Analysis of Reactivity Determination Methods in the Subcritical Experiment Yalina

Version 2005-05-03

Carl-Magnus Persson*, Per Seltborg, Alexandra Åhlander, Waclaw Gudowski

Royal Institute of Technology, Department of Nuclear and Reactor Physics,
SE-106 91 Stockholm, Sweden

Thomas Stummer

Vienna University of Technology, Austria

Anna Kiyavitskaya, Victor Bournos, Jurij Fokov, Sergey Chigrinov…

Joint Institute of Power and Nuclear Research, National Academy of Sciences of Belarus, Minsk, Sosny

Abstract – *Different reactivity determination methods have been investigated at the subcritical assembly Yalina in Minsk, Belarus. The knowledge about on-line monitoring of the reactivity level in a future accelerator-driven system (ADS) is of major importance for safe operation. Since an ADS is operating in a subcritical mode, the safety margin to criticality must be sufficiently large. The investigated methods are the Slope Fit Method, the Area Method and the Source Jerk Method. The results are compared with Monte Carlo simulations performed with different nuclear data libraries. The Slope Fit Method compares well with the Monte Carlo simulation results, whereas the Area Method underestimates the criticality somewhat. The Source Jerk Method is subject to inadequate statistical accuracy.*

* E-mail: calle@neutron.kth.se

I. INTRODUCTION

In order to reduce the radiotoxic inventory of the nuclear waste, accelerator-driven systems (ADS) has been suggested to transmute the transuranic elements accumulated in the waste [1]. In a subcritical core it is possible to use large fractions of "exotic" fuels, apart from plutonium also consisting of the minor actinides americium and curium. Intensive research programs investigating the physics and technology of proton accelerators, spallation targets and subcritical cores are required for the development of a full-scale ADS in the future. This present work is dedicated to the study of reactivity determination methods of a subcritical zero-power core. The development of reliable methods for reactivity determination is essential for the safe operation and for the licensing of a future ADS.

Recently, the comprehensive MUSE program (multiplication with an external source), performed at the MASURCA facility in Cadarache, France [2], was completed. In the MUSE experiments, a neutron generator consisting of a deuteron accelerator and a tritium target, was coupled to a subcritical core operating with a fast neutron energy spectrum. A major part of the experiments in MUSE was devoted to the investigation of methods for reactivity determination of different subcriticality levels [2].

Parallel with the MUSE program, another European program devoted to ADS studies has been running at the Yalina facility outside Minsk, Belarus. This facility has the same basic construction with a neutron generator coupled to a subcritical core, but the spectrum of Yalina is thermal. The fact that the fission chain process relies mainly on thermal reactions implies that the neutronic time scales are in the order of a factor 10^3 larger than in a fast system. The methods used in this study, earlier evaluated in a fast spectrum through the MUSE experiments, will now be investigated in a thermal spectrum with completely different time scales. A pulsed neutron source (PNS) experiment has been analyzed by the Slope fit method [3] and the Area method [4], and a Source jerk experiment [3] has been performed and analyzed. The results have been compared with the results from the MUSE experiments.

II. THE YALINA FACILITY

Yalina is a subcritical assembly operating with a thermal neutron energy spectrum. The external source coupled to the core is provided by a neutron generator consisting of a deuterium accelerator (see Table I) and a Ti-D or Ti-T target. In our experiments, the Ti-T target was used, situated in the center of the core. As fuel, EK-10 type fuel rods with 10% enriched uranium oxide are used in polyethylene blocks for moderation. Under normal conditions, the core is loaded with 280 fuel rods, but this number can easily be modified. The cooling of the core is provided by natural convection of air.

The core (Figure 1) consists of subassemblies surrounding the target and the ion channel up to the side dimensions of 400×400×600 mm. Each subassembly is made of nine blocks (80×80×63 mm) of polyethylene (density 0.927 g/cm^3) with 16 channels for the fuel rods, with a diameter of 11 mm. At the sides and behind the target, these polyethylene blocks are replaced by lead blocks of the same geometry. The purpose of these lead blocks is to convert the monoenergetic (D,T)-neutron spectrum into a more spallation-like spectrum. In the lead zone, neutron multiplication by (n,xn)-reactions and a harder neutron spectrum are achieved. The core is surrounded by a graphite reflector with a thickness of 500 mm. Only at the side facing the accelerator, and its opposite side, no shielding or borated polyethylene is used, for ease of handling. Five axial experimental channels (EC) with 25 mm diameter are located inside the core and reflector [5].

TABLE I
Main Parameters of the Neutron Generator (NG-12-1)

Deuteron energy		100 – 250 keV
Beam current		1 – 12 mA
Pulse duration		0.5 – 100 µs
Pulse repetition frequency		1 – 10000 Hz
Spot size		20 – 30 mm
(D,T)-target	Maximum neutron yield	~$2.0\cdot10^{12}$ ns^{-1}
	Reaction Q-value	17.6 MeV
(D,D)-target	Maximum neutron yield	~$3.0\cdot10^{10}$ ns^{-1}
	Reaction Q-value	3.3 MeV

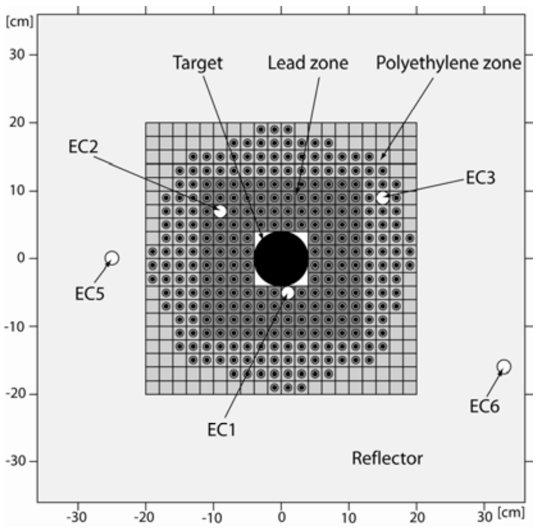

Figure 1. Vertical cross-sectional view of the Yalina core.

Comment [WG1]: "exotic" is not really a technical expression (even if I use it also frequently) in this context. Rephrase to be more technicalm or put "exotic" in quotation marks

III. INITIAL MONTE CARLO SIMULATION

The experimental setup has been analyzed using MCNP version 4c3 [6]. The effective multiplication factor, k_{eff}, and the effective delayed neutron fraction, β_{eff}, have been calculated for three different data libraries; ENDF/B-VI, JEFF3.0 and JENDL3.3. The effective delayed neutron fraction was calculated from the following relation;

$$\beta_{eff} = \frac{N_d}{N_{Tot}}, \tag{1}$$

where N_d is the number of fissions induced by delayed neutrons and N_{Tot} is the total number of fissions [7]. The results for the different libraries are displayed in Table II. For the following analysis, the mean generation time, Λ, is needed. Since MCNP calculates non-adjoint-weighted time parameters, not suitable for kinetic calculations, another approach must be adopted. This is described in the next section.

TABLE II
Effective Multiplication Factor, k_{eff}, and Effective Delayed Neutron Fraction, β_{eff}, Calculated by MCNP.

	k_{eff}	β_{eff} [pcm]
ENDF/B-VI	0.91803 ± 0.00005	788 ± 9
JEFF3.0	0.92010 ± 0.00007	793 ± 9
JENDL3.3	0.92114 ± 0.00006	742 ± 9

IV. REACTIVITY DETERMINATION

IV.A Pulsed Neutron Source Experiment

When a neutron pulse enters a subcritical core, a number of fission chains start to propagate in the fuel. Because of the subcriticality, every fission chain will die out rapidly, which is characterized by a global exponential decay of the neutron flux. By studying the prompt neutron decay after a neutron pulse, it is possible to determine the reactivity of the core in two different ways, either by applying the Slope Fit Method (SFM) or the Area Method (AM), called frequently also as Sjöstrand's method.. These methods have benn be used to analyze the data collected from the pulsed neutron source experiment.
During the experiment, the neutron generator was working at 43 Hz emitting deuterium pulses of duration 2 µs. The ^{3}He-detector was situated in different experimental channels at the core mid-plane. Figure 2 shows the accumulated detector counts after 40 000 source pulses. The inherent source can in all experiments be neglected.

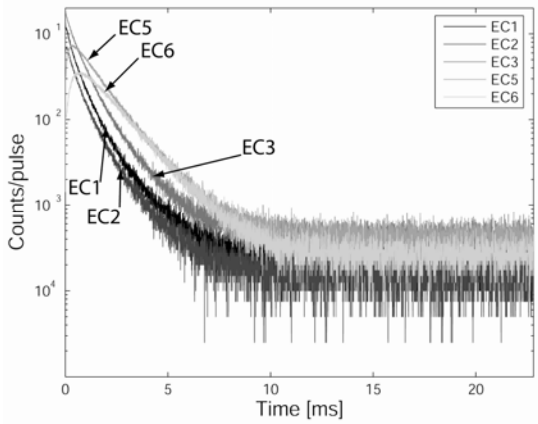

Figure 2. Accumulated ^{3}He-detector counts in different experimental channels after 40 000 pulse insertions.

IV.A.1 Slope Fit Method

According to the one-group point-kinetic model the neutron flux in a multiplying medium induced by a neutron pulse can be described as

$$n(t) = \beta_{eff} \lambda' e^{-\lambda' t} - \rho \alpha e^{\alpha t}, \tag{2}$$

where $\lambda' = \rho\lambda/(\rho - \beta_{eff})$, ρ is the reactivity, λ is the one-group delayed neutron precursor decay constant [2]. α is the prompt neutron decay constant given by

$$\alpha = \frac{\rho - \beta_{eff}}{\Lambda} . \tag{3}$$

Immediately after a neutron pulse injection we assume that $\lambda' t \ll 1$ and the time-dependent behavior is described only by α. By measuring α experimentally, the reactivity, ρ, can be found if β_{eff} and Λ are known.

From Figure 2 it is evident that one single exponential is not enough to describe the complete behavior of the neutron decay. During the first milliseconds, there is a fast transient behavior in the three innermost experimental channels. The flux decreases rapidly until the neutron population has reached a common fission mode energy spectrum. At the same time in the reflector channels, the neutron population builds up from zero, reaches a maximum value and starts to decay.

After approximately 4 ms, the neutron flux reaches a fundamental decay mode, described by the point-kinetic equations, characteristic of the inherent reactor properties. In this region, the fundamental decay mode is described by a single exponential. By fitting a function of the form

Comment [WG2]: Det är väl inget fel att framföra svenska insatser, eller hur?? Feynman method, Sjöstrand's method. Kanske vi skulle vara envisa och använda konsistent namnet "Sjöstrand's method" (called also Area Method", eller hur??

Comment [WG3]: Sant, men ett konstigt uttryck, vad är "common fission mode energy spectrum"

$$f(t)=\sum_{i=1}^{\infty}A_i e^{\alpha_i t} \quad (4)$$

using a minimization code MINUIT [8], it is possible to determine the exponential component which represents the fundamental decay mode. It turns out that four terms are necessary to describe the response function with satisfactory statistical agreement. Two fast exponentials are required for the transient part, one exponential for the fundamental decay mode and finally, one constant for the delayed neutron background. The α-values, describing the fundamental decay mode, are summarized in Table III and displayed in Figure 4. The values are followed by a one standard deviation statistical error. Throughout this study, only statistical errors are considered.

The neutron pulse and the subsequent neutron flux have also been simulated with MCNP, relying on the nuclear data library ENDF/B-VI. The reaction rate with ^{3}He has been tracked in each experimental channel during 10 ms after the neutron pulse. The results for experimental channel 2, 3 and 5 are displayed in Figure 3. This data can be analyzed in the same way as the experimental data, to find the α-values. In this case, the reactivity is already known, through earlier simulations, which means that the mean generation time can be found from Eq. (3).

In the simulated case, three exponents are sufficient to describe the response function with satisfactory statistical agreement. Two fast exponents are required to describe the short-term transient part and one exponent for the fundamental decay mode (obviously there is no delayed neutron background in the simulation). The α-values, one for each experimental channel, and corresponding mean generation time are summarized in **Error! Reference source not found.**.

By using the simulated values of the mean generation time in **Error! Reference source not found.**, it is possible to calculate the reactivity from the experimental values of α (Table III). Here we need some discussion spreading of results over those 5 channels. The same for next tables.

TABLE III
Results from the Slope Fit Method.

	α [s^{-1}]	ρ [pcm]
EC1	-675 ± 13	-8240 ± 260
EC2	-722 ± 19	-9050 ± 260
EC3	-711 ± 11	-8870 ± 160
EC5	-634 ± 21	-8130 ± 310
EC6	-653 ± 2	-8560 ± 30

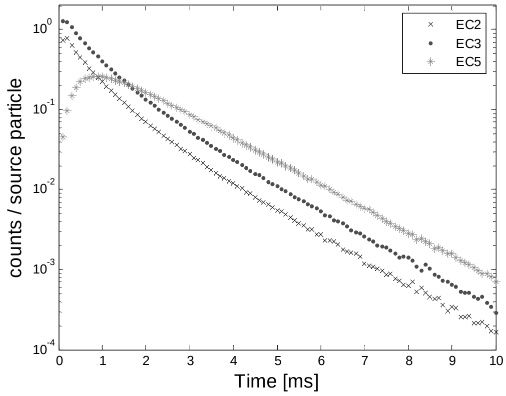

Figure 3. MCNP-simulation of the pulsed neutron source experiment. The curves describe the reaction rate with ^{3}He in EC2, EC3 and EC5.

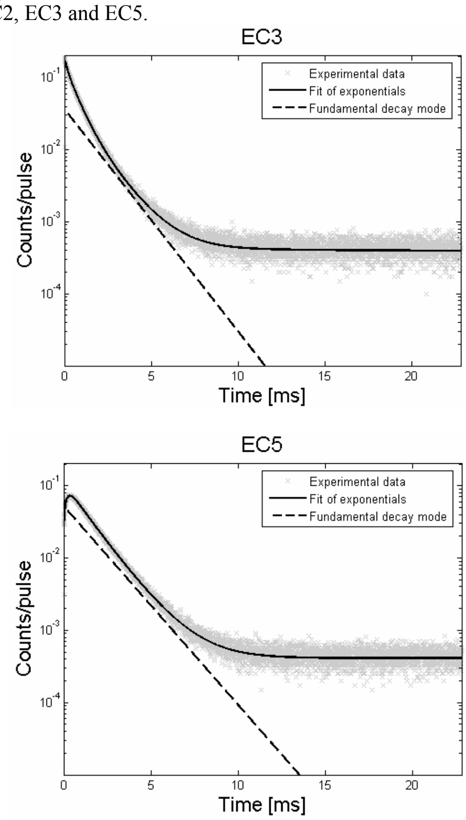

Figure 4. Fit of exponentials to the experimental data in the experimental channel in the core (EC3) and in the reflector (EC5).

IV.A.2 Area Method

TABLE IV
α-values and Corresponding Mean Generation Times, Λ, Determined from the MCNP Simulations of the PNS-experiment.

	α [s^{-1}]	Λ [μs]
EC1	-726 ± 16	134 ± 3
EC2	-713 ± 3	136.3 ± 0.6
EC3	-716 ± 2	135.7 ± 0.5
EC5	-690 ± 7	140.8 ± 1.5
EC6	-679 ± 1	142.2 ± 0.3

Considering the much smaller time scale of the decay of the prompt neutron flux compared to the delayed neutron precursor lifetimes, the delayed neutron flux contribution can be regarded as constant during the studied time interval. If the area under the response function is divided into a prompt neutron area, A_p, and a delayed neutron area, A_d, as illustrated in Figure 5, the reactivity in dollars can be expressed as [4];

$$\frac{\rho}{\beta_{eff}} = -\frac{A_p}{A_d}. \qquad (5)$$

The total area under the experimental data is found by trapezoidal numeric integration and the delayed neutron background is found by averaging the values from the last milliseconds where the curve has flattened out. The results are listed in Table V.

TABLE V
Results from the Area Method.

	ρ [$]
EC1	-13.9 ± 0.1
EC2	-13.7 ± 0.1
EC3	-12.9 ± 0.1
EC5	-13.0 ± 0.1
EC6	-13.5 ± 0.1

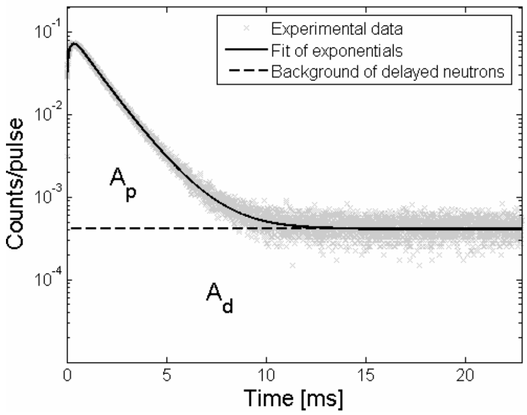

Figure 5. Illustration of the prompt and the delayed areas utilized in the Area method.

IV.B Source Jerk Experiment

The idea behind the Source Jerk Method is to operate the subcritical reactor at steady state, at flux level n_0, and then suddenly remove the neutron source. At this point the system will make a prompt jump to a lower level, n_1, determined by the delayed neutron background. This level is only quasistatic and will decay according to the decay rate of the delayed neutron groups [3]. The reactivity in dollars is given by

$$\frac{\rho}{\beta_{eff}} = \frac{n_1 - n_0}{n_1}. \qquad (6)$$

When running in continuous mode the neutron generator is shut down by dropping an aluminum gate into the deuteron beam line. The neutron flux is measured with a ^{3}He-detector in EC2 (Figure 6). The neutron flux levels are estimated by the flux values before and immediately after the prompt jump, which gives the reactivity ρ = -8.9 ± 1.7 $. The statistical error is large due to low count rate after the source jerk.

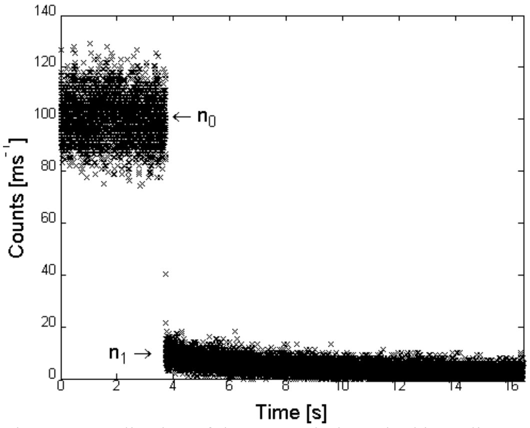

Figure 6. Application of the Source jerk method in Yalina.

IV.C Experimental Estimation of the Mean Generation Time

The neutron mean generation time has been calculated based on Monte Carlo simulations (**Error! Reference source not found.**). However, it is of interest to verify this value experimentally. An estimation of the ratio Λ/β_{eff} can be found by rewriting Eq. (3) as

$$\frac{\Lambda}{\beta_{eff}} = \frac{1}{\alpha}\left(\frac{\rho}{\beta_{eff}} - 1\right), \tag{7}$$

and combining the values of α obtained from the Slope Fit Method with the values of ρ/β_{eff} obtained from the Area method or the Source jerk method. By using the calculated values of β_{eff} it is possible to find an estimate for Λ. From the results listed in Table VI, it can be expected that the true value of Λ is in the range between 140 – 180 µs, by combining the Slope fit method and the Area method. If the Slope fit method and the Source jerk method are combined, the expected value is approximately 110 µs. The results diverge since the different methods give different values of the reactivity. However, this simple method shows that the values obtained by simulations (**Error! Reference source not found.**) should be good estimations of the real value. As a comparison, it can be mentioned that the non-adjoint-weighted mean generation time[1] calculated by MCNP is approximately 370 µs.

[1] The (non-adjoint-weighted) neutron mean generation time is not given explicitly by MCNP. It must be calculated as the ratio between the prompt removal lifetime, l, and k_{eff}, or as the ratio between the prompt fission lifetime, τ, and the number of prompt neutrons per fission, v_p [9,10,11].

TABLE VI
Experimental Estimation of the Mean Generation Time, Λ.

	Λ/β_{eff} [10^{-3} s]	Λ [µs] ENDF/B-VI	JEFF3.0	JENDL3.3
Slope fit + Area				
EC1	22.1 ± 0.4	174 ± 4	175 ± 4	164 ± 4
EC2	20.3 ± 0.6	160 ± 5	161 ± 5	151 ± 5
EC3	19.5 ± 0.3	154 ± 3	155 ± 3	145 ± 3
EC5	22.1 ± 0.7	175 ± 6	175 ± 6	164 ± 6
EC6	22.2 ± 0.2	175 ± 3	176 ± 3	164 ± 2
Slope fit + Source jerk				
EC2	14.5 ± 1.1	114 ± 9	115 ± 9	108 ± 8

IV.D Validity of the Point Kinetic Model

Since all three methods used in this study are based on the point kinetic model, the validity of the model when applied to this system should be investigated. One basic assumption when deriving the point kinetic equations is that the solution can be divided in a time, energy and spatial part [11]. In other words, if the spatial one-energy group flux profile is constant in time, the assumption is valid. By using MCNP, the radial flux profile as a function of time after a neutron pulse has been simulated. Since the core relies mainly on thermal fissions, a one-energy group assumption should be well-founded. The simulation has shown that after approximately 4 ms after the pulse, the relative values of the flux profile do not change in time. At this point the transient mode (component) has relaxed and all points in the system are subject to the same time-dependence. This indicates that the point kinetic model is applicable.

V. DISCUSSION OF RESULTS

V.A Comparison between the Methods

The Slope fit method, the Area method and the Source jerk method have been applied to the same configuration of the Yalina experiments. In comparison with the MCNP calculations, the Slope fit method shows similar results, whereas the Area method and the Source jerk method underestimates and overestimates k_{eff}, respectively. Both the Area method and the Slope fit method produce results with low statistical errors. The Source jerk method, on the other hand, is connected with larger errors due to the large uncertainty in the lower neutron flux level. All results are summarized in Table VII and Figure 7.

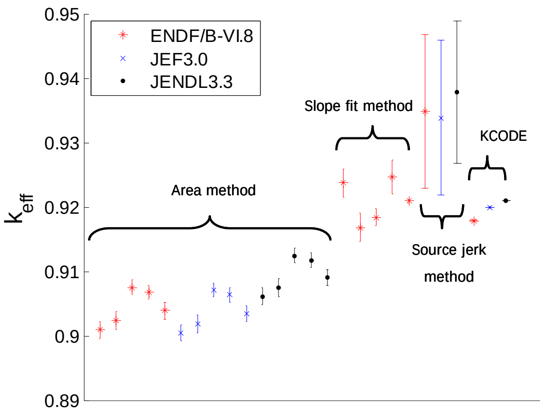

Figure 7. Summary of all k_{eff} obtained by different methods applied on data from 5 experimenta channels. The errorbars corresponds to one standard deviation in each direction. The different libraries refer to the chosen value of β_{eff}.

Comment [WG4]: X-axis must be somewhat labeled,,

TABLE VII
Summary of All k_{eff} Obtained Through Different Methods and for Different Values of β_{eff} from Different Libraries.

	k_{eff}		
	ENDF/B-VI	JEFF3.0	JENDL3.3
	Area method		
EC1	0.9012 ± 0.0013	0.9008 ± 0.0013	0.9065 ± 0.0013
EC2	0.9027 ± 0.0013	0.9022 ± 0.0013	0.9079 ± 0.0013
EC3	0.9076 ± 0.0011	0.9072 ± 0.0011	0.9126 ± 0.0011
EC5	0.9069 ± 0.0011	0.9064 ± 0.0011	0.9119 ± 0.0011
EC6	0.9040 ± 0.0013	0.9036 ± 0.0013	0.9091 ± 0.0012
	Slope fit method		
EC1	0.9239 ± 0.0022	-	-
EC2	0.9170 ± 0.0022	-	-
EC3	0.9185 ± 0.0013	-	-
EC5	0.9248 ± 0.0026	-	-
EC6	0.9211 ± 0.0002	-	-
	Source jerk method		
EC2	0.935 ± 0.012	0.934 ± 0.012	0.938 ± 0.011
	MCNP		
	0.91803 ± 0.00005	0.92010 ± 0.00007	0.92114 ± 0.00006

When applying the Slope fit method on deep subcritical configurations, as in the present case, it can sometimes be difficult to find the correct slope through a fitting procedure. The effects from the transient part have disappeared after about 4 ms, as mentioned in the previous section, and the delayed neutron background starts to influence the shape after approximately 7 ms (Figure 4). Consequently, the fundamental decay mode is the dominating mode during a relatively short time period. The situation is most problematic in the experimental channels in the core, where it is very difficult to distinguish a single slope. In the reflector, on the other hand, the fast exponentials have opposite sign, due to the fast increase in neutron flux during the first millisecond, which makes the fundamental decay mode evident. According to the results, the Area method has a tendency to give lower values of k_{eff} than other methods. Especially the difference between the Area method and the Slope fit method is worth to notice, since they are based on the same measurement. The same tendency was also observed in the MUSE-4 program [2,12].

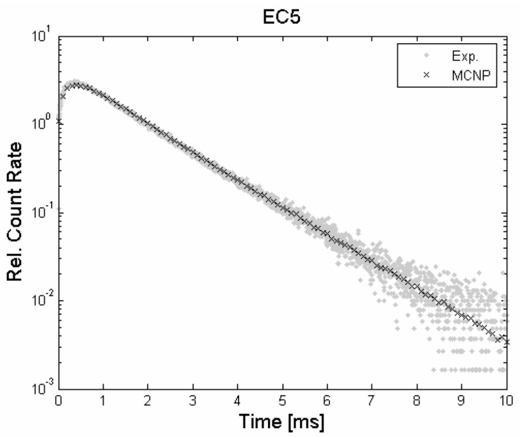

Figure 8. Comparison between PNS experiment and Monte Carlo simulation.

V.B Comparison between the PNS Experiment and MCNP

As mentioned before, no delayed neutron background is achieved when simulating the pulsed neutron source experiment in MCNP. In order to compare the simulation with the experiment, this background must be subtracted from the experimental data. The results are depicted in Figure 8 and indicate good agreement. It is possible to obtain α for the experimental data also in this way, but when substracting the background, an error is introduced which is hard to determine.

VI. CURRENT-TO-FLUX REACTIVITY INDICATOR

In the profound study of different measurement techniques for reactivity monitoring in an ADS, performed within the MUSE-4 program, the method based on the current-to-flux reactivity indicator has appeared to be the major candidate for on-line monitoring [13]. The ratio between the neutron flux, Φ, monitored in a detector in the core and the proton beam current, i_p, can be expressed by the relation;

$$\frac{\Phi}{i_p} \propto \frac{\psi^*}{\rho}, \qquad (8)$$

where ψ^* is the proton source efficiency [14] and represents the average importance of the external proton source, relative to the average importance of the fission neutrons in the fundamental mode. It is closely related to the neutron source efficiency [14,15] φ^*, which is frequently used in the ADS field. φ^* is commonly used in the physics of subcritical systems driven by any external source (spallation source, (D,D), (D,T), ^{252}Cf spontaneous fissions etc.). On the contrary, ψ^* has been defined exclusively for ADS studies, where the system is driven by a spallation source. ψ^* refers to the number of fission neutrons produced in the system for each source proton, $<\mathbf{F}\phi_s>/<S_p>$, and is defined in analogy with φ^*, according to

$$\psi^* = -\rho \frac{<\mathbf{F}\phi_s>}{<S_p>}. \qquad (9)$$

The brackets imply integration over space, angle and energy. Since $<\mathbf{F}\phi_s>$ is approximately proportional to the total power produced in the core, for a given k_{eff}, ψ^* thus relates the core power directly to the proton beam intensity.

According to Eq. (8), assuming that the reactor is operating at a constant reactivity, ψ^* is the proportionality factor between the monitored neutron flux and the proton beam current. By studying the dependence of ψ^* on different possible variations of the target-core properties, a good estimation of the stability of the current-to-flux reactivity indicator can be obtained. Possible transients that might affect ψ^* are, for instance, a change in the beam direction or the beam impact location, the proton energy or the target temperature. Over longer periods, the change in isotopic composition of the fuel due to burnup, or the modification of the core geometry during reloading, might change the source efficiency. In order to assure the reliability of the reactivity indicator, these factors, potentially able to affect the proton source efficiency, should be monitored continually or calibrated (??) on a regular basis.

VII. CONCLUSIONS

Three reactivity determination methods, the Slope fit method, the Area method and the Source jerk method, have been investigated by applying them to the Yalina experiments. Two types of experiments were performed at the facility; pulsed neutron source experiment (PNS) and source jerk experiment. The PNS experiment has also been simulated by MCNP. The simulations provided parameters, such as effective delayed neutron fraction and mean generation time, which were necessary for the evaluation of the reactivity and the effective multiplication constant from the experiments.

From the measurements it can be concluded that:

- The area method underestimates the criticality slightly in comparison with MCNP and the other methods, but gives low statistical error.
- The slope fit method is inconvenient to apply to deep subcritical configurations, but gives reliable results in comparison with MCNP. The cleanest re-

sponse functions are achieved in the reflector channels.
- The source jerk method is connected with large uncertainties.

Moreover, from the simulations it can be concluded that:
- MCNP gives reliable results, which was shown by the simulations in comparison with the experiments. However, care must be taken when calculating the time parameters.
- The point kinetic approximation describes the system well under the studied circumstances.

In this work, differences between different reactivity levels were not studied. Even if the methods do not predict the absolute value of the reactivity absolutely correct, they may be able to predict reactivity changes with good precision. None of these methods have the capability of measuring the reactivity without disturbing the running system. However, the methods can be used to estimate the subcriticality during loading and for calibration of other possible measurement techniques, such as the current-to-flux reactivity indicator.

Although the neutron spectrum of Yalina is thermal and has different kinetic parameters than in MUSE, the results show many similarities.

ACKNOWLEDGEMENTS

The authors would like to thank R. Klein Meulekamp for calculations of effective delayed neutron fractions and for useful discussions. Acknowledge SKB and SKI (full names) for financial support.

Comment [PS5]: Lägg till financial support. Fråga Waclaw vad du ska skriva.

REFERENCES

1. M. SALVATORES et al., "Long-lived, radioactive waste transmutation and the role of accelerator driven (hybrid) systems", *Nuclear Instruments and Methods in Physics Research*, A 414 5-20, 1998.

2. R. SOULE et al., "Neutronic Studies in Support of Accelerator-Driven Systems: The MUSE Experiments in the MASURCA Facility," *Nucl. Sci. Eng*., **148**, 124 (2004).

3. G. R. KEEPIN, *Physics of Nuclear Kinetics*, Addison-Wesley, 1965.

4. N. G. SJÖSTRAND, "Measurements on a subcritical reactor using a pulsed neutron source", Arkiv för Fysik, 11, 13 (1956).

5. S. E. CHIGRINOV and I. G. SERAFIMOVICH, *Experimental and Theoretical Research of Long-lived Fission Products and Minor Actinides in Subcritical Assembly Driven by a Neutron Generator*, Joint Institute of Power and Nuclear Research, National Academy of Sciences of Belarus, Minsk, Sosny, ISTC Project B-070-98. (Unpublished)

6. J.F BRIESMEISTER, editor, MCNP – A general Monte Carlo N-particle transport code, version 4c, LA-13709-M, Los Alamos National Laboratory, USA (2000).

7. S. C Van Der MARCK & R. KLEIN MEULEKAMP, "Calculating the Effective Delayed Neutron Fraction Using Monte Carlo Techniques", PHYSOR 2004, Chicago, Illinois, USA, April 2004.

8. F. JAMES & M. WINKLER, "MINUIT User's Guide", CERN, Geneva (2004).

9. G. D. SPRIGGS et al., "On the Definition of Neutron Lifetimes in Multiplying and Non-Multiplying Systems," LA-UR-97-1073, Brazilian Meeting on Reactor Physics and Thermo hydraulics, Pocos de Caldas Springs, MG, Brazil, 1997.

10. R. D. BUSCH et al., "Definition of Neutron Lifespan and Neutron Lifetime in MCNP4B," LA-UR-97-222, American Nuclear Society Meeting, Orlando, Florida, June 1997.

11. K. O. OTT and R. J. NEUHOLD, *Introductory Nuclear Reactor Dynamics*, American Nuclear Society (1985).

12. D. VILLAMARIN, "Análisis dinámico del reactor experimental de fisión nuclear MUSE-4", Doctoral Thesis, Universidad Complutense de Madrid, Departamento de Física Atómica, Molecular y Nuclear, 2004.

13. Internal MUSE document

14. P. SELTBORG et al., "Definition and Application of Proton Source Efficiency in Accelerator Driven Systems," *Nucl. Sci. Eng.*, **145**, 390 (2003).

15 G. ALIBERTI et al., "Analysis of the MUSE-3 Subcritical Experiment", Int. Conf. Global 2001, France, Paris, September (2001).