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VALIDATION OF THE RFSP-IST CODE AGAINST POWER-REACTOR MEASUREMENTS

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ABSTRACT

This paper provides an overview of the validation of the computer code Reactor Fuelling Simulation Program – Industry Standard Tool (RFSP-IST) version 3.00-05. The validation is structured according to the CANDU[1] physics phenomena identified in the Reactor Physics Validation Matrix. In the validation, lattice-cell properties are calculated using the WIMS-IST-based Simple-Cell Model. The results of some of the completed validation tasks are presented, with each task linked to the relevant phenomena. Finally, additional validation tasks to bridge some of the existing gaps and to fully qualify RFSP-IST for all intended applications are identified and discussed.

1. INTRODUCTION

RFSP-IST [1] is the Industry Standard Tool for finite-core physics design and safety analysis. It was created by merging the OHRFSP code from Ontario Power Generation (OPG) and the RFSP code from AECL. Software Quality Assurance standards in the Canadian nuclear industry require that all IST codes be subjected to quality assurance consistent with CSA Standard N286.7, and in particular that formal code validation be demonstrated.

Concurrent with the validation activities, the nuclear industry agreed to a significant change in physics analysis methodology, namely the replacement of the basic lattice-cell code POWDERPUFS-V [2] by WIMS-AECL [3], and associated changes in device-modelling methodology and full-core simulation methodology consistent with the multi-group structure of the new WIMS-AECL lattice-cell properties. As a result, the development and implementation of the new methodology and validation activities became interwoven.

The RFSP-IST validation plan and tasks were designed to address the modelling of the CANDU reactor-physics phenomena identified as characteristic and/or important in postulated accident transients. This paper presents an overview of the validation plan and methodology, and the current status of the validation activities. In addition, further code development and completion of validation to the requirement of compliance with CAN/CSA-N286.7-99 [4] will be discussed.

[1] CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL)

2. THE RFSP-IST CODE AND WIMS-AECL-BASED METHODOLOGY

The RFSP-IST code is identified as the Industry Standard Tool for finite-reactor (as opposed to lattice) physics analysis and design. It provides a wide range of core-simulation capabilities with different levels of sophistication. The most essential functional requirement of the code is that it provides accurate three-dimensional (3-D) flux and power calculations.

In general, finite-reactor physics analysis is performed in three stages. The entire process relies on a suite of computer codes, each one part of the Industry Standard Toolset for CANDU reactor analysis:

i) the transport code WIMS-IST for the calculation of lattice-cell properties (the IST version of the code is WIMS-AECL release 2.5d, with the 89-energy-group ENDF/B-VI nuclear-data library)

ii) the transport code DRAGON [5] for three-dimensional (3D) modelling of reactivity devices

iii) the RFSP-IST code for full-core analysis

While WIMS-IST is firmly grounded in neutron-transport theory, the computation of bundle-specific lattice-cell properties for use in local-parameter history-based calculations (wherein lattice properties vary through the core, and in time, because of different local conditions such as fuel temperature, coolant density, flux level, xenon concentration, moderator-poison concentration, etc.) is too time-consuming with WIMS-IST. To remove that difficulty, the Simple-Cell Methodology (SCM) has been implemented in RFSP-IST for the calculation of lattice properties [6]. SCM acts as a surrogate for WIMS-IST; it solves the diffusion equation with a simple three-region cell model in annulus geometry and reproduces WIMS-IST results with reasonable accuracy and in a much shorter computation time. For compatibility with the 2-energy-group WIMS-IST-based lattice properties, the device incremental cross sections are now generated with the multigroup transport cell and supercell code DRAGON. The "full 2-energy-group" diffusion equation is now solved in RFSP-IST (as opposed to the "1.5-energy-group" version used with POWDERPUFS-V-based lattice properties).

The major functions and calculation modules in RFSP-IST include:

- design of the time-average 3D flux and power distributions (*TIME-AVER module)
- fuel-management calculations, i.e., simulations of operating-reactor core-tracking histories, including channel refuellings and core burnup steps (*SIMULATE module)
- simulation of random snapshots of reactor-core flux and power distributions (*INSTANTAN module)
- simulations of the effects of reactivity-device movements
- simulations of the asymptotic bulk- and spatial-control functions of the Reactor Regulating System (RRS)
- ^{135}Xe-^{135}I kinetics calculations, i.e., simulations of xenon transients
- three-dimensional kinetics calculations, i.e., simulations of fast transients such as those initiated by hypothetical large-loss-of-coolant accidents (*CERBERUS module), or slow transients with modelling of the reactor regulation system (RRS) response (*CERBRRS module)

With the exception of the flux-mapping calculations, all other calculation methods and options are now functional in the full two-group formalism using WIMS-IST-based lattice properties and DRAGON incremental device properties. Although the option of using the previous POWDERPUFS-V-based methodology is still available in RFSP-IST, all validation tasks and results discussed here focus on the WIMS-IST based two-group methodology.

3. VALIDATION METHODOLOGY

The validation process followed for RFSP-IST is consistent with the industry-wide practices as defined in the Technical Basis Document. The relevant primary phenomena governing the core behaviour during postulated accident conditions were identified:

PH1	Coolant-Density-Change Induced Reactivity
PH2	Coolant-Temperature-Change Induced Reactivity
PH3	Moderator-Density-Change Induced Reactivity
PH4	Moderator-Temperature-Change Induced Reactivity
PH5	Moderator-Poison-Concentration-Change Induced Reactivity
PH6	Moderator-Purity-Change Induced Reactivity
PH7	Fuel-Temperature-Change Induced Reactivity
PH8	Fuel-Isotopic-Composition-Change Induced Reactivity
PH9	Refuelling Induced Reactivity
PH10	Fuel-String-Relocation Induced Reactivity
PH11	Device-Movement Induced Reactivity
PH12	Prompt/Delayed Neutron Kinetics
PH13	Flux-Detector Response
PH14	Flux and Power Distribution (Prompt/Decay Heat) in Space and Time
PH15	Lattice-Geometry-Distortion Reactivity Effects
PH16	Coolant-Purity-Change Induced Reactivity

The validation process started with a Validation Plan, which identified and discussed the applications of the code for modelling each phenomenon, and the experimental or analytical data sets for validation of such applications. A series of validation tasks were then carried out, each focused on testing the code against a specific data set. Note that in a realistic operational manoeuvre or postulated accident transient, very often several phenomena occur simultaneously. The same situation is also reflected in the measurement data used for code validation. One validation task often addresses more than one phenomenon. The results from the various validation exercises have been summarized and collected in the RFSP-IST Validation Summary.

In accident analysis, the key output parameters required from RFSP-IST are:

- Reactivity transient
- Flux distribution
- Bundle and channel power distribution
- Process signals from in-core detectors and out-of-core ion-chambers
- High neutron-power and rate-log trip times, and shutdown-system actuation time
- Device response to perturbations

Implicit in the requirements for application of RFSP-IST to transient analyses is that the starting point of a postulated event has to be properly defined and modelled. Therefore, the simulation of the pre-event steady-state core configuration is part of the application of the code to safety analysis.

A few points are worth noting in the context of RFSP-IST validation:

i) Some phenomena lead to reactivity changes that are related to the underlying characteristics of the lattice design, and the effects are largely uniform in the core. Examples are coolant density-, temperature- and purity-change induced reactivity (PH1, PH2, and PH16); fuel-temperature-change induced reactivity (PH7); fuel-isotopic-composition-change induced reactivity (PH8); and moderator density-, temperature- and purity-change induced reactivity (PH3, PH4 and PH6). Such reactivity effects are principally captured in lattice-cell calculations, and appear in RFSP-IST calculations via variations in the WIMS-IST based lattice-cell properties. Measurements performed in the ZED-2 reactor are the primary source of validation of these phenomena (except for PH8). Since the major determinant of how well the physics toolset models these phenomena

is related to the WIMS-IST performance, these phenomena are explicitly addressed under WIMS-IST validation, with appropriate scaling to power reactor conditions. The WIMS-IST validation is sometimes also supplemented by WIMS-IST/RFSP-IST benchmarks against power-reactor measurement data for typical operating conditions with a full range of fuel burnup. While the conditions during power reactor measurements are often closest to those of interest, they usually involve several phenomena at the same time, and the uncertainties of the measurements are large relative to the controlled ZED-2 experiments. These phenomena and measurements will not be the focus of this paper.

ii) Given the importance of coolant void reactivity, the validation of WIMS-IST modelling of coolant void reactivity has been the subject of an extensive series of measurements and analysis [7]. This phenomenon is addressed separately, involving both WIMS-IST and scaling of the code uncertainties to power-reactor nominal operating and accident conditions, and will not be discussed in this paper.

iii) RFSP-IST models require incremental properties that represent the effect of the reactivity devices. The scope of RFSP-IST validation includes the device reactivity effect and the global flux-perturbation effect due to device movements. However, the generation of the incremental properties and the validation of the local fine-flux perturbation caused by the devices are addressed in the validation of DRAGON.

iv) The Simple-Cell Methodology (SCM) is an integral part of RFSP-IST, and is designed to closely reproduce WIMS-IST results.

It should be noted that any full-core validation task is implicitly a validation of the entire suite of RFSP-IST, WIMS-IST, SCM and DRAGON codes.

The rest of this paper presents illustrative results from comparisons of RFSP-IST against power-reactor measurements from the various tasks performed for the validation of the reactor-physics toolset.

4. KEY VALIDATION RESULTS

All of the diffusion-theory-based major modules of the RFSP-IST code have been tested and validated, including static snapshot simulations as applied in core-tracking calculations, time-average calculations, and time-dependent kinetics calculations. The accuracy of the nominal core-state flux shape calculations has been demonstrated through core-follow simulations, and the accuracy of the delayed-neutron kinetics modelling has been demonstrated through SDS1 trip test simulations. Validation of flux-shape calculations by comparison with in-core detector readings (vanadium detectors and ROP detectors) has been extensive.

4.1 Validation of SCM for the Calculation of Lattice-Cell Properties

The SCM methodology starts from post-processing of WIMS-IST calculation results, and ends with 2-group lattice properties provided to the RFSP-IST core model. The validity of SCM has been demonstrated by comparison with WIMS-IST results (essentially at the lattice-cell level). The validation includes the following tests:

a) A lattice-cell 37-element fuel burnup calculation from 0 to 200 FPD at typical CANDU operating conditions. The k-eff values calculated by SCM and by WIMS-IST agreed to within ±0.2 mk.

b) A 4-hour xenon transient after a reactor shutdown. SCM underestimated the xenon load by 0.7 mk (about 1% of the 65 mk xenon reactivity increase) at the end of the 4-hour transient.

c) Coolant-void reactivity calculations for a fresh 37-element fuel bundle. The results from SCM agreed with those from WIMS-IST to within ±1%.

d) A LOCA transient using a simplified core model with all fresh 37-element fuel bundles. The peak transient reactor power with SCM lattice properties was 2% higher than that with WIMS-IST properties. The agreement in transient reactivity was within ±0.1 mk.

e) A LOCA transient using a realistic CANDU 6 model with a full range of fuel burnup at equilibrium-fuelling conditions, and a stylized thermalhydraulic transient. The agreement for peak channel power and peak bundle power was within ±0.7%. The agreement in transient reactivity was within ±0.1 mk.

The results of these tests show that SCM, when applied either to static or to kinetic calculations, closely duplicates WIMS-IST results.

4.2 Core-Follow Simulations

One of the basic applications of RFSP-IST is to core-follow simulations. This functionality was tested at an early stage of development of the WIMS-IST/SCM/RFSP-IST methodology. A one-year period of Point-Lepreau operating history was tracked, using the (only slightly different) code versions available at the time. Reactivity-device incremental cross sections were generated using a modified MULTICELL version with two-group material cross sections derived from WIMS-IST and a superhomogenization (SPH) scheme. These code versions are expected to produce results very close to those of the current code versions.

The accuracy of the flux-shape calculations performed is demonstrated by comparing the computed and measured vanadium detector fluxes. After an initial period, the standard deviation of the differences between the computed and measured vanadium-detector fluxes at each snapshot simulation is around 3%. Other key results of the core-follow simulations were also examined. The excess reactivity represents a bias in the computed k-eff from criticality, and no divergent trend was observed. The predicted maximum channel and bundle powers were very much consistent with those previously obtained using the conventional PPV/RFSP methodology. The relative consistency of these key parameters (excess reactivity and vanadium fluxes comparisons) demonstrates the code stability and convergence for routine core-follow applications.

4.3 Time-Average Burnup Calculations

An important element in the core-follow application is the tracking of fuel burnup. The average exit burnup is related to the fuelling rate, and thence to the fuelling cost and the economy of power generation. It is therefore essential to confirm the accuracy of the average exit-burnup estimate from the time-average calculations. The validation was performed by comparison of Point-Lepreau data from about one year of operation (1994 March to 1995 April). Most of the refuellings performed in this period used an eight-bundle-shift scheme; the total number of bundles discharged was 5486. For this time-average calculation, the average core conditions, namely the core parameters such as moderator purity, poison concentration, zone fill levels, etc. were obtained by averaging the actual station data as recorded and used in the core-follow production runs at Pt.-Lepreau. These production runs from site (PPV/RFSP) were also re-simulated using the WIMS-IST/RFSP-IST methodology, and the excess reactivity, channel-power and zone-power distributions were averaged over time. The averaged core parameters were used to perform a time-average calculation, with the exit burnup distribution adjusted to match the average excess reactivity and the average channel and zonal power distributions. The computed average exit burnup was then compared to the actual burnup achieved at site over that time period.

The following results were obtained:

i) The discharge burnup from the time-average calculations of 175.7 MW·h/kg(U) compares well with the “measured” average discharge burnup actually attained over the 1-year operating history, which was 177.8 MW·h/kg(U). Thus, the RFSP-IST-calculated time-average burnup is 1.2% less than the average discharge burnup deduced from station data.

ii) The time-average bundle feed rate was 15.34 bundles/FPD, which compares very well with the “measured” fuelling rate for the 12-month operating history, 15.24 bundles/FPD.

iii) Comparisons of vanadium-detector responses show that the simulated flux values from the time-average calculations reproduce very well the year-average vanadium measurements, for each of the 102 individual detectors under consideration. The average difference was about -0.18 %, with a standard deviation of 3.1%, consistent with the results obtained in other core-follow simulations.

4.4 Reactivity-Device Modelling

The modelling of CANDU reactivity devices in RFSP-IST, using device incremental cross sections generated through 3-D neutron–transport calculations with the DRAGON code, has been successfully validated against ZED-2 experiments and against device-calibration tests in power reactors. This paper presents the results from a comparison of reactivity-device modelling with measurements in two different CANDU reactors: light-water liquid zone-controllers (LZC) and adjuster rods in Pickering-A Unit 4, and shutoff rods (SOR) and mechanical-control absorbers (MCA) in Darlington NGS-A Unit 4. All reactivity measurements were performed during Phase-B commissioning with fresh natural-uranium fuel, at low reactor power. All the reactivity-worth calculations were performed in two-energy-group diffusion-theory simulations using the RFSP-IST code, with DRAGON-generated 2-group incremental cross sections used to represent reactivity devices in the core model. All the lattice-cell calculations for the reference lattices without reactivity-control devices present were performed using the WIMS-AECL version 2.5c code with the 89-energy-group ENDF/B-V nuclear data library.

The device-calibration measurements were made by reactivity balance against dissolved neutron poisons in the moderator, and by inter-device reactivity balance. The procedure used to perform calibration tests and the method used to measure the reactivity worth of the control devices were as follows:

1) Batches of moderator poison (gadolinium, boron), designed to add specific “amounts” of negative reactivity, were prepared in advance.
2) The Liquid-Zone Controller System worth was determined by observing the change in average zone level (AZL) which results from the addition of these poison batches.
3) Other reactivity-device worth is determined by observing the change in AZL that results from device insertion or withdrawal.

The reactivity worth of the LZCs in the Pickering-A Unit 4 reactor was measured by recording the change in the average LZC fills required to balance the addition of pre-weighed amounts of boron dissolved in the moderator. The criterion used to assess the accuracy of calculations, referred to as “simulation error”, was computed from the difference between the reactivity changes calculated from changes in moderator boron and changes in LZC fills:

$$\text{Simulation Error (\%)} = 100 \times \frac{\rho_{LZC} - \rho_{Boron}}{\rho_{Boron}} \quad (1)$$

The reactivity worths of the adjusters in the Pickering-A Unit 4 reactor, as well as the reactivity worth of the SORs and MCA rods in Darlington NGSA Unit 4 reactor, were measured by recording the changes in average LZC fills required to balance changes in rod configuration. In these cases, the error in the simulation of the commissioning tests was computed from the difference between the reactivity changes calculated from changes in device configuration and changes in LZC fills:

$$\text{Simulation Error (\%)} = 100 \times \frac{\rho_{ADJ} - \rho_{LZC}}{\rho_{LZC}} \quad (2)$$

Note that the method of measurement does not yield information on the absolute reactivity worth of either the zones or the rods, but only on the reactivity balance between those systems. The absolute reactivity worths of devices are available only from calculated reactivity changes corresponding to configuration changes and calculated moderator poison reactivity coefficients.

The general trend for all device-calibration cases was that the calculated device reactivity worth is lower than the calculated worths of the observed zone-level changes, or the calculated worths of the recorded changes in moderator boron (in the case of liquid-zone-control calibration). For example, the simulation results of LZC reactivity-worth measurements with all adjuster rods inserted in-core and with all adjuster rods out-of-core are presented in Tables 1 and 2 respectively. The average simulation error and standard deviation are –4.2 ± 5.9% in the configuration with all adjusters inserted, and –6.3 ± 2.6% with the adjusters withdrawn.

The calculation accuracy obtained for each of the measured devices from all the individual tests with various device configurations and insertion/withdrawal scenarios can be summarized as follows:

i) The reactivity worth of the liquid zone controllers was calculated to be on average 5.3 ± 2.4% less than the calculated worth of measured changes in moderator boron concentration.

ii) The reactivity worth of adjuster rods was calculated to be on average 5.7 ± 1.0% less than the calculated reactivity worth of measured changes in the average LZC fills.

iii) The reactivity worth of individual shutoff rods was calculated to be on average 6.6 ± 4.2% less than the calculated reactivity worth of measured changes in the average LZC fills. Similarly, the reactivity worth of the mechanical-control absorbers was found to be lower on average by about –3.4 ± 2.3% relative to the reactivity worth of changes in average zone-controller fills.

4.5 Saturating-Fission-Product Reactivity Transients

Fuel-isotopic-composition change, as a function of time, irradiation and reactor power level, is one of the main factors governing variations in reactivity on a time scale of minutes to days, and has a direct impact on the burn-up distribution that determines the flux and power shape at the time of a postulated accident. Also, various reactivity coefficients, including the coolant void reactivity, are dependent on fuel burnup.

One aspect of fuel-composition changes relates to the saturating fission products (SFPs). The number densities of these fission products are a function of power history, and evolve with time and have a direct impact on the system reactivity. For example, during a long shutdown, significant changes will occur in fuel nuclide compositions; most notably, ^{135}Xe decays away. Moderator poison is used to suppress the excess reactivity at reactor restart after a long shutdown. During the restart power ramp-up, the SFPs would start to build up, and the moderator poison concentration would be reduced accordingly to maintain the reactivity balance. Such a shutdown and restart transient (1993 April outage at Point Lepreau) was simulated to test the SFP modelling in RFSP-IST. This analysis served to validate the WIMS-IST/RFSP-IST calculations of the reactivity effect of saturating-fission-products (SFP), as well as moderator–poison-change reactivity effects.

Upon restart, the reactor was first brought to a critical state with all adjusters out of the core. In the next few hours, a portion of the moderator poison was pulled to compensate for adjuster banks re-insertion while still at low power (0.3% FP). The power was then raised slowly back to full power over the following 2 to 3 days. As the power increased and the transient concentrations of ^{135}Xe and other SFPs were evolving, the poison was being pulled from the moderator to maintain criticality and to maintain the zone controllers in their normal operating range.

Start-up simulations were performed from the period just before reactor shutdown to criticality at the restart, and following the power ramp-up transient. The measured poison concentrations and individual zone levels from site were used to perform each of the core-state simulations. The computed core reactivity was compared to a reference value, taken as the k-eff value from the pre-shutdown core-state simulation. Deviation from this reference value is considered as the simulation error in modelling the SFP reactivity transient, together with the error in moderator poison reactivity calculations.

Overall, the agreement in core reactivities is quite good, with an average error and standard deviation of –0.68 mk ± 0.59 mk. The discrepancies are smaller than ± 0.4 mk at the beginning and towards the end of the start-up transient. Evidently, the highest discrepancies are observed when both boron and

gadolinium are present in high amounts in the core, and the concentrations of SFPs are furthest from their full-power equilibrium values.

This good agreement obtained in tracking the core reactivity in a restart transient shows that the SCM history-based method within RFSP-IST predicts very accurately the reactivity balance between changes in SFP concentrations and moderator poison.

4.6 Validation of Kinetics Calculations using a SDS1 Trip Test

The accuracy of the modelling of the prompt and delayed neutron kinetics in a full-two-group formalism in *CERBERUS, the kinetics module of RFSP-IST, has been demonstrated using flux/power run-down data in a SDS1 trip test at Point-Lepreau. The transient covered the flux run-down out to 200 seconds, which corresponds to the length of the recorded data.

The *SIMULATE module was used to simulate the 8-month reactor shutdown and the following power ramp to 75% full power. The *CERBERUS module, with lattice properties calculated with SCM, based on WIMS-IST, was used to solve the time-dependent neutron-diffusion equation in three dimensions and two energy groups, to simulate the trip transient. The *INTREP module was used to calculate the response of each Regional Overpower Protection (ROP) detector, RRS detector, and ion chamber. The RFSP-interpolated fluxes at the ROP detector positions were processed to model the detector dynamic responses and then compared with the measured detector readings during the rundown.

The major uncertainties were related to accuracy of measurements, and assumptions in reactor conditions and modelling limitations:

a) The predicted response of the detectors relative to the measured response is affected by two uncertainty components: (i) the timing of the SOR drop characteristics, and (ii) the uncertainty in the time base of detector measurements. These measurement uncertainties are expected to have a similar magnitude to those typically reported for the annual SDS1 trip tests in PLGS, estimated as ± 42 ms for SOR timing delay, and estimated as ± 32.5 ms for detector time measurements. Thus, in comparing predictions with measurements, the root-mean-square (RMS) combined error of ± 53.1 ms is used to estimate the expected accuracy.
b) The local thermalhydraulic conditions (coolant temperature and density), calculated by the NUCIRC code for each bundle in the core, were held constant throughout the rundown transient, and then set at hot-shutdown conditions after the SORs and MCAs are fully inserted. Similarly, the fuel temperature was kept constant during the power rundown. The fuel-temperature distribution was calculated from bundle powers obtained just before the trip test.

Fifteen-group delayed-neutron precursor groups were used because the simulation lasts over 3 minutes. Delayed fractions and time constants for each precursor group were calculated individually for each fuel bundle, as required by the history-based local-parameter methodology employed in *CERBERUS. Delayed-neutron precursors are assumed at equilibrium with the flux level prior to the transient. The total delayed fraction was 5.685 mk.

The differences between the predicted and measured flux run-down were analyzed in two domains – the flux level difference (signal difference) at a given time, and the time difference (time shift) when the flux ran down to a given level. The signal domain and the time domain statistics were compiled on the signal interval ranging from 20% to 90% of the initial signal. The overall mean and standard deviation in signal difference and time shift for all detector types are presented in Table 3. The error was calculated as (RFSP calculated value – site-measured value). Therefore, a positive signal difference means that the measured detector response is lower than the calculated response, and a negative time error means that the calculated signal leads the measured signal.

In general, the results of ROP-detector (34 SDS1 and 24 SDS2 detectors) calculations show good agreement between RFSP *CERBERUS and the site measurements. The magnitude of the difference ranges from 1.6 to 44.5 ms, well within the measurement uncertainty. On average, the calculated detector response for SDS1 leads the measured signal by 8.48 ms, whereas for SDS2 detectors, the calculated

signal leads the measured response by 22.7 ms. The slope of the calculated and measured rundown are, however, almost identical to each other in every case.

The comparison results for the 28 RRS detectors indicate that RFSP is correctly modelling the delayed components of the detectors. The RFSP results show reasonably good agreement with the average uncompensated RRS-detector readings, with RFSP trailing the measured data on average by 33.3 ms (all detectors were considered except 12A, which showed irregular readings).

In the signal domain, the calculated average linear signal of all 8 ion chambers (SDS1, SDS2, and RRS) show reasonable good agreement with measured data, and for the Log signals (in decades) and Log-Rate signals (in fractional-units/s), the agreement levels are seen to be good. In the time domain, the measured signal leads the RFSP calculated value by 27.6 ms, with a standard deviation of 18.9 ms.

As an example, the comparison between the measured and calculated log-rate signal for an ion chamber is shown in Figure 1. Similarly, the comparison between the measured and calculated value for a SDS2 detector signal is shown in Figure 2.

5. SUMMARY

The validation of the reactor-physics industry standard toolset has progressed to an advanced stage. Results from the validation tasks performed have demonstrated that the migration to the WIMS-IST/RFSP-IST methodology is mostly complete, and the majority of the functionalities required of the code suite have been tested and shown to give reasonable results when compared with site measurement data. This paper has provided some examples of the validation specific to RFSP-IST with benchmark data from power-reactor measurements.

The validation of the reactor-physics code suite will include the separate validation of WIMS-IST, DRAGON and RFSP-IST, and this is now well underway. The next step is to integrate the separate validation results for these three codes, supplemented by MCNP inter-code comparisons and, where necessary, scaling of the uncertainties to power-reactor operating conditions and postulated accident conditions. The integration will lead to a Reactor-Physics Toolset Validation Manual, which will provide a statement on the calculation errors in the modelling of each of the sixteen phenomena when the IST-code toolset is applied to safety analysis.

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Table 1
LZC Reactivity Worth with Adjusters In-Core

Boron Concentration (ppm)	Calculated Boron Reactivity Change (mk)	LZC Fill (%)	Calculated LZC Reactivity Change (mk)	Simulation Error (%)
9.850	0.00	88.4	0.000	-
9.906	0.39	74.3	0.42	+7.7
9.963	0.79	65.0	0.73	-7.6
10.019	1.18	56.1	1.05	-12.4
10.076	1.58	45.0	1.51	-4.4
10.132	1.97	38.5	1.82	-7.6
10.189	2.36	29.3	2.28	-4.2
10.245	2.75	21.0	2.73	-0.7
Average				-4.2 ± 5.9 %

Table 2
LZC Reactivity Worth with Adjusters Out-of-Core

Boron Concentration (ppm)	Calculated Boron Reactivity Change (mk)	LZC Fill (%)	Calculated LZC Reactivity Change (mk)	Simulation Error (%)
11.261	0.00	91.4	0.00	-
11.317	0.39	79.4	0.36	-7.7
11.374	0.79	68.4	0.71	-10.1
11.430	1.18	56.8	1.12	-5.1
11.487	1.57	48.4	1.44	-8.3
11.543	1.97	38.5	1.85	-6.1
11.599	2.36	30.0	2.24	-5.1
11.656	2.75	19.9	2.71	-1.5
Average				-6.3 ± 2.6 %

Table 3
Detector-Response Calculation Accuracy

Detector Type	Signal Domain	Time Domain (ms)
SDS1	-1.5 % ± 3.0 %	-8.5 ± 18.8
SDS2	-3.0 % ± 4.9 %	-22.7 ± 25.5
RRS	-3.9 % ± 2.7 %	33.3 ± 30.0
IC Linear	4.1 % ± 2.9 %	27.6 ± 18.9
IC Log	0.001decade ± 3.0 decade	--
IC Log-Rate	-0.01 %/s ± 1.6 %/s	--

Figure 1
Comparison Between Measured and Calculated Log-Rate Signal: Ion Chamber H

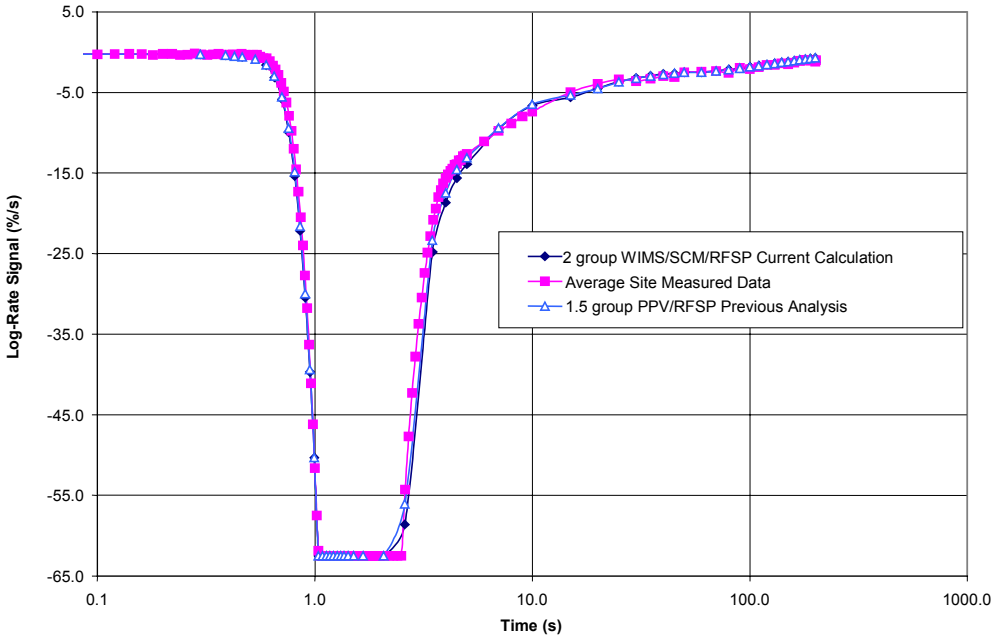

Figure 2
Comparison Between Measured and RFSP-Calculated Signal: SDS2 Detector 5J