WIMS-AECL/RFSP Code Validation of Reactivity Calculations Following a Long Shutdown Using the Simple-Cell History-Based Method

by

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ABSTRACT

The purpose of this analysis is to validate the Reactor Fuelling Simulation Program (RFSP) using the simple-cell model (SCM) history-based method in a startup simulation following a reactor shutdown period. This study is part of the validation work for history-based calculations, using the WIMS-AECL code with the ENDF/B-V library, and the SCM linked to the RFSP code.

In this work, the RFSP code with the SCM history-based method was used to track a 1-year period of the Point Lepreau reactor operating history, that included a 12-day reactor shutdown and subsequent startup. Measured boron and gadolinium concentrations were used in the RFSP simulations, and the predicted values of core reactivity were compared to the reference (pre-shutdown) value. The discrepancies in core reactivity are shown to be better than ±2 milli-k at any time, and better than about ±0.5 milli-k towards the end of the startup transient. The results of this analysis also show that the calculated maximum channel and bundle powers are within an acceptable range during both the core-follow and the reactor startup simulations.

1. Introduction

The Reactor Fuelling Simulation Program (RFSP) [1] is the main tool for physics analysis and fuel management studies of CANDU® reactor cores. The simple-cell model [2] is a one-dimensional, multigroup diffusion-theory lattice-cell calculation that has been linked to RFSP. The purpose of this analysis is to validate the SCM history-based method in a startup simulation following a long shutdown period. This study is part of the validation work for history-based calculations, using WIMS-AECL [3] with the ENDF/B-V library, and SCM linked to RFSP.

For this work, the RFSP code with the SCM history-based method was used to track a 1-year period of the Point Lepreau reactor operating history. The 1-year core-follow

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simulations were performed in order to more accurately represent the history before performing the startup simulations following a 12-day shutdown period (Pt. Lepreau 1993 outage). During the startup transient, measured values of boron and gadolinium concentrations were used in the RFSP code to obtain the system (core) reactivity. These values of core reactivity were then compared with the reference core reactivity, obtained from the core-follow simulations.

The dominant reactivity effects during the startup transient are due to saturating-fission-product changes. Concentrations of saturating-fission-products within the fuel, such as ^{135}Xe, cannot be measured; and therefore the core reactivity, using measured boron and gadolinium concentrations, is used as a means of demonstrating the validity of the results obtained from the SCM history-based method.

In addition, in this analysis maximum bundle and channel powers were evaluated against target steady-state operational values of 882 kW and 6.95 MW (suggested in Reference 4), respectively. These targets account for calculational uncertainty applied to the license limits of 935 kW for maximum bundle power and 7.3 MW for maximum channel power (see Reference 4).

It is important to understand the history-based method and the use of SCM in RFSP, before we discuss SCM/RFSP code validation.

1.1 History-Based Method

In the basic method of representing lattice cells using WIMS-AECL in RFSP, we use fuel tables that consist of cross sections evaluated as functions of irradiation only. But this approach has two weaknesses. First, in addition to fuel irradiation, lattice parameters in fact also depend on other factors, such as moderator density, neutron flux distribution, fuel temperature, coolant temperature and density, moderator poison concentration, etc. The basic method uses average values for these parameters, to calculate the fuel tables, i.e., the calculation assumes the same uniform value throughout the core; for this reason, we refer to this as the uniform-parameter method. Second, in the basic method, when a factor such as moderator poison concentration is changed, the entire fuel table is recalculated with the new moderator poison concentration. That is, the fuel properties are evaluated as if the fuel has been constantly irradiated under the new moderator poison concentration. Therefore, the history of changes in parameters such as moderator poison concentration on fuel compositions is not simulated. In this case, for example, we would need to have fuel tables evaluated in the "perturbation mode", with the moderator poison concentration changing at the appropriate times in the fuel history.

Therefore, the use of pre-calculated, irradiation-dependent fuel tables is not as accurate as the history-based approach. We can replace the use of these irradiation-dependent fuel tables with customized lattice calculations for each bundle in the core. This approach is called the history-based local-parameter method.

1.2 SCM and RFSP

The lattice-code POWDERPUFS-V (PPV) [5], which is based on semi-empirical correlations, is currently used to calculate the lattice-cell properties for the RFSP code. However, PPV is only valid for natural-uranium fuel and for specific fuel designs such as

the 37-element fuel. To overcome these limitations, PPV-based lattice properties are being replaced by WIMS-AECL-based lattice properties. WIMS-AECL is a two-dimensional, multigroup transport code. Using WIMS-AECL for history-based calculations in the RFSP code would be very computer-intensive. Therefore, to help in reducing the computing time without compromising the accuracy of results, the simple-cell model (SCM) was linked to the RFSP code. The SCM is a one-dimensional multigroup method based on diffusion theory, which closely reproduces WIMS-AECL results. For a more detailed description of SCM and its use in RFSP, see Reference 2.

2. Methodology

In this study, normal reactor operation was simulated for an extended period (about one year), to represent the core history for performing startup simulations following the shutdown of the Point Lepreau reactor in 1993 April. Table 1 shows the relevant information on the 1993 shutdown and subsequent startup.

Adjusters are control rods that are arranged in banks, symmetrically placed about the axial, horizontal and vertical mid-planes. All adjuster banks are normally fully inserted into the core. They are used to provide additional reactivity, when needed, by withdrawal from the core. They are also used to shape the thermal flux distribution in the core. In the reactor startup analyzed here, all adjuster banks are withdrawn at the beginning, and then gradually inserted back into the core before significant power levels are reached. The information on the movement of the adjuster banks during startup is presented in Table 1.

During the startup period, a total of 11 poison samples were taken from the moderator for spectroscopic analysis at the Chalk River Laboratories (CRL). The results of these measurements are shown in Table 2.

The normalization factors NF1 and NF2, shown in Table 2, are defined as factors to convert chemical gadolinium and boron concentrations to their neutronically equivalent "natural" gadolinium and boron values, respectively. In other words, the equivalent poison concentrations represent the portion of poison that includes the isotopes that have significant neutron absorption cross sections. Boron, for example, consists of two isotopes, ^{10}B and ^{11}B. Isotope ^{10}B is a strong neutron absorber and therefore burns out in the reactor, whereas isotope ^{11}B has a very small absorption cross section. The normalization factor NF2 takes these factors into account when defining a neutronically equivalent value for boron concentration.

Please note that the isotopic analysis of boron was done only for samples 7 to 11. The boron in samples 1 to 6 was assumed to be natural boron, i.e. the normalization factor was assumed to be 1.0. Isotopic analysis of gadolinium was performed for all 11 samples, as shown in Table 2.

The preliminary work involved preparing the RFSP input files and simulating the operating history from 1992 June 15 to 1993 April 11, using the SCM history-based method. These simulations model the history in preparation for the startup simulations.

Startup simulations of the core were performed from the period just before reactor shutdown at 22:00 on 1993 April 11 until after startup on 1993 April 30, which correspond to 3520.00 full-power days (FPD) and 3525.13 FPD, respectively. The measured moderator boron and gadolinium concentrations were used in the startup simulations. The resulting core reactivities, using measured poison concentrations, were then compared to the pre-shutdown core-follow reactivity of -0.205 milli-k. These results are presented in Table 3 and in Figure 6. These simulations were continued until about one month after startup.

3. Results

The plots for core reactivity, maximum channel power, and maximum bundle power, including the startup simulations up to one month after startup, are shown in Figures 1, 2, and 3, respectively. The reactivity results during the last few months of the core-follow (Figure 1) show that the appropriate reference value for SCM history-based reactivity is -0.205 milli-k, which is the core reactivity immediately before reactor shutdown.

The plots for maximum bundle and channel powers (Figures 2 and 3) show that the peak channel and bundle powers remain mostly below the operational target values of 6.95 MW and 882 kW, respectively. Please note that, to be consistent, the points where poison measurements are not available have been omitted in all the plots, including the plots for maximum bundle and channel powers.

Figure 2 does show some peaks in maximum channel power that exceed the 6.95 MW target. Careful examination has determined that, in the period before the long shutdown, all these peaks occur when the last refuelling is done within a few hours of the end of a simulation time step. Therefore, the origin of these peaks is the low concentration of ^{135}Xe in fresh fuel (the "xenon-free effect"): the saturating-fission-products have not yet reached their equilibrium concentration. These higher values, therefore, are not considered to exceed the relevant limits.

Figure 4 also shows some values of maximum channel power that exceed the 6.95 MW target. In the initial period after startup from the 12-day shutdown, there are two differential effects which would tend to drive the peak powers higher than the target. First, the ^{135}Xe concentration starts at zero for all bundles in the core (both new and old). However, the absence of ^{135}Xe has a differentially greater effect in the inner core than in the outer core, since the ^{135}Xe concentration is normally high in the inner core due to the higher neutron flux. Second, the concentration of samarium isotopes (^{149}Sm and ^{151}Sm) are zero in newly refuelled bundles, while they are high in the older bundles. This is because these nuclides have very long half-lives and therefore they do not decay away during the shutdown. Their concentrations actually increase due to the decay of promethium. These higher values of maximum channel power during the startup transient, therefore, are not considered to exceed the relevant limits.

The plots for maximum channel and bundle powers for the startup period from 1993 April 11 (3520.00 FPD) until 1993 April 30 (3525.13 FPD) are shown in Figures 4 and 5, respectively. These plots show that, aside from the high peak powers caused by the

differential effects that were explained above, the peak channel and bundle powers remain mostly below 6.95 MW and 882 kW, respectively. Therefore, the SCM history-based method predictions during the startup transient are comfortably below the limits. Please note that the startup simulations were performed using measured boron and gadolinium concentrations.

The uncertainty in poison concentration measurements, especially gadolinium, is not known at this time. However, if a reasonable value is assumed for the uncertainty, e.g., 5% in concentration and a nominal worth of ~24 milli-k/ppm for gadolinium, we can conclude that the values of core reactivity are acceptable, considering a typical 1 milli-k measurement uncertainty. The results using measured boron and gadolinium during the startup following a long shutdown indicate that the discrepancies in core reactivity are less than 2 milli-k at any time and that they reduce to less than 0.5 milli-k towards the end of the startup transient. These results are shown in Table 3 and Figure 6.

It is important to note that the discrepancies in core reactivity are highest when gadolinium is present in the core in greater concentration. As the gadolinium concentration is reduced, the discrepancies in core reactivity become smaller. This is evident from Table 3 and Figure 6.

4. Conclusions

The maximum channel and bundle powers obtained using the SCM history-based method and measured poison concentrations show good agreement with the nominal core-follow values, and they remain mostly below the operational target values of 6.95 MW and 882 kW, respectively.

The results of core reactivity calculations using measured boron and gadolinium concentrations indicate that the discrepancy in calculated reactivity is quite acceptable: better than ±2 milli-k at any time, and better than ±0.5 milli-k towards the end of the startup transient. It should be noted that gadolinium is the main contributor to this discrepancy, which has the highest concentration at the beginning of the transient and it reduces to close to 0.0 ppm towards the end of the transient.

5. References

1) B. Rouben, "An Overview of Current RFSP Code Capabilities for CANDU Core Analysis", AECL Report, AECL-11407, 1996 Jan.

2) B. Arsenault, J.V. Donnelly, D.A. Jenkins, "History-Based Calculations Using WIMS-AECL in RFSP", 20th CNS Nuclear Simulation Symposium, Niagara-On-The-Lake, ON, 1997 Sept.

3) J. V. Donnelly, "WIMS-AECL: A User's Manual for the Chalk River Version of WIMS-AECL", AECL Report, AECL-8955, 1986.

4) R. H. Hu, N. Macici, R. Gibb, P.L. Purdy, A.M. Manzer, and E. Kohn, "Fuel Condition in Canadian CANDU 6 Reactors", 5th International Conference on CANDU Fuel, Toronto, ON, 1997 Sept.

5) B. Rouben, "Description of Lattice Code POWDERPUFS-V", AECL Report, AECL-11357, 1995.

Table 1: Power Run-up For PLGS 1993 Startup and Description of Selected Cases

Case Number	Full-Power Days (FPD)	Date	Time	Power Level (%)	Comments / Observations
A. Preparation for Long Shutdown					
SD1	3520.00	93/04/11	22:00	100.	174,148,658 MW.h. All adjuster banks in. Regular production run. Power reduction started for outage at 22:10.
-		93/04/12	00:00	94.6	All adjuster banks in. Power level extracted from reactivity log.
SD2	3520.00	93/04/12	01:00	100.	174,154,230 MW.h. All adjuster banks in.
-		93/04/12	02:00	0.	All adjuster banks out. Reactor is shut down
-		93/04/12	11:00	0.	All adjuster banks back in. Reactor is shut down
SD3	3520.15	Long Shutdown		0.01	All adjuster banks in.
B: Preparation for Startup					
-	-	93/04/24	04:24		Mod. sample taken by chem. lab.(Not used in analysis)
1.	3520.15	93/04/24	05:00	0.3	All adjuster banks out. Reactor critical. Includes shutdown fuelling.
-		93/04/24	07:00		Adjuster bank 7 back in. Zone level with bank 7 in = 27.0 %
-		93/04/24	08:00		Adjuster bank 6 back in. Zone level before bank 6 in = 67.0%. Zone level after bank 6 in = 25.0 %
-		93/04/24	12:00		Adjuster bank 5 back in.
2.	3520.15	93/04/24	14:00	0.3	All adjuster banks in.
C: Raising the Power Level					
3.	3520.15	93/04/25	04:00	0.5	0.56% Full Power. All adjuster banks in.
-		93/04/25	04:27	0.5	Mod. Sample # 1 taken for CRL
-		93/04/25	11:00	7.0	
-		93/04/25	13:00	15.0	
4.	3520.19	93/04/25	18:30	25.0	
-		93/04/25	18:35		Mod. Sample # 2 taken for CRL
5.	3520.28	93/04/25	22:00	50.0	
6.	3520.30	93/04/26	01:00	65.0	
-		93/04/26	01:05		Mod. Sample # 3 taken for CRL
-		93/04/26	02:00		

Continued ...

Table 1: Power Run-up For PLGS 1993 Startup and Description of Selected Cases (Concluded)

Case Number	Full-Power Days (FPD)	Date	Time	Power Level (%)	Comments / Observations
7.	3520.45	93/04/26	06:00	75.0	
-		93/04/26	07:00	87.2	
-		93/04/26	08:00	90.0	
8.	3520.62	93/04/26	10:30	95.0	
-		93/04/26	10:09		Mod. Sample # 4 taken for CRL
-		93/04/26	13:00	97.9	
-		93/04/26	14:00	98.2	
-		93/04/26	18:00	98.8	
-		93/04/26	20:00	99.8	
-		93/04/26	21:00	100.	
9.	3521.10	93/04/26	22:00	100.	
-		93/04/26	22:26		Mod. Sample # 5 taken for CRL
-		93/04/27	02:59		Mod. Sample # 6 taken for CRL
10.	3521.30	93/04/27	03:00	100.	
-		93/04/27	12:00	100.	
-		93/04/27	15:58		Mod. Sample # 7 taken for CRL
11.	3521.80	93/04/27	16:00	100.	
-		93/04/27	23:59		Mod. Sample # 8 taken for CRL
12.	3522.17	93/04/28	00:00	100.	
13.	3522.84	93/04/28	16:00	100.	
-		93/04/28	16:11		Mod. Sample # 9 taken for CRL
-		93/04/29	13:51		Mod. Sample # 10 taken for CRL
14.	3523.76	93/04/29	14:00	100.	
15.	3525.13	93/04/30	23:00	100.	
-		93/04/30	23:04		Mod. Sample # 11 taken for CRL

Note: The date and time for the moderator sample is extracted from the CRL report dated 1994 January 25.

Table 2: Measured Moderator Poison Concentration

Case Number	CRL Sample Number	Date	Time	Power Level (%FP)	Measured Moderator Poison Concentration (ppm)					
					Chem. Gd	Norm. Factor (NF1)	Equivalent Natural Gd	Chem. Boron	Norm. Factor (NF2)	Equivalent Natural Boron
SD1		Apr 11	22:00	100.	(Normal Core-Follow)					
SD2		Apr 12	01:00	100.	(Zone Fills Assumed Same As On Apri1 11 at 22:00)					
SD3		Apr 24	04:00	0.01	(Reactor Shutdown)					
1		Apr 24	04:24	0.3	1.60	0.9955	1.5928	0.08	1.0	0.08
2		Apr 24	14:00	0.3	-	-	-	-	-	-
3	1	Apr 25	04:27	0.5	1.05	0.9955	1.0452	0.08	1.0	0.08
4	2	Apr 25	18:35	25.	0.94	0.9254	0.8698	0.06	1.0	0.06
6	3	Apr 26	01:05	65.	0.91	0.7753	0.7055	0.06	1.0	0.06
8	4	Apr 26	10:09	95.	0.99	0.5151	0.5099	0.14	1.0	0.14
9	5	Apr 26	22:26	100.	1.16	0.2985	0.3462	0.06	1.0	0.06
10	6	Apr 27	02:59	100.	1.24	0.2757	0.3418	0.07	1.0	0.07
11	7	Apr 27	15:58	100.	0.94	0.1659	0.1559	0.55	0.9680	0.5326
12	8	Apr 27	23:59	100.	0.62	0.1184	0.0734	0.70	0.9720	0.6804
13	9	Apr 28	16:11	100.	0.34	0.0762	0.0259	0.69	0.9599	0.6623
14	10	Apr 29	13:51	100.	0.12	0.0691	0.0083	0.51	0.9390	0.4789
15	11	Apr 30	23:04	100.	0.03	0.0448	0.0013	0.25	0.9302	0.2325

Note 1: Sample from Case #1 was taken, but not sent to CRL.

Note 2 : Isotopic analysis of boron for moderator samples 1 to 6 was not done.
A normalization factor (NF2) of 1.0 was assumed in the calculations.

Table 3: Results of SCM History-Based Simulations During 1993 Startup

Case Number	Time (FPD)	Moderator Temperature (°C)	Measured Boron Concentration (ppm)	Measured Gd Concentration (ppm)	Calculated Core Reactivity (mk)	Discrepancy in Core Reactivity (mk)
SD1	3520.00	70.0	----	-----	-0.205	-----
SD2	3520.00	70.0	----	-----	-0.244	-----
SD3	3520.15	37.0	----	-----	-0.189	-----
1	3520.15	37.0	0.08	1.5928	1.954	-----
2	3520.15	45.5	-----	------	0.912	-----
3	3520.15	70.8	0.08	1.0452	1.413	1.618
4	3520.19	70.9	0.06	0.8698	0.756	0.961
5	3520.28	71.4	-----	------	0.715	-----
6	3520.30	71.1	0.06	0.7055	0.794	0.999
7	3520.45	71.0	-----	-----	0.531	-----
8	3520.62	70.9	0.14	0.5099	-1.177	-0.972
9	3521.10	70.8	0.06	0.3462	-0.998	-0.793
10	3521.30	71.5	0.07	0.3418	-1.323	-1.118
11	3521.80	71.0	0.5326	0.1559	-1.696	-1.491
12	3522.17	71.6	0.6804	0.0734	-1.189	-0.984
13	3522.84	71.7	0.6623	0.0259	-0.660	-.455
14	3523.76	70.9	0.4789	0.0083	-0.455	-0.250
15	3525.13	70.8	0.2325	0.0013	-0.563	-0.358

Figure 1: Core Reactivity (mk) vs. Time (FPD) for One-Year Core-Follow (including startup)

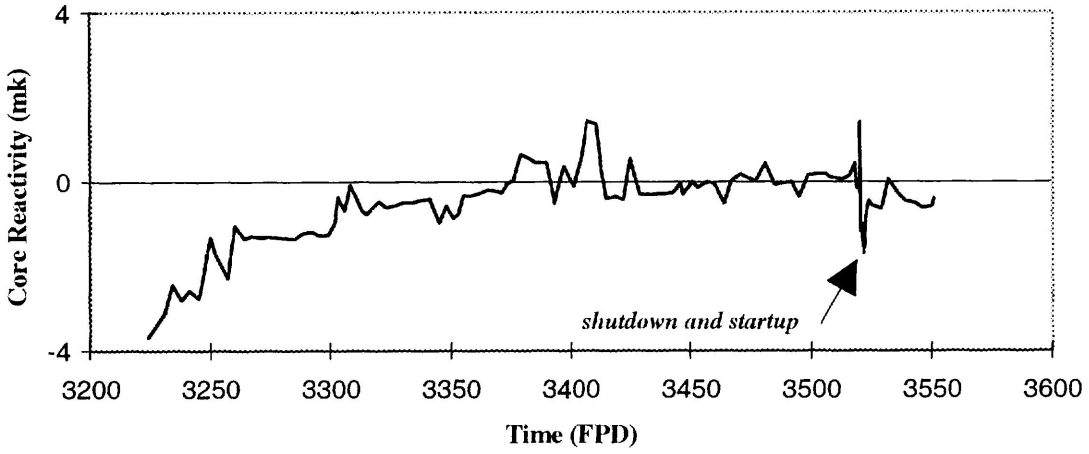

Figure 2: Maximum Channel Power (kW) vs. Time (FPD) for One-Year Core-Follow (including startup)

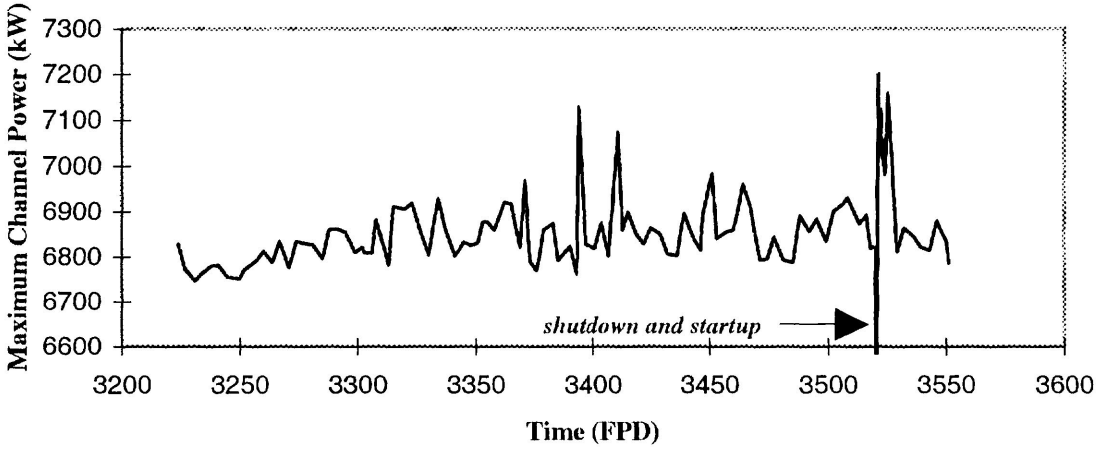

Figure 3: Maximum Bundle Power (kW) vs. Time (FPD) for One-Year Core-Follow (including startup)

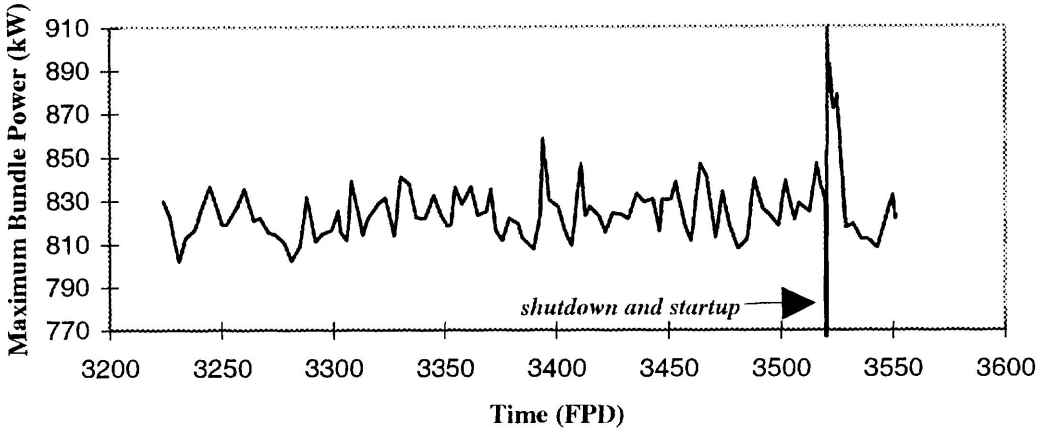

Figure 4: Maximum Channel Power (kW) vs. Time (FPD) for Startup Period When Using Measured Poison Concentrations

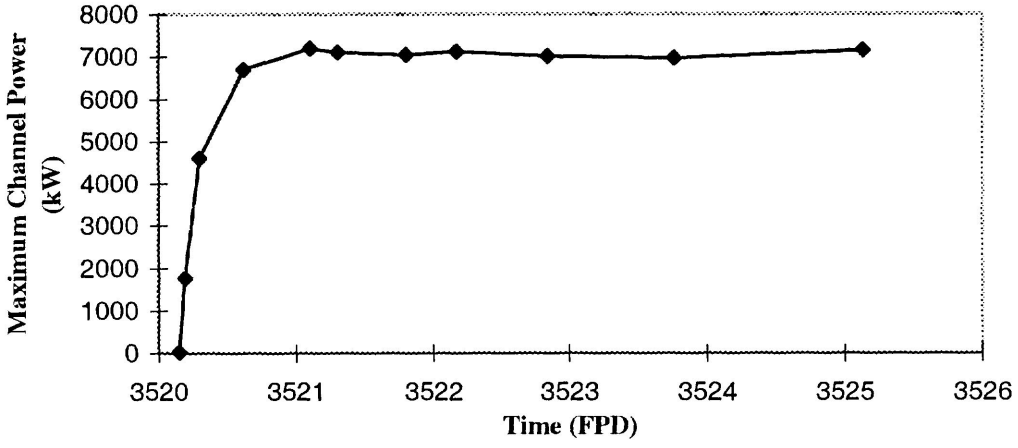

Figure 5: Maximum Bundle Power (kW) vs. Time (FPD) for Startup Period When Using Measured Poison Concentrations

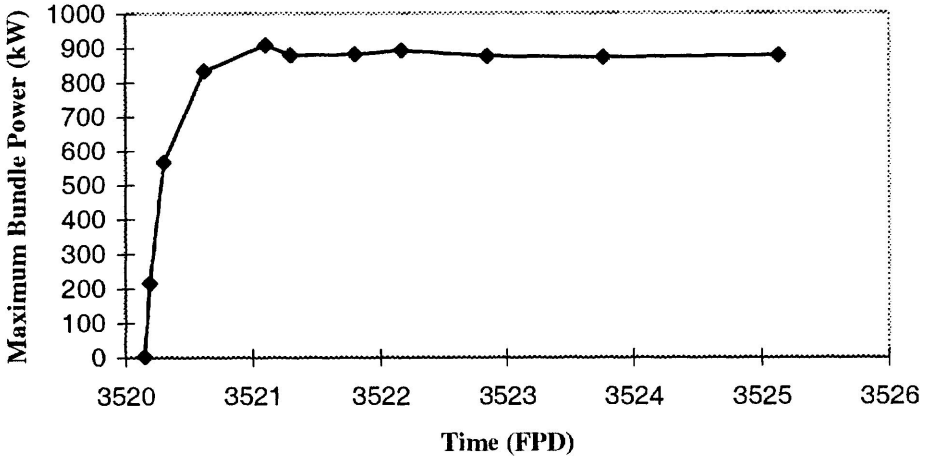

Figure 6: Discrepancy in Core Reactivity During Startup When Using Measured Poison Concentrations (mk)