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Consistent Neutron Kinetics Data Generation for Nodal Transient Calculations

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Abstract

Current three-dimensional transient codes for thermal reactors are mostly based on two-group diffusion-theory nodal models. In the two-group approach no explicit distinction is made between prompt fission neutrons and delayed neutrons. Consequently, effective delayed neutron fractions have traditionally been used in an attempt to compensate for this shortcoming. A fundamentally better approach would be to solve the nodal kinetics equations in a sufficient number of energy groups to explicitly capture neutron emission spectrum effects. However, this would require the availability of a multi-group nodal transient code as well as a lattice code to generate the appropriate multi-group nodal data for the simulator. One such simulator is the PARCS nodal transient code, which is widely used and recognized as representative of the current state-of-the-art. Unfortunately, a proper nodal data preparation path between PARCS and a lattice code is not available. Even though several industrial lattice codes could be considered as candidates, most of them are tailored to producing two-group nodal data and would require modifications to produce multi-group prompt and delayed neutron emission spectra.

In this paper, the particular modifications required to match the TransLAT lattice code and the PARCS nodal transient code for BWR transient applications are reported. Some modifications to PARCS were also required to make two-group and multi-group applications fully consistent. Numerical results are presented both to verify the proper functioning of these modifications and to illuminate the impact of various nodal kinetics data approximations in a selected transient calculation. In particular, the significance of blending rodded and unrodded kinetics data in partially rodded nodes is demonstrated. It is also confirmed that the use of delayed neutron importance factors in two-group calculations notably reduces the differences between two-group and multi-group kinetics calculations.

KEYWORDS: effective delayed neutron fraction, importance factor, delayed neutron spectrum, lattice physics, few-group nodal kinetics calculations

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1. Introduction

The growing interest in applying coupled three-dimensional neutronics/thermal-hydraulic transient calculations to improve the safety and economics characteristics of light water reactor (LWR) fuel and reactor system designs require consistent generation of nodal nuclear data (library of parameterized nodal equivalent parameters) for both steady-state and transient calculations. In addition to the standard two-group nodal cross-sections and flux discontinuity factors the two-dimensional lattice physics codes that are utilized to generate such libraries provide neutron kinetics and delayed neutron data for each relevant nodal material composition. This normally includes delayed neutron yield fractions (betas) and decay constants per delayed neutron precursor family, delayed neutron importance factor(s), and inverse neutron speeds.

An issue that has to be considered in two-group transient calculations is the fact that the difference between prompt and delayed neutron emission spectra cannot be directly accounted for. This effect, which can be clearly seen in a multi-group formulation, is important and as a consequence the two-group transient calculation results may be significantly affected. This situation can be improved for two-group calculations if corrected (effective) values of delayed neutron yield fractions are used. It is precisely for this reason that delayed neutron importance factors [1] for each material composition are produced (either directly or as part of a post-processing activity) by the lattice physics codes.

The use of the importance factors in two-group kinetics calculations represents at best only a crude approximation designed to compensate for the deficiencies of the linearly (flux) collapsed two-group time-dependent diffusion equations. A fundamentally better approach would be to solve the nodal kinetics equations in a sufficient number of energy groups to explicitly capture neutron emission spectrum effects. In order to evaluate the accuracy of the two-group approach (utilizing importance factors) and the potential improvement that might be gained by adopting a multi-group approach in three-dimensional (3-D) transient calculations, applicable codes and data preparation procedures must be available. In particular, appropriate kinetics data must be extracted from lattice code results and transmitted to a nodal simulator that has a multi-group capability. The nodal kinetics data must furthermore be employed (in the simulator) in a manner consistent with the state representation model for the data.

It has been found that many of the existing industrial lattice codes are customized to generating two-group nodal parameters and that delayed neutron emission spectra are not produced at all, with the consequence that it becomes difficult to perform multi-group nodal static and kinetics calculations with consistent nuclear data. The problem is solved by modifying such lattice codes to produce the required multi-group prompt and delayed neutron emission spectra. Likewise, if a specific state representation for the kinetics data must be employed, for example such as that proposed in [1] for the delayed neutron importance factors, then the relevant nodal simulator code must be appropriately modified.

In this paper, we describe the modifications that were implemented in the TransLAT two-dimensional lattice code [2] and the PARCS three-dimensional nodal kinetics code [3] in order to establish the necessary computational platform for an investigation of the two-group versus multi-group kinetics modeling issue. TransLAT was chosen because it includes all the standard features of an industrial lattice code and in addition offers an adjoint spectrum

solution method for computing delayed neutron importance factors. Furthermore, the TransLAT cross-section and kinetics data are based on the latest releases of the ENDF/B-VI data files. PARCS was selected because it is one of the few available nodal transient codes with a multi-group capability. Both codes are representative of the current state-of-the-art.

Numerical results are presented both to verify the proper functioning of the code modifications and to illuminate the impact of various nodal kinetics data approximations in a simplified reactivity insertion type transient.

2. Description of the Actual Work

2.1 Description of TransLAT Code Changes

Although the TransLAT code produces most of the nodal kinetics parameters (e.g. inverse velocities, effective delayed neutron fractions, delayed neutron precursor decay constants) it does not produce prompt and delayed neutron emission spectra as part of its output data. This presents an obstacle to using the code as a nodal data generator for multi-group nodal transient codes. For this reason, TransLAT was modified to edit node-average delayed and prompt neutron emission spectra:

$$\chi_G^{D,l,i} = \sum_{g \in G} \chi_g^{D,l,i} \tag{1}$$

$$\chi_G^{D,l} = \frac{\sum_i \chi_G^{D,l,i} \beta_l^i \sum_j S^{i,j}}{\sum_i \beta_l^i \sum_j S^{i,j}} \tag{2}$$

$$\chi_G^{D} = \frac{\sum_l \chi_G^{D,l} \beta_l^i}{\sum_l \beta_l} \tag{3}$$

$$\chi_G^{P,i} = \sum_{g \in G} \chi_g^{P,i} \tag{4}$$

$$\chi_G^{P} = \frac{\sum_i \chi_G^{P,i} (1 - \sum_l \beta_l^i) \sum_j S^{i,j}}{\sum_i (1 - \sum_l \beta_l^i) \sum_j S^{i,j}} \tag{5}$$

In the above, i is an actinide index, g is a micro-group index, G is a macro-group index, j is a region index, D indicates "delayed", P indicates "prompt", l is a delayed neutron precursor family index, β is a delayed neutron fraction and

$$\beta_l = \frac{\sum_i \beta_l^i \sum_j S^{i,j}}{\sum_i \sum_j S^{i,j}} \quad \text{and} \quad S^{i,j} = V_j \sum_g \nu \Sigma_{fg}^{i,j} \Phi_g^j \tag{6}$$

TransLAT has also been modified to compute node-average delayed neutron precursor family dependent importance factors directly (instead of leaving this to an external operation) from the ratio

$$I_l = \frac{\beta_l^{eff}}{\beta_l} = \frac{\sum_g \chi_g^{D,l} \Phi_g^*}{\sum_g \chi_g^{total} \Phi_g^*} \quad \textbf{(7)}$$

where β_l^{eff} is the effective delayed neutron fraction for family l defined according to the standard definition using the adjoint flux spectrum (see Ref. [4]). The total importance factor is also computed by TransLAT:

$$I = \frac{\sum_l I_l \beta_l}{\sum_l \beta_l} \quad \textbf{(8)}$$

At present only the total importance factor I is passed on to PARCS for utilization in two-group kinetics calculations, the reason being that it was found that the delayed neutron precursor dependency is very weak (<0.1%) and that $I_l \approx I$. In addition to the delayed neutron fractions β_l, the macroscopic prompt and delayed fission spectra (χ_G^P and χ_G^D) and the total importance factor, fission-rate-weighted inverse decay constants for each precursor family and flux-weighted inverse neutron speeds for each macro energy group are calculated by TransLAT and passed on to PARCS.

2.2 Description of PARCS Code Changes

In the standard input cross-section model of PARCS the nodal parameters, including the kinetics parameters, are allowed to be dependent on exposure and a number of instantaneous state parameters. In reference [1] it was proposed to treat the delayed neutron importance factor as a separate nodal parameter because its state dependence could be simplified considerably relative to that needed for a direct state-dependent tabulation of effective delayed neutron fractions. Since the delayed neutron importance factor has not thus far been treated as a nodal parameter by PARCS, it had to be modified to do so. The code was therefore modified to construct the total importance factor per node (or per sub-node in partially rodded nodes) according to the state dependence proposed in [1]:

$$I \approx A_0 + A_1 \left(\frac{\nu \Sigma_{f1} \Phi_1}{\Sigma_{r1} \Phi_1} \right) + A_2 \left(\frac{L_1}{\Sigma_{r1} \Phi_1} \right) \quad \textbf{(9)}$$

The fuel type dependent (and control rod dependent) constants A_0, A_1 and A_2 are provided in the nodal cross section library while the node-wise (not per subnode) fast reaction and leakage rates are calculated by PARCS at the initial steady state as well as at each time step during a transient simulation. PARCS thus also required modifications to read (from the nodal cross section library) the mentioned coefficients and to compute the noted nodal reactions rates (using node-average cross sections and the nodal solution fluxes). Furthermore, the code was changed such that the importance factors are used in two-group transient calculations to modify the "physical" delayed neutron fractions, the latter being extracted in the usual way (per node or per subnode) from the nodal cross section library at relevant local conditions.

It was found that a control rod cusping model (an axial homogenization correction for partially rodded nodes) was implemented only in the two-group analytic nodal model and not in the multi-group nodal expansion method (NEM). Since (two-group) numerical analyses

had shown the cusping correction to be important, it was decided to implement it also in the NEM procedure and to apply it to all nodal cross-section (XS) data, including radial side discontinuity factors (DFs) and inverse neutron speeds. The cusping correction takes the form of a flux-volume weighting correction to the purely volume weighted rodded and unrodded data in a partially rodded node:

$$\Delta\Sigma_g^{node} = \frac{\Sigma_g^{unrod} h^{unrod}}{h^{node}} \left[\frac{\Phi_g^{unrod}}{\Phi_g^{node}} - 1 \right] + \frac{\Sigma_g^{rod} h^{rod}}{h^{node}} \left[\frac{\Phi_g^{rod}}{\Phi_g^{node}} - 1 \right] \quad \textbf{(10)}$$

Here h^{node} is the height of the node while h^{unrod} represents the height of the unrodded part (unrodded subnode) of a partially rodded node. Φ_g^{node} is the node average flux and Φ_g^{unnode} is the average "heterogeneous" neutron flux in the unrodded part as computed for a simple one-dimensional three-node problem with the partially rodded node (with two sub-nodes) as the central node. Likewise for the rodded subnode data. When the heterogeneous flux is assumed to be flat, the cusping correction vanishes. Due to the flux depression in the partially rodded region the cusping correction reduces the control rod worth relative to that obtained without the cusping correction. Note that delayed neutron fractions, precursor decay constants and importance factors are simply volume weighted (i.e. no cusping correction).

It should be noted that in addition to the cusping correction to the node-average data, axial side discontinuity factors are normally also computed as part of the axial homogenization procedure. However, such a calculation has as yet not been implemented for the NEM case (it already exists for the two-group analytic nodal model).

During the course of these code modifications it was discovered that PARCS did not cater for side-dependent discontinuity factors in the event that the so-called "standard" cross-section model is used. It handled side-dependent discontinuity only if the specialized cross-section models for certain pre-defined transient benchmark problems were run. Since the standard cross-section model is the one that is to be used in real-world cases, this deficiency was mended by appropriate code changes. This capability is essential for BWR applications.

3. Results

In this section results illustrating the impact of the modifications implemented in TransLAT and PARCS are presented. The NEM solver of PARCS was exclusively used both in the two-group and the six-group calculations reported here for a simple BWR mini-core kinetics benchmark. This mini-core consists of a 6x6 matrix of identical fresh BWR assemblies with a fuel active height of 365.75 cm (see Figure 1). Reflective radial boundary conditions and vacuum axial boundary conditions are applied. For the transient simulation the initial state (steady-state) was chosen at typical hot-zero-power conditions with the central control rod bank fully inserted and the peripheral eight control rod banks fully withdrawn. No thermal-hydraulic feedback is modeled in this problem and a reasonable power transient evolution is attained simply by control rod movement. The transient is initiated by withdrawing the central control rod (# 2 in Figure 1a) at a speed of 3.81 cm/s and after 12 s into the transient the peripheral rods (#1 in Figure 1a) are inserted at a speed of 7.62 cm/s. The simplicity thus achieved facilitates a clear and meaningful comparative analysis of 6-group vs. 2-group kinetics calculations.

Figure 1a Mini core cross-sectional view and control rod grouping

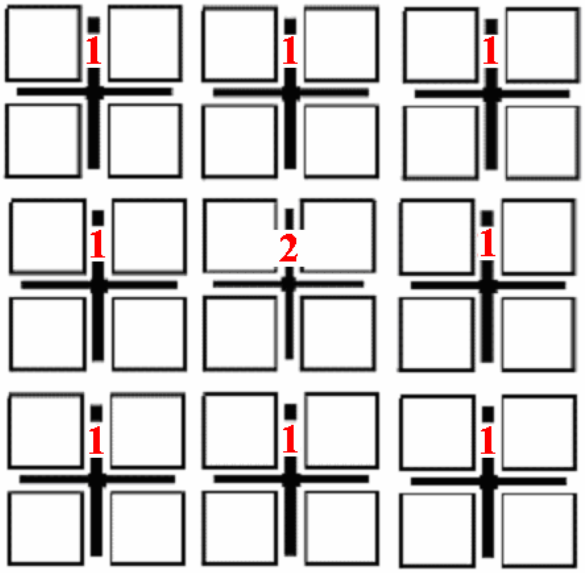
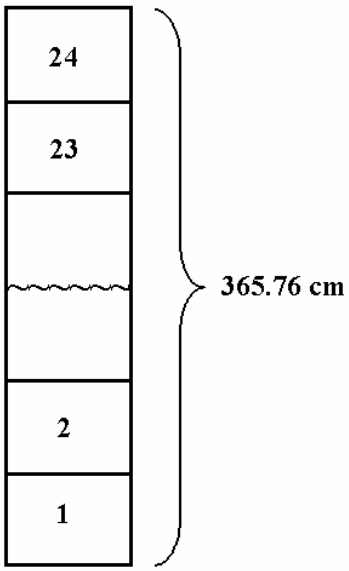


Figure 1b Axial representation of a single assembly



As a demonstration of the features added to TransLAT the (total) delayed and prompt fission spectra produced as output by the code are tabulated in Table 1 for a selected 6-group structure. The data tabulated here are for the unrodded state of the BWR assembly type used in the benchmark problem above. The difference in the neutron emission spectra of prompt and delayed neutrons is clearly seen.

Table 1: 6-group fission spectra for an unrodded BWR assembly

GRP #	Prompt Spectrum	Total Delayed Spectrum	Energy Range (eV)
1	4.747E-01	0.000E+00	1.000E+07 – 1.738E+06
2	4.384E-01	4.658E-01	1.738E+06 – 3.877E+05
3	8.620E-02	5.321E-01	3.877E+05 – 1.503E+04
4	7.188E-04	2.090E-03	1.503E+04 – 3.928E+00
5	4.541E-11	1.108E-10	3.928E+00 – 6.250E-01
6	0.000E+00	0.000E+00	6.250E-01 - 0.000E+00

As a test of the functionality of the TransLAT and PARCS combination, several steady-state calculations of the mini-core problem were performed and are reported in Table 2. In those cases where the original axial boundary conditions were changed from vacuum to reflective and all rods are either inserted or withdrawn a direct comparison between TransLAT and PARCS results is possible (because a unit assembly configuration is attained). The fact that PARCS yields identical results to TransLAT for these cases irrespective of the number of energy groups used in PARCS simply confirms that nodal data preparation by TransLAT and utilization of such data by PARCS is consistent and proper.

When the vacuum axial boundary conditions are re-instated and all rods are again either inserted or withdrawn, a single assembly case with uniform axial leakage is modeled. In this case the 2-group and 6-group PARCS results agree very well, which might be taken as a

further indication that nodal data are properly utilized.

The case where the central rod is inserted illustrates the impact of radial discontinuity factors and here it is seen (last two columns in Table 2) that they do affect the results significantly (about 160 pcm in k_{eff}). The purpose of this comparison is not to demonstrate the utility of discontinuity factors, but simply to confirm that discontinuity factors were implemented as intended (see the discussion at the end of the Section 2.2).

Table 2: Mini-core comparison of TransLAT and PARCS

Boundary Conditions	**Code**	**Groups**	**Rodded** All Rods In k_{eff}	**Un-rodded** All Rods Out k_{eff}	**Central Rod In** with DFs k_{eff}	**Central Rod In** w/o DFs k_{eff}
Reflective everywhere	TransLAT	97	0.827458	1.038651		
	PARCS	2	0.827458	1.038652		
	PARCS	6	0.827458	1.038652		
Reflective radially and vacuum axially	PARCS	2	0.824436	1.034801	1.019213	1.017646
	PARCS	6	0.824477	1.034833	1.018916	1.017326
	PARCS	Δk_{eff} (6 - 2 group)	4 pcm	3 pcm	-30 pcm	-32 pcm
	PARCS	$\max_n(\lvert\Delta P_n\rvert)$ (6 - 2 group)			1.0%	1.1%

To test the implementation (in the NEM solver of PARCS) of the axial homogenization method for partially rodded nodes, the BWR mini-core transient calculation was performed for each of the cases listed in Table 3. The 2-group calculations were performed with three choices for the delayed neutron fractions: "beta physical" implies that the fission-rate weighted betas (for each precursor group) produced by the lattice code and tabulated as a function of state parameters were used, "beta effective" implies that the adjoint spectrum weighted betas produced by the lattice code were used, and "importance factor" implies that the delayed neutron importance factor as reconstructed according to Eq. 9 was multiplied into the "physical" betas. The cases where "unrodded" kinetics parameters are used in all nodes irrespective of their control rod status are interesting from the point of view that such an application is quite common in many 3-D transient codes. The cases labeled with an "a" represent the status in PARCS prior to the changes reported in this paper, with the exception of the radial discontinuity factor implementation. The results for all these cases are presented in Figures 2 through 5. It should be noted that in all cases radial discontinuity factors were used.

Table 3: Mini-core transient cases to test axial homogenization procedure in PARCS

CASE ID					
6-group Beta Physical	**2-group Beta Physical**	**2-group Beta Effective**	**2-group Importance Factor**	**XS & DF axial smearing**	**Kinetics data axial smearing**
1a	2a	3a		Volume	Unrodded
1b	2b	3b		Flux-volume	Unrodded
1c	2c	3c		Volume	Volume
1d	2d	3d	4d	Flux-volume	Volume

The following observations can be made:

1. Using both rodded and unrodded (instead of just unrodded) kinetics parameters has a modest affect on the power peak value (compare "a" to "c") that should certainly not be neglected if the peak value itself and/or the integral of the power evolution is important.
2. The cusping correction to the nodal cross-sections and to the radial discontinuity factors in partially rodded nodes is very significant (it reduces rod worth) and affects both the power peak value and the half-width of the power evolution curve (compare "a" to "b").
3. The use of effective betas in 2-group calculations dramatically reduces the difference between 2-group and 6-group results (see Figure 5) and since the 6-group case represents the delayed neutron emission spectrum explicitly, this must be taken as strong support for the standard practice of using effective betas in 2-group transient calculations. It is also noted that the use of the delayed neutron importance factor nearly reproduces the results obtained with the effective beta (compare case 3d and 4d) and this is an indication that the importance factor methodology has been implemented correctly; it is actually expected that these two approaches should yield different results since the importance factor method accounts for actual local leakage (or spectrum) effects whereas the effective beta method does not.
4. The impact of volume smearing rodded and unrodded *effective* betas (2-group cases) is of similar magnitude but of opposite sign to that obtained with smearing physical betas (compare Figures 3 and 4). The net effect is that the smearing of effective betas lifts the power peak (compare cases 3b and 3d), which is advantageous for reducing the difference between 2-group and 6-group. The smearing of kinetics data in the 6-group case presses the power peak down and thus also has the effect of bringing 2-group (beta effective) and 6-group results closer to each other.

Figure 2: Power evolution comparison for 6-group calculations

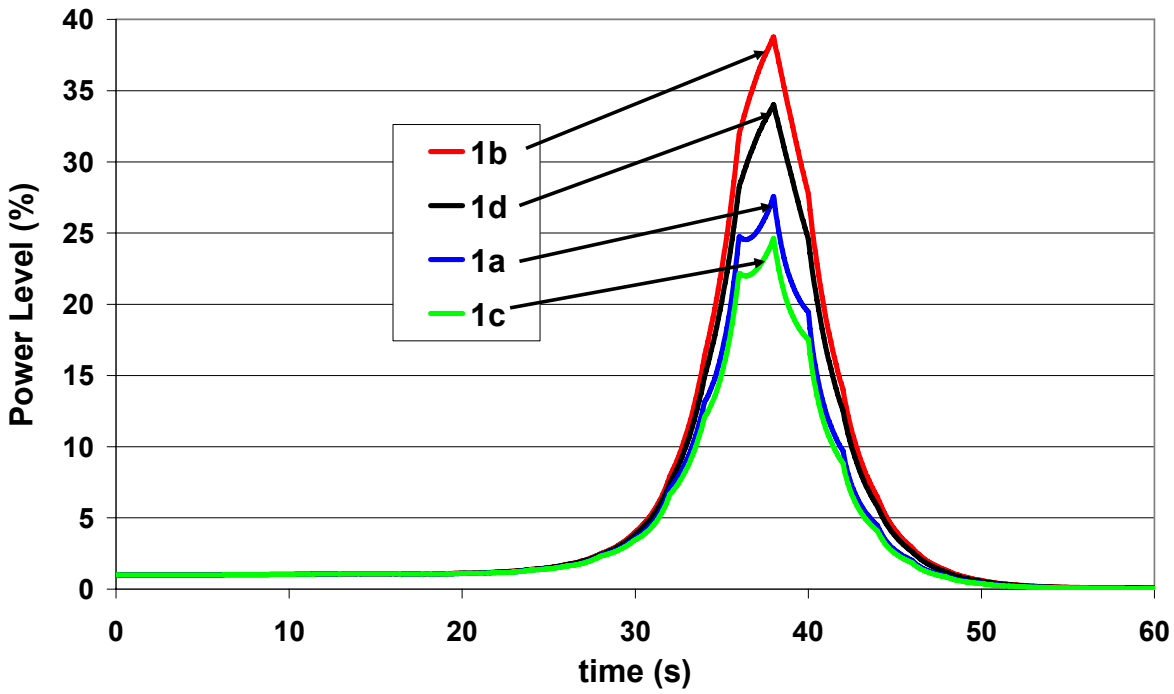


Figure 3: Power evolution comparison for 2-group calculations with physical delayed neutron yield fractions

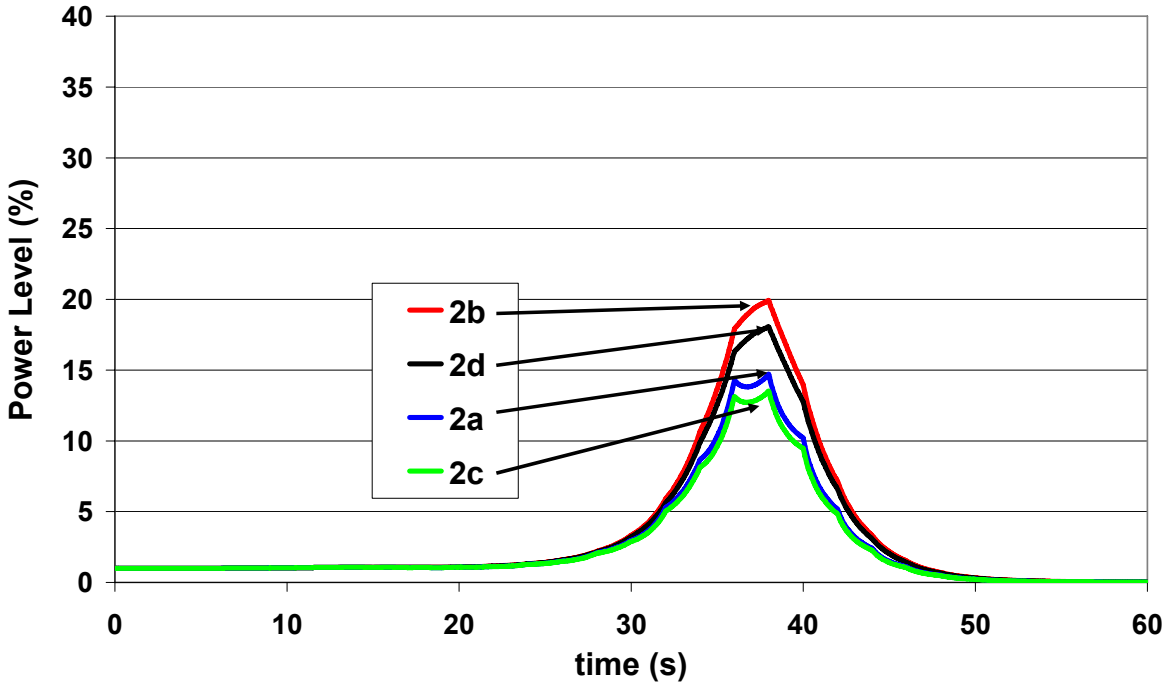


Figure 4: Power evolution comparison for 2-group calculations with effective delayed neutron yield fractions

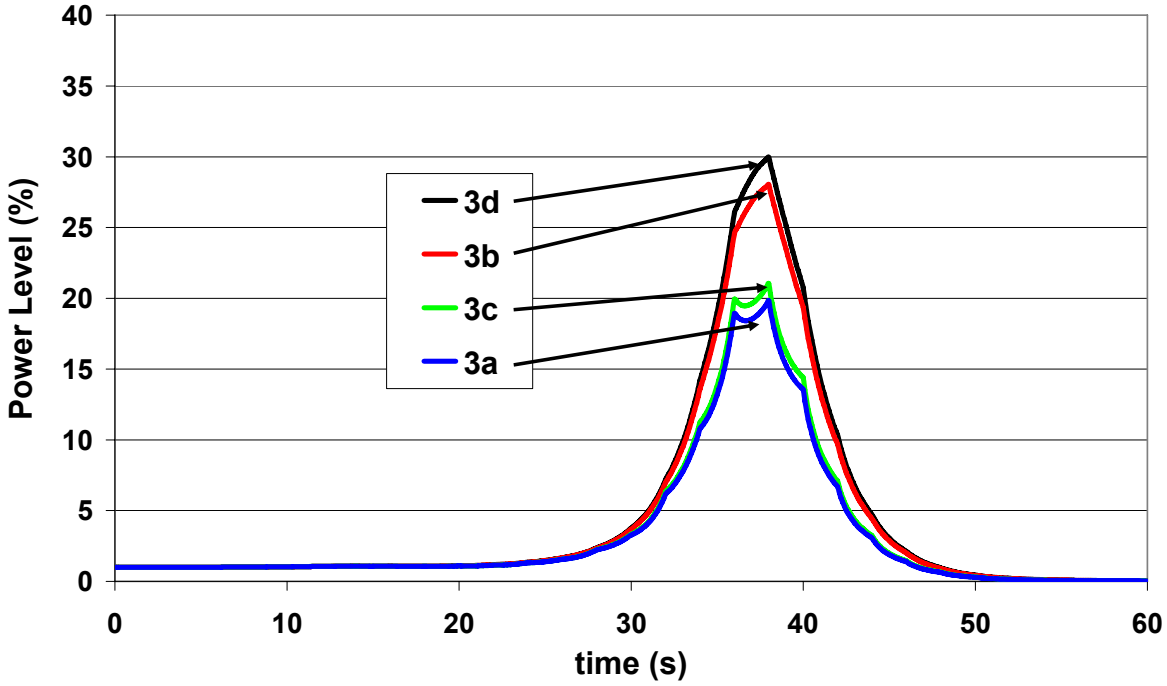
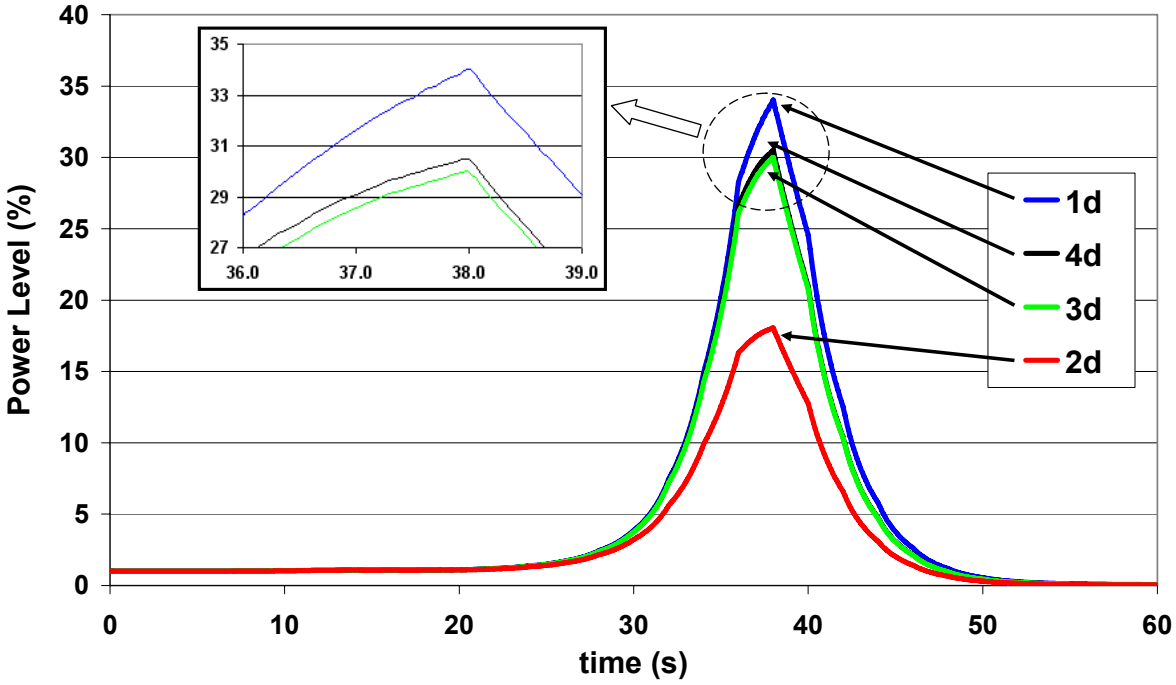


Figure 5: Power evolution comparison for 6-group and 2-group calculations



4. Conclusions

This paper described the modifications to TransLAT and PARCS that were required to establish a 3-D nodal transient methodology that may be employed in multi-group studies. Numerical results were presented to verify that the various modifications function properly and as expected.

Numerical studies were performed to analyze the significance of certain approximations that could be, and often are, applied to the treatment of nodal kinetics data. In particular, it was shown that the simple use of unrodded kinetics data, as is often the case in 3-D nodal transient simulators, could be inappropriate and that blending of rodded and unrodded kinetics data in partially rodded nodes may affect the power evolution during a reactivity insertion transient. It was also shown that the flux cusping correction to the nodal data of partially rodded nodes has an important impact on the power peak and the power evolution in such a transient and that neglect of this correction in transient analyses is certainly improper.

It was further confirmed that the use of delayed neutron importance factors in two-group calculations is needed to bring two-group and multi-group kinetics calculations closer to each other. While this strongly supports current practice it does not imply that it is sufficient.

As far as the remaining differences between two-group and multi-group kinetics calculations are concerned, the neglect of axial discontinuity factors (that should be produced by an axial homogenization procedure, if fully developed) is considered as a possible cause. Therefore, completion of the axial homogenization procedure in PARCS is planned for the very near future.

In conclusion, with the modifications described here, TransLAT and PARCS provide an appropriate multi-group computational platform for an extensive investigation into the adequacy of the standard two-group kinetics approach (using effective delayed neutron fractions). Such an investigation is planned for the near future.

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