COMPARISON OF WIMS/PANTHER CALCULATIONS WITH MEASUREMENT ON A RANGE OF OPERATING PWR

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ABSTRACT

This paper outlines the results of recent validation studies for the WIMS/PANTHER code package for PWR reactors. The paper includes results from two independent sets of analyses which differ in the scope and diversity of validation database, as well as in details of the calculational route.

In all cases the lattice code LWRWIMS has been used to generate the data for the reactor code PANTHER, which has been used to estimate a series of parameters for comparison with measurement. The analysis has been carried out using the latest tabulation of the WIMS data library, which uses data based on the latest JEF2.2 nuclear data tabulation.

The WIMS/PANTHER package validation reported here has been carried out separately by British Energy (BE) and Tractebel Energy Engineering (TEE) on their respective validation databases in order to meet qualification requirements from their respective Safety Authorities. BE is currently using PANTHER for corefollow, reload evaluation and safety analysis for Sizewell B, and TEE, which is using the package under licence, has just started the process of merging WIMS/PANTHER into its own methodology.

1. INTRODUCTION

This paper outlines the results of recent validation studies for the WIMS/PANTHER[1] code package for PWR reactors. The paper includes results from two independent sets of analyses which differ in the scope and diversity of validation database, as well as in details of the calculational route.

In all cases the lattice code LWRWIMS[2] has been used to generate the data for the reactor code PANTHER, which has been used to estimate a series of parameters for comparison with measurement. The analysis has been carried out using the latest tabulation of the WIMS data library, which uses data based on the latest JEF2.2 nuclear data tabulation[3].

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2. OUTLINE OF WIMS FOR PWR LATTICES

LWRWIMS is a general LWR lattice code with features that have been tailored for LWR lattice geometries. Current calculations use a 69 group library based on the JEF2.2 tabulation of nuclear data which is called the '1996' WIMS library. The data for U_{236} has been enhanced on the library to accurately calculate the effect of changing to fuel with a high content of this isotope. The main features of the PWR calculational route are as follows:

WIMS Physical models :
explicit water gap;
grids homogenisation;

WIMS Numerical models :
condensed to 6 groups for pin by pin calculation;
2D XY diffusion theory (GOG);
DMOD option (local transport theory correction for neighbouring rods of perturbing cells such as guide tubes or Gd burnable absorber rods);

3. WIMS CALCULATIONAL ROUTE DEVELOPMENTS

A development of one aspect of the standard route outlined above concerns the method of correcting data for the presence of absorbers or water holes. This has been adapted to improve the behaviour of the correction throughout the depletion. The standard LWRWIMS DMOD treatment, where the diffusion coefficients are adjusted to make diffusion theory agree with transport theory, has one drawback. When a poison burns out the currents in the region of the

poisoned pin become small. This can lead to a singularity in the correction algorithm. To overcome this problem a new route was proposed based on a discontinuity factor correction. This option has been coded into LWRWIMS and results have shown that this treatment removes the problem with the singularity and gives very close agreement with the DMOD treatment in all other cases.

4. OUTLINE OF PANTHER FOR PWRS

PANTHER is a comprehensive, multi-energy group, 3D neutron diffusion program for steady-state, burn-up and transient calculations, in either rectangular or hexagonal geometries. Its primary predictions are steady-state core reactivity, transient power level and steady-state and transient rating distributions.

The code has a highly modular design, and tasks are structured accordingly using a 'controller' language, that allows looping and conditional branching. All data is stored on a database that is accessible for output, manipulation and re-use within the task or subsequent tasks.

PANTHER solves the neutron diffusion equation using either an analytic nodal method[4] in 2 energy groups (which is standard practice for PWR applications), or a semi-analytic nodal method[5] for multi-group applications. In both cases the transverse leakage is approximated by a quadratic polynomial[6]. The transverse leakage term can be adjusted to account for within node burnup and rod tips. Pin power reconstruction is achieved via a two stage process; assembly corner data is interpolated from surface data and then the pin powers are derived within the assembly using the surface and corner data, in conjunction with lattice form factor data.

PANTHER includes a PWR thermal hydraulics function for feedback purposes, and in addition it can link to the EPRI VIPRE code[7] for detailed pin-by-pin or DNBR assessments. The thermophysical properties of the fuel, gap and clad annuli have been derived from ENIGMA[8], a generic UO2 fuel performance code developed by BE for the calculation of fuel temperatures, clad stress/strain and fission product release.

PANTHER has also been linked to the RELAP whole plant analysis code[9], for example for the analysis of a steam line break.

5. DETAILS OF THE CALCULATIONAL ROUTES FOR PWR

The previous section gives details of the code capabilities. This section gives an outline of the application for PWR analysis. BE and TEE have chosen slightly different modelling options in certain key areas, and these are described below. The general model is as follows:

PANTHER model:

whole core 3D calculations
1 node/assembly radially & 12/14 uniform axial layers
no explicit grid representation

Note that BE have used 12/24 non-uniform layers dependent on axial fuel designs.

5.1. NUCLEAR DATA LIBRARY

The nuclear data library in PANTHER is based on cross-sections from a matrix of WIMS calculations, covering dependencies such as irradiation, coolant density, coolant and fuel temperature, boron and rod insertion for PWR applications. All historic effects are treated by a spectral index variation based on a depletion at a perturbed coolant density (representation of individual historic dependencies is possible but is not normal practice). The matrix of lattice calculations need not be complete, where there is no ambiguity about how the missing cases should be filled in. Interpolation is linear and there are no restrictions on the number of parameterisation points for any dependency and cross-dependencies.

The accuracy of the representation is entirely dependent on how many parameterisation points are included and how many cross-dependencies are explicitly represented by WIMS calculations. BE use a library structure based on 650 lattice calculations for a typical material. TEE have included additional cases at a high boron concentration to improve the representation above 2000ppm.

The accuracy of these parameterisations have been assessed by comparing the PANTHER library against explicit WIMS calculations, that cover the range of dependency values encountered in both operational and accident conditions. Table 1 below summarises some typical errors for the TEE library obtained from such systematic assessment while changing conditions or time discretisation for depletion.

Parameter		Mean	Stdev (1σ)
Instantaneous errors on k-inf	pcm	0.2	49
Historic errors on k-inf	pcm	72	415
Errors on			
MTC	pcm/K	0.10	0.88
Boron Worth	pcm/ppm	0.03	0.03
DTC	pcm/K	+0.06	0.16
Xenon Worth	pcm	+19	9
Control-Rod Worth	pcm	-10	15

Table 1: Nuclear Data Library Errors - All Fuel types - Rodded and Unrodded Configurations

This good agreement between PANTHER and WIMS justifies the use of this library for analysis throughout the domain of normal operation and accident conditions. To extend the library to room temperature merely requires additional WIMS calculations at cold and intermediate conditions.

5.2. REFLECTOR REPRESENTATION

Radial reflector data for PANTHER is derived from WIMS transport calculations for a 1-D slice through the core edge and reflector regions. The ‘staggered’ core edge is represented by an 'effective' baffle thickness, ie applying the physical baffle volume to the equivalent core radius rather than using the physical baffle thickness. This model improves the centre-to-edge flux tilt in 121 and 157 assembly cores. The effect on a core with 193 assemblies is much smaller and has been ignored in BE’s analysis. These transport calculations are used to produce smeared effective reflector data which preserve albedoes. TEE also derive data for an explicit model of the axial reflector, whereas BE have used a single extrapolation length. All reflector data is tabulated against moderator density and boron concentration.

5.3. CONTROL ROD REPRESENTATION

All cross-section data for PANTHER, including rodded assembly data, is derived from single assembly lattice calculations with no leakage, other than the critical bucklings used to derive the condensation spectra. This has been shown to consistently under-predict control rod worths by 3% for a 1-in-9 supercell environment, which is more typical of rodded core conditions. The comparison is against a reference WIMS8[10] transport solution. WIMS8 is the successor lattice code to LWRWIMS and provides the most accurate multi-assembly transport solution. Therefore the comparison includes LWRWIMS to WIMS8 differences as well as the effect of the supercell environment. These two effects make contributions to the 3% of a similar size.

6. SUMMARY OF COMPARISONS

A wide range of reactor cores have been analysed by this route, ranging in size in both axial and radial directions. The fuel types vary in both fuel geometry and fuel composition. Some of the reactors include heavily poisoned fuel. The reactors studied are shown below

Reactor	Power MWth	No of Cycles Analysed	No. of Assemblies	Assembly type	Core Height (feet)
Callaway	3411	3	193	17x17	12’
Wolf Creek	3411	1	193	17x17	12’
Sizewell B	3411	3	193	17x17	12’
Tihange1	2652 (1)	20	157	15x15	12’
Tihange2	2775	11	157	17x17	12’
Tihange3	2988	12	157	17x17	14’
Doel1	1192	15	121	14x14	8’
Doel2	1192	14	121	14x14	8’
Doel3	2775(2)	13	157	17x17	12’
Doel4	2988	14	157	17x17	14’

(1) comparison covers cycles uprated to 2867 MWth
(2) comparison covers cycles uprated to 3054 MWth

A range of comparisons of predictions with measurement has been carried out and the resultant agreement is reported in the paper. These measurements included the following

- Measurement at Hot Zero Power (HZP)
    - Critical boron concentrations at various rod bank configurations
    - Bank worth of various rod bank configurations
    - Isothermal temperature coefficients
    - Boron worth
- Measurements at Power
    - Doppler coefficients
    - Flux maps at various power levels

Other more detailed studies have been performed but are not reported here for brevity, eg xenon swing and rod shadow tests, coastdown, turbine trips and general corefollow.

6.1. TEE VALIDATION DATABASE

- core types: Tihange 1-3, Doel 1-4
  121 and 157 fuel assemblies
  8’, 12’and 14’ high
- wide variety of PWR fuel types from different fuel vendors
  lattice type (14x14 15x15 17x17)

- enrichments from 1.90 to 4.25 w%U
- Reprocessed Enriched Uranium (REU) 4.16 to 4.24 w%U
- Gd burnable poisons : 8%-10%Gd_2O_3; 4-8-12-16 rods; on Unat & 2.0 w%U support,
- pyrex discrete poisons (5 fresh cores)
- different cycle strategies
- 8 to 18 months cycles
- core discharge burnups up to 15 GWd/t
- max. assembly burnups up to 51 GWd/t
- out-in, in-out, hybrid and low-leakage loading patterns
- different core conditions
- power uprating, plugging of steam generator tubes

This is a very heterogeneous database for validation. The use of a common calculational route for all reactors and all operating conditions presents a significant modelling challenge.

Summary of the statistical analysis of the TEE validation results over 90 cycles

		Predicted – Measured		Sample
HZP BOC Parameters		Mean	1s	size
Critical boron conc.				
ARO	[ppm]	1	26	96
all rodded config's	[ppm]	-13	19	231
Rod bank worths	Δr[%]	0.4	5.4	295
MTC	[pcm/C]	-2.7	1.3	90
Boron worth	Δr[%]	-2.7	5.5	92

where Δ_r denotes a (PANTHER/Measured -1) difference.

		Predicted – Measured		Sample
HFP Parameters		Mean	1s	size
Critical boron conc.	[ppm]	-10	31	930
2D RRs	Δr[%]			
unfiltered population		0.0	1.5	41427
UO2 fuel pop.		0.0	1.5	32371
fresh Gd fuel pop.		0.2	1.3	2709
Fz det	Δr[%]	-1.2	1.3	930
AO det	Δr[%]	-0.8	1.1	930
FNDH	Δr[%]	0.4	0.9	930
FQ	Δr[%]	-3.5	1.8	930
Natural cycle length	[MWd/t]	-139	365	94

where “2D RRs” denote the relative difference on the axially integrated detector reaction rates from flux maps at Hot Full Power. “Fz det” and “AO det” refer to the axial form factor and the axial offset of core averaged detector traces. “Gd fuel” denotes the fresh Gd_2O_3 poisoned fuels, regardless of type.
Note that FNDH and FQ statistics refer to comparisons with reconstructed power peaking factors from detector response analysis.

6.2. BE VALIDATION DATABASE

- core types: Wolf Creek, Callaway, Sizewell B, Tihange 2
  157, 193 fuel assemblies
  12’ high
- PWR fuel types from different fuel vendors
  lattice type (17x17)
  enrichments from 2.1 to 4.4 w%U
  Gd burnable poisons : 6%-8%Gd_2O_3; 8-16-20-24 rods; on Unat & 2.0 w%U support, Pyrex, WABA discrete poisons, and one cycle with IFBAs
- cycle strategies
  8 to 18 months cycles
  core discharge burnups up to 18 GWd/t
  max. assembly burnups up to 39 GWd/t

This database is much more limited than the TEE experience. Note also that cycles 1-11 of Tihange 2 are common to both the TEE and BE databases.

The results are very similar considering the different scope of the validation studies. The only exception is in the systematic difference on rod worth. Unlike TEE, BE make no correction for the 3% over-prediction seen in supercell calculations (see above), and this bias is seen in the comparison with measurement. On the common Tihange 2 cycles the impact of the different modelling options has been seen to be very low.

Summary of the statistical analysis of the BE validation results over 18 cycles

		Predicted – Measured		Sample
HZP BOC Parameters		Mean	1s	size
Critical boron conc.				
ARO	[ppm]			
all rodded config's	[ppm]	8	21	38
Rod bank worths	Δr[%]	4.2	5.8	96
MTC	[pcm/C]	-2.2	0.9	29
Boron worth	[pcm/ppm]	0	0.35	15

where Δ_r denotes a (PANTHER/Measured -1) difference.

		Predicted – Measured		Sample
HFP Parameters		Mean	1s	size
Critical boron conc.	[ppm]	-2	22	193
2D RRs	Δr[%]			
unfiltered population		0.0	1.3	13398
Fz det	Δr[%]	-0.9	1.5	13398
AO det	Δr[%]	0.0	1.3	231

6.3. ANALYSIS OF TRIPS AND POWER MANOEUVRES

A number of trips and deloads have been analysed using PANTHER in order to determine the accuracy of the power coefficient and short-lived nuclide modelling. Reactivity comparisons are only possible at times when the plant boron level is known accurately, for example from tritrated samples before trips and immediately prior to start-up, or deduced from continuous boration/dilution data during deloads.

An example of the latter occurred during a controlled shutdown at EOC, when no boration/dilution took place during a 10 hour period in which power was reduced from 55% to 0%. Figure 1 shows that the predicted boron levels vary by less than +/-2ppm during this period (from time 0 to 10 hours), which suggests a very low error in the combination of power coefficient, transient xenon and rod worth.

Boron changes over trips are summarised in table below. The difference in boron level prior to the trip has been subtracted from the results. Apart from the long trips, restart reactivity is predicted within -1ppm and +21ppm. The long trips (over 100 hours) are anomalous with differences of -30ppm and -36ppm. PANTHER models the simple xenon and samarium chains explicitly, but takes no account of other nuclides with a transient response, such as rhodium and neptunium. A second model was attempted in which the samarium data was adjusted to allow for these other nuclides; this amounts to a reduction in the samarium response. The impact on the trip modelling is shown in the table. The discrepancy for long trips is reduced, but even the entire removal of the samarium model leaves a bias. This suggests that the net effect of nuclides other than xenon over a period of several days is either neutral or a reactivity *rise*.

Cycle Burnup MWD/te	Trip Length hours	Pre-trip Power percent	Boron Error with Xe and Sm ppm	Boron Error With Xe adjusted Sm ppm	Boron Error With Xe only ppm
52	51	low	3	4	6
591	11	43	7	7	8
2113	35	100	-1	3	17
4310	186	100	-30	-17	16
303	17	100	9	9	16
11387	23	100	21	25	32
162	19	73	2	3	8
1637	132	100	-36	-25	-2
2086	28	100	19	22	32

where errors are given as PANTHER-measurement

7. CONCLUSION

The results show excellent agreement with measurement for the wide range of cores and fuel types studied. Similar results are reported by both BE and TEE, despite differences in the calculational routes and in the scope of the validation databases. This indicates that the WIMS and PANTHER methods are robust and accurate for PWR cores.

REFERENCES

1: P.Hutt et al. “The UK Core Performance Code Package”, Nuclear Energy, J. British Nuclear Energy Society, 30, 291 (1991).

2: M J Halsall, D Powney, M R Moxom, "LWRWIMS, The WIMS Code for Light Water Reactors", International Conference on the Physics of Reactors - PHYSOR 90, Marseille, April 1990

3: C Nordborg, M Salvatores, "Status of the JEF Evaluated Data Library", Proceedings of the Conference on Nuclear Data for Science and Technology, Gatlinburg, Tennessee, May 1994

4: K S Smith, "An Analytic Nodal Method for Solving the 2-Group, Multi-Dimensional, Static and Transient Neutron Diffusion Equation", PhD Thesis, MIT 1979

5: R A Shober, “A Nodal Method for Fast Reactor Analysis”, Proc. Computational Methods in Nuclear Engineering, ANS, Williamsburg, 1979

6: H. Finnemann, F. Bennewitz and M. Wagner, "Interface Current Techniques for Multidimensional Reactor Calculations", Atomkernenergie Vol 30 page 123, 1977

7: C W Stewart et al. “VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores”, Vol 1-5, Rev 3, NP-2511-CCM-A, EPRI 1989

8: R J White et al. “Capabilities and Validation of the ENIGMA Fuel Performance Code”, International Topical Meeting on LWR Fuel Performance, Avignon, Vol 2 p 919, 1991

9: RELAP5/MOD3.2 Code Manual, NUREG/CR-5535

10: M J Halsall, "WIMS8 - Speed with Accuracy", International Conference on the Physics of Nuclear Science and Technology, Long Island, USA, Oct 1998

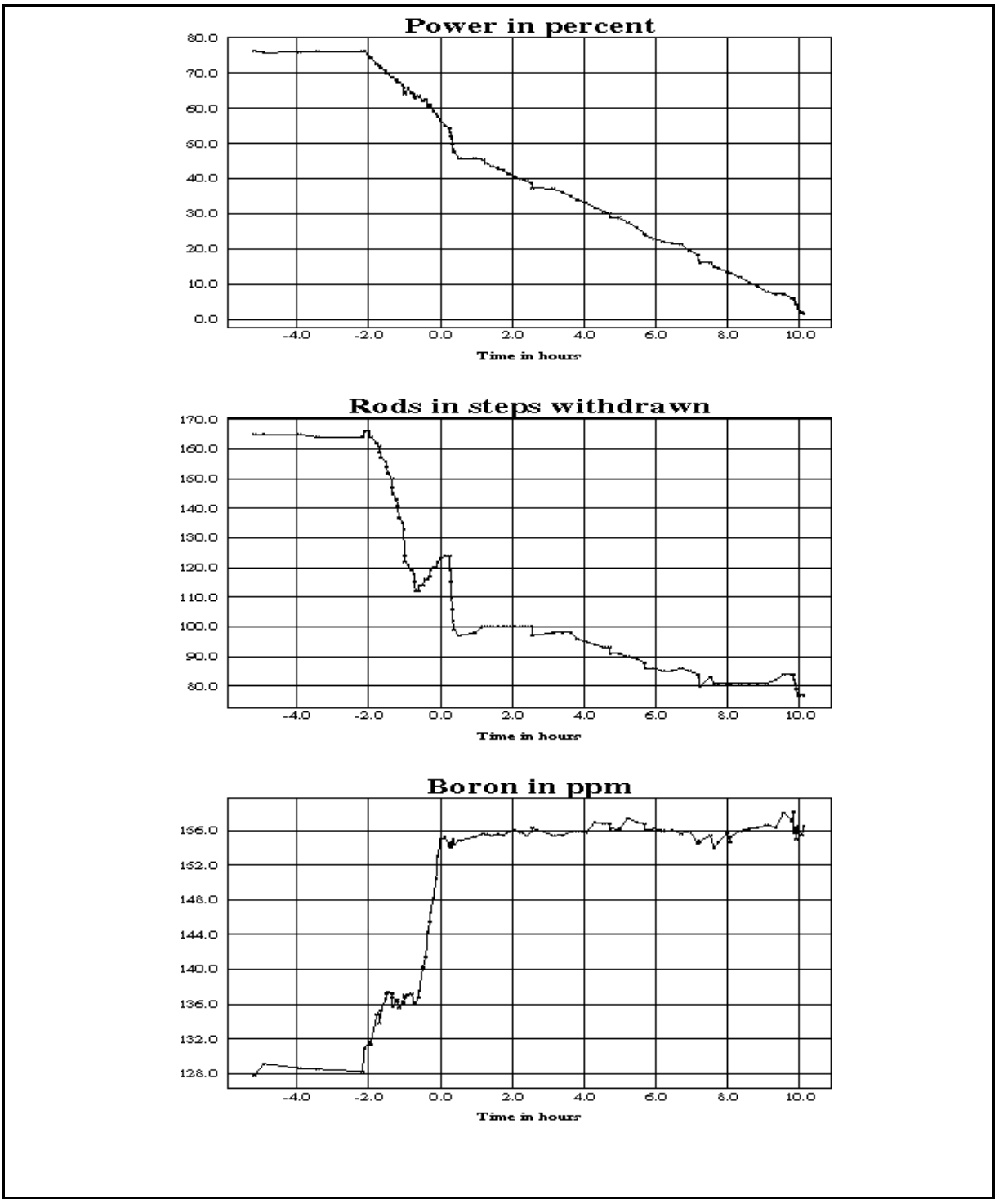

Figure 1: Controlled Power Reduction at End of Cycle