The DCYPWR Code: Fuel Decay Power Calculations for CANDU® Fuel and Reactor Cores

by

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1.0 INTRODUCTION

When a CANDU® * reactor is tripped or shutdown, there is residual thermal energy due to the natural decay of radioisotopes contained in the fuel. This phenomenon, called decay power, is of significant interest when planning maintenance projects during outages. The DCYPWR computer code predicts the decay power from CANDU® fuel after shutdown, for cooling periods from one second to 31 years.

DCYPWR code development began in 1986, and it has been used at PLGS since then as a best-estimate tool to calculate decay power for various applications. The current state of the DCYPWR code is the subject of this paper.

2.0 CODE DESCRIPTION

Decay power calculations in the DCYPWR code are based on two methodologies:

- The American National Standards Institute / American Nuclear Society standard on decay heat power, ANSI/ANS-5.1-1994 [1], and
- An AECL study of channel decay power [2,3]. This method applies a curve based on a safety analysis of the highest value of channel decay power in the core at various times following shutdown.

2.1 ANSI/ANS-5.1-1994 Standard [1]

The ANSI/ANS-5.1-1994 standard accounts for decay heat in reactors with ^{235}U as the initial major fissile material and ^{238}U as the fertile material. The contributions to decay power of the fissile components ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu are treated explicitly. Methods to account for the effect of neutron capture by fission products (using a function G(t)) and to account for the contribution of the heavy elements ^{239}U and ^{239}Np are also used.

* CANDU® and CANFLEX® are registered trademark of Atomic Energy of Canada Limited (AECL).

The overall equation for computing decay power according to the ANS-5.1 standard is:

$$P_d(t) = \left[G(t) \times P_f(t) + P_{max} \times P_{H.E.}(t)\right] \times P_{SD}{}^{\dagger} \qquad (1)$$

where:
- $P_d(t)$ is the fractional decay power at future time, t
- $G(t)$ accounts for neutron capture in fission products (see above)
- $P_f(t)$ is the sum of the fissile decay power fractions
- P_{max} is the maximum power experienced during operation, as a fraction of full power
- $P_{H.E.}(t)$ is the sum of the heavy element decay power fractions
- P_{SD} is the steady state power at the time of reactor shutdown or trip

The function used to represent the effect of neutron capture in fission products, G(t), supports cooling times up to 10^{10} seconds. Below 10^4 seconds, equation 2 was used to determine the values of G(t):

$$G(t) = 1.0 + (3.24 \times 10^{-6} + 5.23 \times 10^{-10} t) T^{0.4} \psi \qquad (2)$$

{Equation 11 of Reference [1]}

where:
- t is the future time of interest, in seconds
- T is the pre-trip operating time, seconds
- ψ is the fissions per initial fissile atom, dimensionless (assumed = 1.0)

From 10^4 seconds and up, the values of $G_{max}(t)$ from Table 13 of the standard were used. These functions are depicted in Figure 1.

2.2 The AECL-5704 Curve [2,3]

AECL-5704 [2] is a safety analysis study to determine the highest value of channel decay power in the core at various times following shutdown. The resulting curve of channel decay power versus time was extended to include cooling periods from 10^5 to 10^8 seconds [3]. A comparison of the extended AECL-5704 curve to that programmed in the DCYPWR code demonstrates very good agreement (correlation coefficient r = 0.9999). Refer to Figure 2.

2.3 Features of the DCYPWR Code

- *Ease of Use* – With a small number of straightforward input and output files, the DCYPWR code is more convenient to use than most industry standard codes (which require extensive user training).

† The DCYPWR code substitutes actual power, in kW, for the terms $P_f(t)$ and P_{max} in the above equation, resulting in an absolute value for decay power, $P_d(t)$ in kW.

- *Modelling of Detailed Power Manoeuvres* – The DCYPWR code can model not only multiple pre-trip power manoeuvres, but also multiple post-trip power manoeuvres. This feature supports operations such as Shutdown System (SDS) Trip-and-Recovery manoeuvres.

- *Individual Bundle Analysis* – Treatment of up to 12 CANDU® fuel bundles on an individual basis.

- *Full Core Analysis* – Two options are available, providing decay powers for various cooling times for each channel, axial zone pairs, the whole core, as well as identifying the hottest bundle and the hottest channel for each cooling time of interest.

    - *Typical Fuel Distribution* – Modelling of an eight-bundle fuelling shift strategy permits the calculation of full core decay power for a typical core with a typical distribution of fuel, i.e. bundle-specific values for average bundle power, exit burnup and fuel type are obtained from the last Reactor Fuelling Simulation Program (RFSP) run prior to shutdown. A sample facemap appears in Figure 3.

    - *Full Core Verification Option* – Full core decay power calculations involve tens of thousands of calculations. Verification of these calculations by hand is clearly impractical. This option was added in compliance with the Canadian Standards Association standard on quality assurance for nuclear analysis codes [4]. An option was added that can compute the decay power for a core with a uniform distribution of fuel, i.e. all bundles in the core having the same average bundle power during irradiation and the same exit burnup at the time of shutdown.

3.0 APPLICATIONS

3.1 Decay Power in a Single Bundle

The decay power for a given bundle can be computed at several times of interest following a shutdown. This feature supports spent fuel shipments by determining when a bundle is safe for off-site transport, which is described in a reactor physics departmental procedure [5]. Applications include shipment of CANFLEX® fuel following the demonstration irradiation at Point Lepreau Generation Station (PLGS), to Chalk River Laboratories for post irradiation examination [6].

This application requires bulk power operating history and bundle-specific burnup, power and fuel-type data just prior to reactor shutdown. These are input in a problem description file, and the user proceeds directly to the Bundle Calculation Menu to calculate the absolute decay power for one or more specific bundles (in kW).

3.2 Axial Profile of Decay Power in a Channel

The axial profile of decay power across a channel (all bundles) can be computed for future times of interest after shutdown. Applications of this feature include the support of outage maintenance activities, such as SLAR (Spring Location and Relocation) and pressure tube diametral creep measurements, by helping to identify the earliest time after shutdown when the Primary Heat Transport System may be opened for maintenance. Refer to Figure 4 for a sample.

This application is described in a reactor physics departmental procedure [7]. Required inputs include bulk power operating history (via a problem description input file) and bundle-specific burnup, power and fuel-type data for the full core just prior to reactor shutdown. Following a full core decay calculation, the user can pass data to the Bundle Calculation Menu to calculate the absolute decay power for the 12 bundles in any channel (in kW).

3.3 Decay Power in the Full Core

The total decay power in the reactor core can be computed at several future times following a shutdown. This feature identifies the shutdown decay heat load over time, so that system engineers can ensure adequate cooling of the reactor. This application is described in a reactor physics departmental procedure [8].

- *Full Core Calculation Option, 8-Bundle Shift Fuelling Strategy* – This option requires bulk power operating history and bundle-specific burnup, power and fuel-type data just prior to reactor shutdown. The DCYPWR code simulates an on-line “8-bundle shift” fuelling strategy. As a result, a different bundle power history is possible for each bundle. This leads to different values of absolute decay power for each bundle (in kW). The full core decay power is the summation of the bundle decay powers. See Figure 3.

4.0 VERIFICATION

The DCYPWR Version 3.5 source code was verified and results were compared to both the ANS-5.1 and the AECL-5704 methodologies.

4.1 ANS-5.1-1994 Methodology

The DCYPWR code Version 3.5 uses the methodology and algorithms presented in the ANS-5.1-1994 standard. Where possible, CANDU®-specific data are used to customize the decay power calculations [9]. The bases for the individual terms in Equation (1) are given in Table 1.

Table 1 – Bases for Individual Terms in Equation (1)

Term	Description	Algorithm	Data
$G(t)$	neutron capture in fission products	ANS-5.1-1994	ANS-5.1-1994
$P_f(t)$	fissile decay power fractions	ANS-5.1-1994	POWDERPUFS-V [9]
P_{max}	max. power experienced during operation	-	plant data
$P_{H.E.}(t)$	heavy element decay power fractions	ANS-5.1-1994	POWDERPUFS-V [9]
P_{SD}	steady state power at shutdown or trip	-	plant data

To compare DCYPWR with the ANS-5.1 methodology, the reactor operating history given in Appendix B of the ANS-5.1-1994 standard was used (see Table 2). The total assumed operating time was (3*300+2*60) days, or 8.8128×10^7 seconds. These examples assumed 1.0 fission per initial fissile atom, as does the DCYPWR code. In the code, the energy released per fission has been assumed to range between 199.74 and 205.30 MeV/fission, depending on burnup. The example in Appendix B assumed a value of 200 MeV/fission (not a significant difference).

Table 2 – Sample Power History

As presented in ANS-5.1-1994, Appendix B, Example 1 [1]

	Fractional Power					
^{235}U	0.80		0.60		0.40	
^{239}Pu	0.13		0.29		0.42	
^{238}U	0.06		0.07		0.08	
^{241}Pu	0.01		0.04		0.10	
	300d	60d	300d	60d	300d	Time →

Values from the DCYPWR code were compared to the results in Appendix B of the ANS-5.1-1994 standard by calculating the relative error, or percent difference. A positive value for relative error indicates that the DCYPWR value was lower than the value from the ANS-5.1 standard. The individual options compared were:

- *Single ANS Calculation Option* – This option provides the intermediate values for the decay power calculation, allowing the user to verify the code by hand calculation. For this option, each step in the operating history in Table 2 was modelled separately, and then the decay power curves were added together. Referring to Figure 5, the overall decay power fractions from the DCYPWR code are in very good agreement with the sample problems. The relative error is less than 1%, except in two instances, which are explained as follows:

  - The deviation at 10^4 seconds occurs at the transition from applying the G(t) equation to applying the conservative upperbound $G_{max}(t)$. The DCYPWR code applies the conservative upperbound $G_{max}(t)$ for $t \geq 10^4$ seconds, whereas the examples in the standard only apply $G_{max}(t)$ for $t > 10^4$ seconds. This application was not clearly defined in the standard, hence this slight overestimate of decay power by the DCYPWR code.

  - The examples in Appendix B of the standard do not report any decay power contributions from the heavy elements for $t \geq 10^7$ seconds. The DCYPWR code does calculate and apply values in this range, resulting in a slight overprediction by the DCYPWR code.

- *Single Bundle Calculation Option versus Hand Calculation* – A single bundle, irradiated for 205 days with an average bundle power of 536.2 kW, and cooled for 256 days yielded a decay power of 113 W using the "single bundle calculation option" (this scenario was not taken from the Appendix B of the standard). The hand calculation for the same bundle (using intermediate values from the "single ANS calculation option" described above) yielded a decay power of 110 W. The relative error is -2.7%, as seen in Figure 5.

- *Full Core Calculation Option, RFSP Data, 8-Bundle Shift Fuelling Strategy* – The results depicted in Figure 6 show very good agreement up to 10^5 seconds (within 10%). Beyond these cooling times, the DCYPWR results are increasingly lower than the ANS-5.1 examples [‡]. This effect is most likely the difference between modelling natural uranium fuel in the CANDU® system compared to enriched-uranium fuel in a light water system. This indicates that decay power in natural uranium may dissipate faster than enriched uranium fuel.

- *Single Bundle Calculation Option, RFSP Data, 8-Bundle Shift Fuelling Strategy* – To confirm the trend seen in the "full core option" above, two bundles were selected with the following characteristics: (a) a bundle of average burnup and low power, hence long dwell time; and (b) a bundle of low burnup and high power, hence short dwell time. In Figure 6, the results show the same general trend as the full core analysis. However, the magnitude of the relative errors is strongly dependant on the history of the specific bundle, demonstrating the complexity of modelling a CANDU® reactor core. If this single bundle analysis were extended to all 4560 bundles in the core, the relative error curve for the full core analysis would clearly represent the average of the 4560 single bundle curves. This would also serve as an overall uncertainty analysis.

‡ Recall that decay powers at cooling times of 10^9 seconds are in the order of 5 kW for the whole core!

4.2 Verification of AECL-5704 Methodology

As indicated in Section 2.2, the results from the equation in the DCYPWR code are in very good agreement with the extended AECL-5704 curve (correlation coefficient r = 0.9999).

4.3 Uncertainties

The uncertainties in the ANSI/ANS-5.1-1994 standard, dependent on cooling time, are stated as follows:

Cooling Time	Uncertainty
$t_{cooling} < 10^3$ seconds	+ 20%, - 40%
10^3 seconds $< t_{cooling} < 10^7$ seconds	+ 10%, - 20%
$t_{cooling} > 10^7$ seconds	+ 25%, - 50%

Historically, DCYPWR code results have been reported with these uncertainties. The comparison exercises have shown that at longer cooling times the DCYPWR code underestimates the examples in the ANS-5.1-1994 standard by an even greater margin. This effect is most likely the difference between modelling natural uranium fuel in the CANDU® system compared to enriched-uranium fuel in a light water system. The overall uncertainty in applying the ANS-5.1-1994 standard to the CANDU® system has not yet been evaluated.

5.0 FUTURE DEVELOPMENT

The following are activities that have been identified for future DCYPWR code development:

- Addition of an option to estimate pin (fuel element) decay power
- Apply or develop further CANDU® specific factors for use with the ANS-5.1 standard
- Use recent isotopic data specific to CANDU® fuel to obtain more accurate power fractions
- Apply the appropriate decay power-to-fission power ratio for decay power calculations
- Identify and apply possible CANFLEX® specific factors for decay power calculations
- Evaluate the overall uncertainty in applying the ANS-5.1 standard to the CANDU® system

6.0 CONCLUSIONS

- The DCPWR code is an easy tool to use that supports spent fuel shipments, reactor shutdown and outage maintenance activities.
- The ANS-5.1-1994 standard is being properly applied by the DCYPWR code, as demonstrated by the close agreement between the hand calculations (using data from the Single ANS Calculation option) and the examples in Appendix B (Figure 5).
- Decay power calculations for a single bundle were confirmed by hand calculation (Figure 5).

- When the whole core is considered, a trend of increasing percent difference with increasing cooling time appears (Figure 6). This indicates that the decay heat load in the CANDU® decreases more rapidly with time than that of a light water reactor with enriched uranium fuel (for which the ANS-5.1 standard was developed). This may be a further confirmation of the inherent safety features of the CANDU® design.
- The single bundle analysis (Figure 6) confirms the trend of rapid cooling in natural uranium fuel in a CANDU® system compared to enriched uranium fuel in a light water system.

7.0 RECOMMENDATIONS

- Further develop the code to improve DCYPWR and reduce uncertainties.
- Validate the DCYPWR code using PLGS site data, collected during the shutdown for Outage 2000.
- Evaluate the overall uncertainty in applying the DCYPWR to decay power calculations for a CANDU® system.

8.0 REFERENCES

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2. A.C. Whittier, D.W. Black and C.R. Boss, "CANDU® Channel Decay Power", AECL-5704, 1977 January.

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4. "N286.7-99: Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants", Canadian Standards Association, Etobicoke, Ontario, Canada, 1999 March.

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6. M.-J. Basque, J.A. Walsworth and R.W. Sancton, "Calculations Supporting the Shipment of Irradiated CANFLEX® Demonstration Fuel Bundles", paper to be presented at 21st Annual Conference of the CNS, 2000 June 11-14.

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8. D.F. Basque and J.A. Walsworth, "Calculation of Shutdown Decay Heat Load", Physics Procedure, Irradiated Fuel Properties PP-03103-07 Appendix 2, Revision 0, ISSUED: 1999 January 27, NB Power PROPRIETARY.

9. C.W. Newman, "Fractional Powers For Use in DCYPWR", NB Power Internal Correspondence to E. Kennedy, Revision 1, 1990 June 11.

Figure 1 - Effect of Neutron Capture in Fission Products - ANS-5.1-1994

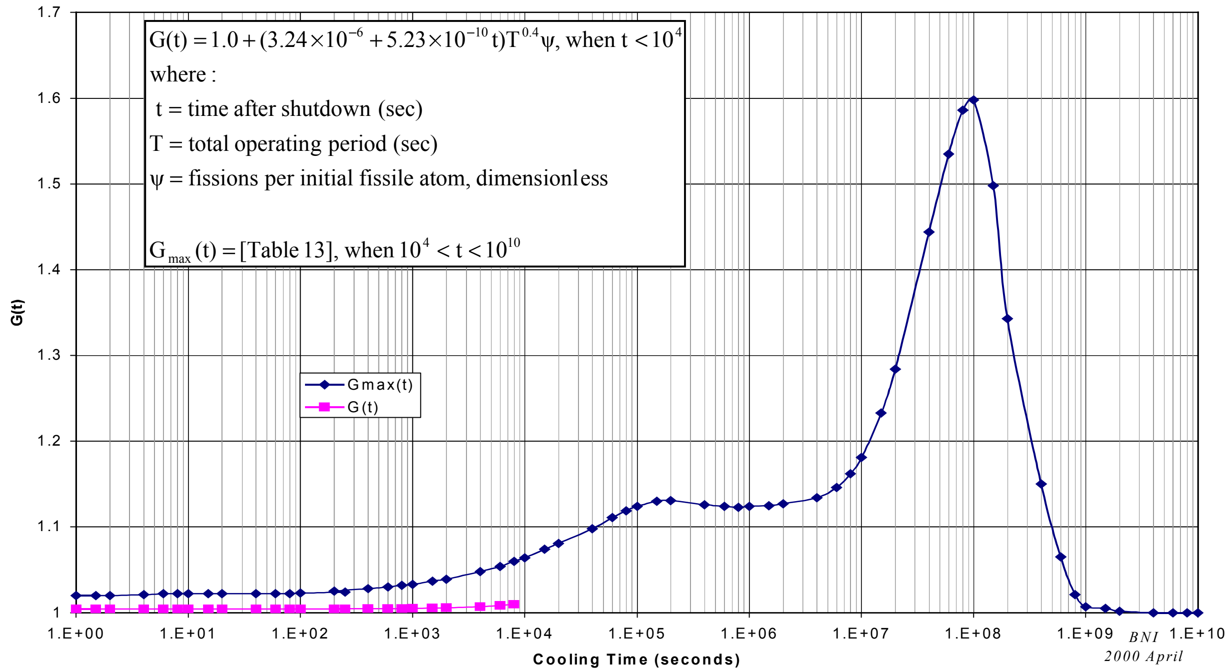

Figure 2 - Relative Decay Power versus Time After Trip

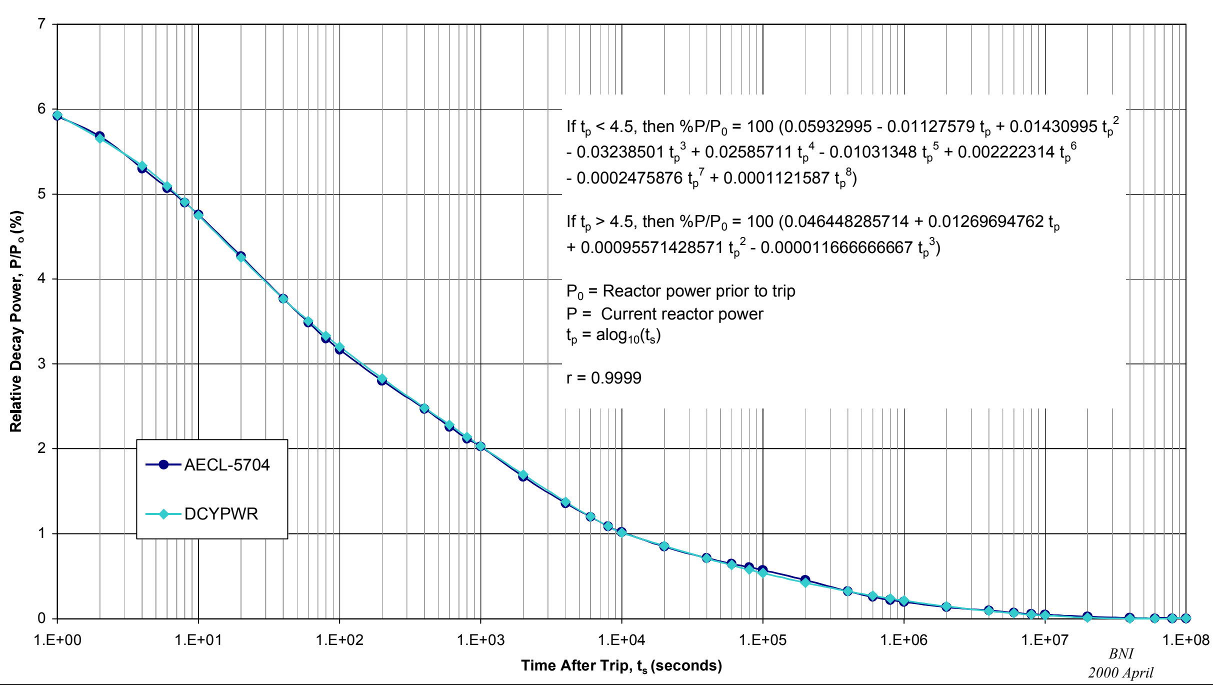

Figure 3 – Full Core Decay Power – 8-Bundle Fuelling Strategy

Reactor Channel Facemap for ANS 5.1 Channel Decay Power (kW), at decay time of 277.778 hours

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15	16	17	18	19	20	21	22
a									8	7	7	8	6	6								
b						7	7	5	9	9	6	9	9	9	9	8	6					
c					8	9	8	4	11	5	10	11	6	9	8	9	8	7				
d				9	6	6	11	11	12	12	12	9	12	12	4	7	10	8	8			
e			7	9	9	5	10	11	10	11	11	10	12	11	12	10	10	7	2	7		
f			9	10	10	12	12	12	9	13	12	6	12	9	13	9	12	6	11	7		
g		8	9	8	10	12	12	12	11	9	10	13	11	13	13	11	9	10	11	9	8	
h		9	3	11	6	12	11	13	12	13	12	13	4	11	7	12	12	11	5	8	9	
j	8	7	7	12	9	13	7	12	11	13	11	12	11	13	13	12	13	10	11	9	9	7
k	7	10	11	11	12	11	11	13	9	10	13	12	8	10	13	11	9	13	12	8	9	6
l	7	6	11	9	13	13	13	12	13	11	5	13	12	13	12	12	12	5	11	12	10	4
m	9	10	12	11	12	11	13	7	9	12	11	13	11	13	9	13	10	13	11	11	10	6
n	9	7	7	12	13	10	12	12	13	13	8	12	10	5	13	11	9	13	11	11	5	8
o	7	9	10	4	8	7	13	13	6	12	12	13	10	12	13	12	13	12	11	7	7	7
p		7	10	9	12	13	10	12	13	10	13	13	13	12	10	12	13	7	12	10	9	
q		8	9	10	10	10	11	13	12	13	10	9	13	6	8	13	10	5	9	7	7	
r			6	9	10	4	12	11	7	13	12	12	11	12	13	11	11	11	9	8		
s			6	8	9	10	8	12	10	11	13	10	13	12	9	12	8	10	6	8		
t				8	7	9	10	10	10	5	11	8	12	10	10	10	9	8	7			
u					6	5	9	10	10	8	10	5	10	10	9	5	7	8				
v						7	6	7	5	9	8	9	9	3	7	8	7					
w									6	7	6	7	7	5								
	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15	16	17	18	19	20	21	22

The bulk thermal decay power was 3649.762 kW.

Figure 4 - DCYPWR - Various Axial Decay Power Profiles at 10^6 sec (277.8 hours)

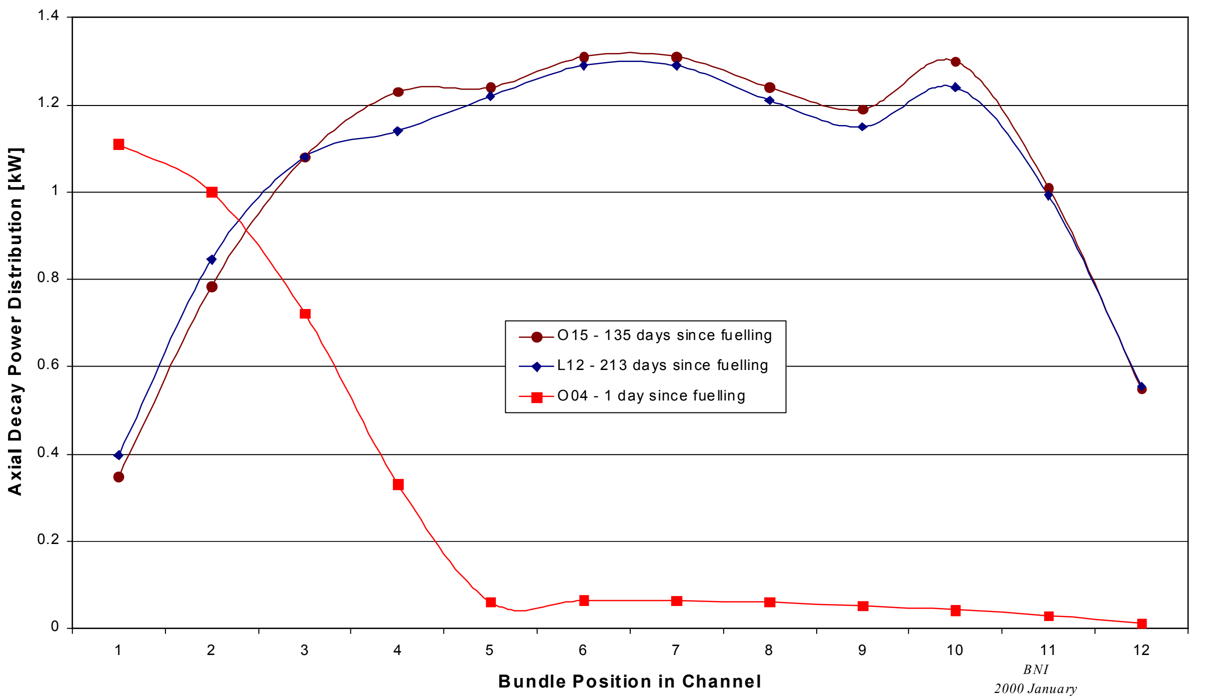

Figure 5 – DCYPWR Code versus ANS-5.1 Examples

Relative Error between DCYPWR Code and Examples in ANS-5.1-1994, Appendix B

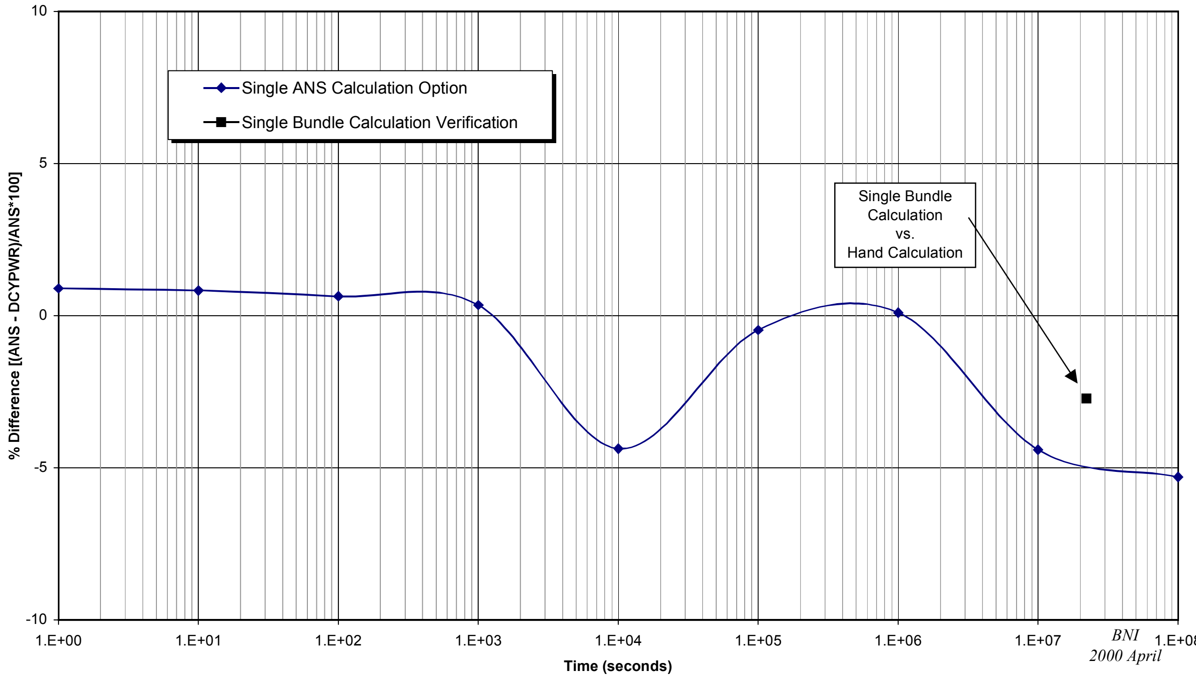

Figure 6 – Decay Power in CANDU® versus Typical Light Water Reactor

Relative Error between DCYPWR Code and Examples in ANS-5.1-1994, Appendix B