Validation of Micro-Depletion Method for CANDU® Reactors for the Core-Tracking Simulations

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Abstract

The WIMS-AECL / DRAGON / RFSP reactor physics code set was used to simulate a core tracking scenario, which constitutes more than 400 full-power days. 102 vanadium detectors were used to record the local fluxes. These cases were run by using the micro-depletion method, embedded in the RFSP code. The calculated diffusion flux at the locations of the vanadium detectors were compared with the site measurement data. The average difference between the calculated flux and the measurement was about 2 %.

1. Introduction

This paper discusses the use of vanadium detector measurement data from the Point Lepreau Generating Station (PLGS) for the period from 1999 Jan 31 to 2000 Aug 19, to compare with a core tracking simulation using the micro-depletion method embedded in RFSP [1]. These vanadium detectors were used to provide a flux-mapping system to synthesize the 3-dimensional flux distribution in the reactor from in-core detector readings (see Figure 1). All the core parameters, including moderator temperature, moderator purity, moderator poison, coolant purity, the power level, and the liquid zone levels (see Figure 2) were recorded. In addition to the validation activity performed in the paper, it should be pointed out that a complete validation would include more measured data, such as discharged fuel burnup and isotopic composition.

2. Micro-depletion Method

In CANDU, the lattice cross sections are calculated with WIMS-AECL [2], and the full core neutron flux and power distributions are calculated with RFSP [1]. Ideally, the lattice code would be coupled directly with the full-core code, and the lattice calculations are performed for each bundle at each time step, so as to treat local parameters and the history of each bundle individually. With cluster capabilities coupled with Message Passing Interface (MPI), it would become practical to run RFSP full-core calculations coupled with WIMS-AECL.

The micro-depletion (MD) method is an improved method compared with the traditional WIMS-AECL grid-based method (also called macro-depletion method in the PWR industry) because it allows the depletion equation to be solved for each fuel region taking into account local conditions. The MD method is a proven technology and is being used widely in the PWR industry. Westinghouse's ANC [3], Studsvik Scandpower's SIMULATE-4, and North Carolina State University (NCSU)'s NESTLE [4] have used the MD method commercially and academically.

The MD method, developed and implemented in RFSP several years ago [5][6], has been verified for faithfulness compared with WIMS-AECL for the Advanced CANDU Reactor (ACR) fuel, 37-

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element natural-uranium (NU) fuel, and Bruce Low Void Reactivity Fuel (LVRF) fuel. The MD method has been validated against the power derating scenario [7].

In the MD method, the two-energy-group lattice-cell macroscopic cross sections are calculated using the following formula (note that the energy-group label is omitted for simplicity):

$$\Sigma_x(\vec{L}) = \Sigma_x(\vec{L}^0) + \sum_l \Delta\Sigma_x^l + \sum_i \left(\sum_m \left(N_m^i - N_{0,m}^i \right) \sigma_{x,m}^i \right) \quad (1)$$

where

x indicates the reaction type,

l is the label for a state parameter (which may be perturbed),

$\Sigma_x(\vec{L})$ is the actual lattice-cell cross section of type x at local condition $\vec{L}$,

$\Sigma_x(\vec{L}^0)$ is the "reference" lattice-cell cross section of type x at the reference condition $\vec{L}^0$,

$\Delta\Sigma_x^l$ is the correction to the "reference" lattice-cell cross section when a single state parameter l is perturbed,

N_m^i is the actual number density of nuclide m averaged over fuel region i,

$N_{0,m}^i$ is the "reference" number density of nuclide m averaged over fuel region i, and

$\sigma_{x,m}^i$ is the microscopic cross section of nuclide m averaged over fuel region i for reaction type x.

If $N_m^i(t)$ denotes the time-dependent number density of nuclide *m* averaged over fuel region *i*, then its rate of change can be generally expressed by the following burnup equation (again, the energy-group label is omitted for simplicity):

$$\frac{\partial N_m^i(t)}{\partial t} = -\left(\Phi_f^i \sum_x \sigma_{x,m}^i(\omega, T_f) + \lambda_m \right) N_m^i(t) + \Phi_f^i \sum_n \left(\sum_x \left(\beta_{n\to m}^x \, \sigma_{x,n}^i(\omega, T_f) \right) N_n^i(t) \right) + \sum_n \left(\beta_{n\to m}^{\text{decay}} \, \lambda_n \, N_n^i(t) \right) \quad (2)$$

where

Φ_f^i is the neutron flux within fuel region i,

ω is the (thermal-neutron) fuel irradiation,

T_f is the fuel temperature,

x is the set of reaction types: capture, fission, and (n, 2n),

$\sigma_{x,m}^{i}\left(\omega, T_f\right)$ is the microscopic cross section of nuclide m for reaction type x, averaged over fuel region i,

$\beta_{n\to m}^{\text{decay}}$ is the yield fraction for production of nuclide m from decay of nuclide n.

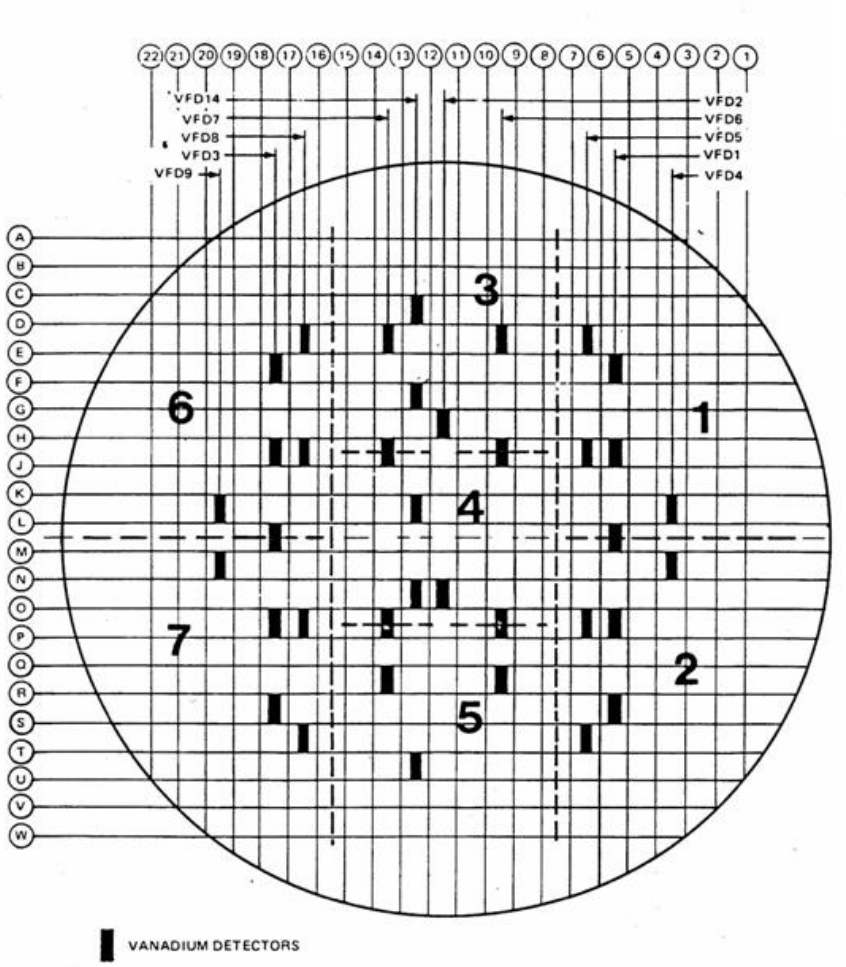

Figure 1 Vanadium Detectors in the Reactor

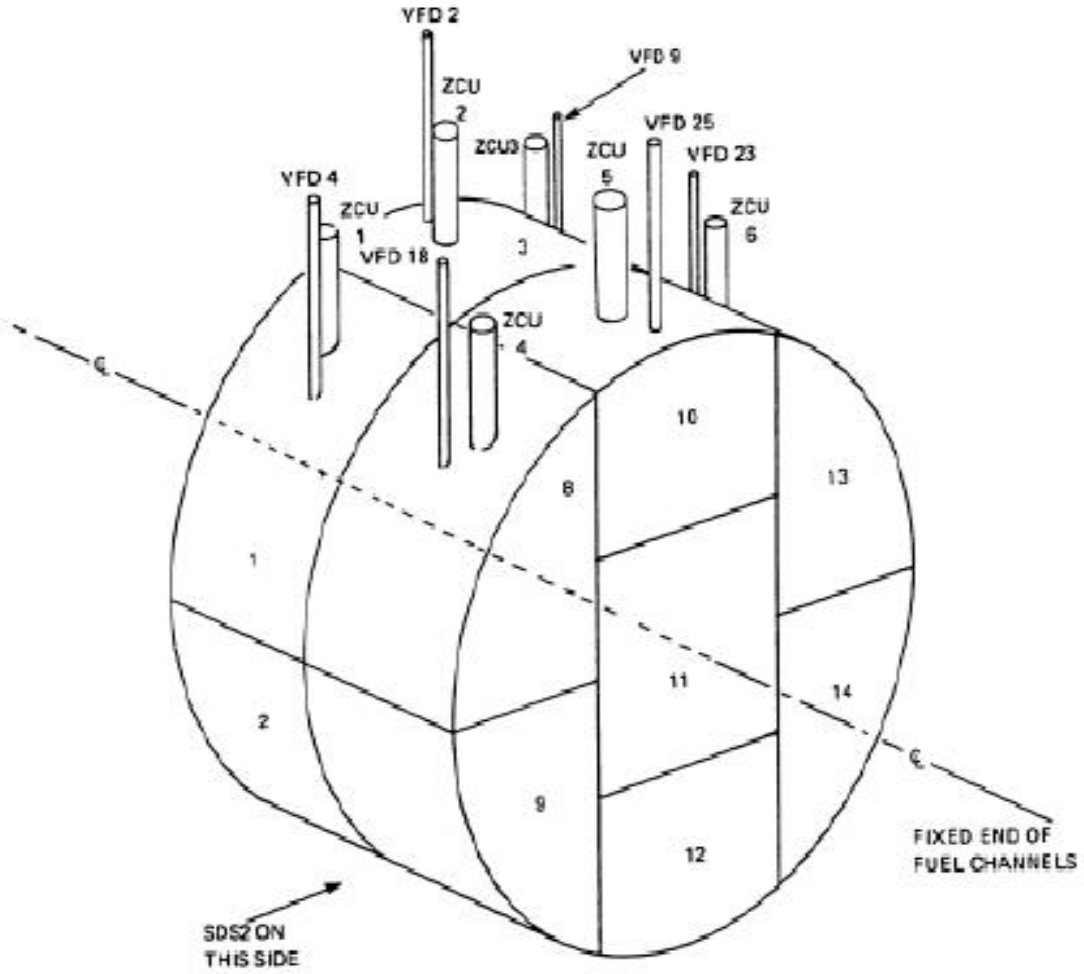

Figure 2 Liquid Zone-Controllers (labelled "ZCU")

3. Validation Model

The reactor physics code set WIMS-AECL / DRAGON [8] / RFSP was used to simulate the power rundown scenario. The lattice-cell cross sections are calculated with WIMS-AECL, and the three-dimensional core neutron-flux and power distributions are calculated with RFSP. The properties for reactivity devices, including the LZC and adjuster rods, are calculated with the 3 dimensional transport code DRAGON.

187 RFSP input files were generated to represent the core tracking period from 1999 January 31 to 2000 August 19. The 187 corresponding measured calibrated detector fluxes at vanadium detector sites

were provided by PLGS. All the measured data were calibrated by using the calibration factor updated to account for aging and shadowing for each detector. Figure 3 shows the power history for this period of time. There were periods of reduced reactor power: one happened between July 29 1999 to August 1 1999, the other one happened around March 11 2000.

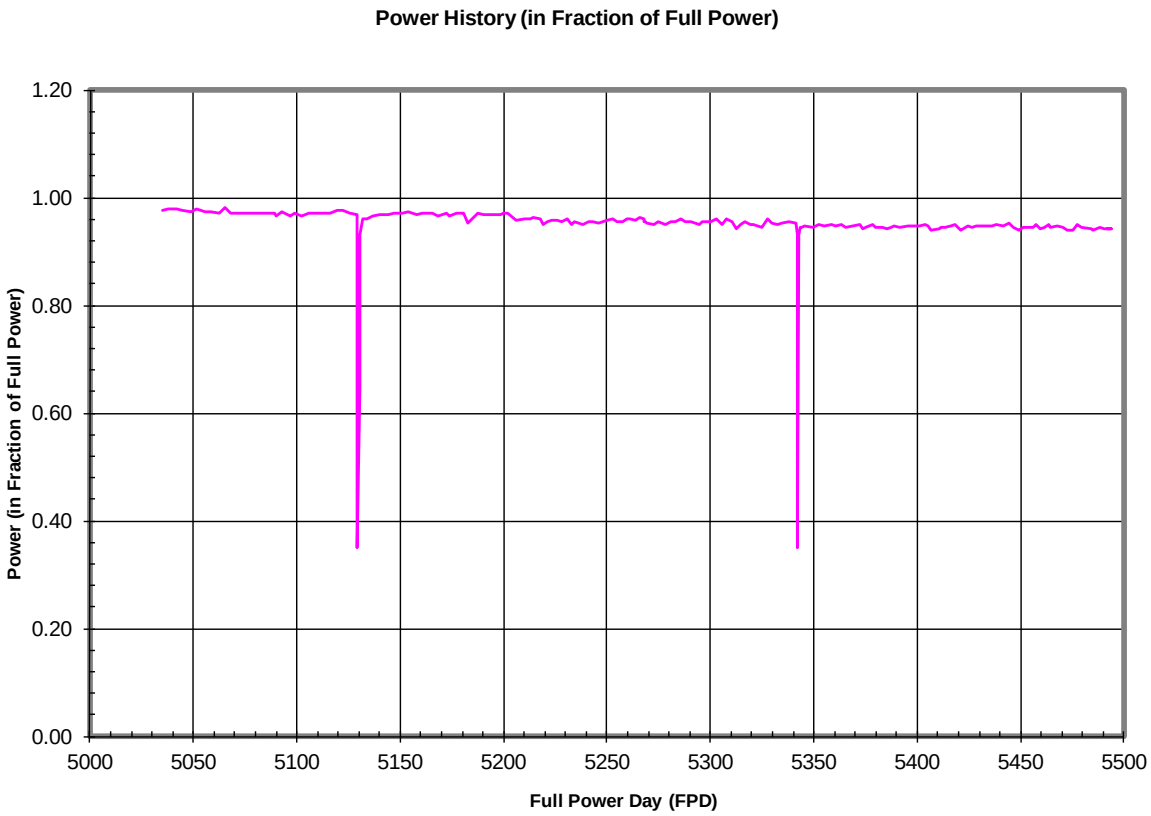

Figure 3 Power History in Fraction of Full Power for the Period Simulated

During this period of time, each ZCU reading was recorded. The ZCU reading was then specified in the core tracking input files at each burnup step. The average zone levels were calculated according to each ZCU reading, as seen in Figure 4.

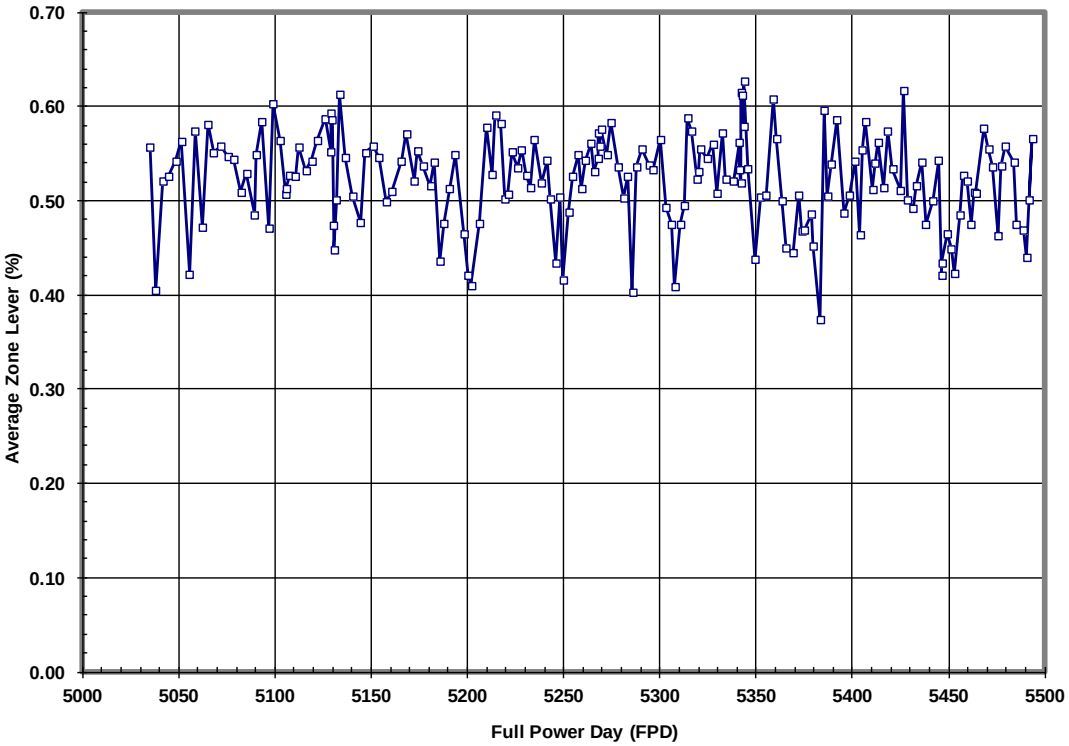

Figure 4 Calculated Average Zone Level (AZL) for the Period Simulated

3.1 Lattice Cross Sections

WIMS-AECL and WIMS Utilities [9] were used to calculate the lattice cross sections in the reactor core for four fuel types. All the fuel tables, including the uniform-parameter fuel/reflector tables and micro-depletion fuel/reflector tables, for these four fuel types were generated by using WIMS-AECL version 2.5d lattice code; while a Type-2 multicell table, which was designed to improve the calculation accuracy for the reactor periphery area was also used for comparison purpose [10]. The Type-2 multicell table was generated by using WIMS-AECL version 3.1, which has a multicell capability.

3.2 Device Incrementals

The adjusters had significantly aged by the time of the transient modelled, after 20 years irradiation. Also the incrementals for LZC has been improved as described in [11]. As such, the new sets of incrementals for adjuster rods and LZCs calculated by 3-dimensional transport code DRAGON, were used in the simulation.

3.3 RFSP Full Core Modelling

At CANDU sites, the main application of RFSP is in tracking the reactor's operating history. This is one of the major functions of the *SIMULATE module in RFSP, which calculates the 3-D flux and power distributions in the reactor at regular intervals. Steps 3 to 4 full-power days are typically taken between simulations. The code takes explicit account of all channel-refuellings. The irradiation distribution is kept up-to-date using the latest flux values. The flux distribution is calculated from diffusion theory, using the two-energy-group finite-difference method with a standard iterative solution scheme.

The burnup on this tape corresponds to the production run at 5032 EFPD (Effective Full-Power Day), on 1999 January 28 at 08:50. The reactor linear power (PLIN) in these production simulations varied from 95% to 98% FP during normal operation.

Figure 5 shows the mesh lines in light grey, the lattice line in dark black, and the notch lines as circles, in the RFSP model.

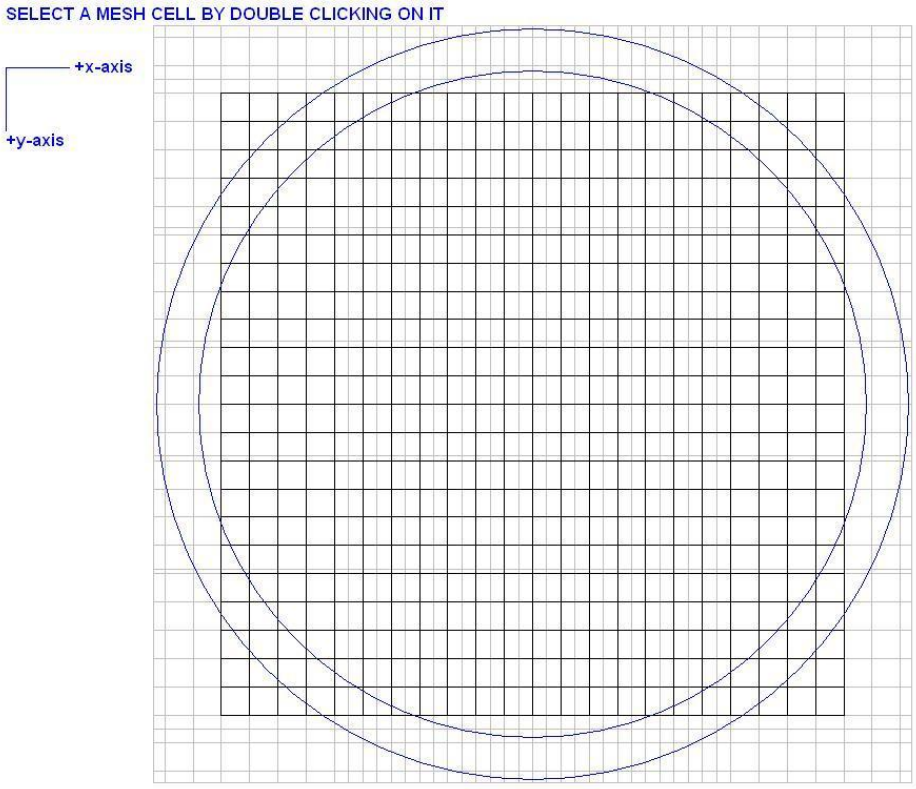

Figure 5 The mesh Lines in X-Y Plane of RFSP Model

Figure 6 shows the specific locations of the Vertical Flux Detector Assembly (VFD) unit in the X-Z plane of the RFSP model. A total of 102 vanadium detectors are distributed within these VFD units to measure the local flux.

There are twenty-two VFD units, each unit comprising a flux detector assembly, a guide tube installation, a thimble installation and the reactivity mechanism deck penetration and seal components, including the guide tube tensioning system.

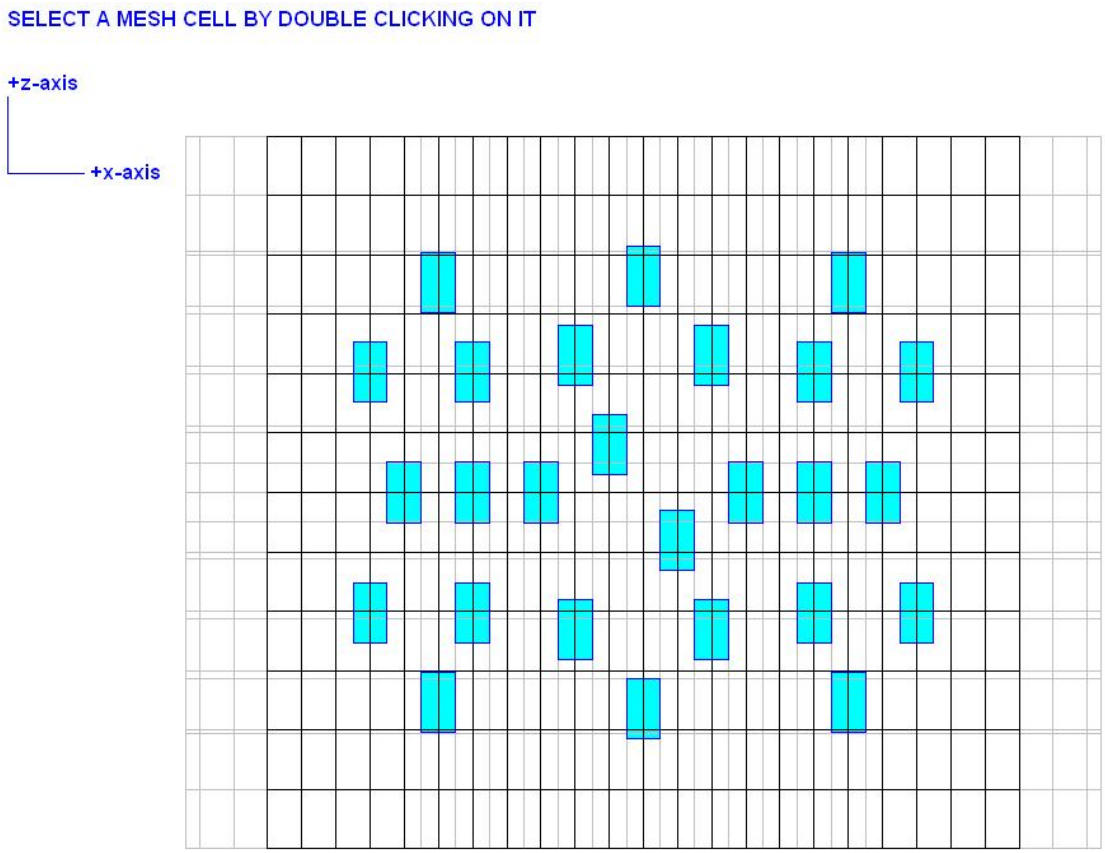

Figure 6 Specific Locations of VFD (Vertical Flux Detector) Unit in the X-Z Plane of RFSP Model

The *FLUXMAP module in RFSP was run to determine the measured calibrated detector fluxes (called Site fluxes) at the vanadium detector sites. The Site fluxes were then compared with the interpolated flux for vanadium detector position (i.e. the diffusion fluxes) by using the *COMPARE module of RFSP.

Two RFSP models were prepared for this core-tracking purpose:

1: Reference Model (with micro-depletion method)
2: Reference Model with 5×2 Type-2 Multicell Corrections

It should be noted that the only difference between these two models was the application of Type-2 multicell correction at the peripheral fuel region of the core in the second model. The same reactivity device incrementals were used for both cases.

4. Comparison of Results

The reactivity, maximum channel / bundle power, and the standard deviation for flux calculation compared with site measurement are all summarized and presented below.

4.1 Reactivity

For the reactivity, Figure 7 shows that about -1 to -1.5 mk sub-criticalities exist for the simulation by using these two models.

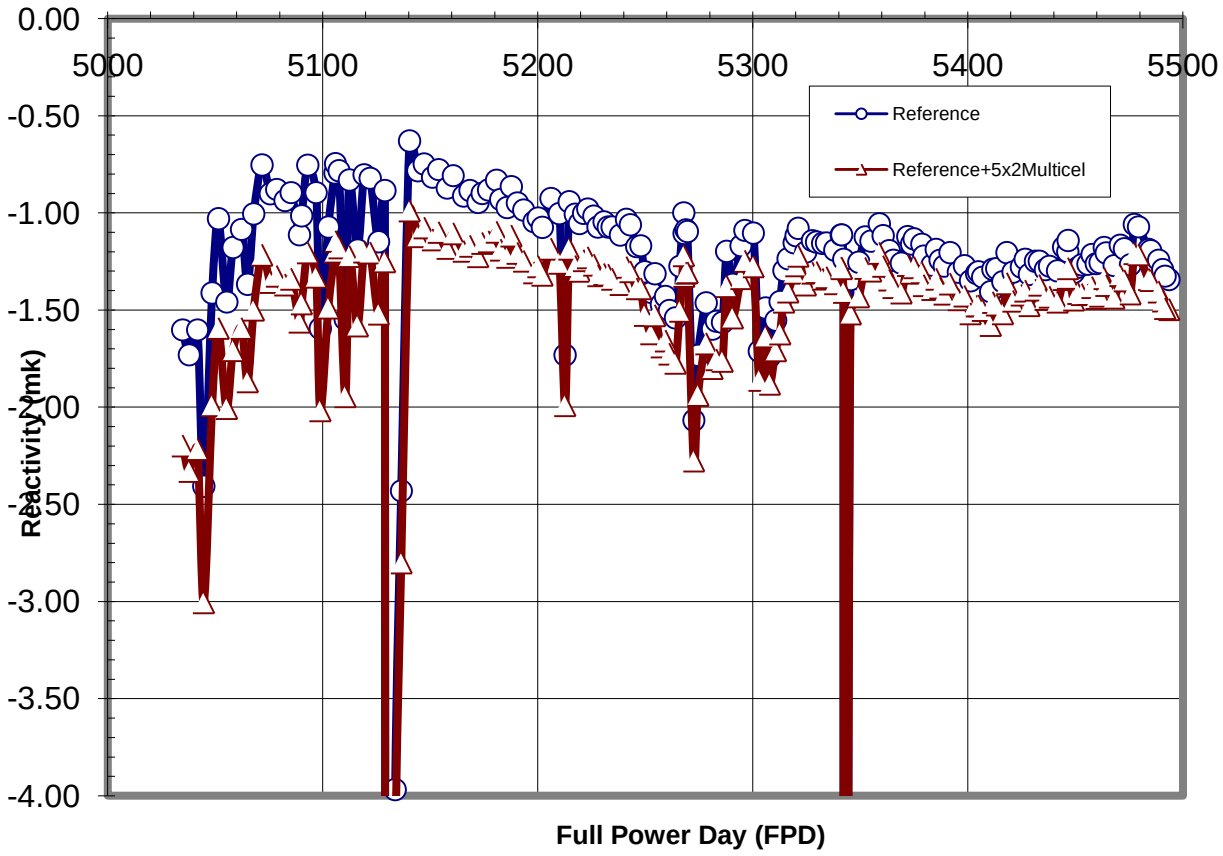

Figure 7 Calculated Reactivities

4.2 Maximum Channel Power

The maximum channel power predicted by multicell model gives slightly higher maximum channel power, with an average difference of 0.34 % and a maximum difference of 1.28 %.

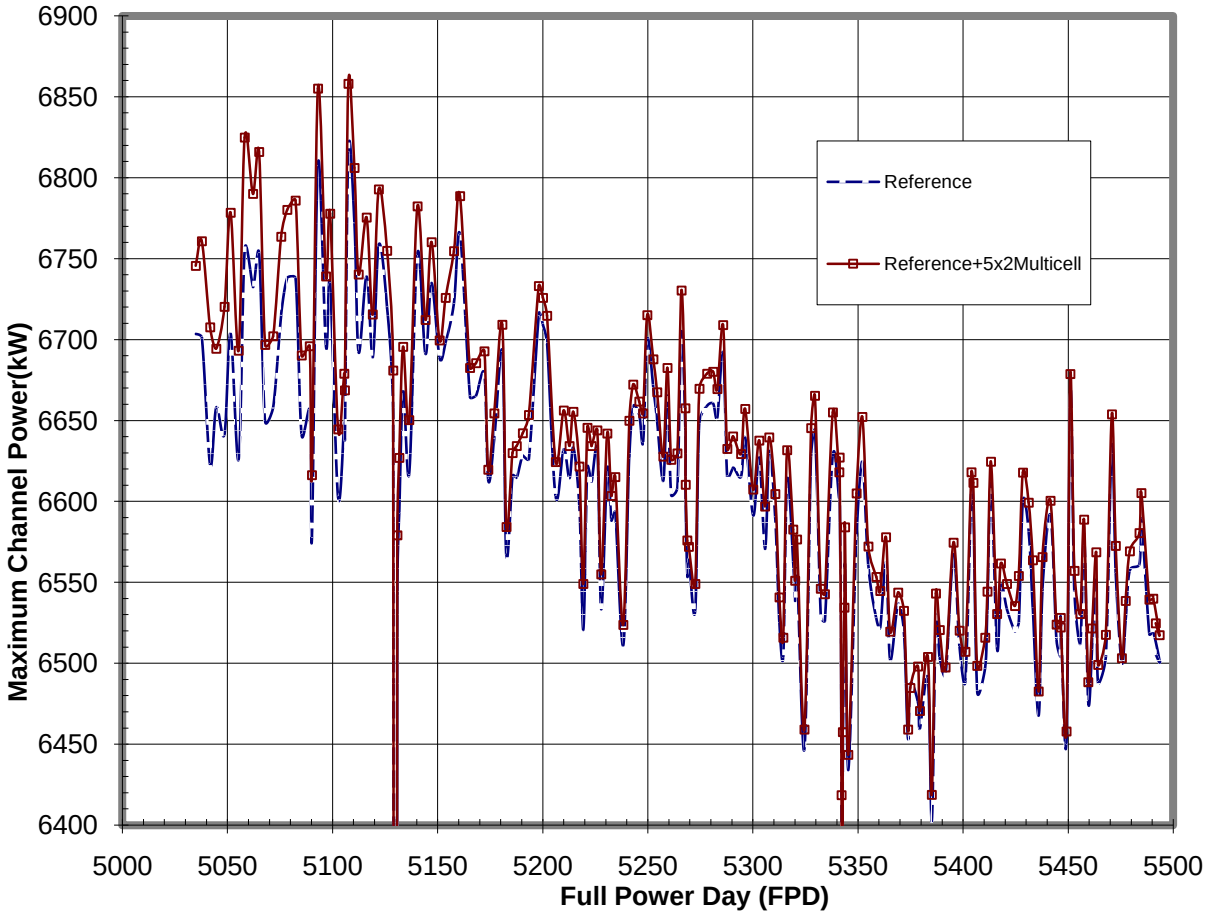

Figure 8 Maximum Channel Power

4.3 Maximum Bundle power

The maximum bundle power predicted by multicell model gives slightly higher maximum bundle power, with an average difference of 0.33 % and a maximum difference of 0.79 %.

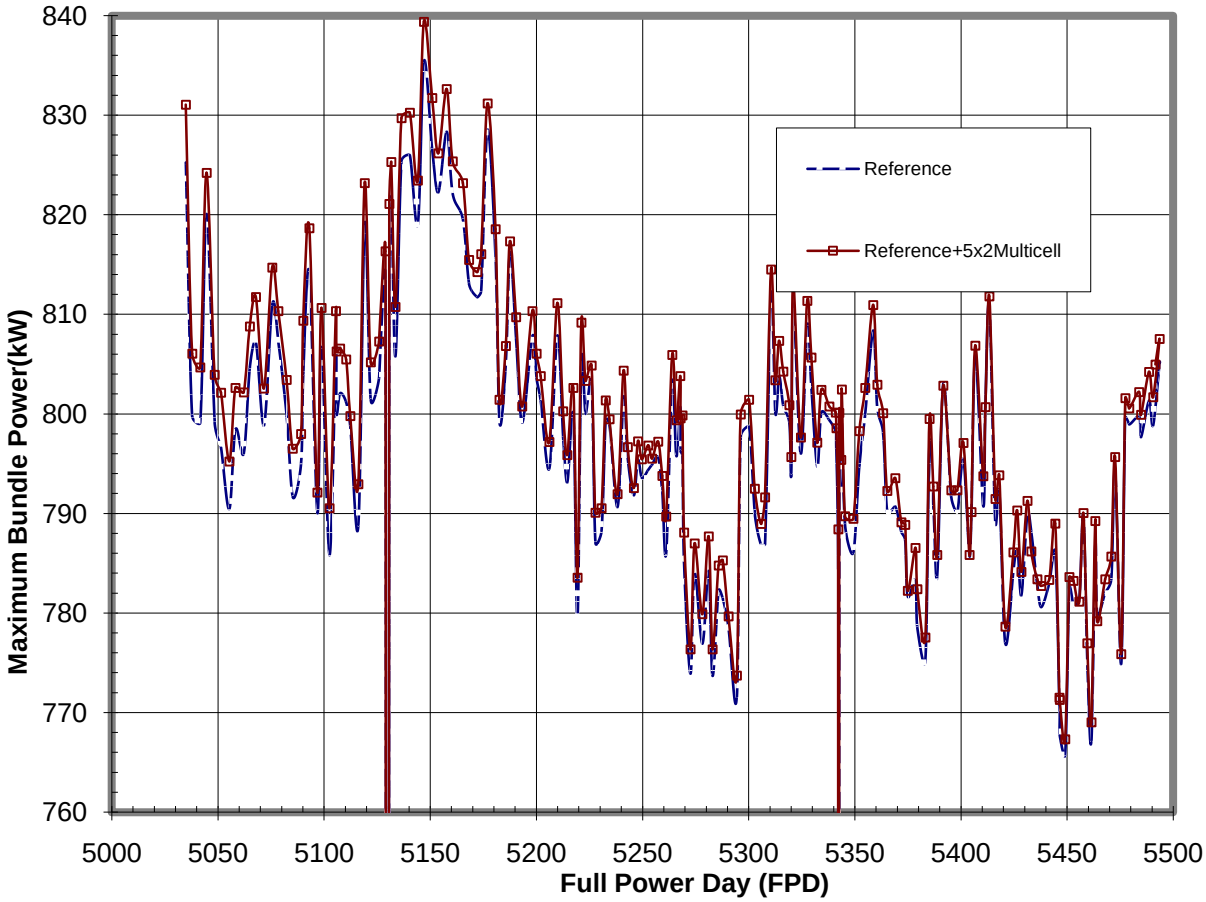

Figure 9 Maximum Bundle Power

4.4 Standard Deviation

Both the standard deviation and the mean percent difference between the calculation and the site measurement are important factors to evaluate the accuracy in flux calculation. Figure 10 shows the standard deviation and the mean percent difference for all the calculation cases. Summarized results comparison is shown in Table 1. The cases for the periods of reduced reactor power were excluded when calculating the averaged standard deviation, since the model used in calculation was suitable to simulate the fast and dramatic reactor power changes. Such scenario as power derating was dealt with in another published paper. [7]

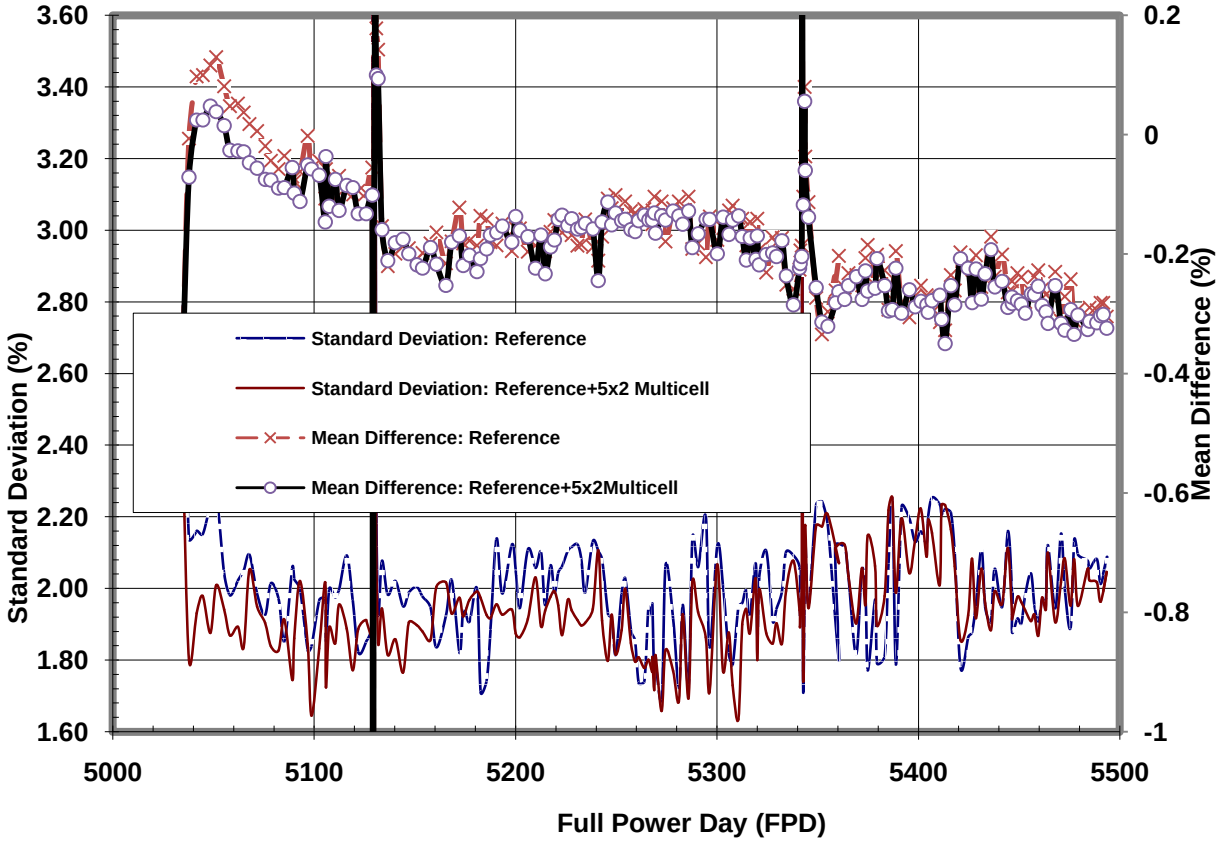

Figure 10 Standard Deviation of Vanadium Detectors (Simulation Vs Measurement)

Table 1 the Standard Deviation Comparison for Difference Models

	Reference	Reference + 5 x 2 Multicell
Maximum Value	2.8	2.4
Minimum Value	1.7	1.6
Average Value	2.0	1.9

The standard deviation given by multicell calculation is slightly better than the one given by the reference model.

Vanadium detector readings were rejected because of irrationality or large differences compared to simulation result. Number of detectors rejected varies a little for different simulation methods. Figure 11 shows the number of detectors rejected at each full power day for the reference simulation.

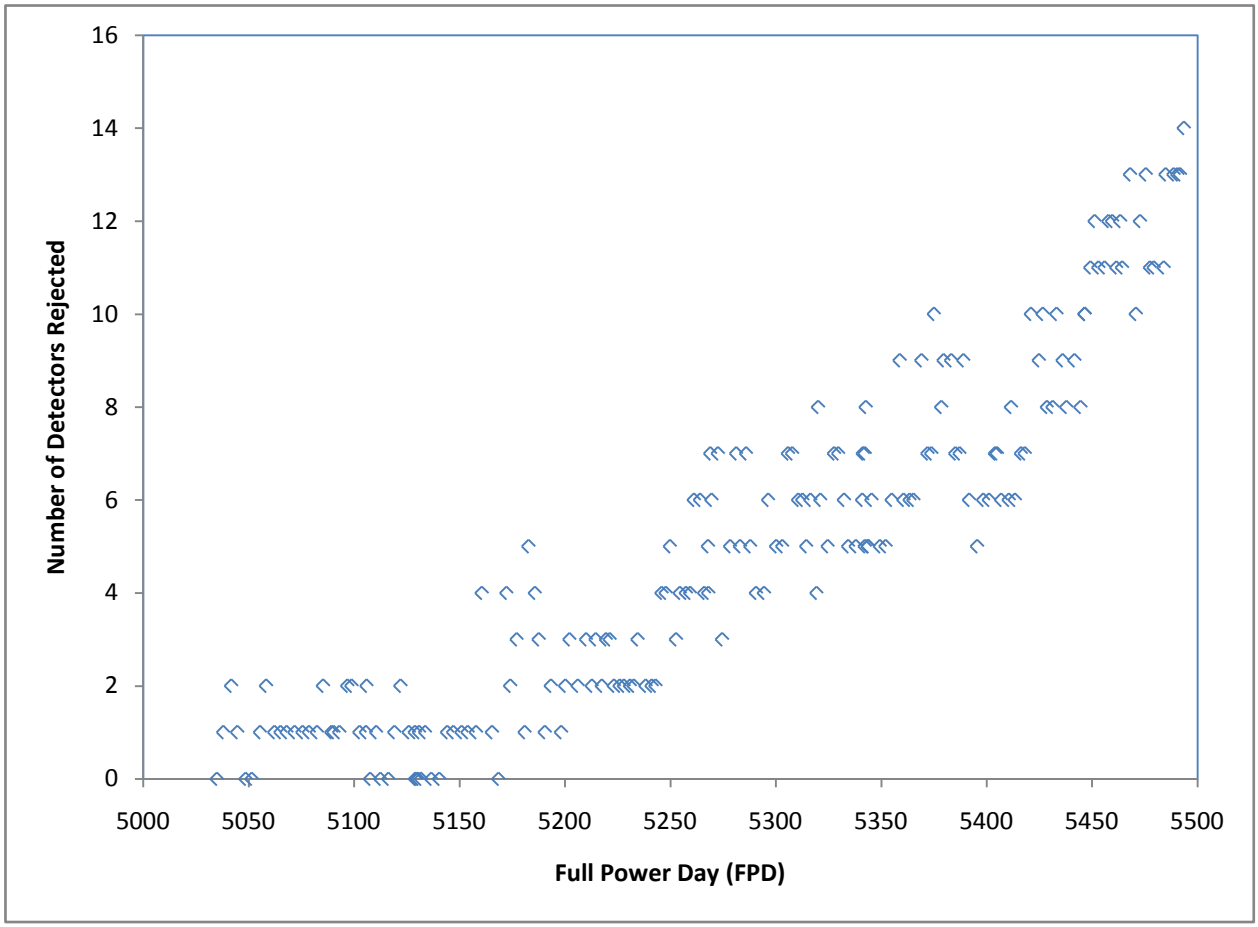

Figure 11 Number of Detectors Rejected for Reference Simulation Calculating Standard Deviation

4.5 Calculated Tilts vs. Measured Tilts

Figure 12 gives the tilts of the simulated fluxes and the site fluxes in the top-to-bottom, east-to-west, and north-to-south directions. The figure shows that the use of the multicell correction did not have much impact on the tilt-differences between the simulated fluxes and the site fluxes.

The agreement of averaged flux at vanadium detector location for the inner core, and the outer core are shown in Figure 13 and Figure 14 for both cases. Except the two periods of reduced reactor power, the differences are all within 1%.

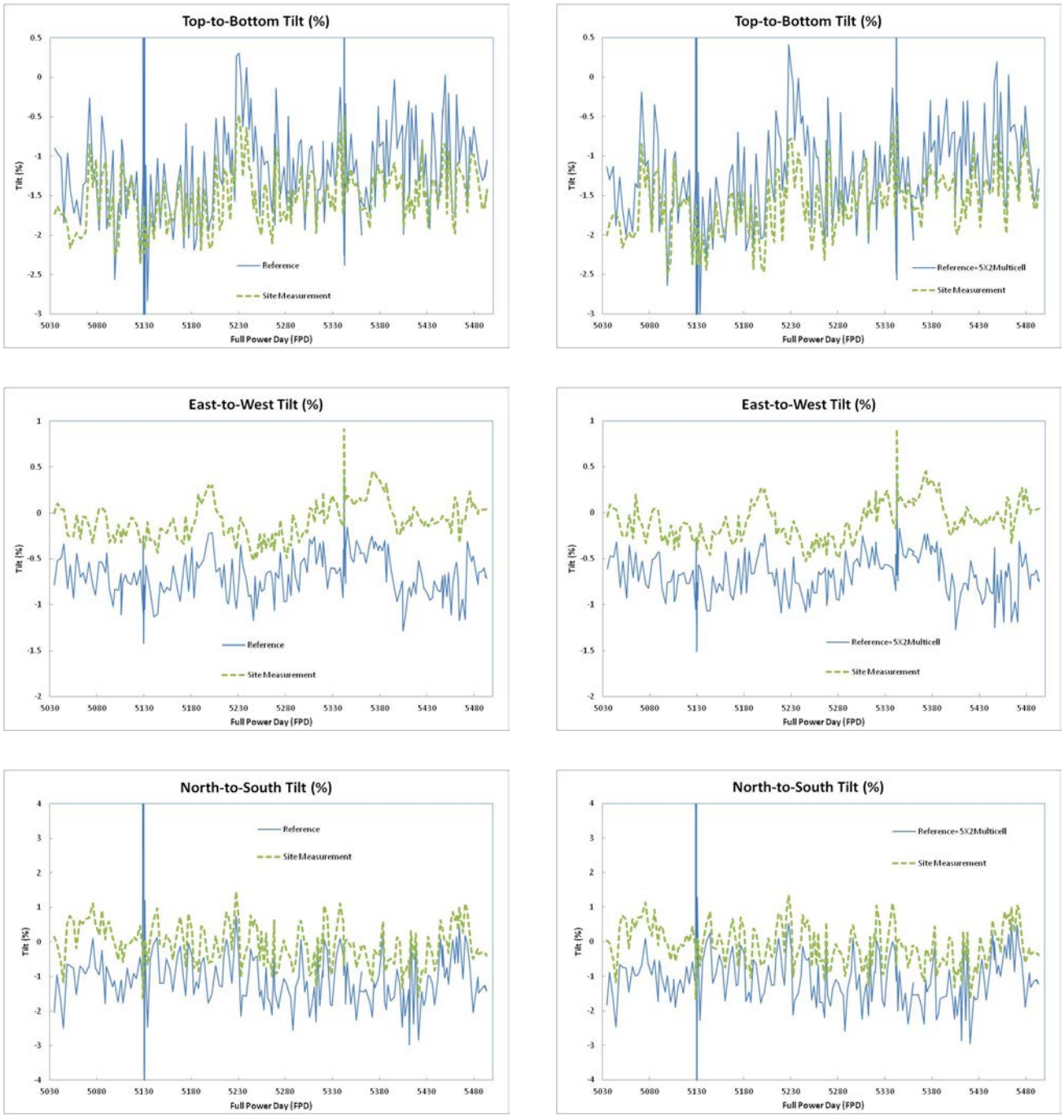

Figure 12 Calculated Tilts and Measured Tilts

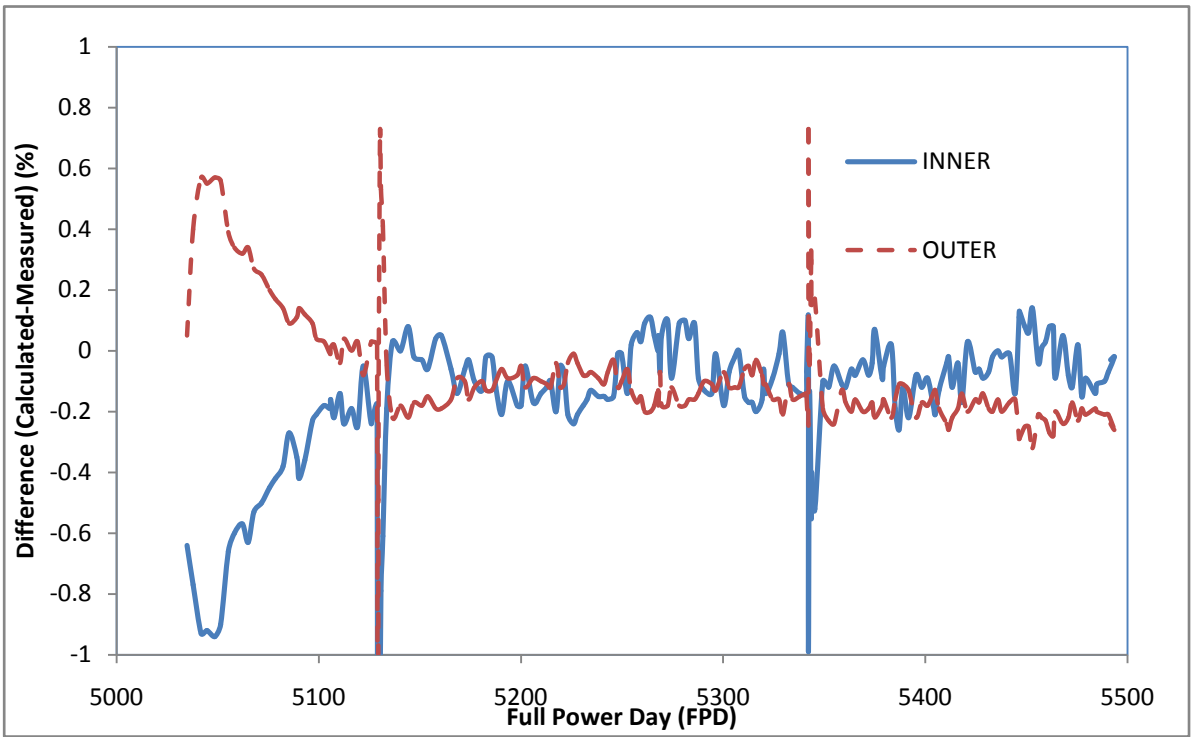

Figure 13 Difference of Averaged Flux at Vanadium Detector Location for Reference Case

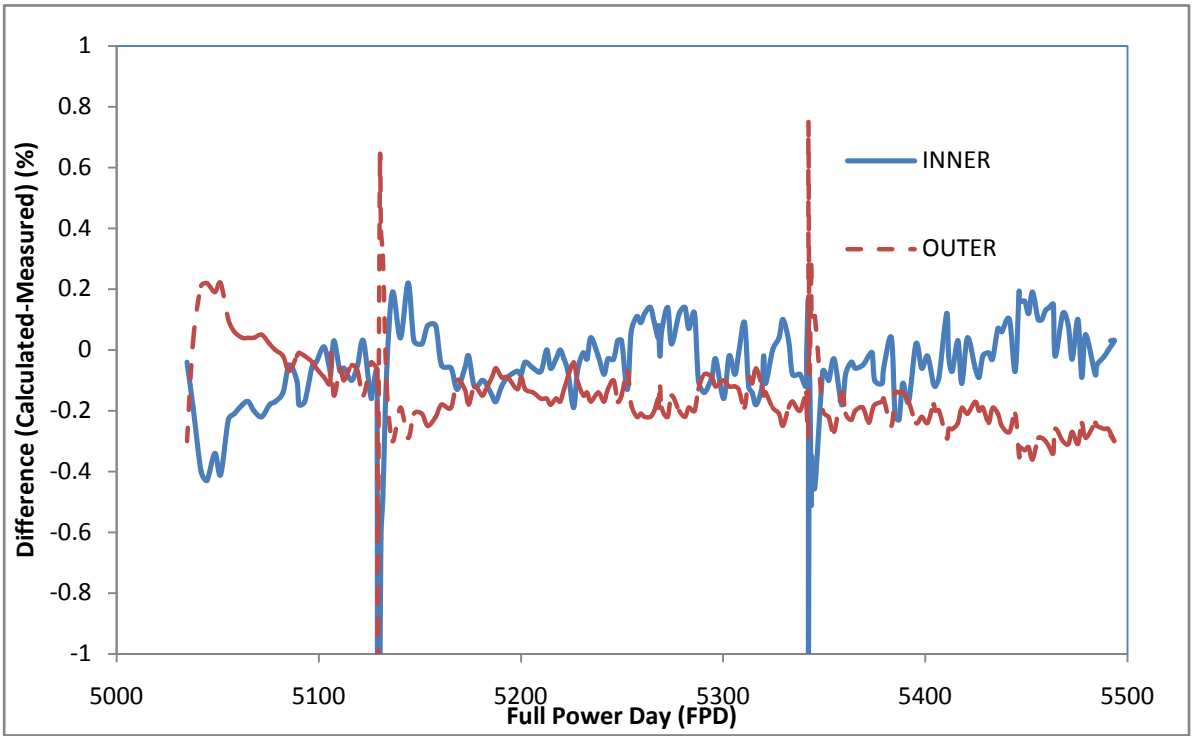

Figure 14 Difference of Averaged Flux at Vanadium Detector Location for Reference with Multicell Correction Case

5. Conclusion

For the core-tracking simulation, the averaged value for the standard deviations for the differences between the RFSP diffusion fluxes and the site fluxes were 2.0%, 1.9 % for the core modelling with reference model (with micro-depletion method), and the reference model with the multicell 5×2 Type-2 correction respectively. The good agreement between the calculated fluxes and site fluxes validates the micro-depletion method for core-tracking scenario.

6. References

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