REVISED DELAYED NEUTRON DATA FOR PICKERING NGS B

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Abstract - Revised delayed neutron fractions and constants were calculated specifically for Pickering NGS B using the latest available delayed neutron data for fissionable isotopes, the currently recommended CANDU delayed photoneutron data, and the formulae provided by Laughton. Burnup-dependent number densities of the fissionable isotopes were computed with WIMS-AECL-IST. Validation and assessment of the new fractions and constants was performed by comparison of historical operating data with point kinetics simulation using the new values. Recommendations are made regarding possible improvements to the delayed neutron data.

1.0 Introduction

Reactor kinetics calculations require delayed neutron data in the form of direct delayed and photoneutron delayed neutron yield fractions and decay constants. Direct delayed neutrons are produced from the decay of various fission products. Photoneutrons appear when gamma rays produced by the decay of fission products interact with the deuterons in the heavy water of CANDU reactors in a (γ,n) reaction.

At Pickering B, currently used delayed neutron data [1] was calculated by the delayed neutron code DND, in conjunction with lattice-cell code POWDERPUFS-V [2-4] which produced the burnup-dependent concentrations of the fissionable nuclides. The equations used to calculate the delayed neutron parameters were presented by Kugler [5]. Recently, as a result of adopting the industry standard toolset (IST) of codes, WIMS-AECL [6] has replaced the POWDERPUFS-V code for lattice cell calculations. One of the reasons for this is that WIMS-AECL more accurately follows the concentrations of the principal fissionable nuclides and therefore fission rates as a function of fuel burnup.

This study attempts to develop a new set of burnup-dependent delayed neutron parameters using the WIMS-AECL code as a source of fission rates. These new parameters were used to perform simulations of various Pickering B outages using the point kinetics code PPK [9]. The simulation results were then compared with historical operating data.

2.0 Calculation Method

A recent summary report on CANDU delayed photoneutron data [7] provides not only recommended delayed photoneutron fractional group yields and decay constants, but also the recommended direct delayed neutron fractional group yields and decay constants of the four main fissionable nuclides (^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu). The review also recommends direct delayed neutron yields and fractions for each those nuclides, and provides total neutron yields per fission (obtained from ENDF/B-VI). All of these data were used in this study.

In his earlier review of CANDU photoneutron data, Laughton [8] provided formulae for calculating delayed neutron parameters suitable for use in point kinetics calculations. These are primarily based on Kugler's method but do not make the assumption that all photoneutrons have the same neutron yield per fission regardless of the fissionable nuclide. Laughton's methods also have a more obvious dependence on fission rate. The formulae suggested by Laughton were implemented in FORTRAN and used in this study.

WIMS-AECL was used to calculate fission rates of the four principal fissionable nuclides listed above for a range of fuel burnups from 0 MWh/kg U to 280 MWh/kg U. Currently, the average Pickering B bundle burnup is 98.97 MWh/Kg U. Within WIMS-AECL, flux calculations were performed using 89 energy groups, and fission rates were calculated by combining rates from the 89 energy groups into one value per isotope.

Tables of delayed neutron data (17 group fractions and corresponding decay constants) were computed for 56 burnup values in the range given above. Prompt neutron lifetime was not re-evaluated.

Laughton's report [8] presents the following irradiation dependant kinetics parameters

Table 1: Kinetics Parameter Equations

$$\bar{\nu} = \sum_j \bar{\nu}^j F^j \tag{1}$$

$$\bar{\nu}_{d,i} = \sum_j \bar{\nu}^j_{d,i} F^j \tag{2}$$

$$\bar{\nu}_d = \sum_{i=1}^{6} \bar{\nu}_{d,i} \tag{3}$$

$$\lambda_i = \frac{1}{\bar{\nu}_{d,i}} \sum_j \bar{\nu}^j_{d,i}\, \lambda^j_i\, F^j \tag{4}$$

$$a_i = \bar{\nu}_{d,i} / \bar{\nu}_d \tag{5}$$

$$\beta_i = \bar{\nu}_{d,i} / \bar{\nu} \tag{6}$$

$$\beta = \bar{\nu}_d / \bar{\nu} \tag{7}$$

where β_i is the delayed neutron precursor yield fraction for group i ,
F^j is the normalized fission rate for the nuclide j,
a^j_i is the fractional group yield for group i of nuclide j,
$\bar{\nu}^j_d$ is the relative yield of delayed neutrons per fission for nuclide j,
$\bar{\nu}^j$ is the relative yield of neutrons per fission for nuclide j,

$\overline{\nu}_{d,i}^{j}$ is the relative yield of delayed neutrons per fission for group,
λ_i is the decay constant for group i,
β is the total delayed neutron precursor yield fraction, and
$\overline{\nu}$ is the total average yield per fission weighted by fuel .

The total delayed neutron precursor yield fraction is therefore taken to be,

$$\beta = \frac{\text{Precursor Atoms}}{\text{Prompt Neutrons} + \text{Precursor Atoms}}$$

Note that F^j is proportional to the total fission rate for the j'th nuclide and is normalized such that $\sum_j F^j = 1$. By combining equations (6), (1), (2), and (5) we can arrive at the resulting equation for the individual group betas

$$\beta_i = \frac{\overline{\nu}_{d,i}}{\overline{\nu}} = \frac{\sum_j \overline{\nu}_{d,i}^{j} F^j}{\sum_j \overline{\nu}^{j} F^j} = \frac{\sum_j a_i^j \overline{\nu}_d^j F^j}{\sum_j \overline{\nu}^{j} F^j}$$

Once the delayed fraction β_i is calculated for a certain group, one can then calculate the appropriate decay constant for that group by using equation (4) in the form

$$\lambda_i = \frac{1}{\beta_i \overline{\nu}} \sum_j \overline{\nu}_{d,i}^{j} \lambda_i^j F^j$$

Note that this is no different from the original form, but makes use of the previously calculated delayed fraction. The same set of equations can be used to calculate delayed photoneutron parameters except the fractional group yields are not dependent on isotope ($a_i^j \rightarrow a_i$). Laughton also provides necessary formulae to collapse the delayed neutron parameters into fewer groups as desired.

3.0 Results

Table 2 presents the delayed neutron parameters for an equilibrium Pickering NGS B reactor.

Table 2: Pickering B Delayed Neutron Data at 98.97 MWh/kg U

Direct Delayed Neutrons

Group	*Delayed Fraction* β	*Decay Constant* λ *(s^{-1})*
1	1.61E-04	1.33E-02
2	9.92E-04	3.16E-02
3	8.28E-04	1.20E-01
4	1.78E-03	3.14E-01

5	9.42E-04	9.41E-01
6	4.21E-04	2.94E+00
Delayed Photoneutrons		
7	1.11E-07	6.26E-07
8	2.31E-07	3.63E-06
9	7.34E-07	4.38E-05
10	5.30E-06	1.17E-04
11	4.69E-06	4.28E-04
12	7.61E-06	1.50E-03
13	1.59E-05	4.81E-03
14	4.51E-06	1.24E-02
15	2.25E-05	3.05E-02
16	1.95E-05	1.11E-01
17	1.95E-05	3.01E-01
Total Beta = 0.00522737		

The total delayed neutron precursor yield fractions *beta* for core burnups at 0 MWh/Kg U and 279.84 MWh/Kg U were calculated to be 0.00748547 and 0.00416584, respectively. The *beta* of 0.0052274 at equilibrium burnup is lower than the current value [1] of 0.0059036.

4.0 Validation of Data

The new delayed neutron parameters produced from this calculation were validated by running PPK [9] simulations and comparing with historical reactor outage data. In order to build representative PPK runs, the following parameters were extracted from the historical database, via the Data Extraction System (DES), for dates and times of the outages,

- Control, shutoff and adjuster absorber positions,
- linear and logarithmic power, and
- moderator poison concentration

Further assumptions made include:

- Inserting all control absorbers into the core is worth -7.834 mk,
- inserting all shutoff absorbers into the core is worth -78.776 mk,
- the extraneous neutron source is 1E-9 per second unless otherwise noted,
- the prompt neutron lifetime is 0.000873 seconds,
- the reactivity worth of Gd moderator poison is -27.2 mk/ppm,
- PPK assumes a point reactor model,

In order to validate the new delayed neutron parameters, short term and long term comparisons were required. For the short term comparisons after tripping a unit by SDS1 or SDS2, higher sampling rate data was necessary. This was obtained from historical data used to test the prompt fractions of in-core flux detectors. Logarithmic power indications were provided every 0.4 seconds from the ion chambers in this data, and were ideal for short term comparisons.

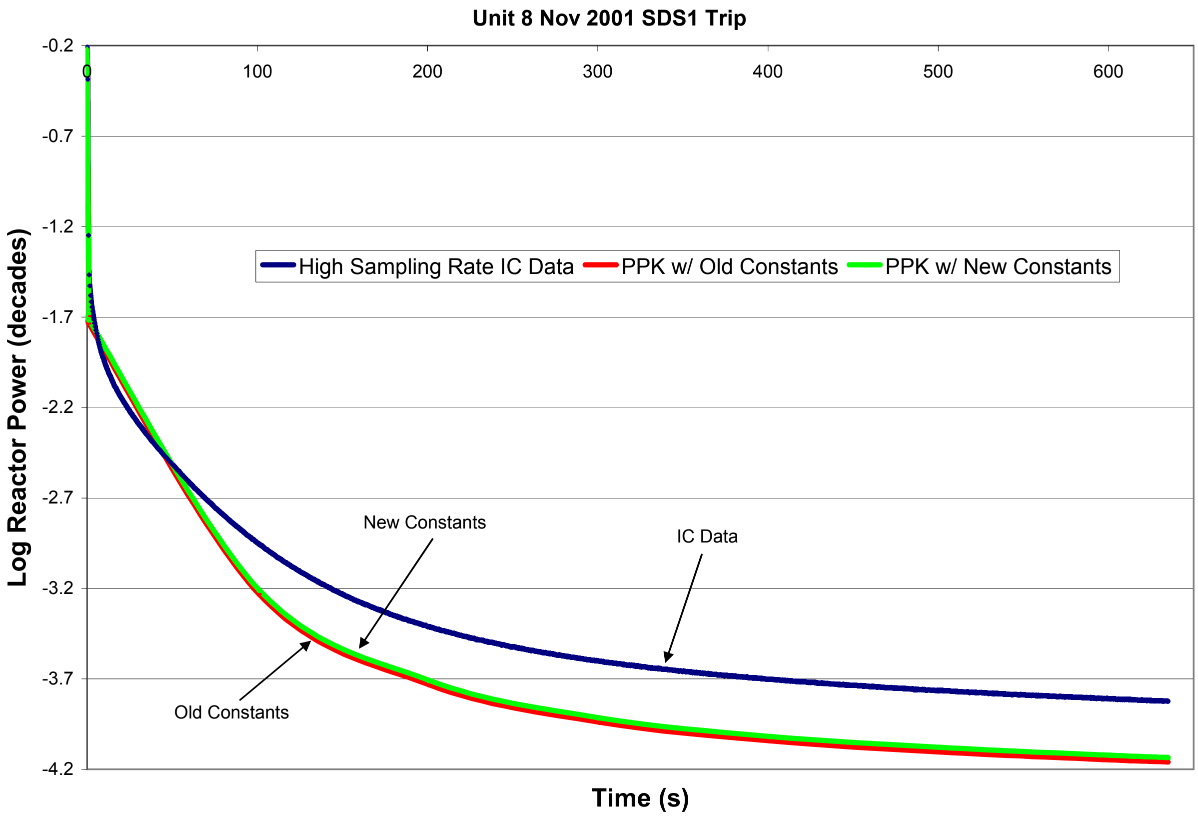
After reactor trip, neutron power will eventually decline to a point at which ion chamber signals of reactor power will become irrational low. Beyond this time, startup instrumentation (SUI) is used to measure reactor power. The SUI readings were calibrated to read the same as the ion chambers at the end of the valid ion chamber data.

4.1 Precursor and Absorber Pair Constants

The PPK code uses the same constants as those derived for use in the TRANSENT code [10] for the time dependence of reactivity from various precursor and absorber pairs of isotopes. There are older as well as a revised set of constants available for use. The old and new sets of constants predict different reactivity for longer outages.

The following plot suggests the new constants are slightly more realistic.

Figure 1: Old vs New TRANSENT Constants Comparison Plot

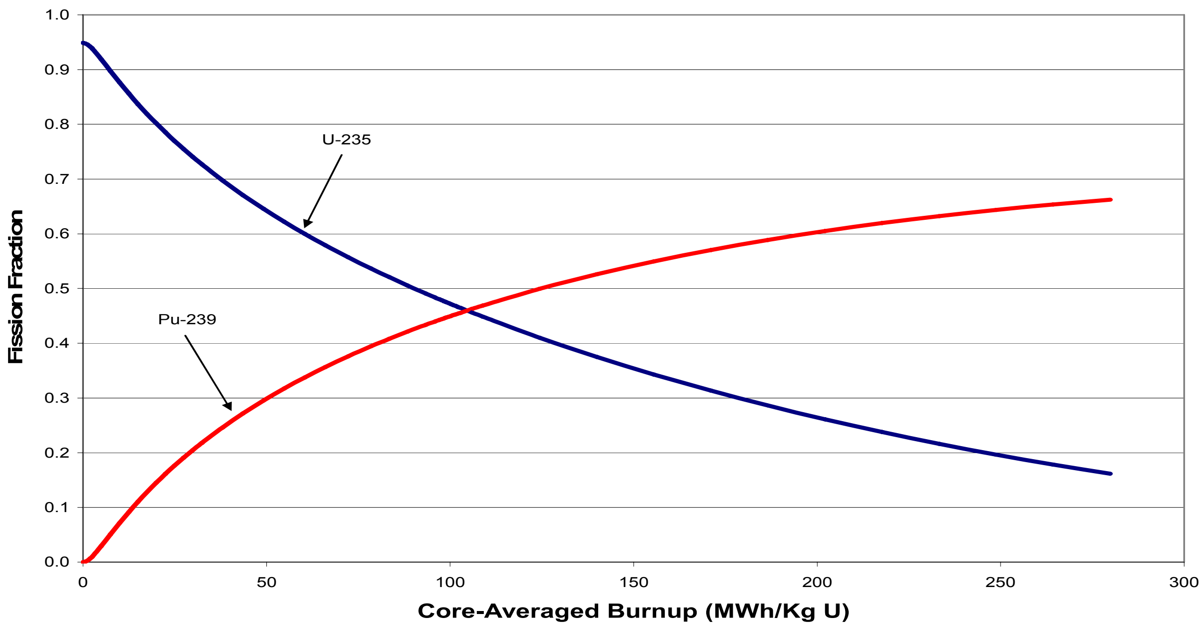


Based on this comparison, all subsequent PPK simulations in this report used the new TRANSENT constants.

4.2 Delayed Parameter Dependence on Irradiation

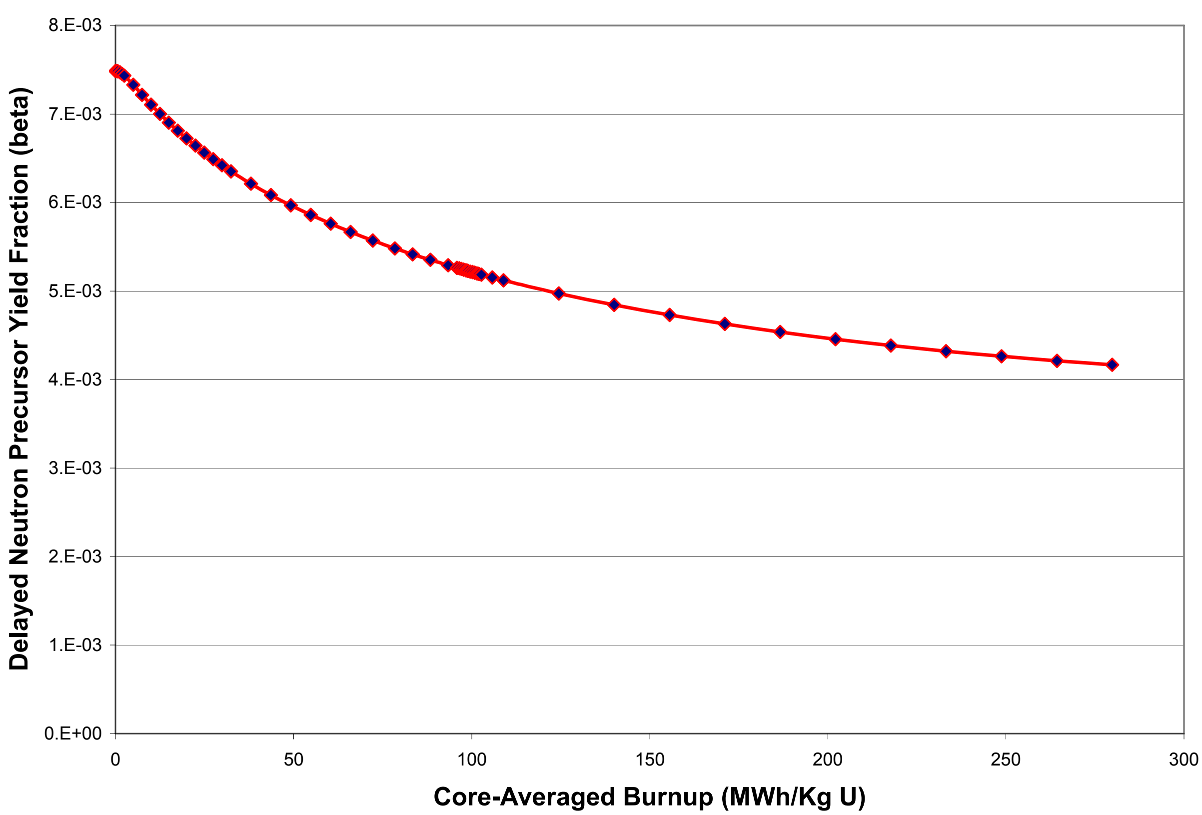
The primary sources of neutrons in a fresh fuel reactor include fissioning in the naturally occurring ^{235}U isotope and delayed neutrons. As the fuel burnup increases over time, the ^{239}Pu fission fraction becomes greater and the ^{235}U fission fraction declines. ^{235}U has a larger delayed neutron yield and this causes a fresh reactor to be more dependent on delayed neutrons.

Figure 2: Fission Fraction Dependence on Burnup



From the following plot, one can see how the precursor yield fraction depends on burnup.

Figure 3: Precursor Yield Fraction *beta* Dependence on Burnup



The following plots show the power decline after an SDS2 trip and suggest that using a core-averaged burnup of 98.97 MWh/Kg U at equilibrium most accurately models the current Pickering B reactors for short and long term (note: outages are designated P/year/unit/number).

Figure 4: P681 Large Time Scale Burnup Comparison

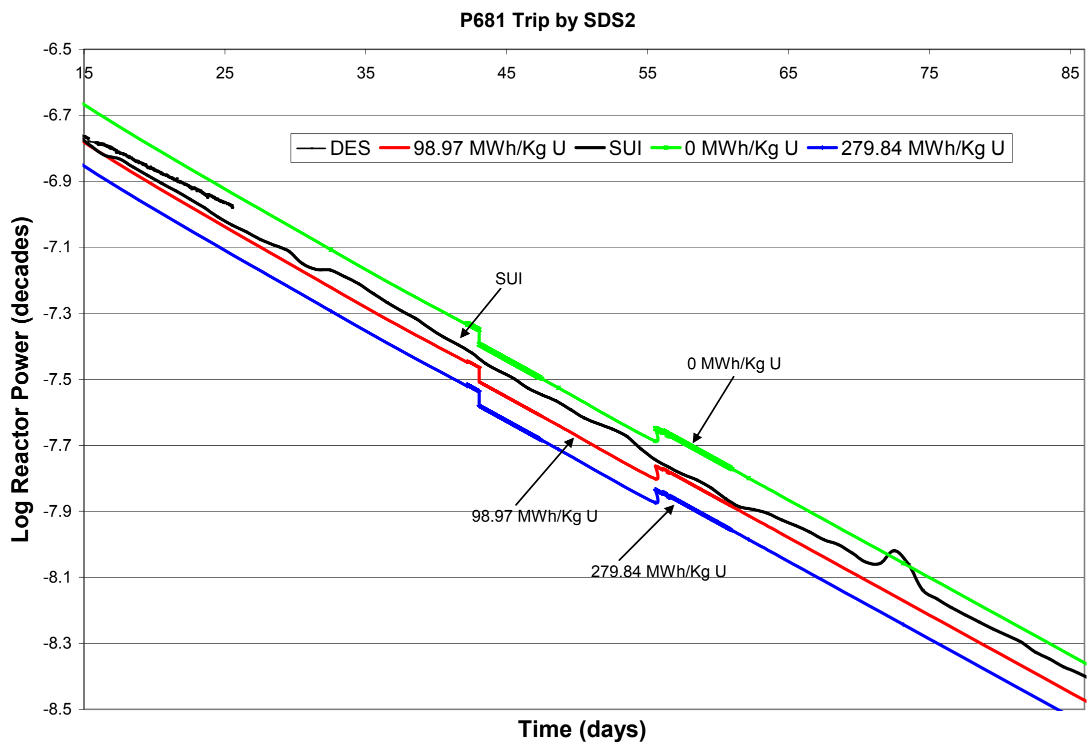
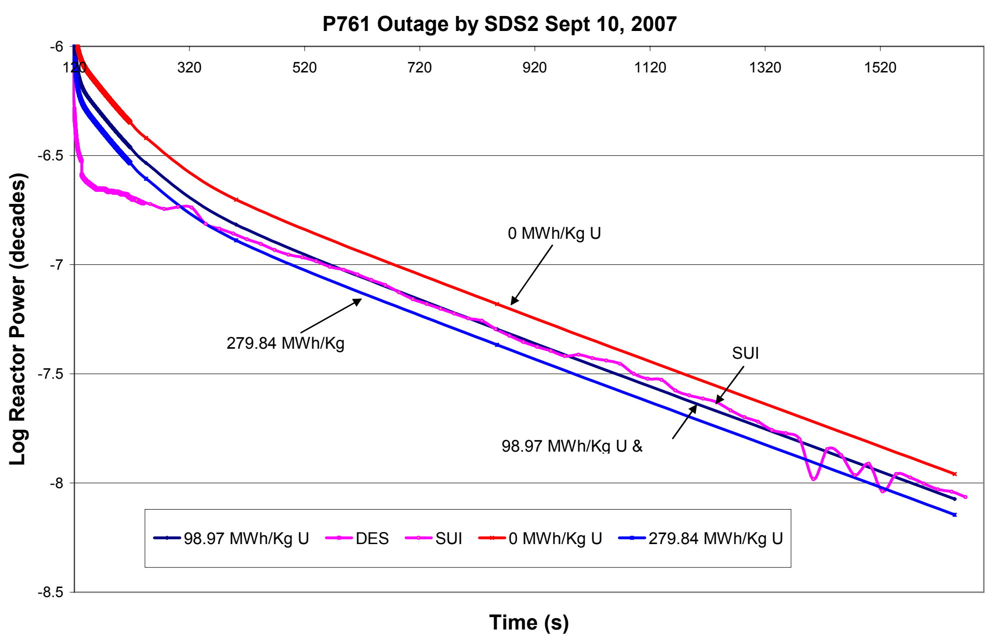


Figure 5: P761 Short Time Scale Burnup Comparison

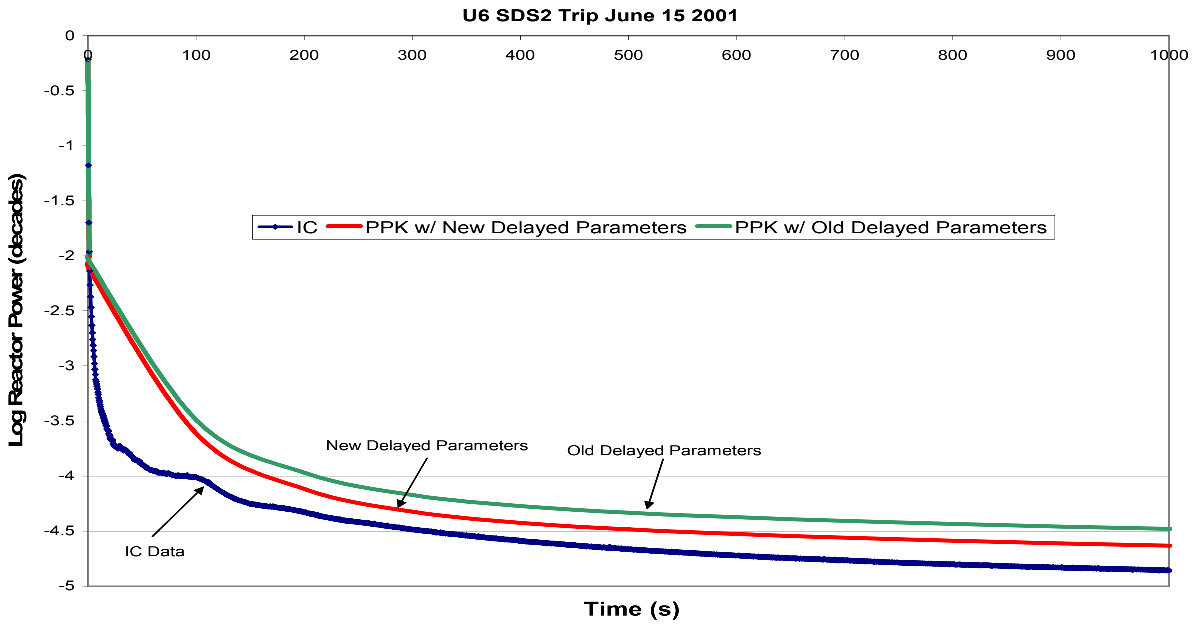


4.3 Short Term Cases

The following figures compare PPK simulations using the new delayed neutron parameters against measured data over relatively short time periods.

Liquid Poison Injection (SDS2) Reactor Trip

Figure 6: SDS2 Trip for P161



Control Absorber and Shutoff Rod Insertion (SDS1) Reactor Trips

Figure 7: SDS1 Trip for P181

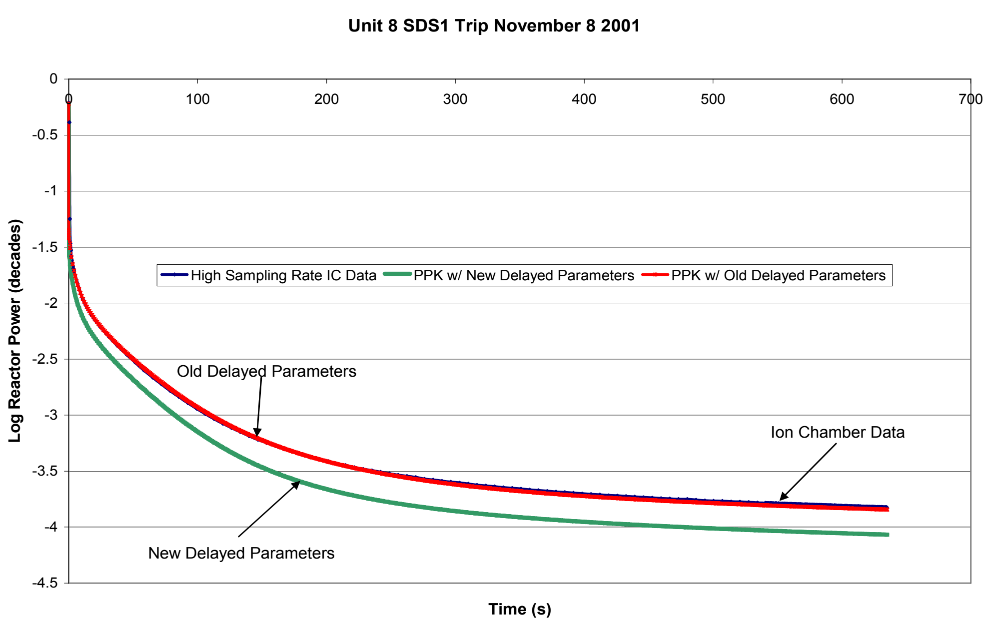
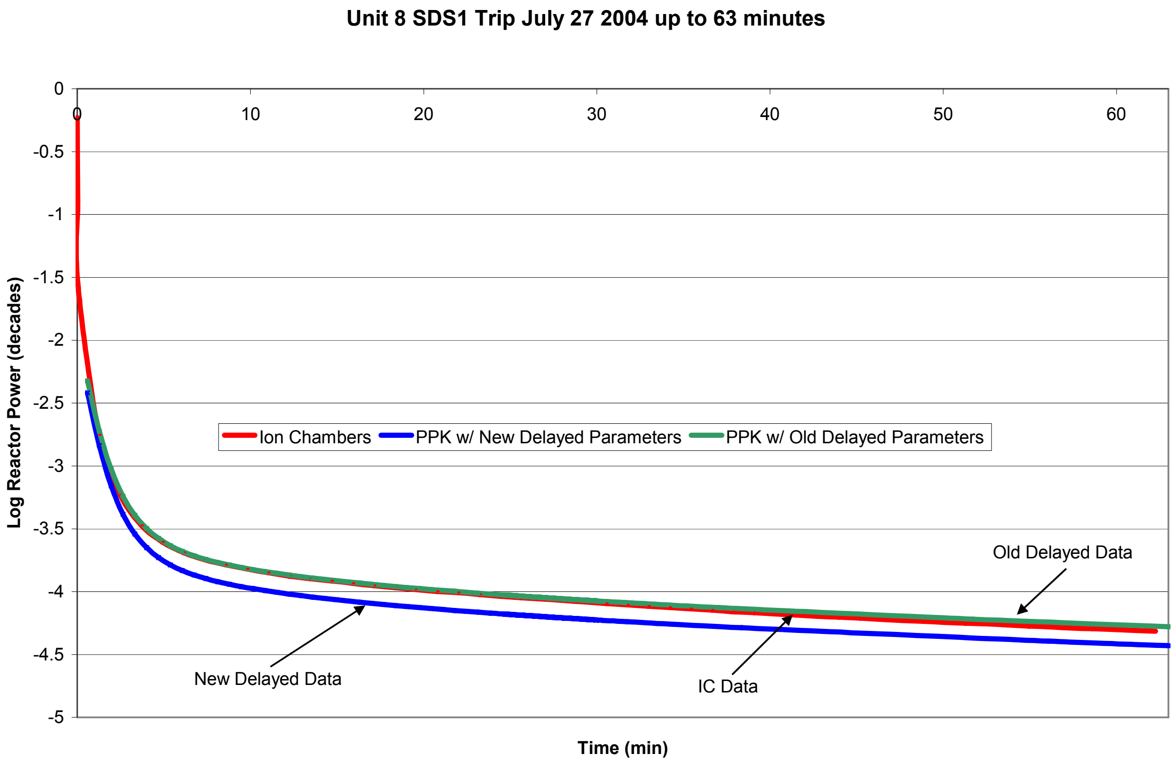
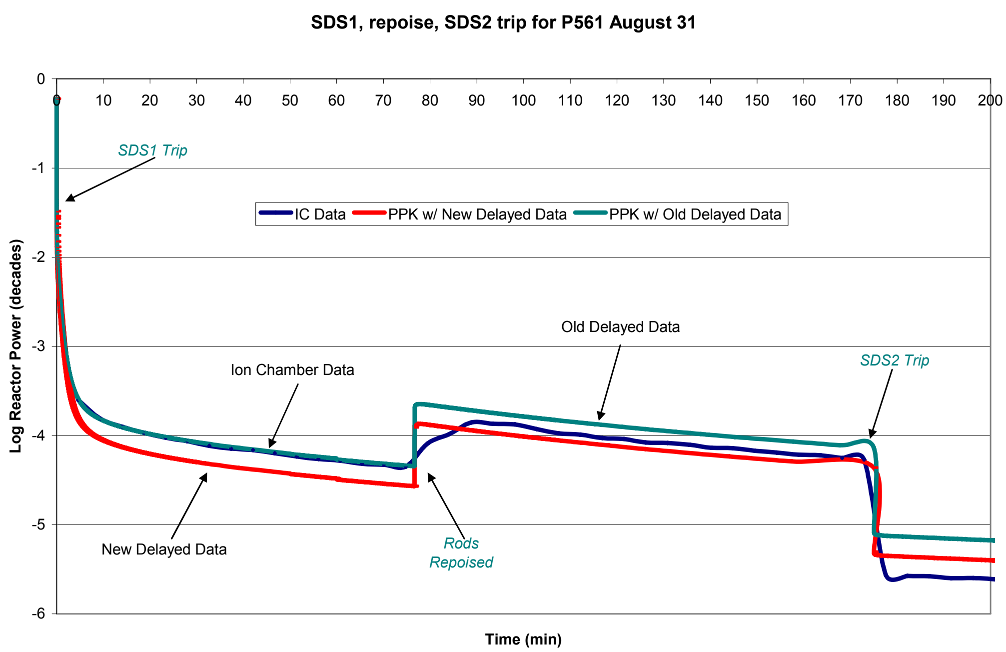


Figure 8: SDS1 Trip for P481 up to 63 Minutes



Mixed Case

Figure 9: SDS1 and SDS2 Trip



4.4 Long Term Cases

The following two figures compare PPK simulations using the new delayed neutron parameters against measured data over long time periods.

SDS2 Trips

Figure 10: P761

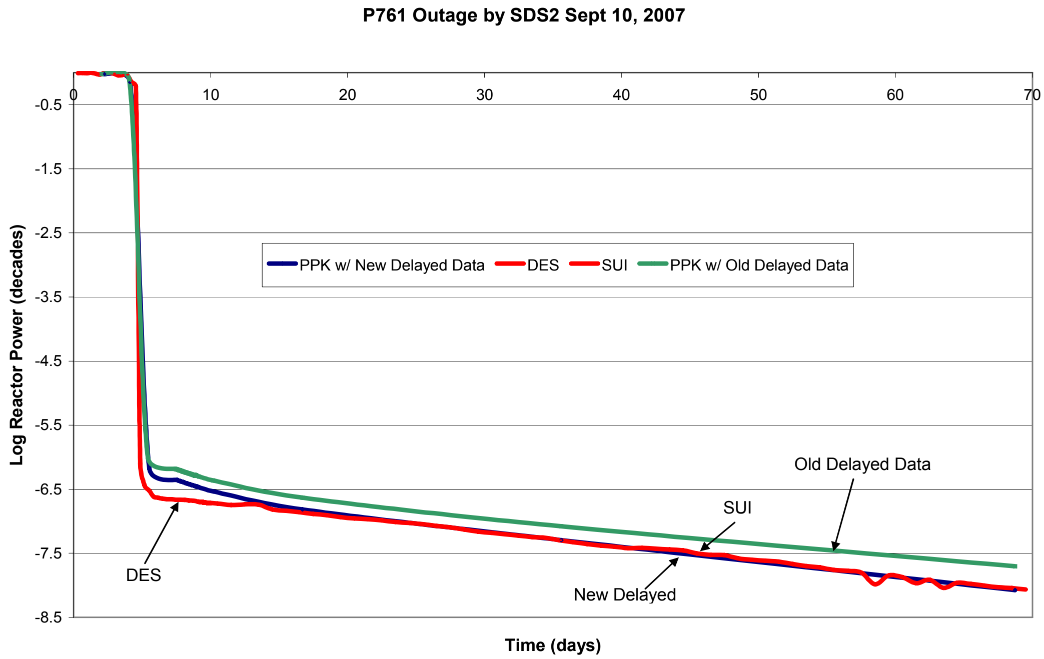
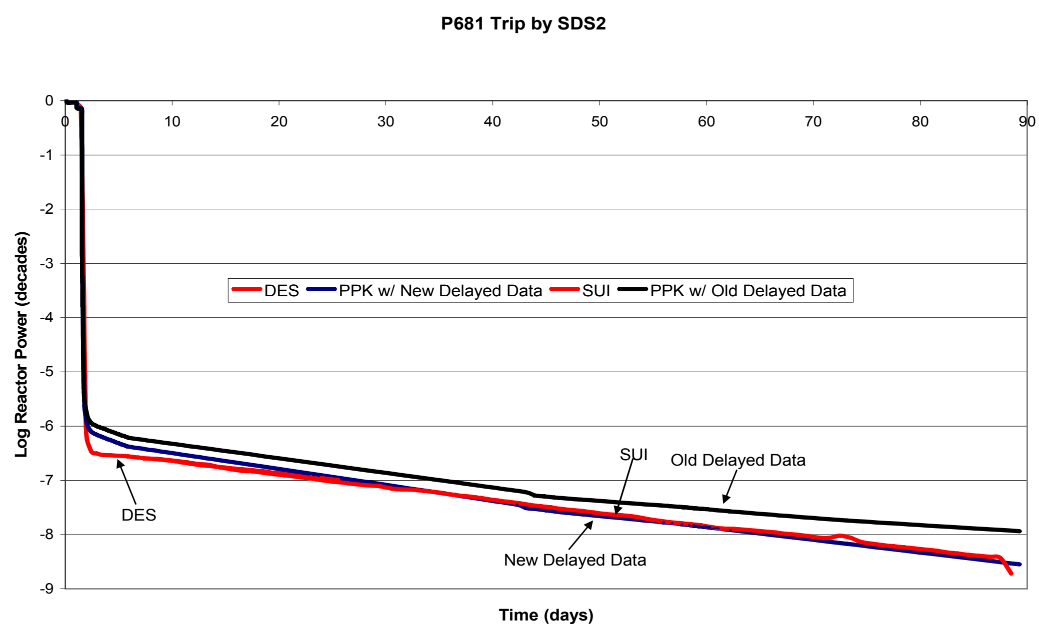
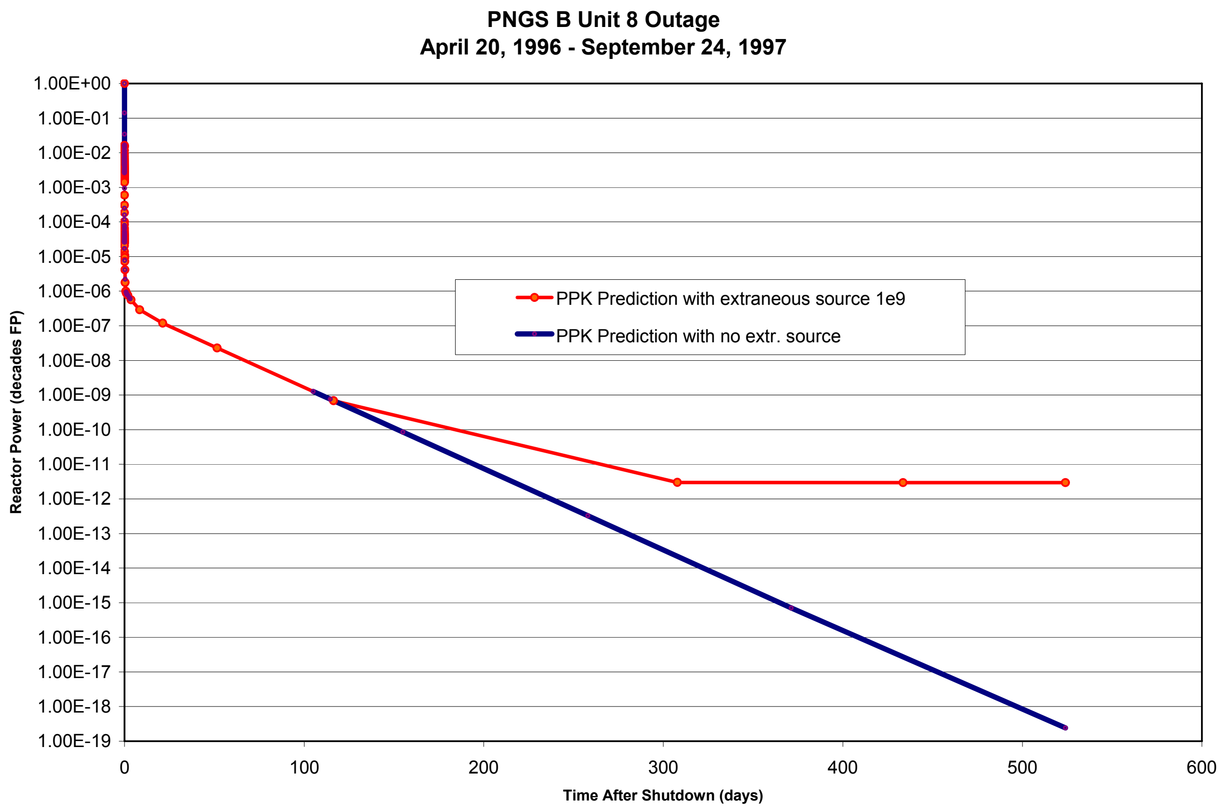


Figure 11: P681



4.5 Extremely Long Term Case

Figure 12: Unit 8 1996 – 1997



In this case, historical SUI data was not available, and the only comparison data useful for this 524 day outage is from a memorandum indicating RRS ion chambers became rational when the core reactivity was at -1 mk, achieved by means of doubling the power by pulling poison. The power was doubled approximately 8.18 times and ion chambers give a power reading of -6.79 decades or approximately 1.62 x 10^{-5} %FP at this point in time (524 days). By backtracking the power doubles appropriately, one can derive the reactor power level just prior to approach to critical

$$P_{present} = \frac{P_{doubled}}{2^{8.18}} = \frac{1.62 \times 10^{-7}}{2^{8.18}} = 5.586 \times 10^{-10}\, FP$$

5.0 Discussion

For many of the PPK-to-outage data comparisons, it became evident that the flux shape had an impact on the ion chamber readings. In cases where shutdown is via the liquid poison injection system, the poison is distributed throughout the reactor including the reflector band of moderator around the outside of the reactor, which is located just before the ion chambers. It serves as an extra neutron absorber and may contribute, in the short term, to an apparent lower power indication than is actually present within the reactor.

Conversely, when a reactor is shut down via insertion of control and shutoff absorber rods, the neutron flux in the center of the reactor becomes much lower, but around the periphery near the ion chambers there are no rods, so the neutron flux remains higher in that region for a period of time. This may contribute to an apparent higher power indication than core-averaged power.

The graph of SDS2 trip from P161 (Figure 6), suggests the new delayed parameters are more accurate than the old. However the SDS1 trips from P181 (Figure 7) and P481 (Figure 8) suggest the old delayed neutron parameters are more accurate. However, since the flux shape dependence on the detector readings cannot be quantified accurately, the old parameters may not necessarily be more accurate. The P561 outage (Figure 9) demonstrates the impact the different flux shapes have; PPK under-predicts indicated power after an SDS1 trip, and over-predicts after an SDS2 trip.

On a large time scale the new delayed parameters are considerably more accurate than the old. In both Figures 10 and 11 the new delayed parameters match the data accurately over months. When considering an extremely long time scale for an outage, over 100 days, the new delayed parameters do not simulate the reactor power level to a good degree of accuracy. Figure 12 shows a 524 day outage, at the end of which PPK (with no extraneous neutron source assumed) predicts the core to be at 5 $x10^{-17}$ % FP while outage data shows the power to be closer to 5.586 $x10^{-8}$ % FP. There are at least two plausible explanations for the discrepancy: the delayed photoneutron data is lacking longer-lived groups, and/or a fixed neutron source exists in the reactor.

Current CANDU photoneutron data is largely based on two ZED-2 experiments [11, 12]. The 1973 experiment [11] had a duration of 45 minutes, while the more recent 2000 experiment [12] lasted for 50 minutes. Given that both of these experiments upon which the photoneutron data is based last less than an hour each, it is reasonable to assume that the scope of the data does not include very long lasting photoneutron groups necessary to capture reactor power behavior for a 524 day outage.

6.0 Conclusions and Recommendations

A new set of 17 delayed neutron group parameters have been calculated and reported in Table 2 and should be appropriate for short and long (less than 100 days) time frames. We leave it to the reader to collapse to fewer groups as desired, and recommend using Laughton's formulas presented in his report [8].

The overall delayed neutron precursor yield fraction *beta* for equilibrium fuel (98.97 MWh/Kg U) of 0.5227% is lower than that reported by Kugler [5] (0.5849%) and the currently used value [1] of 0.5904%.

While reactor power is decaying after a trip, the shutdown system-dependent flux shape presents a small obstacle in short term comparison between the measured data and the PPK simulations. Once enough time has passed, the simulated value for reactor power always approaches the historical data.

To improve the very long term (greater than 100 days) accuracy of the newly calculated delayed neutron data, the CANDU photoneutron groups need to be examined to ensure longer-lived photoneutron groups are included. Also, the existence of a fixed neutron source could be investigated, as this may also contribute to the sustained neutron power seen for very long outages.

7.0 References

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