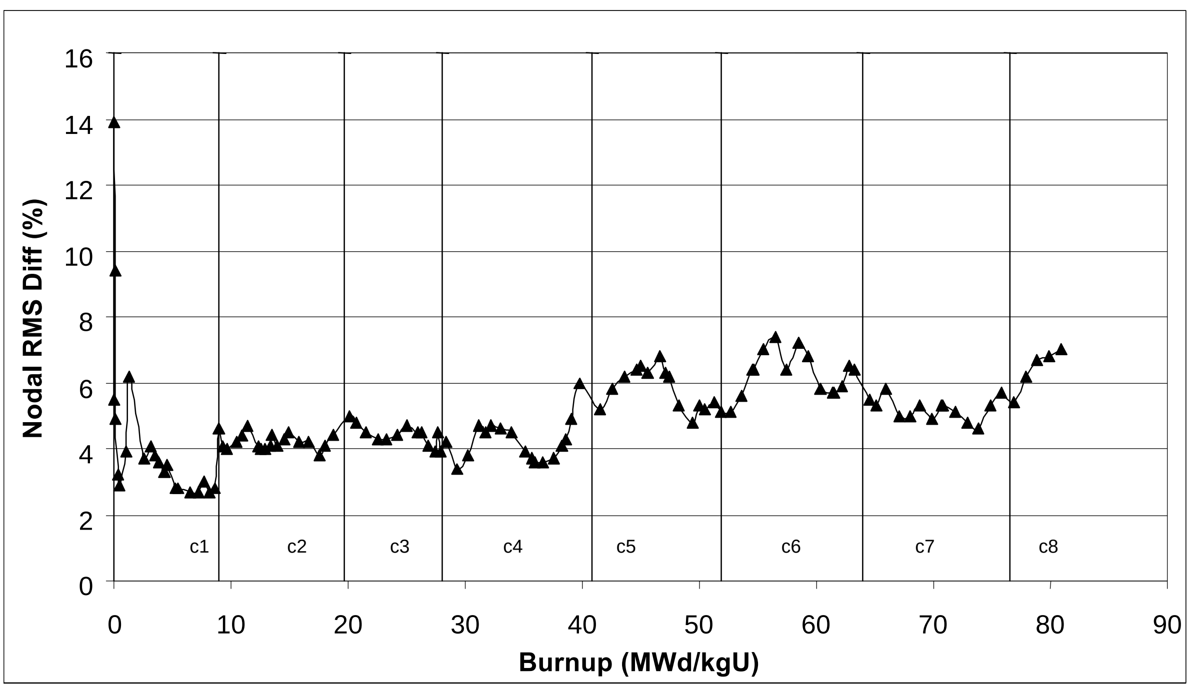

Figure 6: Hope Creek Nodal TIP Results by Cycle

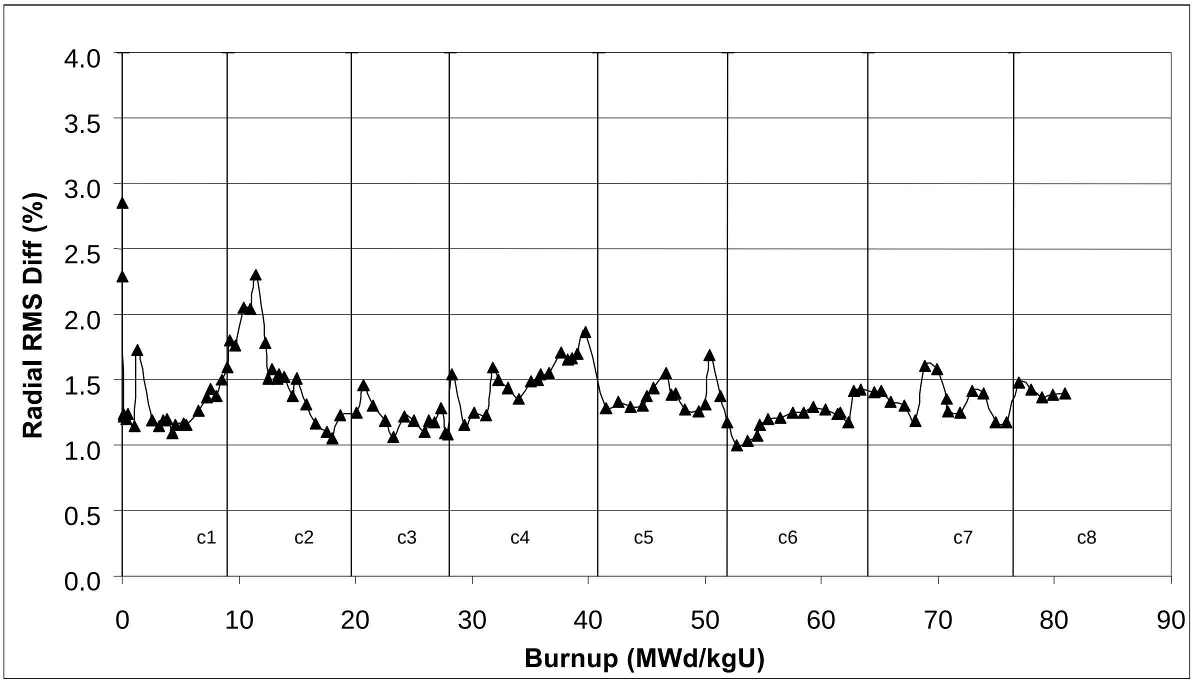

Figure 8: Hope Creek Radial TIP Results by Cycle