On Measurement and Monitoring of Reactivity in Subcritical Reactor Systems

CARL BERGLÖF

Doctoral Thesis in Physics
Stockholm, Sweden 2010
Akademisk avhandling som med tillstånd av Kungliga Tekniska högskolan framlägges till offentlig granskning för avläggande av teknologidoktorsexamen i fysik onsdagen den 12 maj 2010 klockan 14:00 i sal FA32, Albanova Universitetscentrum, Roslagstullsbacken 21, Stockholm.

© Carl Berglöf, april 2010

Abstract

Title: On Measurement and Monitoring of Reactivity in Subcritical Reactor Systems
Author: Carl Berglöf, KTH, School of Engineering Sciences, Department of Physics, Division of Reactor Physics
Language: English

Accelerator-driven systems have been proposed for incineration of transuranic elements from spent nuclear fuel. For safe operation of such facilities, a robust method for reactivity monitoring is required. Experience has shown that the performance of reactivity measurement methods in terms of accuracy and applicability is highly system dependent. Further investigations are needed to increase the knowledge data bank before applying the methods to an industrial facility and to achieve license to operate such a facility. In this thesis, two systems have been subject to investigation of various reactivity measurement methods. Conditions for successful utilization of the methods are presented, based on the experimental experience. In contrast to previous studies in this field, the reactivity has not only been determined, but also monitored based on the so called beam trip methodology which is applicable also to non-zero power systems. The results of this work constitute a part of the knowledge base for the definition of a validated online reactivity monitoring methodology for facilities currently being under development in Europe (XT-ADS and EFIT).

Keywords: ADS, reactivity monitoring, pulsed neutron source methods, neutron noise methods
Till Elin och Elsa
Kärnkraften har länge varit ett hett diskuterat ämne i Sverige och andra länder – inte minst på grund av det långlivade radioaktiva kärnavfallet. Att återanvända använt kärnbränsle och därigenom producera mindre mängder långlivat kärnavfall har varit ett aktivt forskningsområde i åtminstone 40 år. De låga uranpriserna de senaste årtiondena har dock ej drivit på denna utveckling i industriell skala och de flesta länderna återanvänder idag inte sitt använda kärnbränsle. Enligt de senaste prognoserna från flera organisationer såsom OECD står vi inför en kraftig ökning av den globala installerade kärnkraftskapaciteten. Dessutom har OECD uppskattat att den tekniskt och ekonomiskt brytbara mängden uran kommer att räcka som längst hundra år i nuvarande förbrukningstakt. Det är därför troligt att flera länder på sikt kommer att övergå till slutten kärnbränslecykel med återanvändning av använt kärnbränsle. Det kommer att kräva att nya så kallade snabbreaktorer av fjärde generationen tas i drift för att kunna förbränna plutonium och mindre aktinider, som utgör den främsta delen av radiotoxiciteten i det använda kärnbränslet på lång sikt. Det visar sig dock att de mindre aktiniderna, främst americium och curium, utgör ett särskilt problem när kärnbränsle återanvänds, trots att de utgör mindre än 0.1% av den totala avfallsmassan. Endast en begränsad mängd americium kan laddas i en snabbreaktor och mängden curium tenderar att ackumuleras med tiden. På grund av dessa isotopers höga radiotoxicitet skulle förtjänsten att minska slutförvaringstiden gå förlorad om de inte tas omhand på rätt sätt. För att effektivt förbränna även americium och curium har så kallade acceleratordrivna system föreslagits. Dessa system är underkritiska och är ej förknippade med samma begränsningar som snabbreaktorer när det gäller mängden americium som kan laddas i härden. Med acceleratordrivna system i kärnbränslecykeln kan de ämnen som ej tas om hand till follo i vanliga lättvattenreaktorer förbrännas effektivt. Nackdelarna är att de acceleratordrivna systemen måste drivas av en extern neutronkälla eftersom de är underkritiska, vilket gör dem något dyrare att ha i drift, samt att avancerade separations- och upparbetningsmetoder måste tillämpas. Underkriticiteten, marginalen till kritiskt tillstånd, är nödvändig ur säkerhetssynpunkt och kräver ständig övervakning eftersom den kan förändras med tiden. Ämnet för denna avhandling är hur denna säkerhetsmarginal kan mätas och övervakas före och under drift av ett acceleratordrivet
system. Förutom att mätningens kvalitet är viktig för systemets säkerhet medger en god kännedom om underkriticiteten att den kan väljas mindre konservativt, vilket i slutändan är viktigt för systemets ekonomi.

Olika mätmetoder har testats i det underkritiska reaktorexperimentet YALINA utanför Minsk i Vitryssland. I de två uppställningarna YALINA-Thermal och YALINA-Booster har en uppsättning av de bäst lämpade metoderna testats i två olika miljöer för att ge en utökad kändedom om dess tillämpbarhet i framtida acceleratordrivna system. Arbetet är ett led i att definiera och validera ett system för övervakning av underkriticitet för de acceleratordrivna system som är under utveckling i Europa för tillfället (XT-ADS och EFIT).
Ядерная энергетика – это уже давно активно обсуждаемая тема как в Швеции так и других странах не в последнюю очередь из-за долгоживущих радиоактивных отходов атомной промышленности. Переработка отработанного ядерного топлива и тем самым уменьшение производства радиоактивных отходов является активной областью научных исследований вот уже по крайней мере 40 лет. Низкие цены на уран в последние десятилетия не способствовали разветвлению этих технологий в промышленных масштабах, и поэтому большинство государств не перерабатывают отработанное ядерное топливо в настоящее время. Согласно последним прогнозам таких международных организаций как Организация Экономического Развития и Сотрудничества (ОЭСР) мы стоим на пороге значительного увеличения ядерного энергопроизводства в глобальном масштабе. Кроме того, по оценке ОЭСР запасов разведенного урана хватит максимум на 100 лет при современных темпах потребления. Поэтому вполне вероятно, что все больше стран в будущем перейдет к так называемому замкнутому топливному циклу с переработкой отработанного ядерного топлива. Это потребует введение в эксплуатацию так называемых быстрых реакторов 4-го поколения, способных сжигать плутоний и младшие актиниды, которые вносят наибольший вклад в радиотоксичность отработанного ядерного топлива. Однако младшие актиниды, и в первую очередь америций и кюрий, представляют собой значительную проблему при переработке отработанного ядерного топлива несмотря на то, что по массе они составляют менее 0.1% от общей массы отходов. Только ограниченное количество америция может быть загружено в быстрый реактор, а кюрий имеет тенденцию накапливаться с течением времени. По причине высокой радиотоксичности этих изотопов преимущество использования быстрых реакторов для уменьшения сроков хранения отработанного топлива будет сведено на нет. Для того чтобы эффективно сжигать даже америций и кюрий, были предложены системы, управляемые ускорителем. Эти системы являются подkritическими, и поэтому они не связаны такими же ограничениями, что и быстрые реакторы, когда речь идет о количестве америция, загружаемого в активную зону. Включенные в топливный цикл эти подkritические системы
способны эффективно сжигать те изотопы, которые накапливаются при использовании легководных реакторов. К недостаткам подкритических гибридных систем однако следует отнести наличие внешнего источника нейтронов, что делает эксплуатацию таких систем дороже, а также необходимость применять сложные методы обработки и разделения изотопов. Подкритичность, как запас безопасности, необходима с точки зрения безопасности эксплуатации, и требует постоянного контроля, поскольку этот параметр может меняться с течением времени. Темой настоящей диссертации является исследование методов оценки запаса безопасности, т.е. подкритичности, и способов ее измерения до и во время эксплуатации гибридных подкритических систем. В дополнение к точности измерения, которое важно само по себе с точки зрения безопасности, хорошее знание этого параметра позволит выбирать менее консервативный уровень подкритичности, что в конечном счете приведет к улучшенным экономическим показателям.

Различные методы измерения были протестированы в серии экспериментов на подкритическом стенде «Яліна» вблизи Минска в Республике Беларусь. Более подробно, в экспериментах были задействованы две подкритические установки: тепловая сборка (YALINA-Thermal) и быстро-тепловая (YALINA-Booster). На этих двух установках были проверены наиболее распространенные методы измерения подкритичности с целью оценить применимость этих методов в будущих гибридных системах. Данная работа является частью усилий по созданию и тестированию системы мониторинга подкритичности в гибридных реакторах, которые в настоящее время проектируются в Европе (XT-ADS и EFIT).
List of Publications

Included Papers

The following papers constitute the thesis:


1 The author changed his name from Carl-Magnus Persson to Carl Berglöf after Paper II.


**Author’s Contribution**

All calculations of **Paper I**, as well as the writing of the paper, were performed by the author. Concerning **Paper II**, the planning of the experimental program, the development of the data acquisition system, the Monte Carlo simulations, most of the data analysis and the writing of the paper were performed by the author. **Papers III-VII** were produced within the project EUROTRANS Domain 2 ECATS in which the author participated actively in terms of experiment planning, preparation and execution as well as analysis and interpretation of the experimental data. The analysis of the experimental data of **Paper III** was performed in parallel with the co-authors and the final paper was prepared by the author. The data analysis and writing of **Paper IV** were performed by the author solely. In **Paper V**, the author has contributed with the experimental part and assisted in the preparation of the final paper. The theoretical work was initiated based on observations made by the author. The main contributions of the author to **Papers VI-VII** consist, apart from what is already mentioned, of assistance in the analysis of the experimental data.

**Papers not Included**

The following papers are not included in the thesis:


Thesis Related Activities

Apart from the work resulting in the above listed papers the author has actively participated in the first phase of the IAEA coordinated research project on Analytical and Experimental Benchmark Analysis on Accelerator Driven Systems and its collaborative activity Low Enriched Uranium Fuel Utilization in Accelerator Driven Subcritical Assembly Systems. This work resulted in the YALINA-Booster benchmark specification [28].

In the European research programme for the transmutation of high level nuclear waste in an accelerator-driven system (EUROTRANS) the author has participated actively in the subdomain Experimental activities on the coupling of an accelerator, a spallation target and a subcritical blanket (ECATS) with all work packages related to YALINA-Booster.
Contents

Swedish Summary 7
Russian Summary 9
List of Publications 11
Contents 15

The Issue of Irradiated Nuclear Fuel 17
1.1 Introduction 17
1.2 Irradiated nuclear fuel and radiotoxicity 18
1.3 Dedicated burners 20
1.4 The role of ADS in a closed fuel cycle 24
1.5 Reactivity assessment 25

Basic Concepts in Neutron Transport 27
2.1 The neutron transport equation 28
2.2 The point reactor model 29
2.3 Stochastic transport theory and branching processes 30

The YALINA Experiments 33
3.1 YALINA-Thermal 33
3.2 YALINA-Booster 34
3.3 The neutron source 35
3.4 Detectors and data acquisition 36

The Monte Carlo Simulation Tool 37
4.1 Code and nuclear data libraries 37
4.2 Uncertainties 37

Reactivity Measurements 39
5.1 The Sjöstrand area ratio method 39
5.2 The prompt decay fitting method 42
5.3 The source jerk and beam trip methods 45
5.4 The Rossi-\(\alpha\) method 49
5.5 The pulsed Rossi-\(\alpha\) method 51
5.6 The Feynman-\(\alpha\) method 53
Discussion and Conclusions
6.1 Concluding remarks 59
6.2 Transposition to fast systems 62
6.3 Possible application scenarios 63
6.4 General conclusions 66
Bibliography 67
Acknowledgements 71
Chapter 1

The Issue of Irradiated Nuclear Fuel

1.1 Introduction

It is widely understood that the radiotoxicity and required long-term disposal of irradiated nuclear fuel is a main disadvantage of nuclear power compared to other energy sources. Addressing this issue has been a major research field for at least four decades. However, the once-through fuel cycle is still the most common fuel cycle approach in a vast majority of the countries utilizing nuclear power, although this strategy leaves most of the fuel energy content behind.

Making predictions of future global energy needs is always difficult and is associated with large uncertainties. Nevertheless, according to the latest forecast from OECD the installed nuclear capacity might increase by 8-66% by 2030 and by 55-280% by 2050, which is in accordance with other similar studies [1]. If also taking into account that the identified economically recoverable uranium resources will last about 100 years with current consumption rate [2], a transition towards nuclear fuel recycling is likely to occur. This transition will engage fast spectrum breeder reactors to take advantage of the fertile material and burner reactors to enhance nuclear waste volume and radiotoxicity reduction. In other words, the benefits can be found on both the front end and back end of the fuel cycle. The transmutation will not only reduce the volume of the nuclear waste and thereby reduce the required size of the final repository, but also reduce the waste heat load and allow a more dense waste loading strategy.

Burning of plutonium and minor actinides, present in the irradiated nuclear fuel, must take place in dedicated reactors due to safety constrains. The fraction of dedicated burners required in the reactor fleet depends heavily on the upper limit of the allowed americium content to be loaded in the core. Critical fast reactors undergo instability problems when the americium level is too high [3]. When considering transmutation of the waste stock-pile in addition to the fuel cycle waste stream, accelerator-driven systems (ADS) have been proposed as dedicated burners [4,5]. Since such systems are subcritical, the safety constrains are relieved and higher americium fractions can be loaded.
The subcriticality of the ADS is subject to sudden, in case of an accident, and long-term variations due to for instance fuel burnup. Thus, the subcriticality (or reactivity) must be monitored to guarantee that a critical or supercritical configuration, potentially leading to instabilities, does not occur. As shown in this thesis, measuring and monitoring the reactivity of a subcritical reactor system is not a straightforward task. Reactors are heterogeneous constructions, whereas the theory describing them is better applied to homogeneous systems. Moreover, the most applied model, namely the point kinetics model, is limited to one energy group. The discrepancies are often manageable, but can also be large. Consequently, care must be taken when interpreting experimental data with such a simplified model.

In this thesis, several methods for measuring and monitoring the reactivity are tested on experimental data from two subcritical reactor systems: YALINA-Thermal and YALINA-Booster. Through the experiments, the feasibility of the methods in terms of stability, space dependence and accuracy is verified. First, the components of irradiated nuclear fuel are described followed by proposed burners. That leads to accelerator-driven systems and the need of a method to monitor the reactivity. In Chapter 2 the underlying theoretical framework is described followed by a description of the experimental facilities in Chapter 3. The simulation tool is briefly described in Chapter 4, followed by the experimental results in Chapter 5. Finally, the conclusions are discussed in Chapter 6.

1.2 Irradiated nuclear fuel and radiotoxicity

In light-water reactors (LWR), most of the energy comes from fission of $^{235}\text{U}$ which is thereby consumed during the irradiation cycle. At the same time, various isotopes of primarily plutonium, americium and curium are generated through subsequent neutron captures in $^{238}\text{U}$. Due to this process, the nuclear fuel after irradiation consists of, besides fission products and uranium leftovers, a non-negligible amount of transuranium elements (TRU). As can be seen in Table 1.1, the irradiated nuclear fuel consists of almost 95% of unused uranium, 1% of TRU and 4% of fission products. Apparently, 96% of the material can in principle be fissioned and can thus be used for energy production if there are enough incentives to develop the technology for doing so.
Table 1.1. Composition of UOX-fuel (uranium oxide) with 3.7% initial enrichment, burnt to 41.2 GWd/tHM\(^1\), after four years of cooling [6]. The half-life and the effective ingestion dose coefficient [7] for each nuclide are also given.

<table>
<thead>
<tr>
<th>Element or Nuclide</th>
<th>Relative mass [%]</th>
<th>Half-life [a]</th>
<th>Effective dose coefficient [nSv/Bq]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Uranium</strong></td>
<td>94.6</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{235}\text{U})</td>
<td>0.8</td>
<td>7.04·10(^8)</td>
<td>47</td>
</tr>
<tr>
<td>(^{236}\text{U})</td>
<td>0.6</td>
<td>2.34·10(^7)</td>
<td>47</td>
</tr>
<tr>
<td>(^{238}\text{U})</td>
<td>98.6</td>
<td>4.47·10(^9)</td>
<td>45</td>
</tr>
<tr>
<td><strong>Neptunium</strong></td>
<td>0.06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{237}\text{Np})</td>
<td>100</td>
<td>2.14·10(^6)</td>
<td>110</td>
</tr>
<tr>
<td><strong>Plutonium</strong></td>
<td>1.1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{238}\text{Pu})</td>
<td>2.5</td>
<td>87.7</td>
<td>230</td>
</tr>
<tr>
<td>(^{239}\text{Pu})</td>
<td>54.2</td>
<td>2.41·10(^4)</td>
<td>250</td>
</tr>
<tr>
<td>(^{240}\text{Pu})</td>
<td>23.8</td>
<td>6.56·10(^3)</td>
<td>250</td>
</tr>
<tr>
<td>(^{241}\text{Pu})</td>
<td>12.6</td>
<td>14.4</td>
<td>4.8</td>
</tr>
<tr>
<td>(^{242}\text{Pu})</td>
<td>6.8</td>
<td>3.75·10(^5)</td>
<td>240</td>
</tr>
<tr>
<td><strong>Americium</strong></td>
<td>0.05</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{241}\text{Am})</td>
<td>63.8</td>
<td>432</td>
<td>200</td>
</tr>
<tr>
<td>(^{242m}\text{Am})</td>
<td>0.2</td>
<td>141</td>
<td>190</td>
</tr>
<tr>
<td>(^{242}\text{Am})</td>
<td>36.0</td>
<td>7.37·10(^3)</td>
<td>200</td>
</tr>
<tr>
<td><strong>Curium</strong></td>
<td>0.01</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{243}\text{Cm})</td>
<td>1.0</td>
<td>29.1</td>
<td>150</td>
</tr>
<tr>
<td>(^{244}\text{Cm})</td>
<td>92.2</td>
<td>18.1</td>
<td>120</td>
</tr>
<tr>
<td>(^{245}\text{Cm})</td>
<td>5.7</td>
<td>8.50·10(^3)</td>
<td>210</td>
</tr>
<tr>
<td>(^{246}\text{Cm})</td>
<td>1.1</td>
<td>4.76·10(^3)</td>
<td>210</td>
</tr>
<tr>
<td><strong>Fission Products</strong></td>
<td>4.2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(^{79}\text{Se})</td>
<td>0.01</td>
<td>6.5·10(^5)</td>
<td>2.9</td>
</tr>
<tr>
<td>(^{93}\text{Zr})</td>
<td>2.06</td>
<td>1.53·10(^6)</td>
<td>1.1</td>
</tr>
<tr>
<td>(^{99m}\text{F})</td>
<td>2.37</td>
<td>2.11·10(^5)</td>
<td>0.64</td>
</tr>
<tr>
<td>(^{107}\text{Pd})</td>
<td>0.64</td>
<td>6.50·10(^6)</td>
<td>0.037</td>
</tr>
<tr>
<td>(^{126}\text{Sn})</td>
<td>0.07</td>
<td>(\sim 1\cdot 10^5)</td>
<td>4.7</td>
</tr>
<tr>
<td>(^{129}\text{I})</td>
<td>0.50</td>
<td>1.57·10(^7)</td>
<td>110</td>
</tr>
<tr>
<td>(^{135}\text{Cs})</td>
<td>1.09</td>
<td>2.30·10(^6)</td>
<td>2</td>
</tr>
<tr>
<td>Other</td>
<td>93.25</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

\(^1\) Gigawatt-days per metric ton of heavy metal

Different radio nuclides affect human tissue in different ways. The so-called radiotoxicity of a nuclide depends on the type of the exposed tissue and time of exposure. In Table 1.1, the effective ingestion dose coefficients of the nuclides are shown. As can be seen, the dose coefficients of the actinides are generally much higher than those of the long-lived fission products. Figure 1.1 shows the radiotoxic inventory of the spent fuel of Table 1.1 as a function of time, in relation to the radiotoxicity of natural uranium. Concerning the fission products, some short-lived nuclides, such as \(^{90}\text{Sr}\) and \(^{137}\text{Cs}\), have been included since they are main contributors to the radiotoxic inventory in the short term. These fission products have very high activity during the first hundreds years and
constitute the main part of the radiotoxic inventory during this period. When the activity of the fission products have declined, the radiotoxicity is dominated by americium and, after some thousands of years, plutonium becomes the main contributor. Clearly, the long-term radiotoxic issue is caused by plutonium and americium, although those elements constitute only slightly more than one percent of the irradiated fuel. Neptunium is less troublesome since it stays below the reference value. Curium, on the other hand, decays rather quickly to low levels, but would further irradiation of plutonium and americium be considered, the curium production will be an issue that must be taken into account. In an ideal recycling scenario, all plutonium, americium and curium will be transmuted resulting in a curve for the total radiotoxicity following that of the fission products. Consequently, in such scenario, it would be possible to reduce the final disposal time of the spent nuclear fuel from hundred thousands of years to thousands of years.

![Graph showing radiotoxicity compared to natural U](image)

Figure 1.1. Radiotoxic inventory of UOX-fuel of 3.7% initial enrichment after a burnup of 41.2 GWd/tHM, normalized to the radiotoxicity of the amount of natural uranium needed to fabricate the enriched fuel (approximately 7-8 tons per ton 3.7% enriched fuel, depending on the amount of $^{235}$U left in the tail).

1.3 Dedicated burners

Conventional light-water reactors can apparently not burn all the fissionable nuclides present in the fuel. Mainly $^{235}$U and other nuclides with an odd neutron number ($N$),
such as $^{239}$Pu and $^{242}$Am, will be fissioned. This effect comes from the fact that even-N nuclides are in energetically favored states and have therefore low probability for absorption of neutrons. Neutron absorption results in one of the two possibilities fission and capture. The fission probability is consequently defined as

$$ p_f = \frac{\sigma_f}{\sigma_f + \sigma_c}, $$

where $\sigma_f$ and $\sigma_c$ are the microscopic cross section for fission and capture respectively. The fission probability is depicted in Figure 1.2 as a function of incident neutron energy for an even-N nuclide and an odd-N nuclide. The difference between the two types of nuclide can clearly be seen. The fission probability of the odd-N nuclide $^{235}$U is high for all energies, whereas the even-N nuclide $^{241}$Am is fissioned mainly for high energy neutrons above approximately 1 MeV. Therefore, in order to favor fission over capture for all nuclides in the spent fuel and thereby reach high transmutation rate of plutonium and minor actinides and decreased radiotoxicity, the neutron spectrum must be hard. In addition, a hard spectrum is required in order not to build up $^{252}$Cf through neutron captures and subsequent beta decays [8]. The presence of $^{252}$Cf in the irradiated fuel would make the reprocessing cumbersome due to the strong neutron radiation from spontaneous fission events.

![Figure 1.2. Fission probability as a function of incident neutron energy in $^{235}$U (odd-N) and $^{241}$Am (even-N) (data from ENDF/B-VI [9]).](image)

### 1.3.1 Recycling in existing light-water reactors

Recycling of plutonium is already taking place in thermal light-water reactors through so called MOX fuel utilization. By combining plutonium from the LWR waste stream and depleted uranium, a mixed oxide fuel (MOX) is fabricated. The main advantage of this fuel is that the amount of Pu, mainly $^{239}$Pu, can be reduced. However, since the spec-
trum is thermal, americium and curium will be built up through neutron captures in the plutonium and, in the end, increasing the burden of the radiotoxic inventory [6].

It has been proposed to use hafnium cladded minor actinide fuels in boiling-water reactors for transmutation [10]. The hafnium would absorb the thermal part of the neutron flux and thus create a spectrum hardening inside the fuel pin. The study is, however, only a feasibility study, but indicates considerable performance and could be one option in the future.

1.3.2 Fast reactors and associated safety issues

In general, the requirement on hard neutron spectrum rules out conventional use of light-water reactors as efficient TRU burners due to the moderation effect of water. Fast reactors, cooled by for instance sodium, lead or helium, are candidates for TRU burning, but when loading a critical fast core with minor actinides and plutonium instead of uranium, some safety aspects of the core will change drastically. First of all, the Doppler feedback of uranium-free fuels is lower since the effect is most pronounced in $^{238}$U. In addition, the high capture cross section of $^{241}$Am in the range of 100 eV to 100 keV reduces further the Doppler feedback since neutrons tend to get captured in the $^{241}$Am instead of in resonance rich nuclides. For instance, studies have shown that in fast reactors partially loaded with uranium, even a small fraction of americium in sodium- or helium-cooled cores decreases the Doppler feedback significantly [11,12].

Another concern appearing when loading a fast core with large amounts of TRU is related to the low delayed neutron fraction. The delayed neutron fraction is defined as the number of delayed neutrons produced per fission per total number of neutrons produced per fission:

$$\beta(E) = \frac{v_d(E)}{v_p(E) + v_d(E)} = \frac{v_d(E)}{v(E)}.$$  \hspace{1cm} (1.2)

Its value is dependent both on the fissioning nuclide and the incoming neutron energy, as indicated in the above equation. In general it can be stated that the value of the delayed neutron fraction increases with the mass number A for a certain element (constant Z) and decreases for increasing Z (heavier elements). The delayed neutron fraction of $^{235}$U, $^{239}$Pu and $^{241}$Am is depicted in Figure 1.3 as a function of incident neutron energy. It can be noted that the delayed neutron data is associated with much lower accuracy for the higher actinides, such as $^{241}$Am, compared to $^{235}$U and $^{239}$Pu, expressed through both lower energy resolution and larger spread among the different libraries. The delayed neutron fraction is in general constant up to approximately 5 MeV, where it drops to less than 50%. This should, however, not be a major concern for fast systems. The issue is rather related to the low delayed neutron fraction associated to those nuclides where most fissions occur in a uranium-free fuel, mainly $^{239}$Pu.

The energy of the delayed neutrons depends on the precursor from which the neutron originates. In general, this energy is in the order of some hundred keV and is, consequently, less than the average energy of the fission spectrum (1-2 MeV). Therefore, in a thermal system, the delayed neutrons are more efficient in inducing further fissions than the fission neutrons since they have less probability for absorption in $^{238}$U during slow-
ing-down. Taking this efficiency into account, the effective delayed neutron fraction, $\beta_{\text{eff}}$, has been introduced. In general, for a thermal nuclear reactor $\beta_{\text{eff}} > \beta$, and for a fast reactor $\beta_{\text{eff}} \leq \beta$.

![Delayed neutron fractions for $^{235}$U, $^{239}$Pu and $^{241}$Am in the unit of pcm (percent milles, $10^{-5}$) as given by three different nuclear data libraries [9].](image)

**Figure 1.3.** Delayed neutron fractions for $^{235}$U, $^{239}$Pu and $^{241}$Am in the unit of pcm (per cent milles, $10^{-5}$) as given by three different nuclear data libraries [9].

Another effect is related to the coolant reactivity coefficient. In fast reactors, moderation occurs to some extent and causes a slight softening of the neutron spectrum. In a core loaded with a large fraction of even-N nuclides, such as $^{240}$Pu, $^{242}$Pu, $^{241}$Am and $^{243}$Am, this becomes troublesome. As could be seen in Figure 1.2, the fission probability of even-N nuclides increases with incident neutron energy. If there is a neutron spectrum hardening, caused by for instance sudden coolant boiling, the fission rate will increase causing a power increase. Such positive feedback mechanisms must be avoided, thus leading to a limitation to the fraction of even-N nuclides that may be loaded into the core.

The positive feedback can in some designs be balanced by relying on increased neutron leakage due to the lower coolant density and changed lattice geometry due to the temperature increase. A positive coolant temperature coefficient is allowed if it is compensated by the Doppler feedback from the fuel.

In case of total voiding of the core, the situation is more severe. The neutron spectrum becomes very hard and an excess of neutrons will suddenly be available due to the strongly decreased absorption. In fast reactors, this implies a cumbersome situation that may cause an overall reactivity insertion of several dollars (units of $\beta_{\text{eff}}$) [13].
1.3.3 Subcritical systems

In previous sections it has been concluded that fast neutron systems must be employed for efficient transmutation of plutonium and minor actinides and for radiotoxicity reduction. However, fast reactors suffer from deteriorated safety parameters when loaded with large fractions of these elements, in particular $^{241}\text{Am}$. Therefore, it has been proposed to employ fast subcritical source-driven cores as burners of uranium-free fuels [4,5]. The subcriticality makes the core less sensitive to positive reactivity feedbacks, thereby allowing the use of fuels based on large fractions of plutonium, americium and curium. The margin to criticality must be chosen large enough to withstand any reactivity increase that can make the core critical. On the other hand, a large subcriticality level requires a strong source. An effective multiplication factor of approximately 0.97 is foreseen in full-scale designs [14]. However, core voiding may still be a concern even at this subcriticality level, in particular for sodium cooled systems [13].

A constant power level will be retained by coupling a strong external neutron source to the subcritical core. This source will most likely consist of a proton accelerator coupled to a spallation target. Therefore these systems are referred to as accelerator-driven systems (ADS).

1.4 The role of ADS in a closed fuel cycle

Recycling of irradiated nuclear fuel and thereby closing the fuel cycle is the unavoidable path towards a sustainable nuclear development with respect to fuel utilization and waste management. There are several possible fuel cycle scenarios proposed, but the most promising one including ADS is the so called Double Strata fuel cycle [6]. Since traditional light-water reactors are capable of generating electricity to a relatively low cost they will most likely continue their operation at least during their expected life-time. Plutonium recycling with MOX fuel will be implemented to some of these reactors. In parallel to the LWR fleet, a second stratum is implemented that takes care of the remaining plutonium and minor actinides from the LWRs. This stratum will consist of fast reactors that are loaded with MOX fuel. The minor actinides from the LWRs and the fast reactors are, after reprocessing, loaded to an ADS for incineration. In this way, the existing LWR reactors are left as undisturbed as possible and an optimized closure of the fuel cycle is implemented gradually in the second stratum. In the long term, the LWR will be phased out and the fast reactors will become more and more dominant in the fuel cycle. Finally, when all LWRs are decommissioned and the minor actinide stockpile from historical LWR operation has been incinerated, the ADS can be phased out and the fuel cycle will consist of a pure fast reactor fleet with associated fuel reprocessing plants. In that sense, the double strata strategy may be viewed as a transition fuel cycle.
1.5 Reactivity assessment

When operating an ADS loaded by plutonium and minor actinides, criticality must be avoided under all circumstances. Therefore, monitoring of the subcriticality is essential for maintaining safe operation.

Measurements of subcriticality in reactors have been performed since the fifties [15,16]. However, the interest of subcritical systems for irradiated nuclear fuel incineration has increased the need of a stable and reliable subcriticality determination method. Since important parts of the safety system of a future ADS will be based on this designated method, new requirements on the performance are raised, the most important being:

- Capability of online monitoring of the reactivity, i.e. short measurement time.
- Low spatial dependence.
- High accuracy.
- Detector type independence.
- Neutron source independence.

One major study in this field has been performed at the MASURCA facility in Cadarache, France, within the MUSE project [17]. In that study, a number of reactivity methods were investigated and compared to each other in a fast neutron spectrum at zero power. Some further studies were performed within the TRADE [18] and the RACE [19] programs. These studies aimed at higher power levels to include thermal feedbacks in a thermal neutron spectrum, but both projects were cancelled at early stages. In the YALINA experiments in Belarus, studies on reactivity determination have been performed in a thermal and a coupled fast-thermal spectrum. There are two subcritical cores, here referred to as YALINA-Thermal and YALINA-Booster that are coupled to a neutron generator. Paper I deals with experimental results from YALINA-Thermal, whereas Paper II-VII concerns YALINA-Booster.
Chapter 2

Basic Concepts in Neutron Transport

The ultimate goal in reactor physics studies is to determine the neutron distribution in space, energy and time in the reactor. This is achieved by describing the motion of neutrons in the reactor and their interactions with the present materials. The most fundamental quantity in nuclear reactor theory is the neutron density as a function of space, energy and time:

\[ n(r, E, t). \]

The expected number of neutrons of energy \( dE \) around \( E \) in an infinitesimally small volume \( d^3r \) at time \( t \) around \( r \) is \( n(r, E, t)d^3r dE. \)

Another important quantity in reactor theory is the reaction rate density \( F_x(r, E, t) \), where \( x \) denotes symbolically the occurring interaction. From this quantity, the corresponding macroscopic cross section, \( \Sigma_x \), can be introduced:

\[ F_x(r, E, t) = \Sigma_x(r, E) \cdot \varphi_n(r, E, t), \quad (2.1) \]

where \( \varphi \) is the neutron speed.

If having \( n(r, E, t) \), a complete picture of the neutron density distribution in the reactor is known. Unfortunately, no equation is satisfied exactly by \( n(r, E, t) \). Therefore, the angular neutron density, \( N(r, \Omega, E, t) \), must be introduced. The parameter \( \Omega = \frac{\Omega}{\varphi} \) describes the direction of a neutron leaving the position \( r \). Consequently, \( N(r, \Omega, E, t)d^3r d\Omega dE \) is the expected number of neutrons in the volume \( d^3r \) about \( r \), with energy \( dE \) about \( E \), moving in direction \( \Omega \) in solid angle \( d\Omega \) at time \( t \). The products \( \varphi n(r, E, t) \) and \( \varphi N(r, \Omega, E, t) \) occur very frequently in reactor theory and they have therefore been given special names. Thus we introduce the neutron flux

\[ \phi(r, E, t) \equiv \varphi n(r, E, t), \quad (2.2) \]

and the angular neutron flux

\[ \Phi(r, \Omega, E, t) \equiv \varphi N(r, \Omega, E, t). \quad (2.3) \]
The neutron flux and the angular neutron flux are related through

\[ \phi(r, E, t) = \int_{4\pi} \Phi(r, \Omega, E, t) d\Omega. \]  

(2.4)

### 2.1 The neutron transport equation

The distribution of neutrons in a reactor obeys the neutron transport equation, which can be classified as a linear Boltzmann equation:

\[ \frac{1}{v} \frac{\partial \Phi(r, \Omega, E, t)}{\partial t} = -\Omega \cdot \nabla \Phi(r, \Omega, E, t) - \Sigma_a(r, E) \Phi(r, \Omega, E, t) + \int_{0}^{\infty} \int_{4\pi} \Sigma_i(r, \Omega' \rightarrow \Omega, E' \rightarrow E) \Phi(r, \Omega', E', t) d\Omega' dE' \]

\[ + \int_{0}^{\infty} \frac{\chi(E)}{4\pi} \int_{0}^{\infty} \frac{[1 - \beta(E')] v(E') \Sigma_f(r, E') \Phi(r, \Omega', E', t) d\Omega' dE'}{4\pi} \]

\[ + \sum_{i} \int_{0}^{\infty} \frac{\chi_i(E)}{4\pi} \lambda_i C_i(r, t) + S(r, \Omega, E, t) \].  

(2.5)

Here, \( \Sigma_a \) is the macroscopic absorption cross section, \( \Sigma_f \) is the macroscopic fission cross section, \( \chi \) and \( \chi_f \) are the energy spectrum of the prompt and delayed neutrons respectively; \( \Sigma_s \) is the differential macroscopic scattering cross section describing the transfer probability that an incident neutron of initial direction \( \Omega' \) and energy \( E' \) emerges from a possible collision with direction \( \Omega \) and energy \( E \). The delayed neutron precursor density is represented by \( C_i \) with decay constants \( \lambda_i \). Finally, the external source is here given as \( S \). The delayed neutron precursor densities follow the relation

\[ \frac{\partial C_i(r, t)}{\partial t} = \int_{0}^{\infty} \int_{4\pi} \beta_i(E') v \Sigma_f(r, E') \Phi(r, \Omega', E', t) d\Omega' dE' - \lambda_i C_i(r, t), \]  

(2.6)

where \( \beta_i \) is the delayed neutron fraction for the delayed neutron precursor group \( i \) [20].

The two loss terms of Eq. (2.5) come from streaming and neutron absorption, whereas the gain terms arrive from scattering, fissions, delayed neutron precursors and from the external source.

The neutron transport equation without delayed neutrons and external source takes a homogeneous form and can be written in the following way:

\[ \frac{\partial \Phi(r, \Omega, E, t)}{\partial t} = L \Phi(r, \Omega, E, t). \]  

(2.7)

Here, \( L \) is a differential or integro-differential operator depending on the approximation or solution strategy used; for instance diffusion or the neutron transport equation.
essential assumption is that the operator $L$ does not depend on time. One of the common methods to solve this equation is separation of variables as

$$\Phi(r, \Omega, E, t) = \psi(r, \Omega, E)e^{\alpha t}, \quad (2.8)$$

where $\psi$ is referred to as the shape function. It obeys a typical eigenvalue equation:

$$\alpha \psi = L \psi, \quad (2.9)$$

where $\alpha$ is an eigenvalue of the system. There might be countable or uncountable many eigenvalues satisfying the eigenfunction, yielding a set of solutions if assuming discrete spectrum:

$$\alpha_n \psi_n = L \psi_n, \quad n = 0, 1, 2... \quad (2.10)$$

The total flux is therefore a sum of all eigenfunctions

$$\Phi(r, \Omega, E, t) = \sum_{n=0}^{\infty} c_n e^{\alpha_n t} \psi_n(r, \Omega, E), \quad (2.11)$$

where $c_n$ is the amplitude of each alpha-mode. This formulation of the neutron transport equation is convenient when treating spatial dependence.

### 2.2 The point reactor model

Eqs. (2.5) and (2.6) are very difficult to solve, hence a number of simplifying models and assumptions have been proposed. Several common models are as follows:

- multi-group model postulating that the energy $E$ may assume only a discrete number of energy levels,
- one-group model characterized by a single neutron energy and energy independent cross sections,
- diffusion model postulating Fick’s law between the neutron current and the neutron flux, and
- point reactor model assuming separation of space and time variables according to

$$\phi(r, t) = \nu n(t) \psi(r) \quad (2.12)$$

and

$$C_i(r, t) = c_i(t) \psi(r). \quad (2.13)$$

This is also valid for a homogeneous infinite reactor where the shape function is constant.

From now on, the point reactor model will be assumed. One can then derive the following equations [20-22]:

\[
\frac{dn(t)}{dt} = \frac{\rho(t) - \beta_{\text{eff}}}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i c_i(t), \quad (2.14)
\]

\[
\frac{dc_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i c_i(t), \quad i = 1, \ldots, 6
\]

where the neutron density, \( n \), and the delayed neutron precursor densities, \( c_i \), have been introduced through averaging over space, energy and solid angle. The reactivity, \( \rho \), is defined by

\[
\rho = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}} \quad (2.15)
\]

where \( k_{\text{eff}} \) is the ratio of neutron production to neutron absorption. Further, the parameter \( \Lambda \), the neutron reproduction time\(^2\), has been introduced, describing the inverse production rate of neutrons in the system. In Eqs. (2.14), the space dependence has been removed, which means that all points in the reactor are described by the same equations, thereby carrying the name the point kinetic equations. Despite the simplicity of these equations, they are in many cases sufficient to describe the reactor behavior in a satisfactory way but are sometimes inappropriate to use, particularly in large loosely coupled systems. The use of the point kinetic approximation in subcritical systems has been questioned due to the different characteristics of the neutron balance equation. The critical problem is a homogeneous problem for which the perturbation theory of an equilibrium system is well defined, whereas the subcritical problem is inhomogeneous [23]. Nevertheless, it has been shown that in fast subcritical systems characterized by a strong space-time coupling caused by a relatively small core size and long mean free path, the point kinetics approximation performs well compared to a full space-time solution [24].

All reactivity determination methods included in this study can be derived from the point kinetics equations. The derivation of each method is given together with the experimental results later. The methods are:

- the Sjöstrand area ratio method,
- the prompt decay (or slope) fitting technique,
- the source jerk method and the related beam trip method,
- the Rossi-\( \alpha \) method and
- the Feynman-\( \alpha \) method.

### 2.3 Stochastic transport theory and branching processes

Another approach in solving the one-energy group point reactor approximation of the Boltzmann equation is to regard the problem as a stochastic branching process. Let

\[^2\] The traditional name of \( \Lambda \) is the mean neutron generation time [25]. However, it has been proposed to rename this parameter the neutron reproduction time [26].
$P(N,C,Z,t|t_0)$ denote the joint probability of having exactly $N$ neutrons, $C$ delayed neutron precursors and $Z$ detections in a subcritical system at time $t$ given there was one source neutron emitted at $t_0$. Further, assign $\lambda_c$ the rate of neutron captures, $\lambda_f$ the rate of fissions, $\lambda_d$ the rate of neutron detections, $S$ the neutron source strength, $\lambda$ the one delayed neutron group decay rate and $p_f(n,m)$ the probability of emitting $n$ neutrons and $m$ delayed neutron precursors in a fission event. Then, the forward master equation for $P$ is given by [27]

$$\frac{dP(N,C,Z,t|t_0)}{dt} = \lambda_c P(N+1,C,Z,t|t_0)(N+1) + \lambda_d P(N+1,C,Z-1,t|t_0)(N+1) + \lambda_f \sum_{n} \sum_{m} P(N+1-n,C-m,Z,t|t_0)(N+1-n)p_f(n,m) + SP(N-1,C,Z,t|t_0) + \lambda P(N-1,C+1,Z,t|t_0)(C+1) - P(N,C,Z,t|t_0)\left[\lambda_f + \lambda_c + \lambda_d + \lambda C + S\right]$$

with initial condition

$$P(N,C,Z,t_0|t_0) = \delta_{N,0}\delta_{C,0}\delta_{Z,0}.$$  

The second term of the right hand side of Eq. (2.16), indicates that one neutron is lost in the detection process. However, when using fission chambers for neutron detection, neutrons are, in addition, generated in the detection process thus leading to the alternative second term

$$\lambda_d \sum_{n} \sum_{m} P(N+1-n,C-m,Z-1,t|t_0)(N+1-n)p_d(n,m),$$

where $p_d(n,m)$ denotes the probability of the active fission chamber deposit to emit $n$ neutrons and $m$ delayed neutron precursors upon a neutron detection. This leads to a second order effect that is usually not treated in the derivations and will be left aside here as well.

For later application, it is desirable to find the first moments of $Z$, namely the mean value and the variance of the detected neutrons. Therefore, the so called generating functions

$$G(x,y,v,t|t_0) = \sum_{N} \sum_{C} \sum_{Z} x^N y^C v^Z P(N,C,Z,t|t_0)$$

and

$$g_f(x,y) = \sum_{n} \sum_{m} x^n y^m p_f(n,m)$$

are defined [27]. After applying the generating functions above to Eq. (2.16) the derivatives of $G$ can be found. This is of particular interest since the generating functions are defined so that
\[
\frac{\partial G(x, y, v, t \mid t_0)}{\partial x} \bigg|_{x=y=z=1} = \langle N(t) \rangle \tag{2.21}
\]
\[
\frac{\partial G(x, y, v, t \mid t_0)}{\partial y} \bigg|_{x=y=z=1} = \langle C(t) \rangle .
\]
\[
\frac{\partial G(x, y, v, t \mid t_0)}{\partial v} \bigg|_{x=y=z=1} = \langle Z(t) \rangle
\]

Solving for these derivatives in Eq. (2.16) gives directly the point kinetics equations. Further, another useful relation is
\[
\frac{\partial^2 G(x, y, v, t \mid t_0)}{\partial v^2} \bigg|_{x=y=z=1} = \langle Z(t) (Z(t) - 1) \rangle, \tag{2.22}
\]
from which the detector count variance can be deduced.
Chapter 3

The YALINA Experiments

At the Joint Institute for Power and Nuclear Research in Sosny outside Minsk in Belarus, two subcritical cores have been constructed: YALINA-Thermal and YALINA-Booster. YALINA-Thermal (referred to as Yalina in Paper I) started operation in the beginning of this century. In 2005 the fuel of YALINA-Thermal was moved to the new core YALINA-Booster.

Yalina (Яаліна) is the Belarusian word for spruce and refers to the type of environment that is typical for the area of the facility.

3.1 YALINA-Thermal

The subcritical YALINA-Thermal core is loaded with uranium dioxide of 10% enrichment in $^{235}\text{U}$. The fuel pins are situated in a square lattice, depicted in Figure 3.1. The region closest to the central deuteron target is filled with lead. Outside the lead zone, there is a moderating region, filled by polyethylene ($\text{C}_2\text{H}_4$). The reflector is made of graphite with a thickness of about 40 cm. Five experimental channels (EC) are placed at different positions at different radial distances. The experimental channels are positioned in such a way that their relative influence on each other is minimized. As can be seen in Figure 3.1, EC1 is close to the source, EC2 is penetrating the lead zone, EC3 is located in the moderating thermal zone and EC5 and EC6 are located in the reflector. There are in total 280 fuel elements, each of them with a diameter of 11 mm. The spacing between two adjacent elements is 20 mm and the total length of the active fuel is 500 mm. The lead zone is limited in the axial direction and is thus surrounded by polyethylene in all six directions. Originally, the core was designed without this cubic lead block, but it was introduced in a later step of the project with the purpose of investigating its neutronic properties such as scattering and $(n,xn)$-reactions related to ADS spallation target physics.
3.2 YALINA-Booster

YALINA-Booster is a subcritical core with two zones employing a fast and a thermal neutron spectrum respectively. The core consists of a central lead zone, a polyethylene zone, a radial graphite reflector and a front and back biological shielding consisting of borated polyethylene (Figure 3.2). The loading is 132 fuel pins in the inner part of the booster region, containing 90% enriched metallic uranium or 36% enriched uranium dioxide, 563 fuel pins in the outer part of the booster region containing uranium dioxide of 36% enrichment and a maximum of 1145 EK-10 fuel pins containing uranium dioxide of 10% enrichment. The different loadings of the core are described in each paper respectively.

The fast-spectrum lead zone and the thermal-spectrum polyethylene zone are separated by a so called thermal neutron filter, which consists of one layer of 108 metallic uranium pins and one layer of 116 boron carbide (B$_4$C) pins, which are placed in the outermost two rows of the fast zone. Thermal neutrons diffusing from the thermal zone to the fast zone will either be absorbed by the boron or by the natural uranium, or they will be transformed into fast neutrons through fission in the natural uranium. In this way, a coupling of mainly fast neutrons between the two zones is maintained. One of the main goals of this design has been to accomplish two distinguished spectra, fast and thermal, for irradiation purposes.
There are seven axial experimental channels (EC1B-EC4B and EC5T-EC7T) in the core and two axial experimental channels (EC8R and EC9R) and two radial experimental channels (EC10R and EC11R) in the reflector. Moreover, there is one neutron flux monitoring channel in each corner of the core (MC1-4). Three B₄C-control rods (CR) can be inserted in the thermal zone and allow changing the reactivity of the system by about 0.5 $. A detailed description of the core is available in the YALINA-Booster benchmark description [28].

![Figure 3.2. Schematic cross-sectional view of YALINA-Booster (configuration SC0).](image)

3.3 The neutron source

For the pulsed neutron source (PNS) and source jerk experiments, a so called neutron generator was used. The neutron generator is a deuteron ion accelerator coupled to a Ti-T or Ti-D target, located in the center of the core. The D-T or D-D fusion reactions
give neutrons with energy around 14 MeV and 2.5 MeV respectively. The neutron generator can be operated in both continuous and pulsed mode and provides the possibility to generate pulses with frequencies from 1 Hz to 7 kHz with pulse duration of 2 - 130 μs. The maximum beam current in continuous mode is 2 mA, with a beam diameter of about 20 mm, giving a maximum neutron yield of approximately $2 \cdot 10^{11}$ neutrons per second for the Ti-T target and $2 \cdot 10^9$ neutrons per second for the Ti-D target. The deuteron energy is around 250 keV. In continuous mode operation, the accelerator is capable of making repeated short beam trips of a few milliseconds followed by a fast restart of the beam. The frequency of the beam trips and their duration can be adjusted according to the needs.

For the Rossi-α and Feynman-α measurements, various $^{252}$Cf-sources were used.

### 3.4 Detectors and data acquisition

For all measurements in YALINA-Thermal, a $^3$He-detector of 10 mm active length was used. This detector type relies on the (n,p)-reaction in $^3$He, thus being sensitive mainly to thermal neutrons. Data was collected using a multi-scaler (Turbo-MCS).

In the YALINA-Booster experiments a broader range of detectors was utilized. In addition to the above mentioned small $^3$He-detector there were another two larger $^3$He-detectors with active length 25 cm suitable for neutron noise measurements. When operating the D-T neutron source, fission chambers of 1 mg and 500 mg $^{235}$U deposit were used. In Paper II, the experiments were based on the D-D neutron source in pulsed mode resulting in low enough count rate allowing the large He-3 detectors to be used. A counter/timer card was used for recording the arrival time of each detection using a time stamping routine with an accuracy of 12.5 ns. By doing so, a complete record of the experiment was collected and all analysis could be performed afterwards. This is in particular suitable for the noise analysis, since it allows for both a Rossi-α and a Feynman-α analysis on the same set of data.
Chapter 4

The Monte Carlo Simulation Tool

4.1 Code and nuclear data libraries

In order to support the analysis of the experimental results, the experimental setups were analyzed by a Monte Carlo simulation tool. For YALINA-Thermal, the code MCNP4c3 [29] was used, and for YALINA-Booster MCNP5 [30]. The basic principle of a Monte Carlo code is that a huge amount of particle histories is simulated from which an average outcome is deduced. An advantage compared to most deterministic codes is the possibility to use very detailed three-dimensional models and continuous energy nuclear data libraries. In this case, neutrons are transported through a model of YALINA-Thermal [31] or YALINA-Booster [32]. Information concerning interactions with nuclides, such as scattering, capture, fission etc, is given by nuclear data libraries. In the YALINA-Thermal study, the nuclear data libraries ENDF/B-VI, JEFF3.0 and JENDL3.3 were used, whereas in the YALINA-Booster study the libraries ENDF/B-VII, JEFF3.1 and JENDL3.3 were used [9]. After a large amount of transported neutrons, quantities such as effective multiplication factor, neutron flux, and reaction rates can be determined.

4.2 Uncertainties

Results from the Monte Carlo calculation method are always accompanied by a statistical error. On top of that, there are errors propagating from the nuclear data uncertainties and modeling errors.

By simulating a large number of neutron histories, \( N \), the statistical error can be reduced since the relative error, \( e_{\text{rel}} \), follows

\[
e_{\text{rel}} = \frac{1}{\sqrt{N}}. \tag{4.1}
\]
For this reason most of the calculations were performed on a computer cluster with multiple processors operating in parallel.

Errors from nuclear data libraries were in these studies only investigated by changing data library. The identification of uncertainties from individual nuclides is outside the scope of this thesis. Deviations between different libraries were found to be small. Significant differences were found only for JENDL3.3 in the calculation of the effective delayed neutron fraction for YALINA-Thermal.

Modeling errors were a major concern for YALINA-Booster, but not for YALINA-Thermal. In YALINA-Thermal only a few materials were used in relatively small amounts, whereas in YALINA-Booster other construction materials had to be used to support the much heavier construction. Moreover, the materials used in YALINA-Booster that also were used in YALINA-Thermal were used in larger amounts. These materials (stainless steel, lead, aluminum, polyethylene) contain traces of neutron absorbing nuclides. These traces had to be taken into account to achieve reliable results for YALINA-Booster. Sensitivity studies on these traces have been performed on both YALINA-Thermal and YALINA-Booster, but the influence was significant only for YALINA-Booster. In fact, the MCNP calculations of YALINA-Thermal in Paper I were performed without the trace materials, since the traces were discovered later during the work with YALINA-Booster. However, after inserting the trace materials into the YALINA-Thermal model it was found that the contribution to the reactivity was only $160 \pm 20$ pcm. A complete description of the materials including the trace materials can be found in the YALINA-Booster benchmark specification [28].
Chapter 5

Reactivity Measurements

5.1 The Sjöstrand area ratio method

By separating the total neutron density, \( n(t) \), in Eqs. (2.14) into prompt and delayed neutron densities after a source neutron insertion and thereafter integrating over time, the prompt and delayed neutron areas, \( A_p \) and \( A_d \), can be obtained [15]. These areas are depicted in Figure 5.1 and it is straightforward to show that the reactivity in dollars is given by

\[
\frac{\rho}{\beta_{\text{eff}}} = -\frac{A_p}{A_d}.
\]  

(5.1)

This method is sometimes referred to as the Sjöstrand method, named after its inventor, but also as the area ratio method or simply the area method.

Figure 5.1. Prompt and delayed neutron areas used in the area ratio method after repeated source neutron insertions in a subcritical reactor.
5.1.1 Experimental results

The Sjöstrand area ratio method is the most carefully studied method of this thesis. The reason is its simplicity and that it has shown a high degree of stability in previous studies [33,34]. The fact that the observables are integrals makes the method attractive in an accuracy perspective. An important requirement when applying this method is that the fundamental mode overwhelms all other possible modes. However, as will be shown here, there is practically always a contribution from other modes, thus causing what is referred to as spatial dependence.

In both Paper I and Paper II the spatial dependence was investigated by using the same detector at various places in the core. By using the same detector repeatedly instead of using many detectors at the same time, effects from detector influence on reactivity and possible detector differences in terms of for instance efficiency and discrimination levels could be avoided or at least minimized. Thus, the difference in reactivity value stems from the detector position solely.

In Paper II the reactivity was measured at two reactivity levels of 0.975 and 0.965 in YALINA-Booster. The spatial spread was found to be around 6% which is larger than found in other systems [33-35]. Possible causes for the spatial spread can be found by looking at the relative shape of the prompt and delayed fluxes (or areas) respectively (Figure 5.2). As can be seen in the figure, the prompt and delayed fluxes do not have the same distribution in the system. The largest discrepancy is found in EC5T close to the absorber and the fast region, where the prompt flux is affected by higher eigenmodes and the delayed flux is decreased due to absorption to a larger extent than the prompt flux. In the reflector, the prompt flux is slightly more capable of reaching further out than the delayed flux; an effect that can be assigned the different energy spectrum of prompt and delayed neutrons. It is important to remember that the delayed flux is composed of both prompt and delayed neutrons, thus the effect from their different energy spectrum must be small.

![Figure 5.2](image-url)

Figure 5.2. Spatial profiles of prompt flux, delayed flux and reactivity (YALINA-Booster). The points are connected with lines to increase visibility.
In Paper I, the Sjöstrand area ratio method was applied to a deep subcritical state of YALINA-Thermal, namely around 0.91-0.92. At this level of subcriticality, higher eigenmodes become stronger in comparison to the fundamental mode and the underlying assumptions of the area ratio method is violated. As a consequence, the reactivity obtained from the area ratio method starts to deviate from the value obtained from prompt decay methods. This effect has been observed also in other studies [35] and it can also be explained by the non-linear relation between the prompt decay and the reactivity in a reflected deep subcritical system [36]. In YALINA-Thermal a similar shape of the reactivity profile was found as compared to YALINA-Booster. However, the spread is smaller (3%) which might be explained by the lesser extent of heterogeneity of the YALINA-Thermal core.

In Paper III, the measurements of Paper II were extended to include the fast region of YALINA-Booster. It was found that the reactivity as measured in the fast zone could deviate almost 100% from the value obtained in the thermal zone. The explanation is, on one hand, the very strong source mode in the fast zone data, which can be seen in Figure 5.3 during the first microseconds in EC2B, and, on the other hand, the fact that the fast and thermal zones have different spectra and are partly decoupled. In Paper III it is shown through a two-region point kinetics model that the estimated global area ratio reactivity is, for the fast and thermal zone respectively, given by

\[
\rho_{s1} = -\frac{A_p}{A_t} \approx \frac{\rho}{\beta} \left( 1 - \frac{1}{\alpha_1 \tau_1 \alpha_2 \tau_2} - 1 \right) + 1 \quad (5.2)
\]

and

\[
\rho_{s2} = -\frac{A_p}{A_t} \approx \frac{\rho}{\beta} \left( 1 + f \frac{1}{\alpha_1 \tau_1} \right) - f \frac{1}{\alpha_1 \tau_1 \beta_2} \quad (5.3)
\]

In these equations, subindex 1 refers to the fast zone and 2 to the thermal zone, \( \tau \) is the mean neutron lifetime and \( f \) is the return probability describing the probability of a neutron to leave a region and later come back again. The equations above indicate that for both regions, the measured value of the global reactivity is based on the reactivity of the thermal zone with a term and factor describing the coupling of the fast zone. As shown in Figure 5.4, the ratio of the area ratio reactivities of the two zones depends heavily on the return probability, in particular at deep subcriticality.

Converged reactivity values could finally be obtained after applying correction factors achieved through Monte Carlo simulations. These correction factors were obtained by calculating the prompt and delayed fluxes at the detector positions in source mode and by normalizing them to the effective multiplication factor from the criticality calculation.

In Paper III, an investigation of the possible \( k_s \) dependence of the area ratio method was performed. The parameter \( k_s \) describes the multiplication of the source neutrons in contrast to \( k_{eff} \) which describes the multiplication of the fission neutrons. Following long discussions within the community concerning a possible biasing effect it was decided to have in the YALINA-Booster experiments two configurations having the same \( k_{eff} \) but different \( k_s \). This could be achieved by removing highly enriched fuel from the core center, close to the neutron source, and compensating by adding low enriched fuel at
the core periphery. It could finally be concluded that a change in $k_t$ of about 500 pcm did not have any visible effect on the Sjöstrand area ratio reactivity.

![Figure 5.3. PNS histograms for configuration SC3a (CR inserted).](image)

![Figure 5.4. Two-region point kinetics area ratio reactivities and their ratio as a function of return probability.](image)

**5.2 The prompt decay fitting method**

Assuming that the delayed neutrons can be neglected, Eqs. (2.14) read

$$\frac{dn(t)}{dt} = \alpha n(t),$$  \hspace{1cm} (5.4)
where

\[ \alpha = \frac{\rho - \beta_{\text{eff}}}{\Lambda} \]  \hspace{1cm} (5.5)

is the prompt neutron decay constant. By introducing short pulses of source neutrons in a subcritical core, this decay constant can be found [16]. With the knowledge of \( \beta_{\text{eff}} \) and \( \Lambda \), the reactivity, and thereby \( k_{\text{eff}} \), can be obtained. One source neutron pulse is not sufficient for finding \( \alpha \), thus a pulsed neutron source is needed. Since \( \beta_{\text{eff}} \) and \( \Lambda \) are often obtained through simulations it might be useful to compare to some experimental observable. Eq. (5.5) can be rearranged on the form

\[ \frac{\Lambda}{\beta_{\text{eff}}} = \frac{1}{\alpha} \left( \frac{\rho}{\beta_{\text{eff}}} - 1 \right) \]  \hspace{1cm} (5.6)

and thus giving an indirect experimental value of the fraction \( \Lambda/\beta_{\text{eff}} \) based on a PNS experiment and the assumption that the point kinetics approximation holds.

5.2.1 Experimental results

In Paper I, the prompt decay method is applied to a deep subcritical system (YALINA-Thermal). Examples of pulse response are visualized in Figure 5.5. In states far from criticality, the amplitude of the fundamental mode is smaller and other spatial and source modes become more prominent. In the studied case, the fundamental mode could be found in the overall neutron flux decay shortly before the delayed background had been reached. The values obtained in the core region are compatible with each other, whereas the values obtained in the reflector indicate a slower decay. Neutrons reaching the reflector tend to stay there for a relatively long time before being absorbed or leaking out. Thus, the reflector acts as neutron buffer feeding neutrons back to the core. When applying the prompt decay method in the reflector a slower decay is therefore obtained and the margin to criticality will be underestimated. This means a conservative estimation of the subcriticality, which is preferable from a safety point of view, however, this might not be the case in all reflected systems. In this particular case, the reflector consists of graphite which has a low neutron absorption cross section. Other reflector materials, such as sodium, lead or steel, have larger neutron absorption cross sections and might lead to non-conservative conclusions regarding the margin to criticality.

In Paper II, similar results were obtained for YALINA-Booster. However, the pulse response was easier to interpret due to its single exponential nature (Figure 5.6 and Figure 5.7). The major difference compared to the results of Paper I is the smaller spread (<2%) of the results, which is due to the smaller margin to criticality (Figure 5.8). Again, a slightly slower decay in the reflector was observed.

In Paper VII, data from the fast zone of YALINA-Booster was included in the study. It became clear that when studying the pulse response in the millisecond scale, the same prompt decay was observed in both the fast zone and the thermal zone. In the fast zone, the total flux is driven by neutrons leaking through the valve zone. The fast zone is also characterized by a local prompt decay but this decay is too fast to be observed on
the millisecond scale. An interesting effect is that the prompt decay method gives the same results in both the fast and thermal region, whereas the area method results are strongly biased in the fast zone.

Figure 5.5. PNS histograms for YALINA-Thermal (time discretization 50 μs).

Figure 5.6. PNS histogram for the 1132-configuration of YALINA-Booster (normalized to the constant level and with time discretization 5 μs).
5.3 The source jerk and beam trip methods

Assuming a subcritical core at constant power driven by a continuous external source, a neutron flux level somewhere in the core will be $n_0$. Suddenly, the external neutron source is shut-down very quickly. Then, the neutron flux changes rapidly to a semi-stable level $n_1$ consisting of the delayed component of the neutron flux. In the same
manner as for the Sjöstrand area ratio method it can be shown [20] that the reactivity in dollars is given by

\[
\frac{\rho}{\beta_{\text{eff}}} = \frac{n_1 - n_0}{n_1}.
\] (5.7)

When the method is applied in this way it is referred to as the source jerk method. In the YALINA-Booster experiments it was possible to apply this technique in a repeated manner with a constant frequency of 1 Hz and with a beam interruption time of about 30 ms. When applying the method in this way, one must remember that the prompt neutron density recovers directly when the source is restarted, whereas the delayed neutron density is reduced compared to the constant source equilibrium case. Therefore, a correction must be applied that takes into account this effect. The correction factor of \(n_1\) is to a good approximation simply the total flux as if there was no beam interruption divided by the observed total flux with beam interruption. The absolute value of the correction is added to \(n_0\) and \(n_1\) in Eq. (5.7) before evaluating the reactivity.

### 5.3.1 Experimental results

In Paper I, the single source jerk technique was applied. The result is associated with a large uncertainty and illustrates mainly the need of a repeated beam trip methodology, which was finally implemented in Paper IV. Before the experiments of Paper IV started, the control module of the neutron generator was modified to allow current mode operation with short interruptions (beam trips) and with a trigger signal to the data acquisition system for synchronization. In this way thousands of beam trips could be accumulated to obtain the reactivity with low statistical uncertainty. With the beam operating in current mode, a strong oscillation of the flux was observed. The same oscillation could be found in the beam current, but with lower relative amplitude. The oscillation had a frequency of 50 Hz and was a disturbance from the electricity grid that could not easily be eliminated. The oscillation affected the bending magnet in the beam line and caused the beam impact position on the target to move around. Each cycle the beam hit the target close to the boundary were the tritium concentration was lower. Nevertheless, since the oscillation appeared in the millisecond scale and was not present during the beam trips, it did not affect the results of the experiments. The signal from the beam current meter and the fission chamber in the booster region are displayed in Figure 5.9. The 50 Hz oscillations are clearly visible before the beam trip in both the beam current and the detector. The beam current carried additional high-frequent noise that is visible also after the beam trip. It is important to point out that this is a false component of the signal in the sense it is not associated with source neutron generation. In the same figure, the delayed neutron flux, \(n_1\), is clearly visible after the beam trip at \(t = 0\). The total flux level, \(n_0\), must be obtained as an average over an integer number of beam oscillation periods before the beam trip.
Figure 5.9. Detector count rate in EC1B (grey) before and during a beam trip (beam current in black).

An example of beam trip histograms is shown in Figure 5.10, in which nearly 1000 beam trips have been accumulated. When data are added to each other, the oscillations tend to partly cancel out, but it does not affect the results as long as the average value is taken over an integer number of oscillation periods. In the figure, the effect of the control rods can be seen as lower count rate in the detector. By plotting in logarithmic scale, the effect on the prompt decay constant can be viewed (Figure 5.11). In Paper VI it is shown that the same results can be obtained from the beam trip histograms as from the PNS histograms, including the same spatial spread. Thus, having high enough count rate it is possible to use this method as an online reactivity meter. This was tested in these experiments, although the count rate was in the lower range to obtain the reactivity with high enough statistical accuracy. In Figure 5.12, the reactivity was measured once per second using the beam trip method while extracting the control rods during $0 < t < 5$ s. Despite the high uncertainty in each data point, the reactivity increase can be tracked by this method. However, a clear disadvantage of this method is the long stabilization time of the source jerk reactivity. Since it requires that the equilibrium level of the delayed neutrons has been reached, it takes about two minutes to stabilize at the new value. This theoretical reactivity, given by the point kinetics equations is given in grey in the same figure. Important is to point out that during the transition period, the measured reactivity overestimates the margin to criticality, which is a major drawback. Therefore, it would be better to base the reactivity monitoring on the prompt decay constant measured after each beam trip, as shown in Paper VI.
Figure 5.10. Beam trips histograms of SC3a, detector location EC2B with the control rods inserted and withdrawn respectively.

Figure 5.11. Beam trips histograms of SC3a, detector location EC2B for the two cases with control rods inserted and withdrawn (logarithmic scale).
The Rossi-$\alpha$ method

Due to the fission chain reactions in a multiplying medium, detected neutrons may be correlated to each other in space and time. It might happen that two neutrons, detected by the same neutron detector, originate from the same fission chain. In such case, it is likely that these two detections, or events, are close in time. It can be shown that the probability to detect one more neutron after the first neutron decreases exponentially in time with the prompt neutron decay constant $\alpha$. The most intuitive derivation of the Rossi-$\alpha$ method is based on a heuristic approach starting from the probability of having one fission event around $t_0$ within $dt_0$:

$$p(t_0)dt_0 = Ft_0, \quad (5.8)$$

where $F$ is the average fission rate in the system. Subsequent detection events will be either correlated to this first fission event or random. First, a correlated event around $t_1$ within $dt_1$ is considered. The detection probability per initial fission around $t_0$ is

$$p(t_1)dt_1 = \varepsilon \nu p \Sigma_f e^{\alpha(t_1-t_0)} dt_1, \quad (5.9)$$

where $\varepsilon$ is the detection efficiency, $\nu p$ is the number of prompt neutrons emitted per fission and $\nu$ is the average neutron velocity. In the same manner, around a later time $t_2$, $t_2 > t_1$, the probability of a correlated event is
\[ p(t_2)dt_2 = \varepsilon_\nu \left( \nu_p - 1 \right) \Sigma_f e^{a(t-t_0)} dt_2, \]  

(5.10)

where the term \( \nu_p - 1 \) takes into account the neutron loss required to generate the first detection around \( t_1 \). The probability of obtaining one count in \( dt_1 \) followed by one count in \( dt_2 \) originating from the same fission in \( dt_0 \) is consequently the product of all three independent probabilities. Integrating over all possible initial fissions before \( t_1 \) gives

\[ p(t_1,t_2)dt_1dt_2 = F^2 \varepsilon^2 dt_1dt_2 + F \varepsilon^2 \frac{\left( \nu_p^2 - \nu_\nu \right) dt_1dt_2}{-2\nu_p^2 \alpha \Lambda^2}. \]  

(5.11)

Finally, introducing \( t = t_2 - t_1 \) and \( t_1 = 0 \) and normalizing to the count rate in \( dt_1 \) yields

\[ p(t)dt = \frac{\varepsilon D_r}{2\alpha \Lambda^2} e^{\alpha t} dt + F \varepsilon dt. \]  

(5.12)

with the random events included and the common parameters introduced [37,38]. \( D_r \) is the Diven factor defined as [39]

\[ D_r = \frac{\nu_p^2 \left( \nu_p - 1 \right)}{\nu_p^2}. \]  

(5.13)

It should be noted that in this thesis, as well as in the papers, the sign convention of \( \alpha \) has been conserved. In most references, \( \alpha \) is defined with the opposite sign for the noise methods. Finally, by plotting the probability density \( p(t) \) obtained experimentally, the prompt neutron decay constant can be found from the correlated part through fitting of an exponential function. As shown in Paper V, when several alpha-modes are present, the auto-correlation can simply be described by a sum of exponentials.

### 5.4.1 Experimental results

In Paper IV, experimental neutron noise results from the thermal zone of YALINA-Booster are presented. The study was limited to the thermal zone, since the count rate was too low in the fast zone to obtain useful results, even with the largest detector and the strongest neutron source available. The experimental data from the thermal zone revealed not less than three alpha-modes present in the auto-correlation (Figure 5.13). Dual alpha-modes have been reported previously in reflected systems where the core and the reflector contribute with one alpha-mode respectively [40-41]. In the case of YALINA-Booster, there is an additional fast alpha-mode originating from the booster zone. The presence of the higher alpha-modes causes some limitations in the use of the Rossi-\( \alpha \) method. First of all, most of the correlated part of the histogram is occupied by the higher alpha-modes. Thereby, the slowest decaying alpha-mode is best viewed close to the uncorrelated background. Thus, obtaining results with high accuracy is challenging and extremely long measurement times are needed. In addition, having alpha-modes close to each other causes larger uncertainties from the fitting error estimation. In the end, it was not possible to follow a reactivity perturbation of 0.5 $ even after several hours of measurements.
Another interesting observation is that the value of the higher alpha-modes is much less reactivity dependent compared to the slowest alpha-mode. The fastest alpha-mode could to a very good approximation be assumed constant at least in the subcriticality interval 0.85-0.977. The same holds for the first higher eigenmode except at deep subcriticality. It gives an indication that the higher eigenmodes are driven by the surrounding geometry and materials rather than the amount of fuel loaded.

![Figure 5.13. Rossi-α histogram and fitting in linear and logarithmic scale (5 μs time bin). The contribution from individual exponentials is indicated by dashed lines.](image)

### 5.5 The pulsed Rossi-α method

The Rossi-α technique can be applied in the same way to data from a PNS experiment as for a continuous source. It has been shown that the pulsed Rossi-α histogram consists of three terms [42]:

$$p(t)dt = A_1 e^{\alpha t} dt + A_2 dt + A_3 \sum_{n=1}^{\infty} (a_n^2 + b_n^2) \cos(\omega_n t)dt$$  \hspace{1cm} (5.14)

The first term is the ordinary correlated exponential term, also found in the classical formula for a steady state source, and the last two terms are the uncorrelated terms. The last term is a non-decaying oscillating term consisting of a Fourier series of the pulsed neutron source characteristics. The constants of Eq. (5.14) are given in Ref. [42].

#### 5.5.1 Experimental results

Kitamura et al. have described how to analyze the data from a pulsed Rossi-α measurement [42]. Since the source is pulsed, the histogram looks very different from traditional Rossi-α histograms (Figure 5.14).
Figure 5.14. Rossi-α histogram for YALINA-Booster with a pulsed source.

The assumption that the first term of Eq. (5.14) has decayed completely after 80 ms is then made. At that time, the signal only consists of the two uncorrelated terms. Since these terms do not decay in time, experimental data can be chosen from the interval 80-93 ms (the last U-form) and then be deleted subsequently from the previous U-formed intervals. In this way, only the correlated exponential term will be left, as depicted in Figure 5.15. In an ideal case, the constant level after removal will be zero. Finally, the exponent is determined in the same way as was done for the classical Rossi-α case. An example of the final solution is depicted in Figure 5.16.

It was found that the results from the pulsed Rossi-α analysis are in agreement with the results from the PNS fitting method. The errors are larger for the Rossi-α case, due to the more complicated data treatment required to arrive at the result. Compared to the traditional Rossi-α approach, the pulsed method gives results with much higher accuracy. This comes from the fact that the amplitude of the correlated term is much higher in the pulsed experiment, thus increasing the signal-to-noise ratio, and the contribution from higher alpha-modes is lower.
Figure 5.15. Rossi- α histogram with the oscillation term removed.

Figure 5.16. Fitting of exponential to the Rossi- α histogram after the removal of the oscillation term.

5.6 The Feynman- α method

In a nuclear reactor, the counts, \( c \), of a neutron detector will deviate from a true Poisson distribution due to the presence of fissile material. The deviation is denoted \( Y \):
It can be shown that $Y$ is a function of $\Delta T$, the time base used in the measurement. This relation is often referred to as the variance-to-mean or the Feynman-$\alpha$ formula [43]:

$$
Y(\Delta T) = \frac{\varepsilon D_v}{N^2 \alpha^2 \Delta T} \left( 1 + \frac{1 - e^{\alpha \Delta T}}{\alpha \Delta T} \right).
$$

(5.16)

This formula can be derived directly from the Rossi-$\alpha$ method by considering the number of combinations of two events out of a set of events, $c$, within the time interval $\Delta T$:

$$
\binom{c}{2} = \frac{c!}{2!(c-2)!} = \frac{c(c-1)}{2}.
$$

(5.17)

The expectation value is given by integrating Eq. (5.11):

$$
\frac{c(c-1)}{2} = \int_{t_2=0}^{\Delta T} \int_{t_1}^{t_2} p(t_1', t_2') dt_1' dt_2',
$$

(5.18)

which after rearrangement can be written as

$$
\frac{c^2 - c}{c} = 1 + \frac{\varepsilon D_v}{\alpha^2 \Delta T} \left( 1 - \frac{1 - e^{\alpha \Delta T}}{\alpha \Delta T} \right).
$$

(5.19)

The statistical uncertainty of $Y$ is given by the variance of the variance [44] according to

$$
\sigma^2(Y) = (1+Y)^2 \left( \frac{1+Y}{cn} + \frac{2}{n-1} \right),
$$

(5.20)

where $n$ is the number of samples for a given $\Delta T$. Consequently, the statistical accuracy will be higher if the measurement is running for a long time and for low $\Delta T$.

Alternatively, the Feynman-$\alpha$ formula can be derived through the stochastic branching process, Eqs. (2.16-2.22), which was the underlying approach in Paper V. A thorough derivation of this type can be found in ref. [27]. In Paper V, a new Feynman-$\alpha$ formula for an arbitrary number of alpha-modes was derived. It has the following form:

$$
Y(\Delta T) = C_0 \sum_{k=0}^{N-1} \sum_{l=0}^{N-1} C_{k,l} \frac{1}{\alpha_i \alpha_l} \left[ \alpha_i \left( 1 - e^{\alpha_i \Delta T} \right) + \alpha_l \left( 1 - e^{\alpha_l \Delta T} \right) \right],
$$

(5.21)

where $C$ with subindex are constants, and $N$ is the number of alpha-modes. This formula breaks down to Eq. (5.16) in the case of a single alpha-mode.

### 5.6.1 Experimental results

When first analyzing the experimental data (the same data set as for the Rossi-$\alpha$ measurement) a strongly linearly increasing component was identified in the Feynman-$\alpha$ dia-
gram. The origin was shown to be a slow exponential decay of the count rate when performing the experiment. These counts came from gammas from activated construction materials from a previous experiment performed at higher power. The count rate decrease was less than 1% in 17 hours, but gave nevertheless a considerable bias to the Feynman-α data. This parasitic contribution could be removed by dividing the experimental data into subsets with semi-stable count rates and evaluate $Y$ for each subset and then make an average. In the Rossi-α analysis no effect of this kind was observed, although the analysis was based on the same data set. In the case of Rossi-α, the effect would be a base level of the random events deviating slightly from unity, which could not be found and indicates thus a clear advantage of the Rossi-α method in terms of stability to non-stationary count rates. The underlying reason is the fact that the Rossi-α method is based on a time window of a few milliseconds whereas the time frame of the Feynman-α method is a large number of gate widths which can be seconds, minutes or hours.

Not only the reactivity affects the variance-to-mean ratio but also the dead time of the data acquisition system. This has been the subject of numerous papers (for instance [45-51]), however, there are no studies on the case of extending (E) and non-extending (NE) dead times in series as in this specific case. Therefore, an alternative correction path had to be explored to correctly separate the detector dead time contribution from the reactor contribution. The proposed method (Paper IV) is a modification of the method by Hazama [45]. The strength of the method is that it can be applied to any combination of dead times and it is separate from the variance-to-mean model, in contrast to the methods in refs [46,48] where the dead time is embedded in the stochastic transport theory. The drawback of the method is that it requires $\Delta T >> \tau$, where $\tau$ is the dead time.

In Figure 5.17, Feynman-α data before and after correction are displayed. The two data sets are based on the same conditions except the source strength. Both data sets will yield the same resulting plot after correction, which can be seen clearly. Only at very low $\Delta T$ a deviation can be observed. The large difference of the two data sets before correction indicates another weakness of the Feynman-α method in terms of stability and robustness.

Since multiple alpha-modes were identified in the Rossi-α analysis, multiple alpha-modes are expected to appear in the Feynman-α data as well. In Figure 5.18, fittings of one and two alpha-modes to Feynman-α data from YALINA-Booster are displayed. It can clearly be seen that at least two alpha-modes must be applied. At very small gate widths, a small oscillation of the two-exponentials residual can be discerned, thus indicating the need of an additional alpha-mode. However, the condition $\Delta T >> \tau$ makes the model non-applicable at small gate widths and these data must be excluded from the fitting. Therefore, in the cases data are affected by dead time, the Feynman-α method is less useful if the aim is to identify possible higher alpha-modes, which is sometimes required in order to obtain a correct value of the lower alpha-modes.
Figure 5.17. Feynman-α diagrams for the same detector and position but different source intensity, before and after dead time correction.

Figure 5.18. Fitting of one and two alpha-modes to experimental Feynman-α data. The two fittings are hard to distinguish from each other in the figure. Therefore, the residual are plotted as well. The residuals indicate the necessity of applying at least two alpha-modes (exponentials).

Including delayed neutrons into the Feynman-α analysis has been discussed in numerous publications, for instance refs [27,52,53] including Paper V, although its practical usefulness is limited. Rather, the efforts are an expression of the beauty and completeness of the theories developed. Seldom are experimental data analyzed with long enough gate widths to include the delayed neutron contribution, since it is not required when the aim is to obtain the prompt alpha-modes. In Paper V, the effect of the delayed neutrons was included in the study to validate the new Feynman-α formula with delayed
neutrons. In the fitting shown in Figure 5.19, gate widths from 0.15 ms to 2 s have been used, thus covering four orders of magnitude. The delayed neutron contribution can be seen as a small increase in the $Y$-value after about 200 ms gate width. The delayed alpha-mode, obtained through fitting, has a value of approximately $0.1 \text{s}^{-1}$, which well represents a one-delayed neutron group decay rate.

Figure 5.19. Fitting to Feynman-$\alpha$ data including delayed neutrons.
Chapter 6

Discussion and Conclusions

6.1 Concluding remarks

The methods investigated in this thesis can be classified into three groups: methods for reactivity calibration at zero power, methods for reactivity monitoring at zero power and methods for reactivity monitoring at high power. These three groups of methods can, respectively, be assigned a specific source, namely a pulsed source, a continuous source at zero power and a semi-continuous source at high power. Table 6.1 summarizes the main characteristics of the various methods included in this work.

Starting with the pulsed source methods for reactivity measurement at zero power, it has been shown that the Sjöstrand area ratio method can measure the reactivity to a high accuracy in terms of statistical uncertainty, but is sensitive to spatial effects caused by the core heterogeneity and the detector’s close vicinity to the source. If being aware of these effects and applying, for instance, Monte Carlo correction factors, the method is suitable for calibrating the reactivity stepwise during the loading and when the desired loading has been obtained. A possible difficulty is related to the accuracy of the delayed neutron data which will influence the possibility of obtaining a correct value of $k_{eff}$ from the dollar reactivity. For uranium or MOX fuel, this is not a problem since the delayed neutron data is well known, but for the minor actinide bearing fuels, the uncertainties are larger (as indicated in Figure 1.3). By increasing the uncertainty in the final $k_{eff}$, a more conservative margin to criticality must be chosen, which in the end affects the economy of the system.

The pulsed source measurement technique has an advantage that both the dollar reactivity and the prompt decay constant is obtained from the same set of data. The value of the prompt decay constant obtained from the prompt decay fitting technique can be used together with the dollar reactivity to get a value of the ratio $\Lambda/\beta_{eff}$. This experimental ratio can be used for comparison of calculation tools as an additional possibility of validating the calculation model, routine and underlying nuclear data library.
The prompt decay can also be obtained from the same data set through the pulsed Rossi-α method, however, the complicated analysis scheme makes it less attractive.

Continuing with the neutron noise methods, the Rossi-α method has a clear advantage over the Feynman-α method due to the inherent stability towards count rate drift and detector dead time. In addition, the Rossi-α methodology can easier be implemented as an online meter since the analysis time is much shorter. When using the Feynman-α method the data must first be formed into histograms of various time binning for which, each of them, the mean value and variance must be calculated. This calculation scheme requires more time than simply calculating the time difference between the counts, as for the Rossi-α method. Both methods do, as has been shown in this work, suffer from decreased accuracy due to the presence of higher eigenmodes, which causes a limitation, mainly in terms of measurement time, when applied to heterogeneous systems. As was shown in Paper IV, the higher alpha-modes of a Rossi-α histogram can be reactivity independent. Consequently, it is of highest importance to be ensured the correct alpha-mode is monitored during for instance core loading. By comparing with the PNS measurement, the correct alpha-mode can be selected.

At full power the beam trip methodology can be utilized to monitor the reactivity on a semi-continuous basis. The source jerk approach, giving the dollar reactivity at each beam trip, performs well in equilibrium conditions, but gives misleading results in case of transient scenarios. As already pointed out, in case of a step reactivity change there is a stabilization time of about two minutes due to the decay of the delayed neutrons (Figure 5.12). Another example is a sudden source strength modulation, which may be interpreted as a reactivity increase according to the source jerk methodology. In such case the prompt part of the flux will change directly, whereas the delayed part stays almost the same for some seconds and then converges slowly to the new level. The dollar reactivity from the source jerk methodology in case of a sudden source decrease of 10% is shown in Figure 6.1. Also in this case the stabilization time is at least two minutes. In an ADS, such a source modulation can be caused by several reasons, for instance dislocated beam impact position, decreased proton energy and decreased proton source efficiency. However, this kind of source modulation can in principle be accounted for by correlating the source strength with the beam trip reactivity and perform a correction, at least if the source strength is monitored. In any case, when applying the beam trip methodology, the prompt decay after each beam trip will give a more stable result since it does not require any equilibrium condition concerning the delayed neutron precursors.
Table 6.1. Characteristics of the various reactivity measurement methodologies.

<table>
<thead>
<tr>
<th></th>
<th>Pulsed source</th>
<th>Continuous source at zero power</th>
<th>Semi-continuous source at high power</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Sjöstrand area ratio</td>
<td>Rossi-α</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Prompt decay fitting</td>
<td>Rossi-α</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Pulsed Rossi-α</td>
<td>Feynman-α</td>
</tr>
<tr>
<td>Spatial independence</td>
<td>Low</td>
<td>Fairly high</td>
<td>Fairly high</td>
</tr>
<tr>
<td>Statistical accuracy</td>
<td>High</td>
<td>High</td>
<td>Low</td>
</tr>
<tr>
<td>Capability of online monitoring at high power</td>
<td>-</td>
<td>-</td>
<td>Fairly high</td>
</tr>
<tr>
<td>Capability of online monitoring at zero power</td>
<td>-</td>
<td>-</td>
<td>Fairly low</td>
</tr>
</tbody>
</table>

Figure 6.1. Source jerk reactivity after a source strength decrease of 10% at \( t = 0 \).

6.1.1 The current-to-flux reactivity indicator

The most straightforward procedure to detect a reactivity variation in a subcritical power producing system is to compare the input beam current with the output system power or a localized neutron flux [54]. Any reactivity perturbation will alter the current-to-flux ratio and can thus be detected. Since the ratio is just a relative measure of the reactivity, an absolute calibration based on the methods described in this thesis is necessary on a frequent basis. The reactivity can be obtained from the current-to-flux ratio according to

\[
\rho = -c \frac{I}{\phi},
\]

where the proportionality factor \( c \) represents the calibration with other absolute methods [54]. The advantage of this methodology is its simplicity since both the beam current and the neutron flux can easily be obtained based on well-established techniques. However, the robustness and accuracy of the method is yet to be proven. Results from the YALINA-Booster experiments have revealed that in addition to the beam current and the neutron flux, also the source neutron production must be monitored (Paper IX,
Paper XV, Paper XVII), since the source strength can change although the beam current is constant. Thus, an analysis scheme of the current-to-flux methodology taking into account possible system dependent factors must be established, which is outside the scope of this thesis.

6.2 Transposition to fast systems

Since the ADS is foreseen to be a fast system, the transposition of these results, obtained in a thermal and a coupled fast-thermal system, must be done carefully. Each separate case must be judged individually to see what difference would be in a fast spectrum. All methods described are applicable to fast systems, however, the observed disturbing effects may be different.

When applying the Sjöstrand area ratio method in a fast system, prompt neutrons returning from the reflector, after spending there a considerable time, may be interpreted as delayed neutrons. This effect would lead to a conservative estimation of the margin to criticality and is therefore not a major problem. Moreover, this effect will be compensated by the Monte Carlo correction.

The strong reactivity bias found using the area ratio method in the fast booster region (Paper III) was partly caused by the strong heterogeneity of the core. Similar, but less extreme conditions, can be imagined in a heterogeneously loaded ADS with minor actinide bearing targets. In such a configuration, there is a possibility that the detector is accidentally located in a position were part of the delayed flux is hidden by the dedicated targets containing nuclides with large absorption cross sections.

The prompt decay, on the other hand, might be harder to interpret in a fast system, in particular at deep subcriticality, as shown in the MUSE experiments [35]. Thermalized or partly thermalized modes from the reflector or shielding regions might interfere with the exponential prompt decay of the core region, thus making the fitting procedure difficult to perform. Having many detectors in the system will help in understanding such processes. Concerning the PNS methods, the conclusions of MUSE and YALINA are in complete contrast to each other. In this work it has been concluded that the prompt decay method is more stable than the area ratio method, whereas in MUSE the vice versa was concluded and this effect can, most likely, be ascribed the different neutron spectra and the strong heterogeneity of YALINA-Booster.

Concerning the neutron noise methods, they were originally applied to fast systems [38]. The successful outcome of these methods is heavily dependent on the suitable choice of source strength, detection efficiency and detector location rather than the neutron spectrum. Best results are obtained if using as big detector as possible, to increase the count rate, and as weak source as possible, not to overwhelm the correlated counts with random counts. In this context it is interesting to note that the amplitude of the correlated term of Eq. (5.12) is not count rate dependent (unlike the random term). Concerning contributions from higher alpha-modes there is nothing that clearly indicates any worse conditions in general for fast systems, based on experiences from this work and MUSE [35].
Finally, the beam trip methodology is following the characteristics of the PNS results in terms of reactivity and prompt decay values. The major difference in fast systems lies in the time characteristics. The required time the beam is turned off is much shorter in a fast system; probably about 1 ms will be enough. Thus, the power loss in repeated beam trip mode in fast spectrum is lower.

6.3 Possible application scenarios

The subcriticality of the ADS must be assessed during all phases of its operation including the loading and start-up phases, operation at any power level and shut down. Each mode of operation requires an optimum methodology of reactivity assessment which might be different from other modes of operation. Here, a suggestion based on the conclusions of this thesis is presented. It is assumed that the accelerator can operate in pulsed mode, continuous mode and semi-continuous mode with beam interruptions (beam trips).

6.3.1 Instrumentation

The more detectors used, the more information is obtained from the system. Naturally, there is an upper limit on the number of detectors that can be used at the same time. Suitable would be to have detectors close to the spallation target, in the middle of the core, at the core boundary and in the reflector. That would give a complete picture of the source pulse propagation and the spatial dependence of the different methods when applied to the specific system.

All work in this thesis is based on detectors in pulsed mode operation, meaning that each detection causes an electric pulse that is passing a discriminator. A square pulse is then generated that is registered by the data acquisition system. This is suitable only at low neutron flux with relatively low count rates. When the count rate is too high for the system to handle, which is the case in for instance nuclear power reactors at high power, detectors in current mode operation are used. In such detectors, the integrated charge output from the detector is measured directly after amplification. The measured current (or voltage) from the detector is directly proportional to the neutron flux. Studies performed at YALINA-Booster, not included in this thesis, have shown that the beam trip methodology applied with current mode detectors gives equivalent results compared to pulsed mode detectors (Paper XI and Paper XVI).

The choice of detector is also a crucial point. At low power large pulsed mode detectors are needed for the noise techniques and system start-up. Thus, both pulsed and current mode electronics chains will be needed. Most likely, fission chambers will be used since the gamma induced counts can easily be discriminated. For detection of high energy neutrons from the spallation target, fission chambers based on lead might be an option.
6.3.2 Core loading

During core loading, the goal is to add fuel bundles stepwise from empty core to the final configuration foreseen. In this process, it is of uttermost importance to avoid supercriticality and, in the case of ADS, criticality with possible subsequent power excursion. As in all start-up procedures performed in critical reactors, the inverse counting method with a start-up source will be utilized. With this method, an extrapolation of the inverse detector count rate to zero constitutes the basis for estimating the critical fuel mass. In this way, criticality can be avoided, but the drawback is that the value of $k_{\text{eff}}$ cannot be determined. In parallel with the inverse counting method, the data should therefore be analyzed in the Rossi-$\alpha$ manner. Thereby, a picture of the set of alpha-modes will become clear. Having the prompt decay constant, the subcriticality of the core is still unknown since the neutron reproduction time and the delayed neutron fraction are unknown. These parameters are possible to estimate by computer simulations, but the uncertainties associated might be large, in particular when high fractions of minor actinides are present. Therefore, it would be recommended to make use of the accelerator and run it in pulsed mode to obtain the dollar reactivity and an additional estimation of the prompt decay constant. The combined results of the dollar reactivity and the prompt decay constant gives an estimate of the ratio of the neutron reproduction time and the effective delayed neutron fraction that can be used for validation of calculated parameters. Once the simulated ratio agrees with the measured, the effective delayed neutron fraction can be used for calculation of the effective multiplication factor from the dollar reactivity. Thereafter, a cross-check should be performed by comparing the obtained $k_{\text{eff}}$ with the simulated. In addition, the effective delayed neutron fraction can be measured with some of the existing techniques [36,55-57]. One should also keep in mind that the definition of the effective delayed neutron fraction is ambiguous in source-driven systems since its definition is based on the critical adjoint function [58].

During the core loading procedure, it is important to use several detectors to achieve a global understanding of the spatial effects of the core to avoid a possible overestimation of the subcriticality. Naturally, it is important to remember that during the loading procedure, a detector located outside the core in the reflecting volume, will later be located inside the core. This will affect the value of the prompt decay and possible higher eigenmodes.

6.3.3 Power increase

Once the foreseen loading has been reached the power will be increased by operating the accelerator beam in current mode. The power increase must be slow enough to avoid thermal stress to the construction materials. The maximum temperature increase per unit time must be determined from the material properties of the various core components, in particular the vessel. The maximum temperature increase rate will constitute the basis for the maximum beam increase rate allowed. The temperature increase in the core will change the subcriticality due to temperature feedbacks, such as Doppler feedback, coolant temperature feedback and increased neutron leakage. This reactivity change must be under control, therefore it is suggested to operate the accelerator in repeated beam trip mode. The imposed beam trips must have a high enough frequency
to follow possible reactivity changes. Most likely the beam current will be increased stepwise to carefully pre-determined currents that do not violate the maximum temperature increase rate. At each step the beam trip reactivity is measured when the system has reached steady state conditions. During those moments when the system has not reached steady state conditions, the prompt decay can be used as reactivity indicator by applying beam trips. Before the temperature is increased the beam trip reactivity has been calibrated against the Sjöstrand area ratio reactivity at zero power.

6.3.4 Full power operation

At full power, when the system has reached thermal equilibrium, reactivity changes might still appear during normal operation due to fuel burnup. In general, the reactivity swing rate will be designed to be negligible or slow but will in any case require new reactivity calibration with the beam trip technique from time to time.

More important is, however, the ability to track possible abnormal reactivity related events or accidents. Such sudden events can be traced by the beam current-to-flux ratio. One must keep in mind that other effects might affect the current-to-flux ratio such as the neutron production rate in the spallation target and the beam impact position. Since the current-to-flux methodology will indicate a relative reactivity shift, an absolute calibration must be performed on a regular basis using beam trips. In a fast system the beam interruption time, \( d \), does not need to be longer than 1 ms. The period between the beam interruptions, \( T \), must be short enough to maintain a reliable calibration of the current-to-flux reactivity indicator, but long enough not to lose too much effective power. The power of a system undergoing frequently imposed beam trips will be associated with the following power duty factor:

\[
 f_{fr} \approx 1 - \frac{d}{T} \left(1 - \frac{1}{1 - \rho_s}\right). \tag{6.2}
\]

Assuming a beam interruption time of 1 ms and a period of 1 s, the factor is approximately 0.999 and is practically independent of the reactivity level. Thus, the power fraction penalty due to the imposed beam trips is \(~1\‰\) and should not constitute a major economical concern.

An alternative is to measure the prompt decay during the beam trip. The advantage would be that it works even if the delayed neutron background has lost its equilibrium level due to longer unexpected beam interruptions and, in addition, the beam interruption time could be about a factor three shorter. However, a combined use of these two analysis methods would be preferable to be on the safe side.

Since the fuel composition will change with burnup, the effective delayed neutron fraction will change in time. However, this variation is slow and new measurements of this parameter can be done much less frequently compared to the imposed beam trips. A drawback is, however, that based on existing methods [36,55-57], this must be done at zero power.
6.3.5 Slow shut-down
To avoid thermally induced stresses in materials the system power should be decreased slowly, in the same manner as during system start-up. Thus, the temperature decrease rate should not exceed a certain limit. Since most reactivity feedback mechanisms are designed to be negative with respect to a power increase, the subcriticality of the system will decrease when the power and the temperature are decreased. Consequently, the reactivity must be carefully monitored during the power decrease. As for the system start-up, the beam trip method cannot be used in terms of dollar reactivity measurement since it requires steady state conditions, but it can be applied for measuring the prompt decay. The procedure would therefore be the same as for the power increase, with increased carefulness since the feedbacks act in the unfavorable direction.

6.3.6 Emergency shut-down
In case of unexpected events or accidents the system must be shut-down quickly by turning off the beam completely. Naturally, the system must be designed to remain subcritical under the resulting temperature decrease. With the external source shut down and having a low power, noise techniques can be applied to monitor the subcriticality in terms of the prompt decay. In particular, the Rossi-$\alpha$ method can be applied since it has been shown to be faster and insensitive to non-stationary count rates, in contrast to the Feynman-$\alpha$ technique. Before restarting the system, a new reactivity calibration based on a pulsed neutron source must be performed.

6.4 General conclusions
A set of reactivity measurement methods have been applied to the subcritical assemblies YALINA-Thermal and YALINA-Booster to investigate their stability, accuracy and capability of online reactivity monitoring. The experiments have revealed some inconsistencies with previous studies. For instance, the Sjöstrand area ratio method gave results with strong spatial spread in the heterogeneous YALINA-Booster, whereas the prompt decay fitting method gave more stable results. The Rossi-$\alpha$ method showed some advantages compared to the Feynman-$\alpha$ method in terms of dead time sensitivity, power drift and higher alpha-mode determination. The online monitoring capability of the beam trip technique was demonstrated. The dollar reactivity and the prompt decay obtained at each beam trip followed the same spatial spread as the corresponding result from the pulsed neutron source measurement, which could be corrected through Monte Carlo correction factors.

Possible application scenarios were proposed based on the conclusions from the experiments. Before a validated online reactivity monitoring methodology is defined, the methods need further investigations in a fast neutron system. This will be accomplished in the next phase of the European project in which this work has been performed.
Bibliography

15. N.G. Sjöstrand, Measurement on a subcritical reactor using a pulsed neutron source, Arkiv för fysik 11, 13 (1956).
17. F. Mellier (coordinator), The MUSE experiments for sub critical neutronics validation, Deliverable no 8 – Final report (2005).
18. C. Rubbia et al., TRADE: A full Experimental Validation of the ADS Concept in a European Perspective, AccApp’03, June 1-5 2003, San Diego, California, USA (2003).
32. Y. Fokov, YALINA-Booster MCNP input file, Joint Institute for Power and Nuclear Research, Sosny, Minsk, Belarus, 26 October 2006.
36. R.Y.R. Kuramoto et al., Absolute measurement of $\beta_{\text{eff}}$ based on Feynman-$\alpha$ experiments and the two-region model in the IPEN/MB-01 research reactor, Annals of Nuclear Energy 34, 433-442 (2007).
40. G. Kistner, Rossi-$\alpha$ theory for assemblies with two prompt neutron groups, Nukleonik 7, 2, 106 (1964).
47. K. Hashimoto et al., Experimental investigation of dead-time effect on Feynman-$\alpha$ method, Annals of Nuclear Energy 23 (14), 1099-1104 (1996).
48. Y. Kitamura et al., General formulae for the Feynman-$\alpha$ method with the bunching technique, Annals of Nuclear Energy 27, 1199-1216 (2000).


56. G. Perret & Y. Rugama, Détermination de la fraction de neutrons retardés $\beta_{\text{eff}}$ et de la constante de décroissance prompte $\alpha$ par la technique du bruit de pile en mode courant dans le programme MUSE-4, CEA NT SPEX LPE 03-002 (2003).

57. S. Okajima et al., Summary on international benchmark experiments for effective delayed neutron fraction ($\beta_{\text{eff}}$), Progress in Nuclear Energy 41, no. 1-4, pp. 285-301 (2002).

During the progress of this work many people have crossed my path on a more or less daily basis and deserve my deepest gratitude due to their efforts to help me in the right direction. In particular I would like to thank Prof. Wacław Gudowski, who encouraged and inspired me to do my thesis work in this topic, and Prof. Janne Wallenius, who has been very supportive throughout the whole project. The supervision of Dr Vasily Arzhanov has been very useful in the moments I got lost.

This thesis would have been impossible to accomplish without the support and help from my colleagues at the Joint Institute for Power and Nuclear Research in Sosny, Belarus. Victor Bournos has not only taught me how experiments are performed, but has also allowed valuable words of wisdom to tunnel through the language barrier. The kindness and inexhaustible persistency of Yurii Fokov is highly appreciated and respected. If something is lost in the airport, Yurii is the one to ask for help. All support from Ivan Serafinovich, Sergey Mazanik, Christina Routkovskaia, Ala Kulikouskaya and Dr Anna Kiyavitskaya are not to be forgotten. If there is one second of my time in Sosny I will never forget, it includes Anna, a pencil, cotton, pure alcohol and a bad throat. Did someone else mention Inez too?

Muchas gracias are devoted to my Spanish friends Dr David Villamarín, Dr Manuel Fernández-Ordóñez and Vicente Bécares at CIEMAT. Your company during innumerable Minsk-dinners and Sosny-lunches added extra value to the time in Belarus. The kind advises of Dr Enrique González-Romero were very much appreciated. Other visiting scientists during the experiments helped in keeping my mind in positive mode: Dr Filip Kondev, Dr Bruno Merk, Varvara Glivici-Cotrută, Yuriy Romanets, Dr José Luis Sousa Neves, Prof. Jerzy Janczyszyn, Dr. Władysław Pohorecki, Pavel Gajda and Maarten Becker. Many thanks to Tania Kiyavitskaya, for her commitment to show me the Belarusian culture and cuisine, and Dr Tamara Korbut for making any nearby person cheerful.

A particularly warm appreciation is directed to my colleagues at the reactor physics division and the nuclear physics group at KTH in making an enjoyable environment with science and friendship in symbiosis. Andrei Fokátu has always had a special role in both Belarus and Sweden. Will the perfect code ever be written? Can a diamond be com-
pletely polished? The help from Janne, Vasily, Jitka and Milan in proof-reading this thesis is deeply acknowledged. Some people that have left the department for new challenges have also helped me a lot. Among them are Dr Daniel Westlén, for always giving good advice, Dr Per Seltborg who was my supervisor during the initial part of my work and Dr Alberto Talamo who set up the computing cluster on which most of my calculations were performed.

My deepest and warmest gratitude is devoted my wife Elin and my daughter Elsa. Your love and friendship is an infinite source of inspiration and energy, making life enjoyable and work pleasant. Last but not least Gudrun, Göran and Ulrika are honored for always keeping me in mind and supporting me whenever possible and Marjatta and Kjell who are helpful when needed and making the writing of this thesis less stressful.

The financial support from SKB, IP-EUROTRANS (contract no. FI6W-CT2005-516520), the Sweden-America Foundation and the support from the Swedish Institute through the Visby program is very much appreciated.