Comparative Analysis of Operation and Safety of Subcritical Nuclear Systems and Innovative Critical Reactors

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Посвящается моим близким
/ A mes proches /

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Index

Acknowledgments/Remerciements **5**

Index **7**

Abbreviations **11**

Variables **12**

1. Introduction **15**

1.1. Challenges facing nuclear power 16

1.2. Inherent safety approach and deterministic safety analysis 17

1.2.1. Inherent safety 17

1.2.2. Deterministic safety analysis 18

1.2.3. Ways to achieve the deterministic safety 19

1.3. Hybrid (subcritical) Systems 20

1.3.1. Artificially enhanced neutronics and core subcriticality 20

1.3.2. Critical reactor 21

1.3.3. Systems with independent external neutron source. Accelerator Driven Systems 22

1.3.4. Coupled Subcritical Systems 23

1.3.4.1. Delayed Enhanced Neutronics (DEN) systems 23

1.3.4.2. Accelerator Coupled Systems 24

1.3.4.3. DENNY concept 25

1.4. Molten Salt Systems 25

1.4.1. Safety-related properties of molten-salt reactors 25

1.4.2. Molten salt cores utilized for safety study 27

1.5. Subjects of the present study 27

2. Dynamics of subcritical systems. Simulation of unprotected transients in advanced MSRs 31

2.1. Introduction 31

2.2. Mathematical model for transient simulation 32

2.2.1. Reactor model 32

2.2.2. Reactor neutronics 34

2.2.3. Thermo-hydraulics and feedbacks. Fast spectrum system 36

2.2.4. Thermo-hydraulics and feedbacks. Thermal-spectrum system 37

2.3. Tool of transient simulation 38

2.4. Choice of the subcriticality level 39

2.4.1. The subcriticality level aiming to achieve the deterministic safety 39

2.4.2. Unprotected transients 40

2.4.3. Subcriticality caused by necessity of the tight neutronics enhancement 41

2.5. Reference cores for transient simulation 41

2.5.1. Fast-spectrum core 41

2.5.2. Thermal-spectrum cores 42

2.5.2.1. AMSTER-WISE-type option, Th-fuel ... 42
2.5.2.2. AMSTER-WISE-type option, U+TRU-fuel ... 42
2.6. Unprotected transients in the fast-spectrum systems ... 45
2.6.1. Unprotected Transients Over Power/Transients Over proton Current ... 45
2.6.2. Unprotected Loss Of fuel Flow ... 47
2.6.3. Unprotected Gain Of fuel Flow transients ... 47
2.6.4. Unprotected Loss Of Heat Sink transients ... 48
2.6.5. Combination of unprotected accidents: ULOF followed by UTOP/TOC ... 48
2.6.6. Combinations of unprotected accidents: UGOF followed by ULOF and, later on, by UTOP/UTOC ... 49
2.6.7. Combinations of unprotected accidents. ULOHS followed by ULOF and, later on, by UTOP/UTOC ... 50
2.7. Unprotected transients in the thermal-spectrum thorium-feed systems ... 52
2.7.1. Unprotected TOP/TOC transients ... 52
2.7.2. Unprotected LOF transients ... 53
2.7.3. Unprotected LOHS transients ... 54
2.7.4. Unprotected GOF transients ... 54
2.7.5. Summary for the thermal-spectrum Th-fuelled system ... 54
2.7.6. Unprotected transients in the systems with large margin of subcriticality ("tight" neutronics enhancement) ... 55
2.8. Unprotected transients in the thermal-spectrum uranium- and transuranics-fuelled systems ... 56
2.8.1. Unprotected TOP/TOC transients ... 56
2.8.2. Unprotected LOF transients ... 57
2.8.3. Unprotected LOHS transients ... 58
2.8.4. Unprotected GOF transients ... 58
2.8.5. Unprotected Over-Cooling (UOVC) transients ... 58
2.9. Discussion ... 59
3. Core subcriticality as a tool of safety enhancement ... 61
3.1. Foreword ... 61
3.2. Study of a DEN-system based on the AMSTER core ... 61
3.2.1. Introduction ... 61
3.2.2. Major parameters of the reference molten salt cores ... 62
3.2.3. Reactivity variations in the reference MSR ... 63
3.2.4. Simulation of Unprotected Transients ... 65
3.2.5. Conclusions ... 69
3.3. Kinetics of DEN-systems ... 70
3.3.1. Introduction ... 70
3.3.2. New kinetic features of coupled hybrid systems ... 71
3.3.3. On "artificial" group of delayed neutrons and its delay time ... 71

3.3.4. "Heat removal" model versus "delayed argument" model ... 75
3.3.5. Kinetic response to variation of source efficiency in the one-group approximation ... 76
3.3.6. Discussion. Comparison with the case of reactivity insertion ... 80
3.3.7. Conclusions ... 81
3.4. On behavior of ADS when feedback effects are degraded ... 82
3.4.1. Subcriticality level, necessary to compensate feedback degradation ... 82
3.4.2. On the grace time of ADS with a positive feedback ... 84
4. A hybrid system with intrinsic improvement of the delayed neutron fraction and of the power feedback: DENNY concept ... 87
4.1. Introduction ... 87
4.2. Generalities of hybrid systems ... 88
4.3. Accelerator-core coupling modes in the case of ACS ... 88
4.4. Principle of the operation – DENNY concept ... 90
4.5. Results and discussion ... 92
4.6. Conclusions ... 96
5. Preliminary safety study of the REBUS-3700 concept ... 97
5.1. Introduction ... 97
5.2. Goals of the concept ... 97
5.3. Description of the System ... 98
5.4. Safety Analysis ... 100
5.4.1. Reactivity coefficients and reactivity effects ... 100
5.4.2. Reactor model ... 101
5.4.3. Reactivity balance equation for the circulating-fuel system ... 101
5.4.4. Quasi-static Analysis of Unprotected Transients ... 106
5.4.4.1. Unprotected Transient Over Power ... 106
5.4.4.2. Unprotected Loss Of Flow ... 107
5.4.4.3. Unprotected Loss Of Heat Sink ... 108
5.4.4.4. Unprotected Over-Cooling ... 110
5.5. Conclusions and outlook ... 111
6. External Neutron Sources: revision of candidates ... 113
6.1. Realization restrictions for hybrid systems ... 113
6.2. Photoneutron source as a candidate for applications in subcritical systems: comparison with spallation reactions ... 114
6.2.1. Introduction ... 114
6.2.2. Analysis. Inter-comparison: photoneutron sources versus spallation sources ... 114
6.2.3. Results and conclusions ... 117
6.3. Thermo-nuclear fusion as a candidate for external neutron source in hybrid systems ... 117
6.3.1. Introduction ... 117
6.3.2. Fusion reactions ... 118
6.3.3. Neutron analysis. Inter-comparison: ADS versus a fission-fusion hybrid ... 119

6.3.4. Conclusion 123
7. Conclusions........ 125
8. References 131
9. Appendix. Solution of the kinetic equation for coupled hybrid system in one-group approximation 137
9.1. Inhour equation for the coupled hybrid system 137
9.2. Approximation of one group of delayed neutrons 138
9.2.1. The one-group kinetic equation for coupled hybrid system........ 138
9.2.2. Properties of the roots of the modified inhour equation in the one-group approximation ... 138
9.2.3. The solution of kinetic equation for coupled hybrid system in one-group approximation ... 139

Abbreviations

ACS	Accelerator Coupled System
ADS	Accelerator Driven System
AMSTER	Actinides Molten Salt TransmutER
ARTEN	ARTificially Enhanced Neutronics
DDE	Delayed Differential Equations
DBA	Design-Basis Accident
DEN	Delayed Enhanced Neutronics system
DENNY	Delayed Enhanced Neutronics with Nonlinear neutron Yield
DID	Defense In Depth
FPD	Full Power Day
LOCA	Loss Of Coolant Accident
MA	Minor Actinides
MSR	Molten Salt Reactor
MCNP	Monte-Carlo N-Particle code
MCNPX	Monte-Carlo N-Particle code with high energy X-section data files
NP	Nuclear Power
PSA	Probabilistic Safety Analysis
REBUS	fast molten salt reactor with uranium fuel (from Russian: REactor Bystryi Uranovyi Solevoi)
TRU	TRansUranic elements
UDBA	Ultimate Design-Basis Accident
ULOF	Unprotected Loss of Flow
ULOHS	Unprotected Loss of Heat Sink
UTOP	Unprotected Transient Over-Power
UTOC	Unprotected Transient Over (Proton) Current
UOVC	Unprotected Over-Cooling
WISE	Waste-free, Intrinsically Safe, and Efficient

Variables

$\mathbb{A}$	-	normalized (core) power feedback coefficient
$\mathbb{B}$	-	normalized (source) power feedback coefficient
$c_p^{(g)}$	-	specific heat capacity of graphite (thermal spectrum systems)
$c_p^{(s)}$	-	specific heat capacity of salt
C_p	-	core heat capacity
D	-	salt flow
$\mathfrak{D}$	-	transient suppression parameter
$\mathcal{D}$	-	determinant of quadratic equation
d	-	salt flow, normalized to its nominal value
E_{th}^{out}	-	thermal energy, generated in the reactor core
E_e^{out}	-	generated electric energy.
f	-	fraction of the output reactor power feeding the external neutron source
G_b	-	energy amplification coefficient of an subcritical blanket
H	-	heat transfer coefficient for the heat-exchanger
h	-	heat transfer coefficient for the heat-exchanger, normalized to its nominal value
H_{sg}	-	heat transfer coefficient from salt to graphite and vice versa
I_n	-	intensity of the external neutron source
I_p	-	intensity of the proton beam
k_{eff}	-	neutron multiplication factor of a reactor core
K_D	-	Doppler constant for the fast-spectrum system
$\tilde{K}_D$	-	Doppler constant for the thermal-spectrum system
ℓ	-	prompt neutron lifetime
ℓ_{eff}	-	effective neutron lifetime, taking into account delayed neutrons
L_c	-	core linear dimension (height)
l_c	-	length of the tube, leading from heat-exchanger to reactor core
L_h	-	heat-exchanger linear dimension (height)
l_h	-	length of the tube, leading from reactor core to heat-exchanger
m_{eff}	-	entire neutron multiplication factor of a DEN system
$M_c^{(s)}$	-	mass of graphite in the core
$M_c^{(s)}$	-	mass of salt in the core
$M_h^{(s)}$	-	mass of salt in the heat-exchanger
$M_t^{(s)}$	-	mass of salt in a tube
N_g	-	number of delayed neutron groups
P	-	core power
p	-	core power, normalized to the nominal core power
r	-	source reactivity
r_0	-	nominal subcriticality level
S	-	contribution of the external neutron source

t - time
T - temperature of salt in the core
T_g - temperature of graphite in the core (thermal spectrum systems)
T_h - temperature of salt in the heat exchanger
T_k - temperature of the heat sink (condenser)
$T^{\dagger}$ - core disruption temperature
u - ratio of the prompt neutron lifetime to the mean neutron lifetime
v_c - fuel velocity in the core
v_h - fuel velocity in the heat-exchanger
V_{core} - volume of salt in the core
V_{out} - volume of salt and out of core
V_{tube} - volume of salt in a tube
W_i - contribution of delayed neutron precursors of i^{th}-group to core power
W - contribution of delayed neutron precursors to core power in one-group approximation
W^{+} - contribution of the artificial group of delayed neutron precursors to core power
Y_n - neutron yield from spallation target
y_n - neutron yield from spallation target normalized to energy of incident charged particle
z - electric energy cost of a source neutron, i.e. electric energy spent to produce one source neutron

$\alpha_{feedback}$ - total thermal feedback coefficient
β - total fraction of delayed neutrons
β^{*} - fraction of delayed neutrons, corrected to take into account precursors' decay outside the active core region
β_i - fraction of delayed neutrons of i^{th}-group
β^{+} - fraction of the artificial group of delayed neutrons
γ, μ, σ - empiric coefficients for neutron yield from spallation target
δ_{ind} - fraction of the independent external neutron source
η_a - accelerator efficiency
η_e - efficiency transformation of thermal energy to electric energy
$\tilde{\eta}_e$ - efficiency transformation of thermal energy to electric energy in the fusion reactor
η_R - reactor efficiency
$\eta_{P\to Q}$ - local source effectiveness
φ^{*} - importance of the source neutrons
$\varkappa$ - fraction of the beam energy deposited in the system
λ - decay constant of delayed neutron precursors in one-group approximation
λ_i - decay constant of delayed neutron precursors of i^{th}-group
λ^{+} - effective decay constant for the artificial group of delayed neutrons
Λ - prompt neutron generation time
θ - difference between core temperature and condenser temperature
Θ - asymptotic reactor period
ϱ - density of the salt

ρ	-	(core) reactivity
τ_{core}	-	time interval during which circulating fuel spends inside the core
τ_{out}	-	time interval during which circulating fuel spends outside of the core
ξ_R	-	ratio of electric energy, necessary to satisfy the inherent reactor energy consumption needs, to produced electric energy

1. Introduction

Les systèmes nucléaires innovants, notamment les systèmes dédiés à la transmutation des actinides mineurs, peuvent souffrir de la dégradation significative des paramètres déterminant leur sûreté. Par exemple, la fraction de neutrons retardés peut diminuer considérablement par rapport aux réacteurs nucléaires conventionnels. Un autre problème qui peut apparaître dans de tels systèmes est la réduction des effets de contre-réaction, notamment de l'effet Doppler. Cette dégradation des paramètres de sûreté aboutit à rendre le pilotage de tels systèmes très délicat. Ainsi la sûreté accrue de systèmes innovants peut être indispensable pour leur réalisation pratique.

Les réacteurs à sel fondu, malgré quelques désavantages, sont reconnus pour être particulièrement favorables à la sûreté déterministe, appelée aussi intrinsèque ou inhérente. Ceci grâce à des propriétés intrinsèques, notamment la pression interne basse, de petites réserves de réactivité grâce au retraitement en ligne, etc. Cependant, quelques autres propriétés intrinsèques, comme un feedback positif du modérateur en graphite ou le petit effet Doppler ne sont pas toujours optimisées. De plus, une diminution de la fraction de neutrons retardés dûe à la circulation de combustible, l'insertion de la réactivité positive dans le cas d'arrêt de circulation et la solidification éventuelle du combustible peuvent poser des problèmes pour atteindre la sûreté déterministe.

Une solution innovante qui vise à traiter ces problèmes pourrait être une amélioration « artificielle » de la neutronique des systèmes nucléaires : une source externe de neutrons ajoutée à un cœur permet à ce dernier de fonctionner dans un état sous-critique (réacteurs nucléaires hybrides). Cette criticité du cœur peut être utile au moins dans deux cas : si le bilan neutronique est « serré » ou pour l'amélioration de la sûreté quand les propriétés neutroniques, déterminant la sûreté de tels systèmes sont dégradées.

En ce qui concerne l'amélioration de la sûreté, l'approche intrinsèque est un des moyens fondamentaux pour atteindre ce but. Dans le cadre de l'approche déterministe il est considéré que tous les accidents provoqués par un impact externe doivent être exclus en utilisant des propriétés intrinsèques des composants du réacteur.

Dans ce contexte, l'objectif principale de cette étude est d'examiner le rôle de la sous-criticité en tant que moyen d'améliorer la sûreté des réacteurs nucléaires innovants, notamment des réacteurs à sel fondu, dédiés à la production d'énergie et à la transmutation/incinération des déchets nucléaires. La sûreté intrinsèque est considérée comme but ultime de cette amélioration. Les aspects suivants seront examinés dans ce travail de thèse :

I. *Des études antérieures ont montré que la sous-criticité est un moyen utile d'améliorer la sûreté. Cependant ces études étaient souvent concentrées sur l'analyse de systèmes particuliers sans porter l'accent sur la sous-criticité en tant qu'une nouvelle option dans la conception des réacteurs nucléaires qui permet atteindre cet objectif. Dans notre étude nous allons essayer de montrer « Ce que » et « Comment » la sous-criticité peut apporter pour l'amélioration de la sûreté.*

II. *Il y a divers régimes de fonctionnement d'un cœur sous-critique avec une source externe : la source de neutrons peut être soit indépendante de la puissance du coeur soit couplée à la production de neutrons dans le cœur. Les conséquences pour la sûreté seront différentes pour ces deux types de couplage. Dans ce travail nous allons effectuer une comparaison directe de ces deux modes de fonctionnement. Outre cela, nous proposerons un moyen de combiner leurs avantages inhérents.*

III. *Pour atteindre l'objectif d'amélioration de la sûreté, la source externe de neutrons dans un système sous-critique doit être suffisamment performante ; cela suscite des contraintes économiques et technologiques. Dans ce contexte une analyse des différentes sources de neutrons (photo-nucléaires, thermonucléaire) est effectuée.*

IV. *L'extension de l'approche déterministe à des systèmes hybrides. (Cette approche, initialement proposée pour l'analyse de réacteurs critiques, ne prend pas en compte les nouvelles opportunités offertes par les systèmes hybrides).*

V. *L'interférence entre les exigences pour les paramètres du cœur et celles pour les paramètres de la source externe de neutrons : l'identification des cœurs qui permettent d'atteindre la sûreté déterministe dans le contexte des aspects I à IV. (Même si la sous-criticité est un moyen utile d'améliorer de la sûreté, son rôle reste auxiliaire. Ainsi, pour répondre aux exigences de la sûreté intrinsèque, les propriétés du cœur doivent être adoptés).*

Ce travail est organisé de la façon suivante : chaque chapitre commence par une introduction indépendante suivie d'une description de la méthodologie utilisée et éventuellement de la description d'un modèle. Après cela, les résultats principaux et leurs analyses seront présentés. Enfin, des conclusions sont formulées soit à la fin de chaque section (partie) ou/et à la fin de chaque chapitre. En outre, chaque chapitre peut être considéré comme une étude indépendante dont les objectifs et les résultats sont à chaque fois présentés. Cependant, les sujets couverts par ce travail appartiennent bien à une seule thématique : cette étude présente une analyse du potentiel de la sûreté déterministe des systèmes critiques et sous-critiques, notamment des réacteurs nucléaires à sel fondu, ayant pour l'objectif l'estimation du rôle de l'amélioration artificielle de la neutronique.

1.1. Challenges facing nuclear power

The current stagnation of the Nuclear Power (NP) is evident, but one may anticipate that this tendency has temporal character. Indeed, actually there is still no real alternative for replacement of traditional fuels, principally hydrocarbons, which will be exhausted in the near future (Bauquis, 2004). Moreover, pollution of the environment by combustion products of fossil fuels, mainly by CO_2, becomes extremely menacing for the global-scale ecological equilibrium.

Even in these "favorable" conditions, the future nuclear technology has to meet some exigent requirements to be publicly acceptable and to take its own place in the pattern of the world's energy production. Nowadays these requirements are the subject of the consensus of NP specialists all over the World. These requirements for the future nuclear technology (so-called nuclear reactors of the 4-th generation) are explicitly formulated in the framework of the International Generation IV Forum (U.S. DOE, 2000) and Russian BREST reactor conceptions (Adamov, Orlov *et al.* 2001). To qualify for this role such a technology has to conform the following criteria:

(I) *Safety/reliability.* The future reactor design has to supply inherent safety and fault tolerance. It is imperative to exclude severe accidents with large radiation releases under conditions of any equipment failure, external impact or human error by using natural properties in nuclear reactor and their components.

(II) *Long term sustainability.* A quasi-unlimited availability of fuel resources may be achieved due to the use of plutonium from spent nuclear fuel, through efficient use of natural uranium and, subsequently, thorium. The plant is to be designed to deliver a maximum possible amount of useful energy.

(III) *Minimal nuclear waste.* The radiotoxicity of the waste issued from nuclear industry has to be minimized, ideally to the level comparable with the natural radiotoxicity of the extracted uranium/thorium (principle of preserving of the natural radiation balance).

(IV) *Economic competitiveness.* It should be economically effective when compared with other competitors, in terms of low costs and fuel availability (e.g., fuel breeding), high efficiency of thermodynamic cycle, etc.

(V) *Proliferation resistance.* Future fuel cycles should aim to minimizing the inventories and accessibility of weapon-useable materials.

Above issues are extremely interconnected and, consequently, have to be treated taking into account complexity of the entire structure of the future NP. In this study we focus our attention on safety aspect as the enhanced safety remains one of the decisive and desirable properties of prospective NP. So-called *inherent safety* is one of the fundamental approaches to achieve this goal (Wade, 2000).

1.2. Inherent safety approach and deterministic safety analysis

1.2.1. Inherent safety

Strategies of minimization of risks related to nuclear energy production have been developed and refined over many years for conventional (critical) reactors. The release of radioactivity caused by reactor core disruptions is considered as the most significant among all other risks.

Traditionally, minimization of risks is based on the *defense in depth* (DID) safety principle such that any single failure will not defeat the strategies for meeting basic safety functions[a]. Multiple barriers (fuel cladding, primary coolant boundary and reactor containment building) are used to prevent a radiation release even in accidental conditions (e.g. Ref. Libmann, 1996). Highly reliable, diverse and redundant systems for controlling and terminating the chain reaction and for fission and/or decay heat removal are provided. Finally, rigorous maintenance, training and certification of in-plant personnel are used to minimize the possibility of human errors.

This traditional approach is able to minimize risks, nevertheless, it can not, generally, eliminate them, what particularly concerns the most severe unacceptable accidents which probability remains non-zeroed and very uncertain due to the reasons discussed below.

[a] These functions include (Wade, 2000): the containment of the radioactive materials within controlled space to protect humans from radiation damage; the reliable heat evacuation from the fissioning medium as well as decay heat evacuation from the fission products and transuranics in the fuel for all times subsequent to the fission event; the maintain of a balance of neutron production and destruction rate in the fissioning fuel, etc.

As for heavy accidents, the damage related to these events could be unacceptably high. Therefore, their risk has to be either precisely defined and accepted or, even better, it has to be zeroed. In this case, all depends on the safety means being used for reactor safety. When only "active engineering" and organizing safety means are used then the problem of their reliability becomes a decisive one. In practice, for the quantitative assessment of risks the Probabilistic Safety Analysis (PSA) is the only suitable instrument to be used. However, being applied to reactor accident analysis, the PSA has an important shortcoming in assessment of real risks: as discussed in Refs. (Slessarev, 1992; Adamov, Orlov *et al.* 2001; Slessarev *et al.*, 2004) there is neither sufficient operating experience nor convincing theoretical data to support it. This becomes particularly dangerous if NP is going to be widely used in the future.

Therefore, it seems that the large-scale NP may not include those reactor concepts which safety (particularly, protection against severe accidents) is based only on the PSA. Nowadays an alternative *deterministic* (also, *inherent or natural*) *safety* approach is proposed (Orlov and Slessarev, 1988; Slessarev, 1992). According to this approach declared by A. Weinberg (1984) just after the Three-Mile-Island accident, the intrinsic safety principle is one of fundamental means to build the deterministically safe NP: all *severe* accidents (e.g., prompt criticality excursion, loss of coolant, fires, steam and hydrogen explosions) "should be deterministically excluded under conditions of any human errors, failures of or damage to equipment and safety barriers, *by the use of intrinsic physical and chemical properties and behavior of the fuel, coolant and other reactor components*" (from Ref.: Adamov, Orlov *et al.*, 2001, p. 21). In other words, the safety functions have to be achieved by exploiting the natural laws of physics and inherent characteristics of the nuclear installation (Wade, 2000).

Only those designs where all unprotected heavy accidents do not cause unacceptable core damages could be considered *deterministically safe.* The deterministic level of safety seems to be ideal (maximal) respecting safety certainty to be achieved.

In this paradigm the above approaches may be considered as complementary means: the deterministic safety approach has to be applied to exclude *severe* accidents, whereas DID has to be used to minimize the residual risks, which may be evaluated applying the PSA methods.

1.2.2. Deterministic safety analysis

The deterministic safety requirement implies that *all* potential accidents in natural safe reactor caused both by internal and external impacts are treated as Design-Basis Accidents[b] (DBA). The Ultimate Design-Basis Accident[c] (UDBA) should not lead to fuel failure and radioactive releases such that would require evacuation of people from the territory around the plant (Adamov, Orlov *et al.*, 2001).

[b] Design Basis Accidents (DBA) are postulated accidents to which a nuclear plant, its systems, structures and components must be designed and built to withstand loads during accident conditions without releasing the harmful amounts of radioactive materials to the outside environment.

[c] Ultimate Design-Basis Accident (UDBA) is the accident which covers any event resulting from human errors or multiple failures of equipment, including loss of forced cooling, failure of the scram function, insertion of full reactivity margin, damage to outer barriers (containment and reactor vessel).

In other words, deterministic safety analysis is based on study of all possible (with respect to nature laws) unprotected events independently on their probability, as well as their multiple combinations. All severe accident scenarios has to be considered as *unprotected* events where failure of all active safety means and all possible human errors does not cause the reactor core disruption.

Therefore, the approval of the required safety is somewhat simplified (when compared with conventional PSA techniques) as an analysis might be limited (Slessarev, 1992) by consideration of:

- all important and technically possible initiators of accidents (including human errors);
- the deterministic analysis of only principal accident scenarios, which might lead to core disruption; there is no needs to consider the "post-disruption events" – they have to be deterministically excluded – this leads to important economy in R&D.

Finally, analysis of hypothetical (non-credible) accidents is an optional job performed to obtain ultimate estimates (Adamov, Orlov *et al.*, 2001).

A system is considered to be *deterministically safe* if all unprotected events do not break through the "domain" of the acceptable parameters (or the domain of *viability*). It means that temperatures, pressures, etc. during transients will not lead to the destruction of the system.

1.2.3. Ways to achieve the deterministic safety

To achieve the deterministic safety the system has to be designed in an appropriate way using the intrinsic safety features: fuel, coolant, structure materials with the sufficiently large margins to loose their basic properties; passive safety means based on nature laws and corresponding designs. Some measures, aiming to reach this goal are formulated, for example, in Refs. (Wade, 1986; Slessarev, 1992; Adamov, Wade, 2000; Orlov *et. al.*, 2001). They include (among others) the following recommendations:

- favorable inherent reactivity feedbacks to keep heat production and removal in balance;
- large margins to damage temperatures (i.e. large viability domain);
- passive control and cooling features, feedbacks governed by a large negative temperature coefficient and a high level of natural circulation of the coolant to prevent dangerous temperature growth under off-normal conditions;
- minimal reactivity margin (reactivity reserves necessary to compensate in-core reactivity effects), ideally smaller than the fraction of delayed neutrons; this excludes a fast runaway of reactor power under condition of any erroneous actions or failures in the reactivity control system;
- minimal accumulated non-nuclear energy: excluded possibility of fires and of explosions, a chemically inert coolant with high boiling temperature, low internal pressure.

The above recommendations were proposed for traditional (critical) reactors. The use of so-called *hybrid reactor* systems, where a *subcritical core* is associated with an *external* intensive *neutron source* (e.g. Accelerator-Driven Systems, fusion-fission hybrids), may offer a new opportunity to reach deterministic safety even for the systems for which the inherent safety-related drawbacks do not permit to attain this goal in critical configuration.

1.3. Hybrid (subcritical) Systems

1.3.1. Artificially enhanced neutronics and core subcriticality

Many problems of the current NP could be solved if more neutrons were available per every fission taking place in the reactor fuel or if one might affect the temporal characteristic of the fission neutron yield. A supplementary neutron yield per fission would allow the use of natural fuels with "tight" neutronics such as natural uranium (U) and thorium (Th) without isotopic enrichment. It would also simplify or eliminate the necessity of fuel reprocessing, expand fuel reserves enormously and improve resistance to proliferation (Salvatores *et al.*, 1999). Safety could also be enhanced with the help of supplementary neutrons, if these neutrons were delayed with respect to prompt fission neutrons (Salvatores *et al.*, 1995, Gandini *et al.*, 1999; Slessarev *et al.*, 1999; Gandini *et al.*, 2000). Taking into account that the fraction of delayed neutrons in a critical reactor, and particularly in actinide transmutors, is small, some supplementary delayed neutrons would be sufficient to enhance safety drastically.

Hybrid reactor systems offer some promising options in resolving the current problems of nuclear power: long-lived radioactive wastes, safety enhancement etc. In these hybrid concepts the subcritical core may play either principal or subsidiary role. Thus in the case of the fusion-fission hybrids a subcritical blanket is foreseen for utilization on fusion neutrons and therefore its role may be considered as an auxiliary; whereas in the case of the ADS the external neutron source plays supporting role. In this context Slessarev and Bokov (2004) introduced the notion of so-called systems with ARTificially Enhanced Neutronics (ARTEN) as the *category of hybrids where the external neutron source is foreseen with the objective to artificially improve the neutron yield and/or its characteristics (e.g. time and energy spectra).*

ARTEN systems are quite attractive with respect to fuel cycle simplification, expanding fuel reserves, non-proliferation and long-lived waste reduction. Preliminary studies (Slessarev *et al.*, 1999; Slessarev *et al.*, 2004) have demonstrated that a NP park, based on ARTEN-type systems, could be developed according to the required rate of energy production. In addition, it would last for thousands of years until all reserves of plutonium (Pu), uranium (U) and natural thorium (Th) will be exhausted. The subsequent "shrinkage" stage of NP will require fuel reprocessing and it will lead to some limited amount of wastes with acceptable radiotoxicity. On the other hand, the safety potential of such ARTEN systems is not sufficiently known.

ARTEN-systems can be considered as potentially capable resolving many of the principal issues associated with prospective NP. A production of supplementary neutrons can be made in hybrid systems where fission reactions take place in a subcritical core simultaneously with supplementary external neutron production via spallation reactions, thermo-nuclear fusion or photonuclear reactions, etc. Certainly, the net energy output in ARTEN-type system will decrease, however, it still might be acceptable, in particular, when the supplementary neutrons provide this system with some improved properties (e.g., deterministic safety).

As will be demonstrated in present work, from the point of view of safety issues in the framework of the deterministic safety approach, ARTEN-type systems may be divided into two categories based either

- on an *independent* external neutron source or
- on a "dependent" external neutron source, i.e. *coupled* with the core neutronics.

The term "*independent external neutron source*" in our considerations means that there is no *intrinsic* and *imminent* dependence of the neutron production in the external neutron source on fission rate in the subcritical blanket (though the intensity of the external source may be eventually corrected depending on core power, for example, to assure constant reactor power), whereas the term "*coupled system*" signifies an inverse situation, i.e. that the intensity of the external neutron source depends *intrinsically* on the core power.

Below we discuss particularities of each system in detail.

1.3.2. Critical reactor

In the present study critical cores with its parameters (dimensions, nominal power, coolant flow etc.) and properties (in-core feedbacks, heat capacity etc.) will play the role of reference and of starting point for our analysis. Moreover, the critical core may be considered as a limit case of the subcritical system, where the subcriticality level is quite small (quantitatively, when compared to delayed neutron fraction β). Indeed, it is known that commercial critical reactors are also supported by a very small (not intense) external neutron source. The difference between critical reactors and hybrid systems lies in the intensity of the source, i.e. its capability to drive the system power, and in particularity of "driving mechanisms". In contrast to ARTEN systems, in critical reactors a small, independent and time-constant neutron source is unable to drive/affect the core power.

One may explain the distinction of different realizations of ARTEN systems from corresponding critical system utilizing an *energy transfer diagram* schematically presented schematically in Figure 1. The thermal energy E_{th}^{out}, generated in the *reactor core* is transmitted by the *heat transfer system* to the *electric energy production device* (in our studies it is assumed that there is no heat losses in the heat transfer system). The energy production device (e.g., turbine) transforms the heat to electricity with some transformation efficiency η_e and generated electric energy $E_e^{out} = \eta_e E_{th}^{out}$ goes to the *power grid*.

In the neutron multiplying blanket of the critical reactor the *self-sustaining chain reaction* takes place. Consequently, this system needs no external energy supply to maintain this reaction, except some energy necessary to keep reactor running (i.e. to feed heat transfer system, safety system etc.). One may take into account the decrease of net energy output by introducing the entire reactor efficiency

$$\eta_R^{(critical)} = \eta_e \left(1 - \xi_R\right), \tag{1.1}$$

where the factor ξ_R denotes the ratio of the energy, necessary to satisfy the inherent reactor energy consumption needs, to the total produced electrical energy. Obviously, in the steady state conditions we may pass on from energy to power: $E_e^{out} \to \overline{P}_e^{out}$, $E_{th}^{out} \to \overline{P}_{th}^{out}$, $\overline{P}_e^{out} = \eta_e \overline{P}_{th}^{out}$, i.e. formulate above considerations in terms of steady-state power $\overline{P}^{out}$.

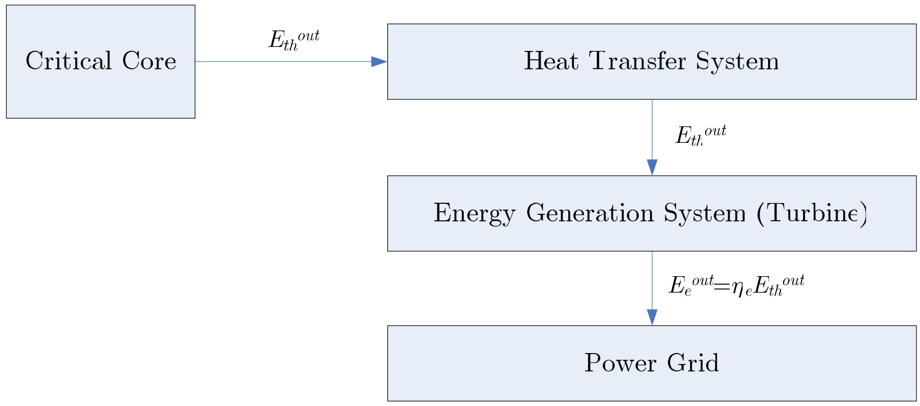

Figure 1. Energy (power) transfer diagram for the critical system.

1.3.3. Systems with independent external neutron source. Accelerator Driven Systems

Accelerator-Driven Systems (ADS) may be one of realistic types of subcritical ARTEN systems (see Refs.: Takahashi, 1995; Rubbia *et al.*, 1995; Bowman, 1995). In definition by Wade (2000), "the term ADS comprehensively includes all non-self sustaining fissioning neutron multiplying assemblies, which are driven by external neutron source provided by a charged particle accelerator and a neutron production target". The ADS concepts connive that the power feeding the external neutron source, namely the proton accelerator, originates from the external power grid. Therefore, Accelerator-Driven System may be considered as a particular case of ARTEN system with "independent-source".

The energy transfer diagram for ADS is evidently different from this one for the critical system (see Figure 2). As noted above, in this case the reactor core is subcritical, consequently the self-sustained chain reaction is not possible. Therefore the subcritical reactor blanket operates as *neutron and energy amplifier*: the output thermal energy E_{th}^{out} of the core is the energy E_b of the charged particle beam, originated from an *accelerator*, times the energy amplification coefficient G_b. The energy E_e^{in}, needed for accelerator to create and to accelerate charged particles, is issued from the power grid. This energy depends on the accelerator efficiency η_a: $E_e^{in} = E_b / \eta_a$. Finally, one can introduce the fraction f of produced electrical energy serving to feed the accelerator:

$$f = \frac{E_e^{in}}{E_e^{out}} = \frac{1}{\eta_e \eta_a G_b}. \tag{1.2}$$

Apparently, the entire reactor efficiency decreases, when compared with corresponding critical reactor, down to the value:

$$\eta_R^{ADS} = \eta_e \left(1 - \xi_R - f\right). \tag{1.3}$$

In coming Chapters we will return to the Eq. (1.2), namely we will write the explicit expression for the energy amplification coefficient G_b.

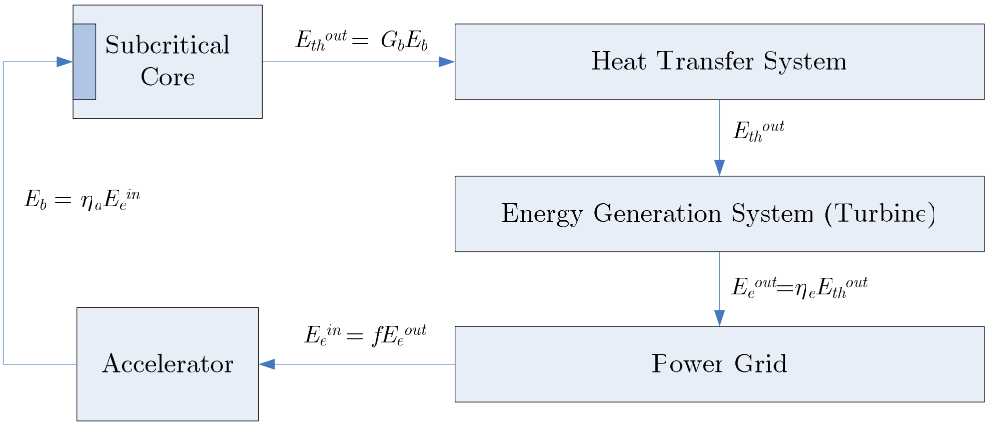

Figure 2. Energy transfer diagram for an Accelerator-Driven System (ADS).

As for the safety of the ADS, a large subcriticality level ($k_{eff} \cong 0.95 \div 0.97$) mitigates the negative consequences of the degradation of safety parameters. However, such significant subcriticality level requires powerful and expensive particle accelerator. Moreover, a large subcriticality decreases the beneficial influence of the in-core thermal feedbacks. Hence, a choice of the core subcriticality level becomes the compromise between the benefits offered by the subcriticality and the possible drawbacks that this subcriticality may yield.

1.3.4. Coupled Subcritical Systems

1.3.4.1. Delayed Enhanced Neutronics (DEN) systems

Another way, oriented to artificial delayed neutron production (Delayed Enhanced Neutronics concept or briefly DEN), may be realized by the direct transformation of part of the fission energy into electricity and, finally, to external neutrons using a special neutron production mechanism (spallation, bremsstrahlung-photonuclear, nuclear fusion, etc.). These supplementary neutrons can be naturally or artificially “delayed” when compared with prompt neutrons of fission. Such a system with subcritical core operates with increased total fraction of delayed neutrons. This fraction consists of the delayed neutrons of two kinds:

– originating from fission product decay (so-called “natural” delayed neutrons as in conventional critical reactors) and
– originating from a supplementary neutron production mechanism with their particular neutron spectrum and spatial characteristics. Unlike the first kind, their delay depends on the engineering design of the installation and can be optimized by the designer. These neutrons can be considered as a group of “artificially” created delayed neutrons.

Without these supplementary neutrons, reactor core would remain subcritical. Together with these neutrons, cores become critical, i.e. the external neutron source produces a quantity of neutrons, necessary to sustain the chain reaction in the core. Hence, a DEN-system (Slessarev and Bokov, 2004) can be referred to "coupled" ARTEN systems, operating in a critical/self-sustaining[d] mode and therefore achieving both goals discussed earlier in this work: increasing the *total neutron yield* per one fission as well as the *delayed neutron yield*. The physical background of this concept is rather simple: an intermediate process "hides" neutrons (of some neutron generation) temporarily to recover them later. This allows the neutron life time be artificially increased and, in this way slowing-down dangerous transients.

This particular property of the DEN system, if compared with the conventional critical reactors, can improve the reactor dynamics significantly. Moreover, the DEN system operates in a critical mode and consequently unlike the ADS, it takes advantage of favorable temperature feedbacks, existing in these systems.

The dynamics peculiarity of the DEN-systems (Slessarev *et al.*, 1999) is the following: the supplementary neutron production depends upon reactor thermo-hydraulics. The spallation neutron delay would have similar time-characteristics as the thermo-hydraulic processes in a nuclear power plant. Hence, in the case of a thermo-hydraulic unprotected transient or any failure of the heat removal mechanism, DEN self-regulates the power level in similar rates as the thermo-hydraulic processes. Unlike DEN, ADS power is much less sensitive to the thermo-hydraulic transients and power "shut-down" via proton-beam cut-off is the only mechanism to protect against these accidents.

Below we discuss in detail one of possible realizations of the DEN concept – so-called Accelerator Coupled System (ACS).

1.3.4.2. Accelerator Coupled Systems

Gandini *et al.* (2000) proposed to use a fraction f of the output reactor power to feed the external neutron source (accelerator) with the goal to increase the coupling between ADS neutronics and thermal-hydraulics. This fraction is fixed at any instant of time and may be adjusted in order to compensate the possible reactivity swing. Here we may note that in comparison to a critical reactor a new opportunity appears: the reactor power may be controlled not only by control rods, but also (via external neutron source) by the fraction f. This particularity is also addressed in the present study.

As proposed by Gandini *et al.* (2000), the practical realization of this ACS-coupling would consist in the splitting of the secondary coolant loop into production one and the coupling one, generating the electricity feeding the external neutron source (see Figure 3). In this case the neutron production in the core and in the external source becomes *intrinsically coupled*.

The energy transfer diagram for ACS is similar to this one for the ADS (see Figure 2). The only difference is that the power feeding the accelerator does not originate from independent power grid, but from the energy producing system of the same reactor. Obviously, Eqs. (1.2)-(1.3) remain valid in the case of ACS.

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[d] The common agreed terminology does not exist yet. The term "self-sustaining" is preferable, as the term "critical" is reserved for the reactor blankets with self-sustaining fission reaction. But we would like to stress the similarity between conventional critical reactors and DEN-systems. For that reason we make use of notation "operates in critical mode" to characterize DEN.

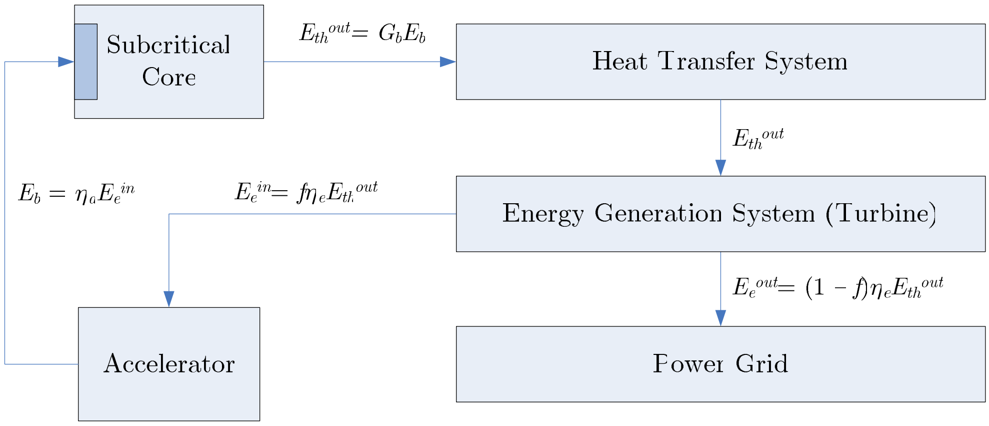


Figure 3. Energy transfer diagram for an Accelerator Coupled System.

1.3.4.3. DENNY concept

In this work we will discuss one more type of coupled systems – so-called DENNY-system (Delayed Enhanced Neutronics with Non-linear neutron Yield), proposed in Refs. (Bokov *et. al.*, 2004a,b). The DENNY system is a further development of the ACS concept. This concept aims to fusion the inherent safety-related advantages of Accelerator-Driven Systems and of Accelerator Coupled Systems. As ACS, DENNY operates in the critical regime with subcritical core, but the mode of coupling is changed – it is proposed to apply energy of incident charged particles as the coupling parameter instead of the beam current. Details and advantages (from the point of view of reactor safety) of such a coupling are discussed in Chapter 4.

As will be demonstrated in subsequent Chapters, the core subcriticality is an attractive option for the safety amelioration. Nevertheless, to achieve the deterministic safety level, the reactor core properties have to be suitable, i.e. the system has to meet as good as possible the conditions, mentioned in Subsection 1.2.3. Among others, molten salt reactors are good candidates to achieve this goal. In the next section we discuss this kind of nuclear systems in more detail.

1.4. Molten Salt Systems

Some radically innovative concepts of prospective NP (Gat, 1987; Slessarev *et al.*, 2001; Lecarpentier, 2001; Nuttin, 2002 and many earlier works) use mobile (circulating) fuel. Mobile fuel is rather attractive to be applied in conventional critical cores; however, sometimes its deterministic safety potential is limited.

1.4.1. Safety-related properties of molten-salt reactors

As is demonstrated by many authors, the Molten Salt Reactors (MSR) have the extraordinary potential to reach natural safety (Blinkin and Novikov, 1978; Gat, 1987; Novikov and Ignatiev, 1990; Gat and Dodds, 1997). Indeed, many recommendations listed in Section 1.2.3 may be easily realized in Molten Salt Systems due to their inherent properties. Let us identify and summarize these safety-related properties of MSR.

In general, low non-nuclear energy is present in the molten salt reactors, i.e., low internal pressure in the reactor core as well as chemical inertness of the salt with respect to air or moisture results in absence of fire hazard or of explosion hazard.

As the molten salt fuel is in the same time the heat carrier, the Loss Of Coolant Accident (LOCA) has no sense; nevertheless, some means for the residual heat evacuation have to be foreseen in the reactor design.

Excess reactivity and reactivity margin may be quite low, given that the reactivity swing due to burn-up may be minimized due to breading and fuel (quasi-) continues reprocessing.

The possibility of the emergency fuel draining may be foreseen in design. In the case of accident the fuel may be drained by gravity into dump tanks that are assured to retain subcriticality and have sufficient natural cooling to assure cooling of the fuel (Gat and Dodds, 1997; Lecarpentier, 2001).

As fuel is molten in MSR, hence the core melting accident has no sense. The safety criteria are no more based on, for example, fuel melting, but on the salt properties. Nevertheless, the main danger of this kind of accident in solid-fuel systems is a potential re-criticality. This menace would be excluded in MSR if the salt remained homogeneous in any circumstances. It means that the chemical stability of the salt has to be guaranteed in all range of functioning and accident conditions.

Despite above beneficial properties of MSRs, there are some concerns with respect of their safety. Hereafter we mention the most pertinent of them.

A drawback of mobile fuel system is partial loss of delayed neutrons. Since fuel circulates and spends some time beyond active core region, as a consequence, precursors of delayed neutrons decay partially outside of core without contributing to neutron balance. In addition, any fuel stop or even decrease of its flow in the core leads to the reactivity insertion. Therefore, this particularity of mobile fuel systems is twice penalizing for their safety.

Distinctiveness of mobile fuel reactors and, in particular, molten salt reactors is the absence of the first containment barrier (fuel cladding). This can be considered as a disadvantage of these reactors in respect to safety requirements. However, another inherit property of liquid fuel system – on-line reprocessing – is able (to some extent) to compensate this drawback. In fact, the on-line reprocessing removes the gaseous and volatile part of the source term being the inventory of radioisotopes in the reactor and potentially available for dispersion to the environment. This part of radioactive isotopes is most likely to be dispersed to atmosphere when there is breach of containment. Anyway, multiple barriers can by designed to compensate for this lack of first safety barrier.

Finally, in some cases, MSR cores may have an unacceptable from the safety point of view positive feedback, mainly due to contribution of the graphite moderator component. (Lecarpentier, 2001; Lecarpentier *et al.*, 2003). In this case such a reactor is unstable with respect to power fluctuations, and therefore its control becomes problematic.

Subcritical (hybrid) MSR might be one of potential solutions to above safety-related problems, especially if it does not require cumbersome design and if it does lead to high economic penalties.

1.4.2. Molten salt cores utilized for safety study

In the present study diverse molten salt reactor cores have been chosen as reference for analysis. They cover different strategies for fuel cycles and correspond to both fast and thermal spectrum systems. *Fast spectrum* versions of molten salt homogeneous cores are similar to the concepts WISE (Slessarev *et al.*, 2001), TASSE (Salvatores *et al.*, 1999) and REBUS (Mourogov and Bokov, 2004). *Thermal spectrum* molten salt cores with graphite moderator are similar to the concepts: AMSTER (Vergnes *et al.*, 2000; Lecarpentier, 2001), TASSE (Salvatores *et al.*, 1999) or TIER (Bowman, 1999), RSF (Nuttin, 2002).

1.5. Subjects of the present study

Now we may summarize all above aspects in order to formulate the problems under consideration, the goals and the subjects of the present study (see Figure 4 for detail).

It is known, that **advanced nuclear systems** and, in particular, systems devoted to the Minor Actinide (MA) transmutation, may suffer from the significant degradation of safety characteristics. For example, such important parameter as fraction of delayed neutrons may decrease by several times compared to the conventional nuclear reactors. Another serious problem, arising in such systems, is the reduction of in-core feedbacks effects, namely Doppler-effect - the fastest and the most important temperature feedback effect in the reactor core. This degradation of safety properties makes the control of such systems rather delicate. Therefore, the ensuring of the safe operation of these innovative systems may become crucial for the practical realization of these novel concepts.

The **molten salt reactors**, despite some safety-related drawbacks, are recognized to be particularly convenient for the natural safety due to its intrinsic properties, namely low internal pressure, low fuel-coolant inflammability, small reactivity margin due to on-line reprocessing, etc. However, other inherent properties, such as a positive feedback effect of the graphite components of core and a small Doppler-effect, are not yet optimized. Besides, a number of other physical properties, in particular, a partial loss of the delayed neutron fraction due to fuel circulation, a positive reactivity insertion in the case of circulation stop and the fuel solidification, may raise some problems on the way to the deterministic safety. The insertion of some quantities of MA in the core definitely would degrade the situation further.

A novel solution aiming to handle the above problem may be the artificial enhancement of system neutronics: an external neutron source added to the core permits the system to operate with subcritical core (so-called **hybrid nuclear reactors**). Core subcriticality can be helpful at least in two cases: either for neutronics enhancement of cores when neutron balance is too tight or/and for safety improvement purposes when feedback effects or other physical parameters are degraded.

When the safety amelioration is discussed, the **intrinsic safety** principle is one of the fundamental means to reach it. In the framework of the deterministic safety approach is assumed that, all potential accidents provoked by either external or internal impact must be excluded utilizing inherent properties of reactor components.

In this context, **the principal goal of this study is to investigate the role of core subcriticality for safety enhancement of advanced nuclear systems, in particular molten salt reactors, devoted to both energy production and waste incineration/transmutation. The inherent safety is considered as ultimate goal of this safety enhancement**.

The following aspects are addressed in the present work.

I. Earlier studies have demonstrated that the subcriticality is a helpful mean of safety amelioration. However, these studies were often focused on analysis of a particular system without marking out the core subcriticality as new option in reactor design allowing to reach this objective. *In the present study we will try to qualify «What» and «How» the core subcriticality may bring to safety improvement.*

II. There are diverse regimes of operation of subcritical core with external neutron source: this neutron source may be either independent on core power (e.g. ADS) or coupled to the neutron production in the core (e.g. ACS). It is obvious that safety issues vary for these different regimes. *In the present work we carry out a direct inter-comparison of subcritical systems operating in different regimes. Moreover, we will propose the way to combine their inherent advantages.*

III. To qualify for this role of a safety enhancement tool, the external neutron source has to be suitably efficient, what leads to some constraints of both economical and technological nature. Hence, *an analysis of the external neutron source characteristics is addressed in this context.*

Finally, other important problems are also discussed in the present work, however additional studies are necessary in these cases (see below):

IV. *Expansion of the deterministic safety approach to hybrid systems* (this approach was initially proposed for analysis of critical reactors, but it did not take into account new opportunities offered by hybrid systems);

V. *Interference of the requirements on core parameters with parameters of the external neutron source, i.e. identification of the class of reactor cores for which the core subcriticality permits to attain the deterministic safety level in the context of above aspects I-IV* (Even if subcriticality is a helpful tool of safety amelioration, its role remains auxiliary. Hence, to meet the deterministic safety requirements, the core properties have also to be convenient).

This thesis work is organized as follows. Each chapter starts with an independent introduction, which is followed by a required methodology and model description. Further, the major results and their analysis are provided. Finally, some conclusions are drawn either at the end of some sub-sections or at the very end of the chapter. Therefore, each chapter can be viewed as an independent study with its problem formulation, goals and results.

On the other hand, the subjects covered in this work can be well described by a single thematic. In brief, this study presents an analysis of the deterministic safety potential of critical and ARTEN systems, in particular systems with molten salt fuel, with the goal to assess the role of artificially enhanced neutronics.

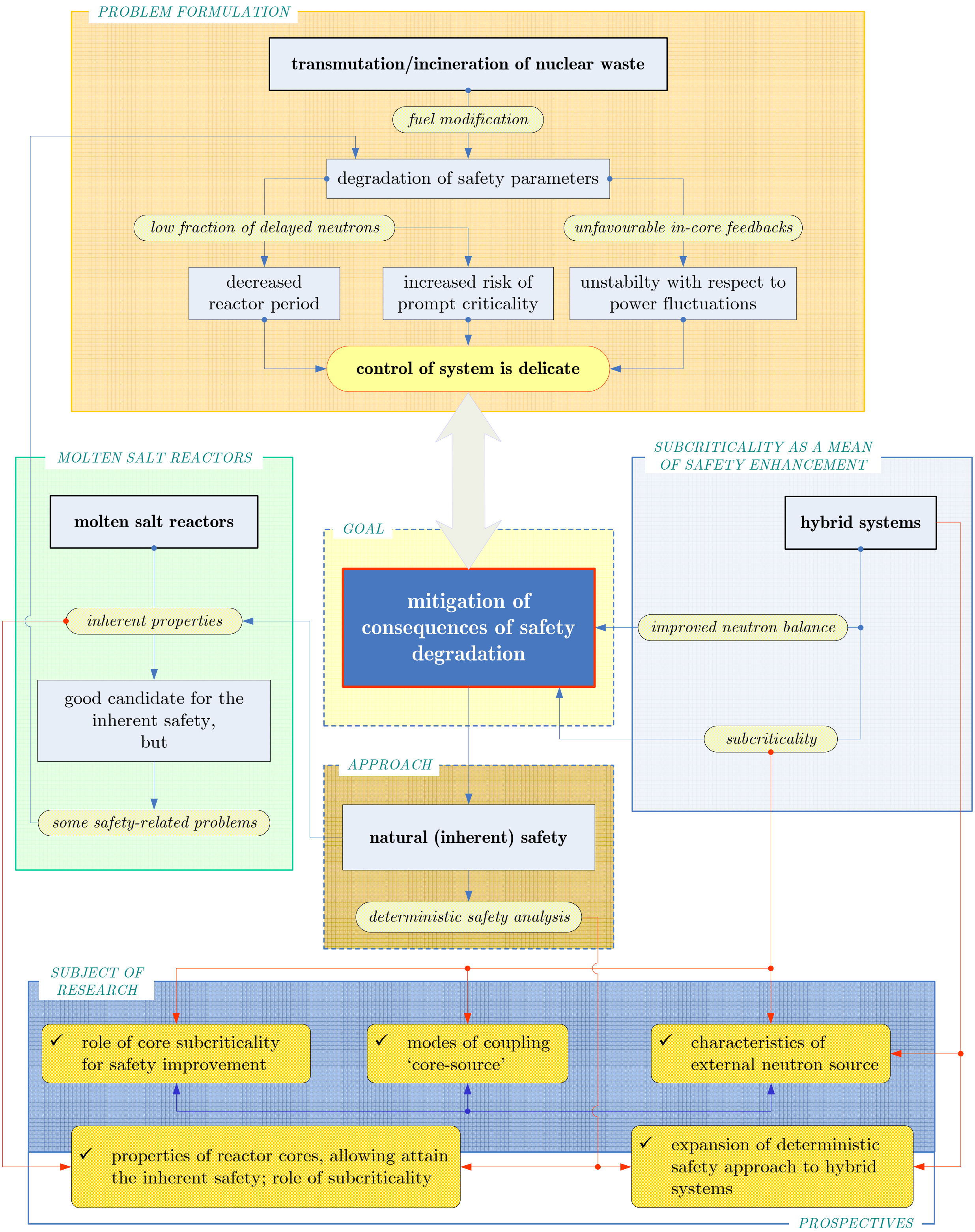


Figure 4. A schematic presentation of the problem formulation, the goals, the subjects of research and the approaches applied in the present study.

2. Dynamics of subcritical systems. Simulation of unprotected transients in advanced MSRs

Résumé *– Dans ce Chapitre une analyse comparative du potentiel de sûreté déterministe des systèmes à sel fondus innovants avec des cœurs critiques et sous-critiques est présentée pour des systèmes rapides et thermiques. Cette analyse inclut deux aspects : le choix du niveau initial de sous-criticité et l'étude des transitoires non protégés qui ont été simulés à l'aide d'un schéma couplant la cinétique point et la thermo-hydraulique. Une étude des systèmes hybrides avec un cœur à sel fondu support thorium (coefficient de contre-réaction légèrement positif) et support uranium (coefficient de contre-réaction négatif) est effectuée. Cette étude inclut une analyse des réserves de réactivité dans ce système, la simulation de transitoires non protégés pour différents niveaux de sous-criticité (y compris le niveau de sous-criticité égal à zéro – système critique). Cette étude a démontré que la sous-criticité permet d'adoucir les transitoires, d'augmenter le délai de grâce, même si le coefficient de contre-réaction est défavorable. Les résultats montrent que même un petit niveau de sous-criticité (2-3 dollars) peut significativement améliorer la sûreté.*

2.1. Introduction

This Chapter is devoted to simulation and analysis of unprotected transients in the critical and corresponding subcritical systems (both for ADS and for the DEN-systems). The goal of this study is to identify and to illustrate the principal features of the dynamics of subcritical systems as well as the role of the core subcriticality. It is worth to stress from the very beginning that the study, carried in this Chapter, has the objective not to prove the safety of particular MSR concepts, but to demonstrate general tendencies related to subcriticality.

MSRs with both fast and thermal spectra are considered. The "generalized" cores chosen for analysis correspond to fast spectrum version of molten-salt homogeneous cores similar to the concepts WISE (Slessarev *et al.*, 2001; Slessarev *et al.*, 2004) and TASSE (Salvatores *et al.*, 1999) and thermal spectrum molten salt cores with graphite moderator similar to the concepts AMSTER (Vergnes *et al.*, 2000; Lecarpentier, 2001), TASSE (Salvatores *et al.*, 1999) or TIER (Bowman, 1999). The integral parameters (dimensions, feedback coefficients, etc.) of these systems were addressed for transient simulation.

Several configurations will be studied in this section: critical reactors (it will be denoted "CRT"); subcritical cores with an independent neutron source (ADS); subcritical cores with coupled fission-spallation processes (DEN); and a combined system, where a part of the external neutron source has an independent energy source and another part is coupled with fission (denoted in this Chapter as "HYB"). A HYB-system realizes, in fact, a combination of ADS and DEN.

A base model of the reactor is chosen and a simple mathematical model describing physical processes in the reactor is introduced in next Sections.

2.2. Mathematical model for transient simulation

2.2.1. Reactor model

Point-kinetic approximation of core neutronics and the simplified "two-point" thermo-hydraulics in the cooling/energy generating circuit have been chosen as a simplified model (Slessarev and Bokov, 2004). The so-called "*external cooling*" scheme for reactor is assumed. The external cooling signifies that the molten salt, being simultaneously a fuel and a heat carrier, circulates in the primary loop passing through the heat exchanger where it transmits accumulated heat to the secondary loop (see Figure 5). The thermo-hydraulic model of the first cooling circuit includes two spatially separated elements: the core and the heat-exchanger connected by the tubes of lengths l_h and l_c correspondingly. Note that these parameters may affect reactor dynamics and, especially in the case of the DEN-system, since the DEN core supplies an accelerator with energy and the delay of energy delivery depends upon parameters of a system (geometry, salt flow rate, etc.).

Core thermal power, core and heat-exchanger linear dimensions, fuel flow velocities and other parameters of the systems under consideration are summarized in Table I. They are coordinated in the way to obtain the nominal temperatures at the core and at the heat-exchanger. It is assumed in the present study that there is no heat loss through the tubes.

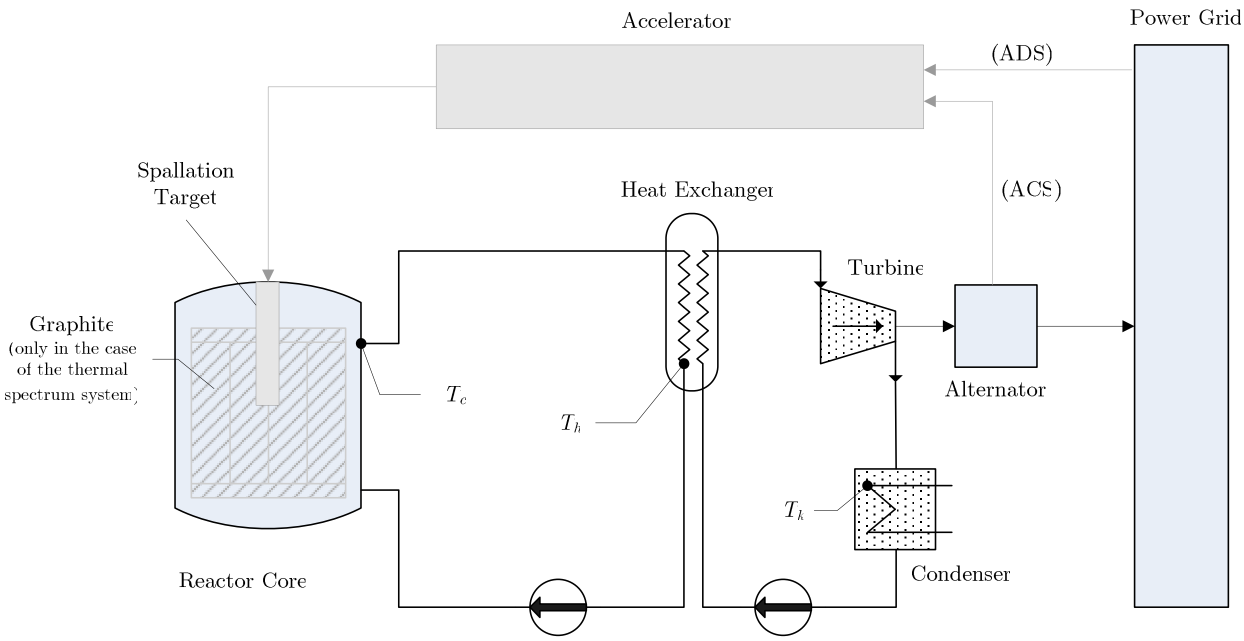


Figure 5. The simplified "two-point" thermo-hydraulics in the cooling/energy generating circuit.

Table I. Thermo-hydraulic parameters for the fast spectrum and the thermal spectrum systems.

Spectrum	Configuration	Nominal power, MW(th)	Steam condenser temperature, °C	Device	Matter	Heat transfer coefficient MW/°C	Length, m	Mass, t	Velocity, m/s	Specific heat capacity MWs/(t °C)	Initial temperature [a], °C
Fast	1	1500	20	core	salt	—	4.0	100	0.33	2.0	650
				heat-exchanger	salt	—	4.0	100	0.33	2.0	560
				tubes	salt	—	2.0	8.3[a]	2.0	2.0	650/560[a]
	2	1500	400	core	salt	—	4.0	100	0.33	2.0	650
				heat-exchanger	salt	—	4.0	100	0.33	2.0	560
				tubes	salt	—	5.0	20.6[a]	2.0	2.0	650/560
Thermal	Th fuel	300	400	core	salt	15.1	8.6	100	0.33	2.0	650
					graphite		8.6	1035	—	1.45	650
				heat-exchanger	salt	—	8.6	100	0.33	2.0	610
				tubes	salt	—	2.0	3.8[a]	2.0	2.0	650/610
	U+ TRU fuel	300	400	core	salt	15.1	8.6	100	0.33	2.0	650
					graphite		8.6	1035	—	1.45	650
				heat-exchanger	salt	—	8.6	100	0.33	2.0	610
				tubes	salt	—	2.0	3.8[a]	2.0	2.0	650/610

[a] follows from initial conditions

2.2.2. Reactor neutronics

The mathematical model describing physical phenomena in the system consists of the coupled system of simplified point kinetic equations with one group of delayed neutrons, reactivity feedback effects, thermo-hydraulic equations and corresponding initial conditions.

The point kinetic equation of a subcritical hybrid system can be written as the ordinary differential, time-dependent equation for core power (e.g. Refs. Hetrick, 1971; Ash, 1979; Waltar and Reynolds, 1981; Rozon, 1992):

$$\frac{\partial P(t)}{\partial t} = \frac{\rho(t) - \beta^*}{\Lambda} P(t) + \lambda W(t) + S(t), \tag{2.1}$$

where P is the core power; $\rho = \left(k_{eff} - 1\right) / k_{eff}$ is the reactivity of core (k_{eff} is the neutron multiplication factor); $\Lambda = \ell / k_{eff}$ is the prompt neutron generation time (ℓ is the prompt neutron lifetime); β^* is the corrected fraction of delayed neutrons, this correction takes into account decay of delayed neutron precursors outside the core (see explanation below); W describes a contribution of delayed neutron precursors to core power (value proportional the concentration of delayed neutrons); λ is the one-group decay constant of the precursors.

Equation (2.1) has to be combined with the equation of the evolution of the delayed neutron precursors. One of the specific features of a core with circulating fuel is a reduced concentration of the delayed neutron precursors inside the core. In mobile fuel systems, delayed neutron precursors leave the core and return back partially decayed after passing through heat exchangers and reprocessing lines (if the reprocessing is foreseen). Hence, the kinetic equation for precursors of delayed neutron $W(t)$ has to be corrected to take into account this particularity. This equation could be expressed in the one-group approximation for delayed neutrons in the following way:

$$\frac{\partial W(t)}{\partial t} + \frac{D(t)}{V_{core}} \left[W(t) - W\left(t - \tau_{out}(t)\right) \exp\left(-\lambda \tau_{out}(t)\right) \right] = \frac{\beta}{\Lambda} P(t) - \lambda W(t), \tag{2.2}$$

where β is the total fraction of delayed neutrons appearing due to precursors decay with the corresponding decay constant λ; τ_{out} is the time that circulating fuel spends out of core. At the nominal ($t \le t_0$) condition, this leads to reduction of the concentration of delayed neutrons when compared with non-circulating fuel as well as to reduction of the delayed neutron fraction:

$$\beta^* = \beta \vartheta_0, \tag{2.3}$$

where a correcting factor $\vartheta_0 \le 1$ is introduced:

$$\vartheta_0 = \left(1 + \frac{\left[1 - \exp\left(-\lambda \tau_{out,0}\right) \right]}{\lambda \tau_{core,0}} \right)^{-1}. \tag{2.4}$$

Note, that here and later the subscript "0" denotes initial or nominal value of the corresponding parameter.

In Eq. (2.1) the term S describes the contribution of the external neutron source. The explicit expression for this term depends on the realization of the hybrid system (coupled or independent source). Thus, for ADS, one can assume (after normalization to the nominal reactor power P_0):

$$S^{(ADS)}(t) = \frac{r(t)}{\Lambda} P_0, \quad \text{where} \quad r(t) = r_0 + \Delta\rho_{\mathrm{TOC}}(t). \tag{2.5}$$

In Eq. (2.5) the parameter $r_0 = -\rho_0 = \left(1 - k_{eff,0}\right) / k_{eff,0}$ is the *nominal subcriticality level*, whereas the term $\Delta\rho_{\mathrm{TOC}}$ describes the perturbation of the external neutron source due to variations of the proton beam current.

Note, that the use of the *subcriticality level*, which has positive values in subcritical systems instead of the reactivity (negative parameter for subcritical systems), is more convenient for analysis. In present work, we will often apply both notations.

In the DEN-systems the intensity of the external neutron source is proportional to the output power $S^{(DEN)}(t) \propto P^{out}(t)$. In our study it is supposed that electric energy is produced immediately after the first cooling loop. As it will follow from consequent results and corresponding analysis, this simplification leads to some underestimation of the safety potential of the DEN-system. Newton cooling model is used for description of the heat exchange with the environment. This yields (after normalization to nominal parameters, corresponding subscript '0' applied):

$$P^{out}(t) = P_0 \frac{\left(T_h(t) - T_k\right)}{\left(T_{h,0} - T_k\right)}, \tag{2.6}$$

where T_h is the temperature of salt in the heat-exchanger and T_k is the temperature of the heat sink (e.g. the temperature of steam in a condenser).

Hence, one may write the explicit expression for the contribution of the external neutron source in the DEN-system in the following way:

$$S^{(DEN)} = \frac{r(t)}{\Lambda} P_0 \frac{\left(T_h(t) - T_k\right)}{\left(T_{h,0} - T_k\right)}. \tag{2.7}$$

In addition to the systems with net independent external source (ADS) and coupled source (DEN) one may also imagine intermediate case, i.e. a system where a part of energy needed to feed the external neutron source arrives from the power grid and the rest is the energy produced by the same installation. It is of interest to analyze the behavior of such ADS-DEN hybrid. In this case the reactor power due to the external (spallation) neutrons is expressed as

$$S(t) = \frac{r(t)}{\Lambda} P_0 \left[\delta_{ind} + (1 - \delta_{ind}) \frac{\left(T_h(t) - T_k\right)}{\left(T_{h,0} - T_k\right)}\right] \tag{2.8}$$

Here, δ_{ind} is the fraction of the independent external source. If $\delta_{ind} = 1$, the source is independent of core power (ADS). On the contrary if $\delta_{ind} = 0$, the system source is coupled completely (DEN). An intermediate case corresponds to the combination of ADS and DEN denoted above as HYB.

Finally, the time-dependent total reactivity ρ is the sum of the initial subcriticality level [see also Eq. (2.5)], the reactivity perturbations $\Delta\rho_{\mathrm{TOP}}$, and reactivity changes due to thermal feedback effects $\Delta\rho_{feedback}$, i.e.

$$\rho(t) = -r_0 + \Delta\rho_{\mathrm{TOP}}(t) + \Delta\rho_{feedback}(t). \tag{2.9}$$

To describe explicitly the in-core feedbacks as well as dependence of the external neutron source for the DEN-system [Eq. (2.7)], a model of heat transfer in the reactor is necessary. Below we introduce separately two slightly different models for heat transfer and for thermal feedbacks (for fast-spectrum and thermal-spectrum configuration), which will be utilized for transient simulations.

2.2.3. Thermo-hydraulics and feedbacks. Fast spectrum system

In fast spectrum systems under consideration the reactivity variation due to thermal feedbacks includes only the Doppler-effect (due to great core dimensions, and in the absence of any internal core structure other feedback effects are negligible), which is equal to

$$\Delta\rho_{feedback}^{Doppler}(T) = \int_{T_0}^{T} \frac{K_D(T_0)}{T} dT = K_D(T_0) \ln\left(\frac{T}{T_0}\right). \tag{2.10}$$

where K_D is the Doppler constant, T is the fuel temperature and T_0 is the fuel temperature at nominal conditions.

The time-dependent core temperature T and the heat-exchanger temperature T_h are described by the following system of thermo-hydraulic equations:

$$c_p^{(s)} M_c^{(s)} \left\{ \frac{\partial T(t)}{\partial t} + \frac{D(t)}{V_{core}} \left[T(t) - T_h\left(t - \tau_{h\to c}(t)\right) \right] \right\} = P(t); \tag{2.11}$$

$$c_p^{(s)} M_h^{(s)} \left\{ \frac{\partial T_h(t)}{\partial t} + \frac{D(t)}{V_h} \left[T_h(t) - T\left(t - \tau_{c\to h}(t)\right) \right] \right\} = -P_0 \frac{\left(T_h(t) - T_k\right)}{\left(T_{h,0} - T_k\right)}, \tag{2.12}$$

where $c_p^{(s)}$ is the fuel specific heat capacity; $M_c^{(s)} = \varrho V_{core}$ and $M_h^{(s)} = \varrho V_h$ are the fuel masses in the core and in the heat-exchanger correspondingly; V_{core} and V_h are volumes of salt in the core and in the heat-exchanger, ϱ is the density of salt; $\tau_{c\to h}$ and $\tau_{h\to c}$ are the time intervals, required the fuel to be delivered from one device ($c-$ "core" or h – "heat exchanger") to another. In this study is assumed that these tubes have equal dimensions (i.e. lengths and cross-sections), hence one may take that $\tau_{h\to c} = \tau_{c\to h} \equiv \tau_{tube}$.

Note, that the delays $\tau_c(t)$, $\tau_h(t)$ as well as the parameter $\tau_{out}(t)$ [see Eq.(2.2)] can be determined from the following implicit equations:

$$V_x - \int_{t-\tau_x(t)}^{t} D(t)dt = 0 \text{, where } x = 'core', 'out', 'tube', \tag{2.13}$$

where V_{tube} is the volume of salt in a tube and $V_{out} = V_h + 2V_{tube}$ is the volume of salt out of core. Note, that in our study we assume that the fuel flow $D(t)$ is a definite function of time.

The initial conditions for Eqs. (2.1) - (2.13) include: the nominal thermal power P_0; the nominal temperature of the core T_0; the nominal subcriticality level r_0; the nominal fuel flow D_0; the effective fraction of delayed neutrons, $\beta^* = \beta\vartheta_0$ with the correcting factor ϑ_0 defined by Eq. (2.4) where $\tau_{x,0} = V_x / D_0 \left(x = 'core', 'out', 'tube'\right)$; the initial concentration of precursors of delayed neutrons

$$W_0 = \frac{P_0\beta\vartheta_0}{\Lambda\lambda}; \tag{2.14}$$

the steady-state temperature of the heat-exchanger

$$T_{h,0} = T_{c,0} - \frac{P_0}{\varrho c_p^{(s)} D_0}. \tag{2.15}$$

Other parameters may be evaluated from technical parameters of the system: device dimensions, fuel velocity distributions etc., presented in Table I, taking into account the equation of the mass conservation for the liquid fuel:

$$D_0 = \frac{V_{core} v_{c,0}}{L_c} = \frac{V_h v_{h,0}}{L_h} = \frac{V_{tube} v_{tube,0}}{l_{tube}}, \tag{2.16}$$

where $v_{c,0}$, $v_{h,0}$, $v_{tube,0}$ are nominal velocities of salt in the core, in the heat-exchanger, and in the tubes correspondingly; and L_c, L_h and l_{tube} are linear dimensions of these components.

2.2.4. Thermo-hydraulics and feedbacks. Thermal-spectrum system

In comparison with the fast reactor concept, the thermal spectrum system contains significant masses of graphite within the core region which absorbs a part of released heat and slows down the transients. All equations concerning power, precursors of delayed neutrons as well as the equation of mass conservation remain valid.

For the thermal-spectrum system the above thermo-hydraulic equations can be rewritten in the following way:

$$c_p^{(s)} M_c^{(s)} \left\{ \frac{\partial T(t)}{\partial t} + \frac{D(t)}{V_{core}} \left[T(t) - T_h\left(t - \tau_c(t)\right) \right] \right\} = P(t) - H_{sg}\left[T(t) - T_g(t) \right];$$

$$c_p^{(s)} M_h^{(s)} \left\{ \frac{\partial T_h(t)}{\partial t} + \frac{D(t)}{V_h} \left[T_h(t) - T\left(t - \tau_h(t)\right) \right] \right\} = -P_0 \frac{\left[T_h(t) - T_k \right]}{\left[T_h(t_0) - T_k \right]}; \tag{2.17}$$

$$c_p^{(g)} M^{(g)} \frac{\partial T_g(t)}{\partial t} = H_{sg}\left[T(t) - T_g(t) \right],$$

where H_{sg} is the heat transfer coefficient between the salt and the graphite; $M^{(g)}$ and $c_p^{(g)}$ are the mass and the specific heat capacity of the graphite respectively (see also Table I).

The total thermal feedback effect in this case is a sum of the Doppler-effect $\Delta\rho_{feedback}^{Doppler}$ as well as the salt $\Delta\rho_{feedback}^{salt}$ and graphite $\Delta\rho_{feedback}^{graphite}$ thermal expansion effects (Lecarpentier, 2001):

$$\Delta\rho_{feedback}(t) = \Delta\rho_{feedback}^{Doppler}(t) + \Delta\rho_{feedback}^{salt}(t) + \Delta\rho_{feedback}^{graphite}(t)\,. \tag{2.18}$$

In the system with thermal spectrum, the reactivity variation due to the Doppler-effect can be evaluated in accordance with the following expression:

$$\Delta\rho_{feedback}^{Doppler}(T) = \tilde{K}_D(T_0)\int_{T_0}^{T} \frac{dT}{\sqrt{T}} = 2\tilde{K}_D(T_0)\left(\sqrt{T} - \sqrt{T_0}\right). \tag{2.19}$$

The thermal expansion feedback for the molten salt and the graphite is supposed to be linear with the coefficients $\alpha_{expansion}$ and $\alpha_{graphite}$ correspondingly in accordance with Ref. (Lecarpentier, 2001):

$$\Delta\rho_{feedback}^{salt}(t) = \alpha_{expansion}\left(T(t) - T_0\right),\ \Delta\rho_{feedback}^{graphite}(T_g) = \alpha_{graphite}\left(T_g(t) - T_{g,0}\right). \tag{2.20}$$

In this subsection we suppose that there is no heat release in graphite, hence the initial condition for the graphite temperature is $T_{g,0} = T_0$.

2.3. Tool of transient simulation

As we can see from Eqs. (2.1)-(2.20), the dynamics of the reactors is described by a coupled system of nonlinear Delayed-argument Differential Equations (DDE). This system of DDE may be resolved numerically. The Runge-Kutta method (Korn and Korn, 1967) slightly modified to take into account particularities of DDE was applied as the numerical implementation. As some of above kinetic differential equations are so-called *stiff differential equation* (because of the small parameter Λ being the prompt neutron generation time) a special attention was paid to guarantee and to verify stability of the numerical scheme.

A special computational code was created for transient simulation. For this Fortran-90 programming language was utilized. Finally, the numerical scheme and the code were verified, utilizing exact (analytical) solutions in some idealized problem statements, admitting analytical solutions.

2.4. Choice of the subcriticality level

The choice of initial subcriticality level r_0 is very important with respect to economics as well as to the technological feasibility of a system: the smaller subcriticality the better economics of a hybrid system and feasibility of the external neutron source (accelerator + target). There are two roles of core subcriticality to be considered in this Chapter:
(i) the safety improvement keeping in mind the deterministic safety as an ultimate goal;
(ii) the “tight” reactor neutronics enhancement if it is required.

These two cases will be analyzed below.

2.4.1. The subcriticality level aiming to achieve the deterministic safety

It is evident that the level of subcriticality has a significant influence on transients. As it will be shown below, in the majority of unprotected transients, subcriticality reduces both power oscillations and the increase of core temperatures. In addition, this favorable effect expands the *grace time* depending on feedback effects and on anticipated reactivity insertions. If unprotected transients do not menace core integrity during significant time, then one can use the term “temporarily limited” deterministic safety. With this term, subcriticality can be considered as an important factor to achieve the temporarily limited deterministic safety. If the total feedback reactivity effect is positive, cores eventually leave the domain of acceptable parameters (temperatures, power, pressures, etc.) earlier or later, depending on the initial subcriticality level. In such cases, the added subcriticality is only able to delay the core disintegration. On the other hand, a negative reactivity feedback effect (even a small one) allows expanding the grace time considerably if subcriticality is applied. Moreover, as it will be shown later, DEN has a potential to keep its core inside of the domain of acceptable parameters asymptotically achieving the so-called “time unlimited deterministic safety level”.

If all reactivity related transients were sufficiently slow, a significant improvement of the safety potential would be expected in terms of the extended grace time. The subcriticality would have such a potential if the following basic choice of the subcriticality level is applied: *the total sum of absolute values of all independent in-core reactivity effects* ($\Delta\rho_{tot}$) *including their uncertainties plus the maximum reactivity insertions* $\Delta\rho_{\mathrm{TOP}}^{\max}$ *does not exceed the nominal level of subcriticality* $r_0 = \left(1 - k_{eff,0}\right)/k_{eff,0}$, *i.e.*

$$r_0 > \Delta\rho_{tot} + \Delta\rho_{\mathrm{TOP}}^{\max} . \tag{2.21}$$

Such a choice does not guarantee unlimited deterministic safety of the core. However, it defines the conditions when transients are slow. The examples of such choices for different molten salt cores under consideration are presented in Tables II-IV. They include the traditional list of phenomena leading to reactivity variation (corrected to take into account particularities of MSRs):

- Doppler reactivity feedback effects;
- Temperature effects of all core components;
- Neptunium (Np) or/and Protactinium (Pa) effects;
- Fuel mass variations in cores in the case when fuel circulation fails or on-line fuel reprocessing results in a reactivity insertion.
- Other reactivity effects play a less important role in the case of MSR.

2.4.2. Unprotected transients

When the preliminary choice of the subcriticality level is done, the study of the safety potential requires simulation of anticipated unprotected transients caused by different realistic transient initiators, such as failures of cooling systems, reactivity insertion, etc. as well as their combinations.

Among these transients (see Tables II-IV), the Unprotected Transient Over Power (UTOP) and/or the Unprotected Transient Over Current (UTOC) are the most dangerous. The reasons of such potential events are multiple reactivity and/or beam current "reserves" (necessary for the normal operation) which could be released as unexpected control rod actions or/and proton beam current intensity variations in the case of subcritical systems. Fortunately, at the nominal regime, many of these reserves become exhausted and it minimizes the value of inserted reactivity. Besides, there are uncertainties related to the spallation neutron production (total neutron yield, energy spectra and angular distribution of neutrons, contribution of secondary reactions, etc.) and these uncertainties also have to be taken into account with respect to the peculiarities of a given system.

The control of subcritical systems has the following peculiarity – these systems can use either special mechanical rods which change reactivity, or the proton current variation mechanisms. In both cases, different failures could provoke some transients. The presence of control rods requires the subcriticality level correction, while the use of proton current correction does not cause the reactivity change and the subcriticality level r_0 is allowed to be smaller.

The study has to include other potentially dangerous events such as Unprotected Lost Of Flow (ULOF), Unprotected Lost Of Heat Sink (ULOHS), Unprotected core Over-Cooling (UOVC) or core overheating (if positive feedback effects are dominating), Unprotected Gain of Flow (UGOF), in-core fuel mass oscillations, the Np/Pa reactivity effects, etc. Moreover, non-nominal regimes have to be also examined. However, this requires to know the detailed design of the installation (it is worth to remind that a "generalized" model systems are considered in this Chapter) and therefore we will restrict our study to the most important events. Thus, for example, reactivity transients caused by the Np/Pa effects as well as core fuel mass fluctuations are very slow and have not been analyzed, supposing that these phenomena may be compensated by continuous adjustment of the fuel content.

Following the logic of the deterministic safety approach, during our simulations we looked for the most dangerous conditions for transients. For example, different characteristic times for the pumps halt (this event leads to ULOF transient) in order to find the most menacing situation. These "most pessimistic" conditions will be utilized below for analysis.

Finally, the following recommendation of Lecarpentier (2001) the salt boiling temperature of 1300°C was chosen as the disruption criterion for the molten salt systems, i.e. $T^{\dagger} = 1300°\mathrm{C}$ is assumed to be the maximal limit of acceptable core temperature. The lower limit of acceptable parameters is the temperature of fuel solidification of 450°C.

2.4.3. Subcriticality caused by necessity of the tight neutronics enhancement

In this situation, a large subcriticality plays an important role in improving of the neutron balance. For example, in the Th-fuelled core of the WISE concept (Slessarev *et al.*, 2001, Slessarev *et al.*, 2004), one is obliged to keep r_0 as high as $20 \div 25\beta$. Here again the safety can be improved much more easily due to the larger margin to the core criticality.

2.5. Reference cores for transient simulation

A model core with a fast-spectrum (in two configurations) and two thermal-spectrum cores with graphite moderator were utilized for transient simulations. The fast spectrum core corresponds, in fact, to a "generalized" fast-spectrum option of the WISE-concept core (Slessarev *et al.*, 2001, Slessarev *et al.*, 2004), whereas thermal spectrum models correspond to the thermal-spectrum options of the WISE-concept and to the "TRU-incinerator with support-uranium" and "self-generator with support-thorium" core options of the AMSTER concept (Vergnes *et al.*, 2000; Lecarpentier, 2001).

2.5.1. Fast-spectrum core

The parameters of systems under consideration are given in Table I. Configurations "1" and "2" correspond to two different cases: with a low condenser temperature ($T_k = 20\,°\mathrm{C}$) and an elevated ($T_k = 400\,°\mathrm{C}$) temperature to prevent fuel from solidification.

The total temperature feedback effect in the core has a negative value and, hence, core cooling inserts the dangerous positive reactivity starting with the nominal regime (see Table II). The Doppler coefficient of reactivity K_D, Np/Pa reactivity effects and the fraction of delayed neutrons β (see Table V for detail) have been evaluated in accordance with Refs. (Slessarev and Tchistiakov, 1997; Adamov, Orlov *et al.*, 1997). Then, following the recommendations of Section 2.4.1, the subcriticality levels both for ADS and DEN (Table II) has to be approximately chosen: $r_0 = 2.1\beta$ if the reactivity reserves are preserved on control rods; $r_0 = 1.4\beta$ (or $k_{eff} = 0.995$) if all reactivity reserves are replaced by the proton current variation $\Delta\rho_{\mathrm{TOP/TOC}} \approx (0.75 \div 1)\beta$. Therefore, the subcriticality value $r_0 = 2\beta$ has been chosen for analysis of unprotected transients.

2.5.2. Thermal-spectrum cores

2.5.2.1. AMSTER-WISE-type option, Th-fuel

In this case the total temperature effect in the core has a positive value and, hence, core heating inserts a positive reactivity (see Table V). The Doppler constant of reactivity K_D, Np/Pa reactivity effects, and the fraction of delayed neutrons β have been evaluated in accordance with Refs. (Slessarev *et al.*, 1999; Lecarpentier, 2001). Then, following the recommendations of Section 2.4.1, the subcriticality level both for ADS and DEN (see Table III) is chosen as follows: $r_0 = 4.8\beta$ if all reactivity reserves on control rods are foreseen; $r_0 \approx 3.8\beta$ if all reactivity reserves are replaced by the proton current variation. The mean value of the subcriticality level (to be considered as pessimistic) $r_0 = 4\beta$ has been chosen for analysis of the TOP/TOC transients. The range of the postulated maximum reactivity insertion has been chosen to be $\Delta\rho_{\mathrm{TOP/TOC}} \approx \beta$.

2.5.2.2. AMSTER-WISE-type option, U+TRU-fuel

In this case the total temperature reactivity effect of the core is negative (see

Table V). The Doppler coefficient of reactivity K_D, the Np reactivity effects and the fraction of delayed neutrons β (see Table V) have been evaluated in accordance with Refs. (Slessarev *et al.*, 1999; Lecarpentier, 2001). Then, in accordance with the recommendations of Section 2.4.1, the subcriticality level for both ADS and DEN (see Table IV) have been chosen in the following way: $r_0 \approx 3.4\beta$ if all reactivity reserves are preserved on control rods; $r_0 \approx 2.4\beta$ if the proton current variation is foreseen. The subcriticality value: $r_0 = 3\beta$ has been chosen for analysis of the TOP/TOC transients, while the range of the postulated maximum reactivity insertion is $\Delta\rho_{\mathrm{TOP/TOC}} \approx \beta$.

Table II. In-core reactivity effects and reactivity reserves in the fast spectrum system (core nominal average temperature: 605°C; the low temperature limit is 450°C).

In-core reactivity effects/Reserve at nominal conditions	$\Delta\rho$ (pcm)	$\Delta\rho_{\mathrm{TOP/TOC}}$ (pcm)
Homogeneous core cooling (878K $\rightarrow$ 723 K) – Doppler-effect	100[a]	0
Fuel mass fluctuations	±150	150
Fuel stop ($\approx \beta / 2$)	200	0
Np/Pa effects (reduced flux)	10/150	0
Uncertainties $\Delta\rho_{un}$	50	—
Uncertainties related to spallation $\Delta\rho_{sp}$	—	100
Operational reserve	—	50
Total $\Delta\rho_{total} = \sum \Delta\rho$	510/650 ($1.25/1.6\beta \approx 1.4\beta$)	300 ($\approx 0.75\beta$)

[a] Empty at nominal regime

Table III. In-core reactivity effects and reactivity reserves in the Th-fuelled thermal spectrum system (core nominal average temperature is 630°C; lower and upper limits for the fuel temperature are 450°C and 1300°C correspondingly).

In-core reactivity effects/Reserve at nominal conditions	$\Delta\rho$ (pcm)	$\Delta\rho_{\mathrm{TOP/TOC}}$ (pcm)
Homogeneous core heating (903 K $\rightarrow$ 1573 K)		
Doppler-effect	-1350[a]	—
fuel expansion	1306[a]	—
graphite	804[a]	—
Total temperature effect of core cooling	760[a]	0
Fuel mass fluctuations	±200	200
Fuel stop ($\approx \beta / 2$)	175	0
Pa effect (reduced flux)	50	0
Uncertainties $\Delta\rho_{un}$	150	—
Uncertainties related to spallation $\Delta\rho_{sp}$	—	100
Operational reserve	—	50
Total $\Delta\rho_{total} = \sum \Delta\rho$	1335 ($\approx 3.8\beta$)	350 ($\approx 1\beta$)

[a] Empty at nominal regime

Table IV. In-core reactivity effects and reactivity margins in the U+TRU-fuelled thermal spectrum system (core nominal average temperature is 630°C; lower and upper limits for the fuel temperature are 450°C and 1300°C correspondingly).

In-core reactivity effects/Reserve at nominal conditions	$\Delta\rho$ (pcm)	$\Delta\rho_{\mathrm{TOP/TOC}}$ (pcm)
Homogeneous core cooling ($903\,\mathrm{K} \rightarrow 723\,\mathrm{K}$)		
Doppler-effect	556[a]	—
fuel expansion	-261[a]	—
graphite	216[a]	—
Total temperature effect of core cooling	511[a]	0
Fuel mass fluctuations	±250	250
Fuel stop ($\approx \beta / 2$)	225	0
Np effect (reduced flux)	50	0
Uncertainties $\Delta\rho_{un}$	50	—
Uncertainties related to spallation $\Delta\rho_{sp}$	—	100
Operational reserve	—	50
Total $\Delta\rho_{total} = \sum \Delta\rho$	1086 ($\approx 2.4\,\beta$)	400 ($\approx 1\,\beta$)

[a] Empty at nominal regime

Table V. Integral parameters characterizing the safety physics of the molten salt cores (from Refs.: Slessarev and Tchistiakov, 1997; Adamov, Orlov *et al.*, 1997; Slessarev *et al.*, 1999; Lecarpentier, 2001).

Spectrum	Fast		Thermal	
Configuration	"1"	"2"	Th fuel	U+TRU fuel
Doppler coefficient K_D	-5×10^{-3}	-5×10^{-3}	-7×10^{-4}	-8.8×10^{-4}
Salt expansion coefficient $\alpha_{expansion}$,°C^{-1}	—	—	1.95×10^{-5}	1.45×10^{-5}
Graphite expansion coefficient $\alpha_{graphite}$,°C^{-1}	—	—	1.2×10^{-5}	-1.2×10^{-5}
Delayed neutron fraction β	4×10^{-3}	4×10^{-3}	3.5×10^{-3}	4.46×10^{-3}
Precursors decay constant λ , s^{-1}	0.08	0.08	0.08	0.08
Prompt neutron generation time Λ , s	1×10^{-6}	1×10^{-6}	4×10^{-4}	4×10^{-4}
Subcriticality level r_0	8×10^{-3}	8×10^{-3}	1.4×10^{-2}	1.35×10^{-2}

2.6. Unprotected transients in the fast-spectrum systems

2.6.1. Unprotected Transients Over Power/Transients Over proton Current

Insertion of the reserve of reactivity, foreseen for the compensation of multiple reactivity effects, is the reason of transients with a considerable and rapid increase of core power (UTOP). These unprotected transients could be finally terminated without reactor damage if there are sufficient prompt negative feedback effects due to, for example, core heating. For UTOP/UTOC studies, the corresponding transients were simulated by the linear insertion of the total reserve of reactivity/current in the period of 1 s (see Figure 6). In Figure 6 (on the right), upper curves correspond to the core outlet temperatures, while the lower ones correspond to the core inlet temperatures. The same notation will be used in all figures (Figure 6 to Figure 25).

In **critical systems** (**CRT**) with unprotected TOP, there is a narrow and significant power jump with the maximum amplitude higher by factor of 30 in the magnitude compared with the nominal power followed, by a mellow power oscillations for 300 s after the reactivity insertion. Finally, an asymptotic power at the end of the transient is achieved if the total feedback temperature effect is negative. The behaviour of the core temperature is similar but with wider oscillations. The maximum core temperature (~2200°C) exceeds the upper temperature limit ($T^{\dagger} = 1300°\mathrm{C}$) considerably. The asymptotic core temperature depends directly upon the Doppler-effect value and this temperature is expected to be too high (~1800°C) for the core. It means that this accident will lead to core disruption and it is not acceptable in terms of deterministic safety criteria. Moreover, strictly speaking, the curves, after the salt temperature exceeds 1300°C, have no physical meaning.

TOP-behavior of the **ADS** is much smoother: both the asymptotic power and the core temperature are weakly dependent on feedback effects and are defined by the reactivity jump value. For this particular case, the maximum power jump amplitude does not exceed the factor of 2 of the nominal power without power oscillations. The asymptotic power is about 1.5 of the nominal power, while the maximum core temperature does not exceed 1000°C. All these parameters do not challenge core integrity and can be considered as acceptable.

DEN behavior takes an intermediate position between critical reactors and ADS: similar to ADS, there is a small power jump. Neither power nor temperature oscillations are observed. DEN eliminates short but dangerous fluctuations of power and of the core temperature in the beginning of transients. Meanwhile, DEN works in a “critical mode”. As a result both the asymptotic power and the core temperature follow the asymptotic parameters of the corresponding critical core: their values are defined also by the Doppler feedback effect. The difference is evident: DEN transients slow down due to the delay of spallation neutrons. However, later during the transient, asymptotic parameters become dangerous as in the case of critical reactors.

Hence, one can conclude that unprotected TOP behavior is the most favorable for **ADS**, less favorable for **DEN** (non-acceptable after 300 s of the transient) and unacceptable for this critical reactor.

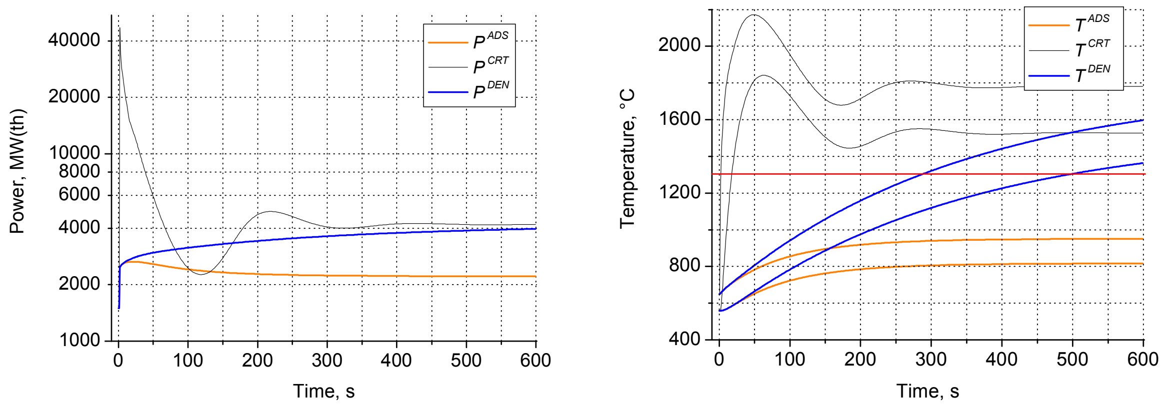


Figure 6. Unprotected TOP ($\Delta\rho_{\mathrm{TOP}} = \beta$ in the period of 1 s) transient in fast-spectrum systems (Th-fuelled).

The direct inter-comparison between UTOP and UTOC for subcritical core systems (see Figure 7) outlines advantages of hybrids controlled by proton current variation. It also confirms that transients of UTOC-type are less dangerous compared with UTOP at the same subcriticality level. For example, it leads to a supplementary core temperature reduction in **DEN**: about 200°C in 10 minutes after the UTOC has started.

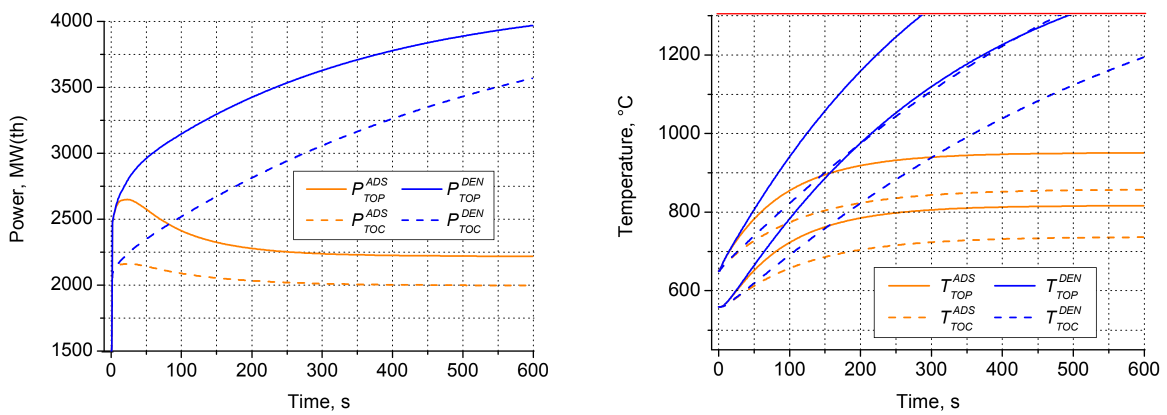


Figure 7. Inter-comparison of unprotected TOP/TOC transient ($\Delta\rho_{\mathrm{TOP/TOC}} = \beta$ in the period of 1 s; Th-fuelled; fast-spectrum systems; r_0 has the constant value).

Similarly as in the case of UTOP, UTOC transients in **DEN-**systems are expected to be slower compared with critical reactors (case of UTOP). However, the *asymptotic* values of power for both **critical system** and DEN will remain similar.

ADS demonstrate the safer behavior regarding TOC transients. As for DEN, the interval of the acceptable response is sufficiently large, up to 500 s after the reactivity insertion. Despite the significant reduction of the transient temperatures and the increase of the grace time for **DEN**, the deterministic safety conditions are fulfilled only for ADS.

2.6.2. Unprotected Loss Of fuel Flow

Loss Of fuel Flow (ULOF) accidents were simulated by significant flow reduction from the nominal value down to 10% of nominal value in the period of 10 s supposing that remaining flow can be continuously supported later on due to the fuel natural circulation (see Figure 8).

The following effects take place because of fuel flow slowing down:
(i) the increase of delayed neutron fraction in the core and, hence, the insertion of reactivity;
(ii) core overheating followed by consequent feedback effects.

Within the models considered in this chapter, one obtains the following results.

In **critical reactor** there are important oscillations of power (factor of 1.4 in the power amplitude during the first 30 seconds) and of fuel temperature (a rise up to 1050°C). There is the danger that the heat-exchanger will become overcooled for both low (20°C) and elevated (350°C) "heat sink temperatures" T_k. However, this threat can be avoided by further elevation of heat sink temperature, say, up to 400°C.

ADS itself (without beam halt) is unable to reduce its power sufficiently (feedback effects do not play such important role as it does in critical reactors) and there is an "asymptotical" growth of the core temperature which can exceed finally (in approximately 10 minutes) the temperature limit.

The behavior of power and temperatures is more favorable in the case of **DEN.** As one could expect, due to the reduction of accelerator power, core power is significantly reduced by 40 % of the nominal level. Finally, the increase of core temperature up to 900°C still keeps the system away from the limiting conditions.

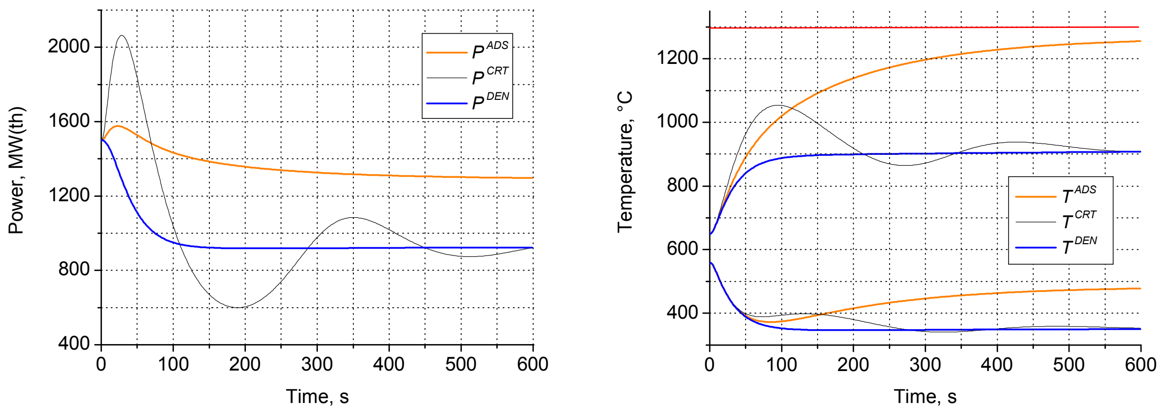


Figure 8. Unprotected LOF (pump power fall of 90 % in the period of 10 s) transient in the fast spectrum system (configuration 1).

2.6.3. Unprotected Gain Of fuel Flow transients

Overcooling of cores can also provoke transients followed by significant power and temperature changes. The corresponding case can be simulated by the rapid increase of the flow if the pump power increases suddenly. Figure 9 presents Unprotected Gain Of fuel Flow (UGOF) transients when the fuel flow rate is doubled linearly in the period of 30 s.

During UGOF critical reactors exhibit important oscillations of power while subcritical system transients (both ADS and DEN) are negligible when compared to the critical ones. In addition, temperature transients will not produce serious troubles for any systems. With respect to UGOF, the mobile fuel system seems to have the unlimited grace time.

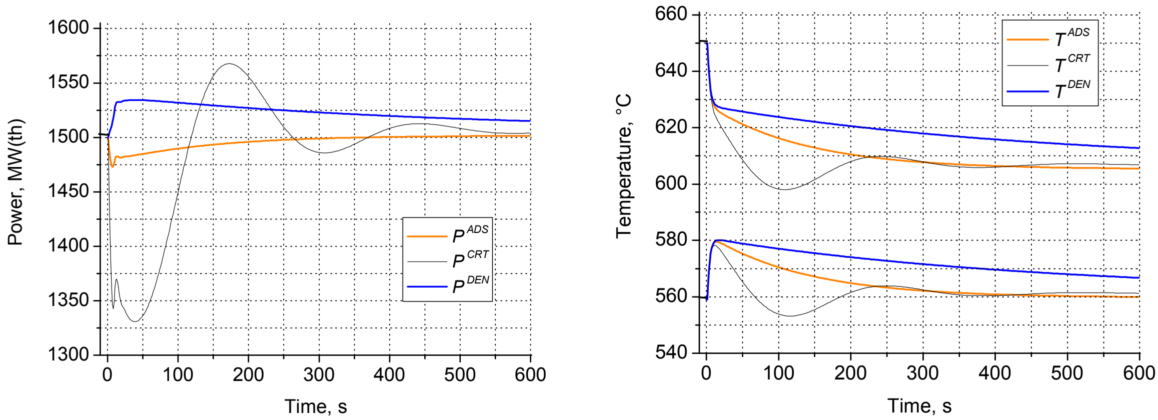


Figure 9. Unprotected GOF (pump power jump by 100 % in the period of 30 s) transient in the fast spectrum system (configuration 1).

2.6.4. Unprotected Loss Of Heat Sink transients

These transients have been simulated by the “linear” (in the period of 3 s) stop of the heat transfer through the heat-exchanger causing the rapid core overheat (see Figure 10). As was mentioned earlier in this work, feedbacks of **critical reactor** reduce core power rapidly, while the core temperature, after a negligible growth, returns back close to the nominal level in about 500 s.

ADS is not able to reduce sufficiently its power (feedback effects are not effective for ADS). This leads to high temperatures: one observes the continuous increase of the core temperature and, after 5 minutes, it exceeds the limit of the viability.

Unlike ADS, **DEN** behaviour is again more favourable compared with others due to the prompt reduction of accelerator power thanks to the coupling and, hence, of total power which approaches to about 10 % of its nominal value. Core and heat-exchanger temperatures remain around the nominal core temperature level.

Hence, DEN exhibits the most favorable and deterministically safe behavior (critical system behavior is also acceptable), however ADS transients are dangerous.

2.6.5. Combination of unprotected accidents: ULOF followed by UTOP/TOC

The deterministic safety study has to include the analysis of all possible combinations of unprotected accidents. Let us consider the most severe of them in the following new conditions: we assume the elevated temperature (up to 400°C) of the heat sink in order to avoid fuel solidification.

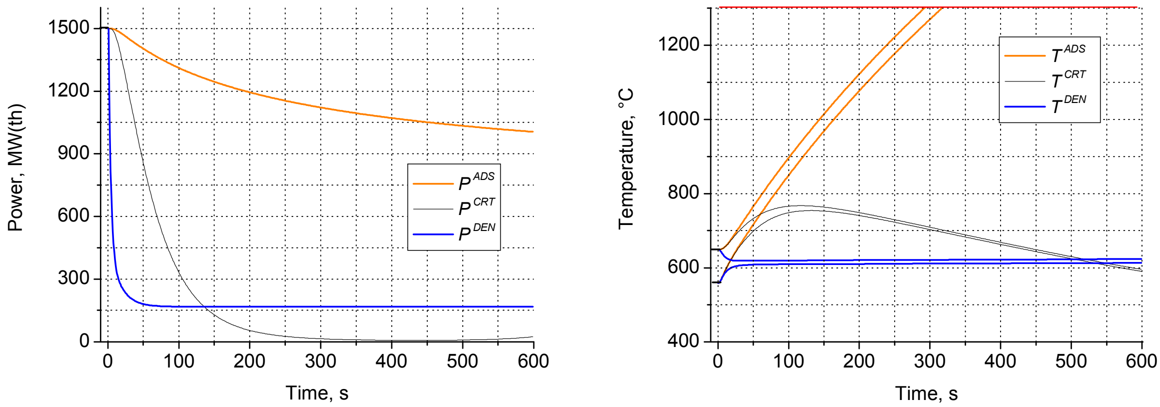


Figure 10. ULOHS transient (heat sink falls by 90 % in the period of 3 s) in the fast spectrum system (configuration 1).

One of the most severe combinations of transients could be realized according to the following scenario: ULOF started at $t = 1$ s leading to a reactivity insertion due to the fuel slowing down (see Figure 11). This reactivity insertion reaches its maximum at 30 s after ULOF started. At this time, UTOP (for critical reactor) or corresponding UTOC (for subcritical systems) leads to the insertion of the maximum reactivity of 0.75β. Corresponding transient curves are presented in the Figure 11.

Analysis shows that the power of the **critical reactor**, manifesting significant oscillations, exceeds the nominal power by a factor of 10, while its core temperature exceeds the limiting value of 1300°C even at the beginning of UTOP. Asymptotic temperatures also exceed the limiting temperature.

ADS has much smoother power behavior, however, its core temperature exceeds the limiting value in about 100 s after the beginning of ULOF. On the other hand, **DEN** will not exceed the temperature limit at least during the first 10 minutes of this transient as shown in the same Figure 11.

2.6.6. Combinations of unprotected accidents: UGOF followed by ULOF and, later on, by UTOP/UTOC

Three sequential accident initiators are simulating the following possible events: at $t = 1$ s, the pumps are doubling linearly their power in the period of 10 s and later, because of their failure, the fuel flow decreases significantly (down to 10%) initiating a ULOF. At the most dangerous time ($t \approx 200\,\mathrm{s}$) of ULOF it is followed by UTOP/UTOC. The corresponding curves of power and temperatures are presented in Figure 12.

Conclusions are very similar to the previous cases: **DEN** demonstrates the best safety behavior for at least the first 10 min.

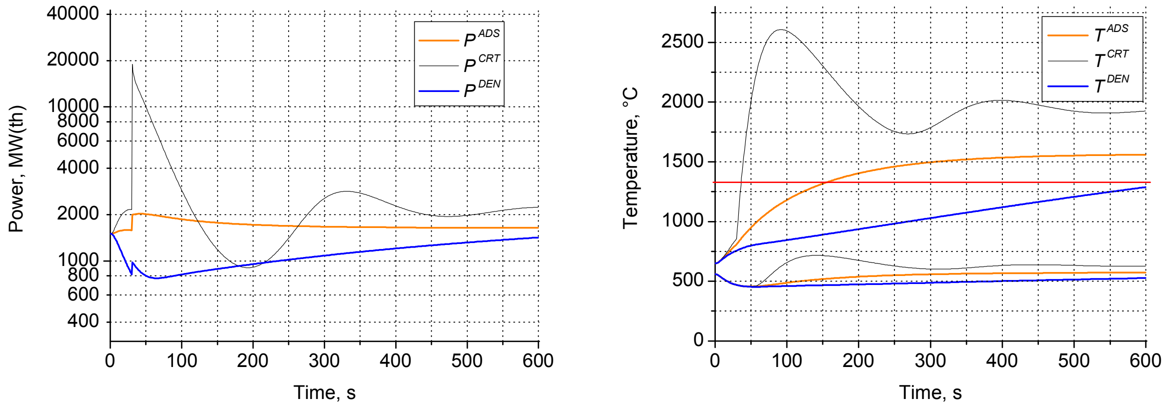


Figure 11. Combined ULOF (pump power falls: by 90 % in the period of 10 s starting at $t = 1$ s) with UTOP/UTOC in the fast spectrum system (configuration 2). Reactivity $\Delta\rho_{\mathrm{TOP/TOC}} = 0.75\beta$ is inserted in the period of 1 s at $t = 30$ s.

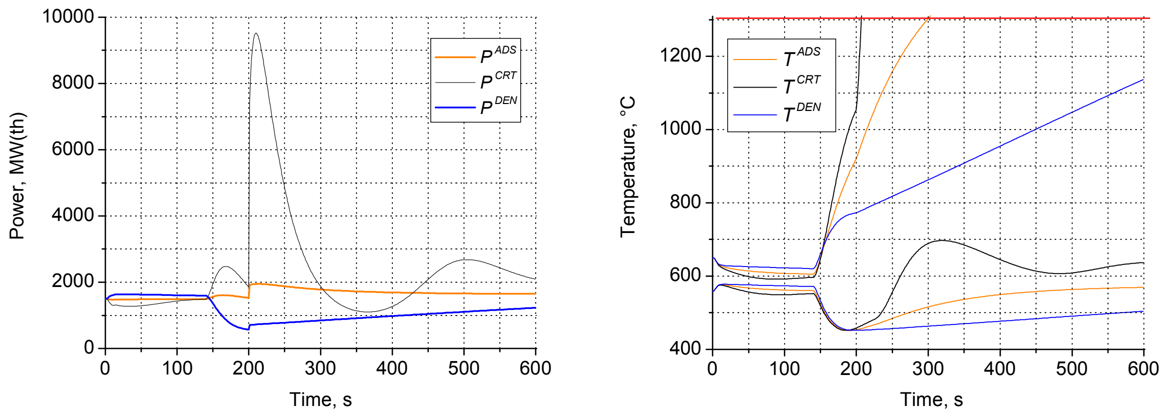


Figure 12. Combined unprotected transient in the fast spectrum system (configuration 2): GOF (salt flow increases from 100 % to 200 % in the period of 10 s) followed by LOF (pump power falls from 100 % to 10 % in the period of 10 s at $t = 140$ s) and finally TOP/TOC occurs with $\Delta\rho_{\mathrm{TOP/TOC}} = 0.75\beta$ for 1 s at $t = 200$ s.

2.6.7. Combinations of unprotected accidents. ULOHS followed by ULOF and, later on, by UTOP/UTOC

This paragraph presents the simulation of the following aggravated events: ULOHS calls the temperature increase with its maximum in the vicinity of $t = 100$ s (see Figure 10); at this moment, ULOF starts with maximum reactivity insertion around $t = 125$ s (see Figure 8) and, right at this moment, UTOP/UTOC stars.

Figure 13 shows that the **critical reactor** is unable to resist the temperature jump which is far above of the limiting value $T^{\dagger}$. In addition, the core power exhibits large oscillations. **ADS**, retaining a high level of the core power, has an unacceptable temperature growth and within 200 s it overcomes the limiting temperature of 1300°C. **DEN** is able to retain its suitable power and temperature regimes during longer time (at least, more than 10 minutes from the moment when transients have been initiated).

Similar situations have been verified with different sequences of the “most severe” UTOC, ULOF, ULOHS events: unfavorable UTOC+ULOF (see Figure 14) and UTOC+ULOF+ULOHS (see Figure 15).

These studies indicate that DEN “behaves” in a more acceptable manner than its competitors: the effect of smoothing the thermo-hydraulic transients (the benefit of the fission-spallation coupling) leads to the reduction of the maximum temperature in the core.

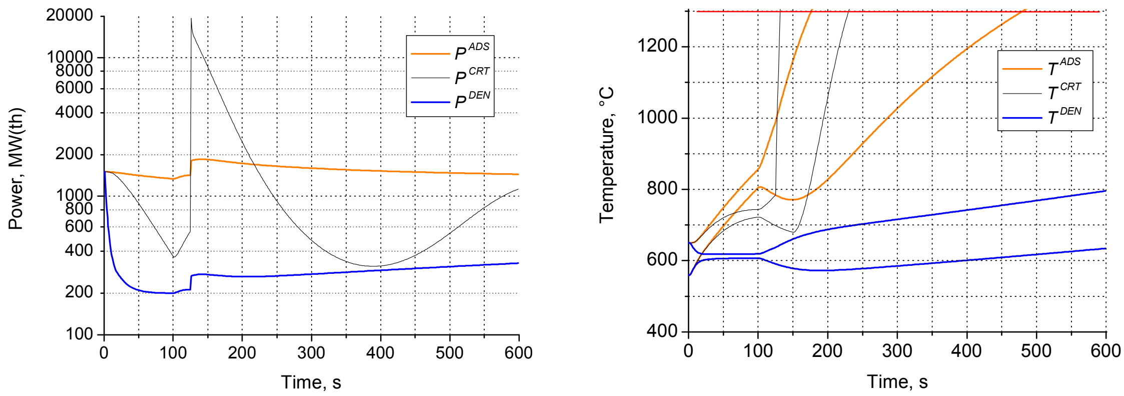


Figure 13. Combined unprotected transient in the fast-spectrum system (configuration 2): LOHS (heat sink fall: by 90 % in the period of 3 s) followed by LOF (pump power fall from 100 % to 10 % in the period of 10 s) at $t = 100$ s and finally TOP/TOC occurs with $\Delta\rho_{\mathrm{TOP/TOC}} = 0.75\beta$ in the period of 1 s at $t = 125$ s.

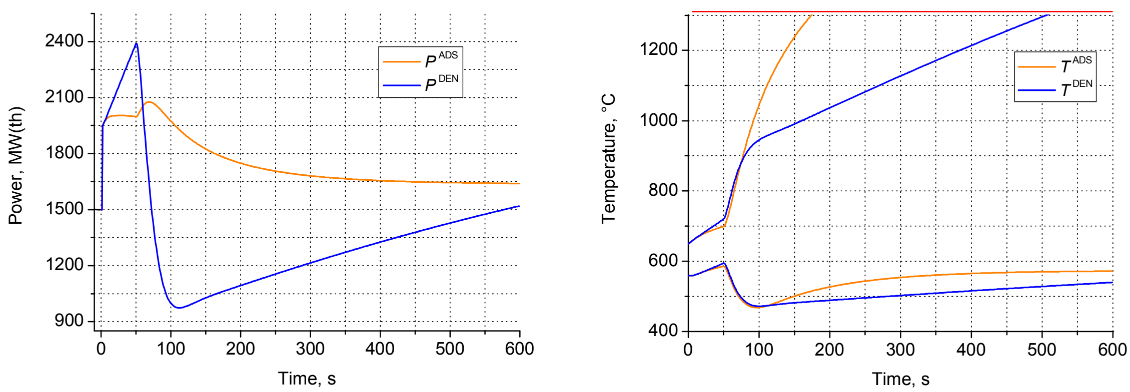


Figure 14. Combined unprotected transient in the fast spectrum-system (configuration 2): TOC ($\Delta\rho_{\mathrm{TOP/TOC}} = 0.75\beta$ is inserted in the period of 1 s) followed by LOF (pump power fall from 100 % down to 10 % in the period of 10 s) at $t = 50$ s.

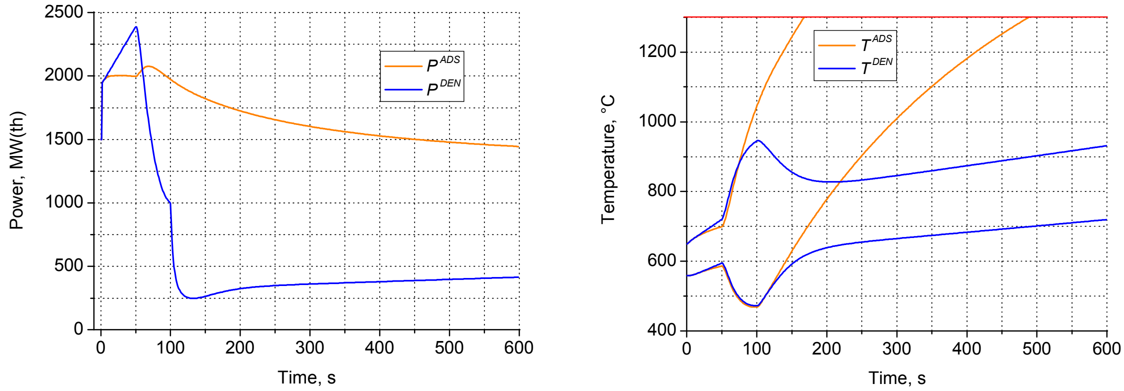


Figure 15. Combined unprotected transient in the fast-spectrum system (configuration 2): TOC ($\Delta\rho_{\mathrm{TOC}} = 0.75\beta$ is inserted in the period of 1 s) followed by LOF (pump power falls from 100 % down to 10 % in the period of 10 s) at $t = 50$ s and LOHS (heat sink fall: 90 % in the period of 3 s) at $t = 100$ s.

2.7. Unprotected transients in the thermal-spectrum thorium-feed systems

Unlike for the fast-spectrum systems, the safety features of some designs of conventional thermal molten salt system with graphite moderators (both Th and U fuelled) are (Lecarpentier, 2001) insufficient to achieve the deterministic level. At least, as it will be shown hereafter, the deterministic safety could be attained only temporarily. The following analysis will supply us with the information about the most dangerous anticipated unprotected events.

As it was indicated, significant positive feedback effects require some prudence with respect to the choice of the subcriticality margin. Hereafter, the subcriticality level of about $r_0 = 4\beta$ has been chosen for transient studies (see Table IV).

2.7.1. Unprotected TOP/TOC transients

Similarly as for fast spectrum systems, **ADS** demonstrates the safest behavior despite insertion of a considerable proton current reserve (see Figure 16 for details).

Critical reactor (**CRT**) power has an unlimited prompt jump and the core temperature exceeds the limit of an acceptable temperature in less than a few seconds (when 1β in the period of 1 s is inserted) despite the thermal inertia effect related to a significant thermal capacity of graphite. Hence, this concept is fully unacceptable with respect to UTOP. **DEN**, due to an important delay of spallation neutrons, resists to UTOC for much longer time (more than one hour). However, later during the transient, the core boiling becomes possible. **Hybrid of ADS and DEN** (with $\delta_{ind} = 50\%$) exhibits even better resistance to this type of transient.

In conclusion, all systems, except **CRT**, have a significant (more than 1 hour) grace time with respect to UTOP/UTOC as clearly seen from Figure 16.

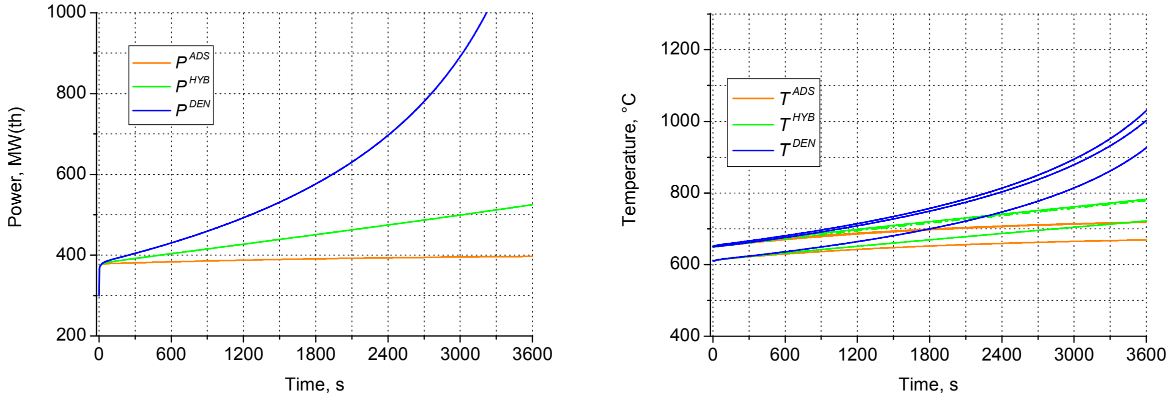


Figure 16. UTOP/UTOC transients in the thermal (Th-fuelled) spectrum systems.

2.7.2. Unprotected LOF transients

The reactivity insertion caused by the increase of the delayed neutron fraction, combined with the positive feedback effect, leads to a rapid power jump for the **critical system** (Figure 17). In about 100 s (when the fuel flow falls from 100% down to 10% in the period of 10 s), the core temperature exceeds its limit. On the contrary, power behavior of all subcritical core systems is much smoother (e.g. the core temperature increases gradually by 200°C/hour in DEN). The best (the most favorable) behavior is inherent to **DEN**, while it is more dangerous – for **ADS** due to somewhat higher power level.

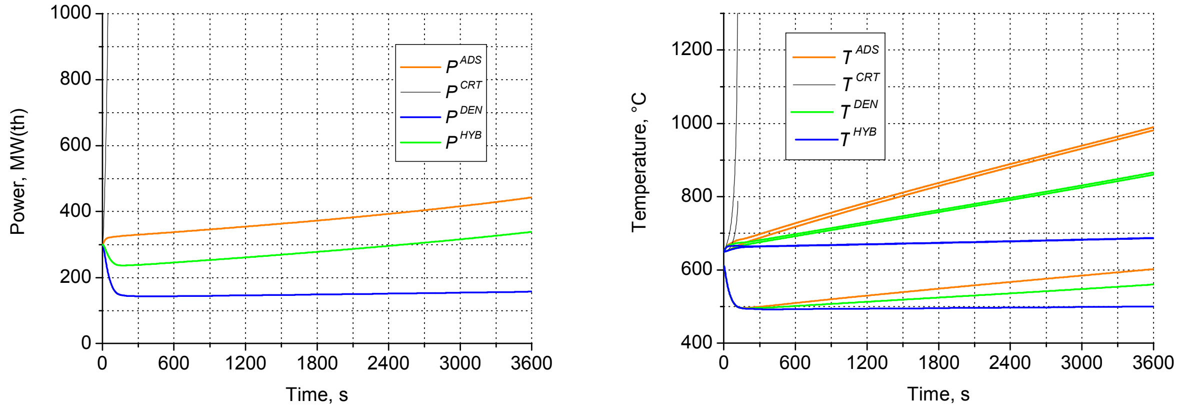


Figure 17. ULOF transients in the thermal-spectrum (Th-fuelled) systems.

It seems that positive feedback-effects do not allow realization of the deterministic safety behavior during infinite time. Nevertheless, subcritical core systems have much longer grace times (at least, tens of minutes).

2.7.3. Unprotected LOHS transients

The character of curves after unprotected loss of heat sink is similar to that of ULOF: the behavior of **DEN** is more preferable regarding others, because this tranient almost terminates the accelerator feeding. In the case of the **critical reactor** the core temperature exceeds the temperature limit already after approximately 400 s (after when ULOHS has started) and the **ADS** temperature – only after 65 minutes, while the DEN core temperature remains acceptable all the time (see Figure 18).

2.7.4. Unprotected GOF transients

In this case our calculations demonstrated that the temperature characteristics of all systems will not exceed the corresponding limits and, therefore, this transient is acceptable for all systems.

2.7.5. Summary for the thermal-spectrum Th-fuelled system

With respect to all unprotected anticipated transients discussed above, unfavorable positive feedback effects are the most important reason for the degradation of the safety features of both critical and subcritical thermal-spectrum Th-fuelled systems. Certainly, inherent properties of subcritical core systems slow-down discussed unprotected transients.

DEN or combined **DEN+ADS** hybrids demonstrated the potential of a self-protection allowing a comfortable grace time without the necessity of any manual intervention.

It can be concluded that these unfavorable feedback effects would not lead to the deterministic safety asymptotically even by choosing some low subcriticality level. It is also clear that an increase of subcriticality margin is favorable with respect to the UTOC transients; however, this can be unfavorable regarding the ULOF or ULOHS transients.

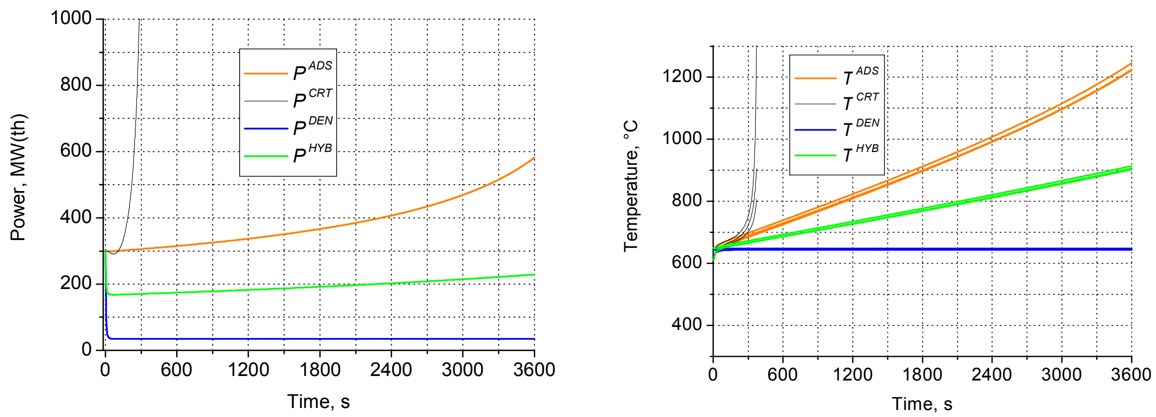


Figure 18. ULOHS transients in the thermal (Th-fuelled) spectrum systems.

2.7.6. Unprotected transients in the systems with large margin of subcriticality ("tight" neutronics enhancement)

According to the deterministic route of safety analysis (see Section 2.3.1), let us try to increase the subcriticality level in order to enhance the safety potential. As an extreme case one can choose $r_0 = 20\beta$ as the largest value that can be realistic if the Th-fuelled system would be used in the prospective NP (Slessarev *et al.*, 2001) because of its "tight" neutronics. The corresponding transients can be compared directly with those which have been obtained for the lower subcriticality level: $r_0 = 4\beta$.

Unprotected TOC transients (with the same reserve inserted) are rather flat both for power and temperatures despite the significant *positive* feedback effects (see Figure 19). **ADS** behavior is preferable, while **DEN** temperatures are continuously (although very slowly) increasing. Within one hour after TOC started, they remain far away from the temperature limit. This is much better compared with excursion-type transients in a critical reactor.

As for the ULOF transients, the reactivity insertion due to the fuel flow reduction does not play an important role for subcritical systems. Now, the **ADS** behavior is less preferable taking into account the increase of core temperature due to positive temperature feedbacks (see Figure 20). At the same time, a positive feedback effect is not able to overcome the large initial subcriticality level and the temperature increases slowly: in one hour after the ULOF started, the core temperature does not exceed 900°C. The **DEN**-system decreases the spallation neutron intensity by reduction of the accelerator power. As a result both the core power and temperatures are very stable.

ULOHS behavior of the ADS is more threatening (however, to much smaller extent when compared with critical reactors) than the DEN behavior due to similar reasons: the power increases (although slowly) this leading to a continuous temperature increase followed by the insertion of supplementary reactivity (see Figure 21).

One concludes that subcritical systems are safe even in the case of positive feedbacks if the level of subcriticality is sufficiently large.

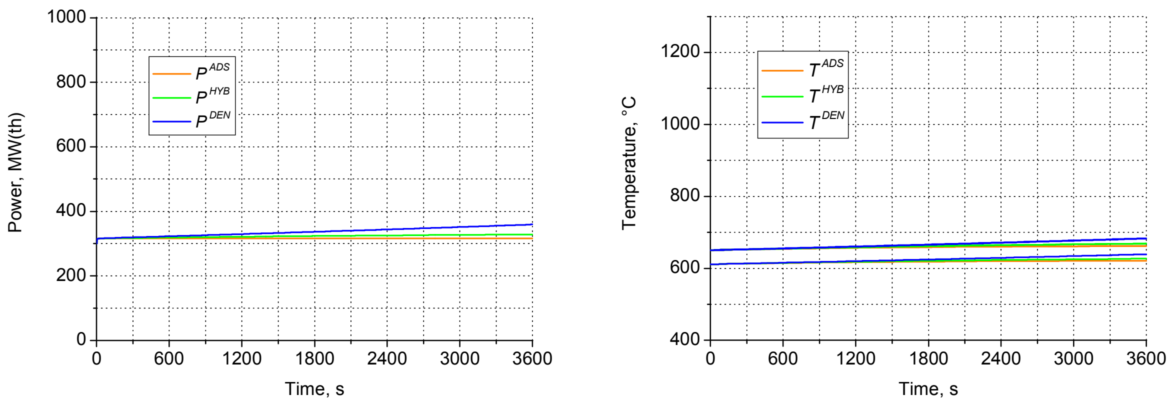


Figure 19. UTOC transients in the thermal spectrum (Th-fuelled) systems with a large subcriticality level

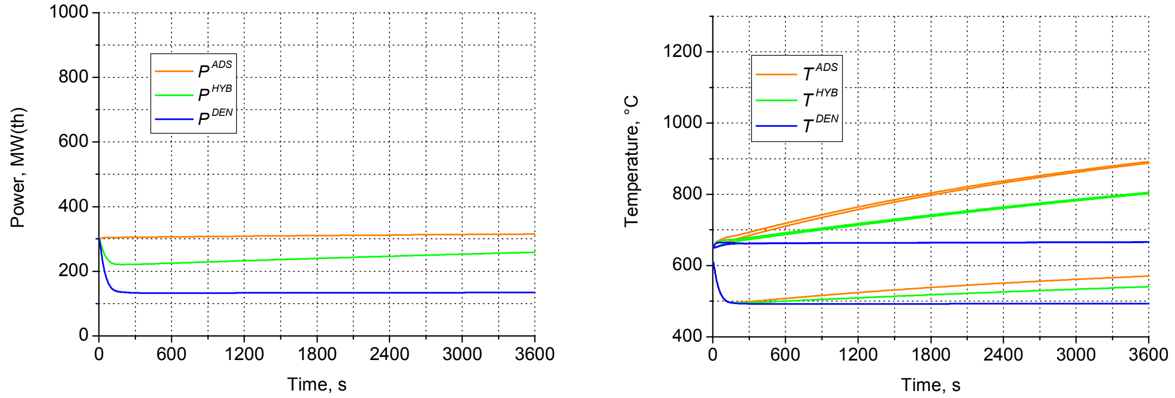


Figure 20. ULOF transients in the thermal spectrum (Th-fuelled) systems with large subcriticality level.

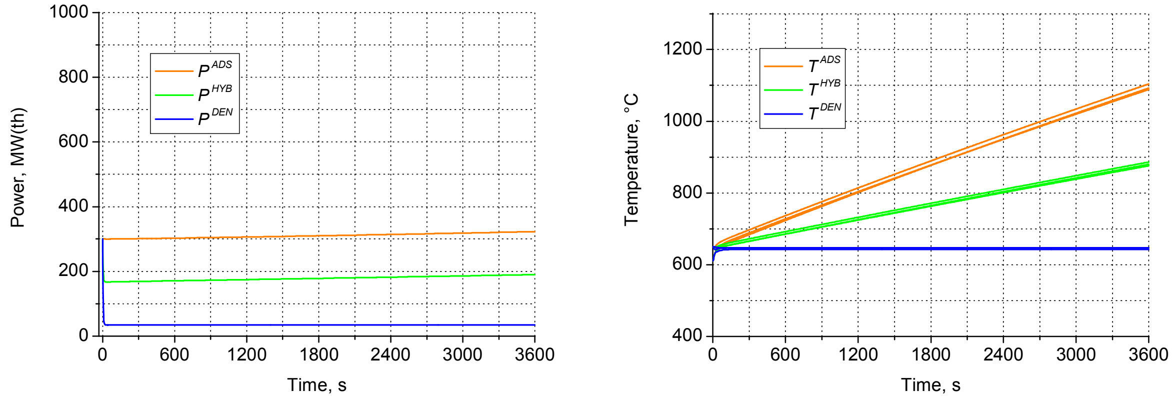


Figure 21. ULOHS transients in the thermal spectrum (Th-fuelled) systems with a large subcriticality level.

2.8. Unprotected transients in the thermal-spectrum uranium- and transuranics-fuelled systems

Analysis of these systems is interesting because feedback effects are much more favorable compared with the Th-fuelled option, analyzed in the previous Section. Their characteristics allow choosing the "standard" recommendation (see Section 2.3.1) for the subcriticality level, namely: $r_0 = 2.4\beta$.

2.8.1. Unprotected TOP/TOC transients

Figure 22 represents the behavior of the core power and core temperatures in the all systems with the maximal reactivity margin insertions. Note, that in the present Section the temperatures of both fuel and graphite are presented in figures.

In **critical systems**, a significant Doppler-effect leads to a large core power and temperatures despite the large thermal capacity of the core (due to the presence of graphite). The maximal temperature of fuel is achieved quickly (in a few seconds) and this situation lasts for a long time (about 200 s). This high temperature level is aggravated by the subsequent oscillations. It means that **critical reactor** is not capable for the self-protection. On the other hand, the asymptotic temperature level is not too high (due to the significant negative feedback effects) and this gives a chance for **DEN** system to survive (as usually, UTOC is the weakest point of **DEN**). This analysis has shown that all subcritical systems have a remarkable deterministic safety potential during the UTOC events.

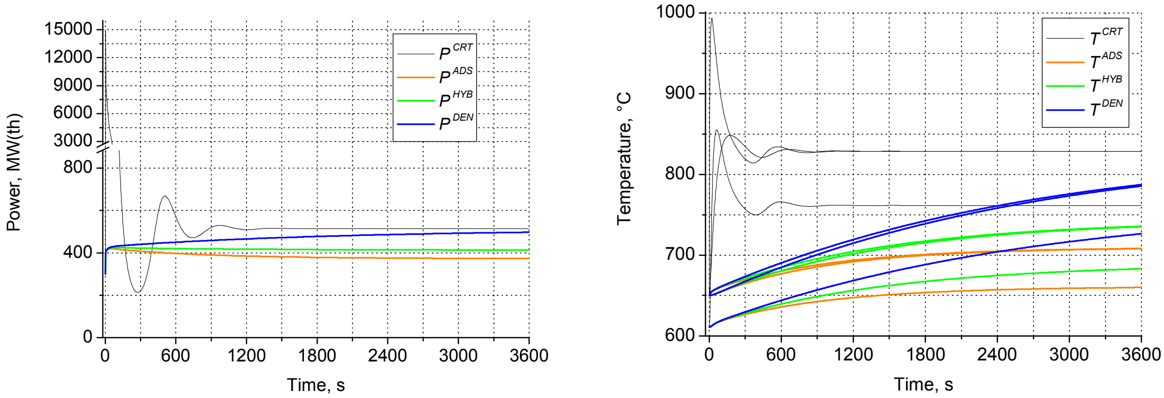


Figure 22. Unprotected TOP/TOC transients in the thermal (U+TRU-fuelled) spectrum systems; $\Delta\rho_{\mathrm{TOP/TOC}} = \beta$, $r_0 = 2.4\beta$.

2.8.2. Unprotected LOF transients

ULOF events for the **critical system** demonstrate undesirable oscillations of power. On the other hand, the temperature regimes for all systems (including critical system) are acceptable with respect to the deterministic safety requirements as seen from Figure 23.

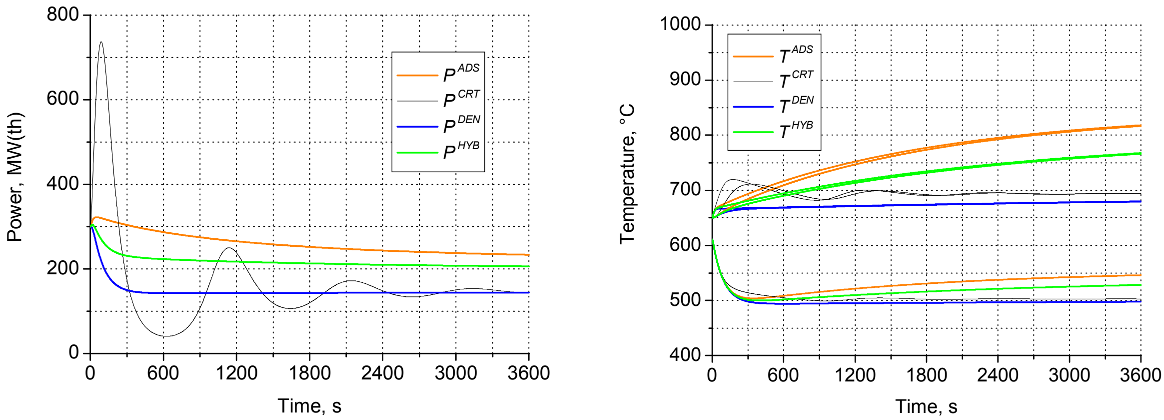


Figure 23. Unprotected LOF (pump power fall from 100% to 10% in the period of 10 s) transients in the thermal (U+TRU-fuelled) spectrum systems. The subcriticality level $r_0 = 2.4\beta$ was used.

2.8.3. Unprotected LOHS transients

Despite the "specific" character of the power transient in the **critical reactor** (cooling of the graphite moderator causes some oscillations), all regimes of the critical mode seems to be acceptable (see Figure 24). On the other hand, the **ADS** behaviour is not without some important doubts: the core temperature increases monotonically. Behavior of the **DEN** system is the most preferable with respect to the deterministic safety requirements as shown in the same Figure 24.

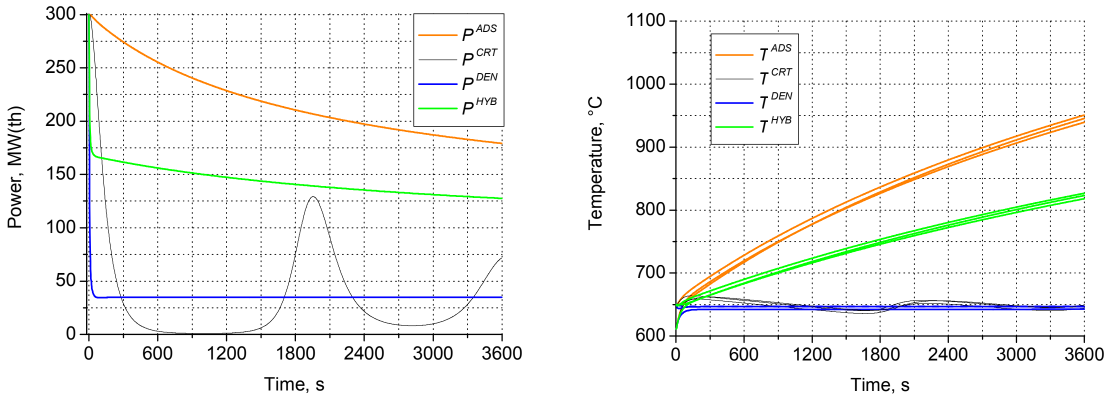


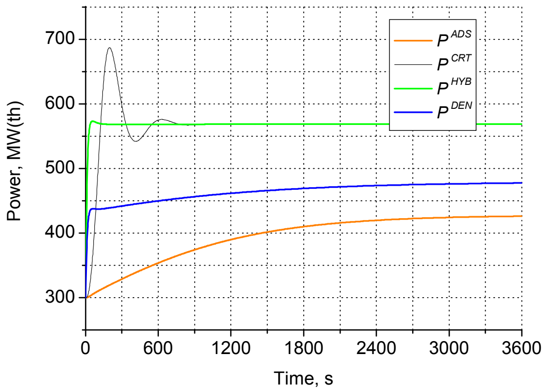
Figure 24. Unprotected LOHS (loss by 90% in the period of 3 s) transients in the thermal (U+TRU-fuelled) spectrum systems; subcriticality level $r_0 = 2.4\beta$ was used.

2.8.4. Unprotected GOF transients

Our results show that the UGOF events do not lead to serious consequences for any of the above systems, i.e. both the critical and sub-critical ones. Although, some power oscillations take place in the case of critical system the core temperatures behave smoothly for all systems.

2.8.5. Unprotected Over-Cooling (UOVC) transients

As it was shown above, the temperature level T_k in the condenser plays an important role in preventing the molten salt from solidification during different transients. To avoid salt solidification it was proposed to increase T_k from 20°C to 400°C. At these new conditions, a rapid decrease of T_k would lead to the core cooling and eventually to the unprotected UOVC transients. Figure 25 presents these transients in the case of the U+TRU fuelled systems. It can be seen that the subcriticality leads to a decrease of the power oscillations, which are inherent to critical systems. Moreover, the DEN-system supplies the temperature regime with an important stability and still far from the fuel solidification point.



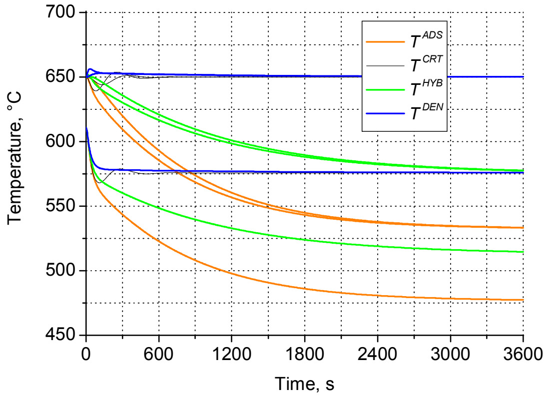


Figure 25. Unprotected overcooling (decreasing of heat exchanger temperature down to 20°C in the period of 10 s) transients in thermal (U+TRU-fuelled) systems.

2.9. Discussion

For the first time, a comparative analysis of the dynamics of different types (ADS and ACS/DEN) of hybrid systems was carried out. The above analysis of multiple unprotected events yields some general conclusions with respect to the possibility of achieving the deterministic level of safety both for fast and thermal spectra systems.

Despite the very favorable safety physics potential of the molten salt critical cores (small reactivity margins, the limited change of reactivity, etc.), critical systems have the limited potential in achieving the deterministic safety level, particularly, when the combinations of reactivity insertions and fuel circulation stop are assumed. Significant oscillations of power and fuel temperatures are the main drawbacks of unprotected transients even with favorable negative reactivity feedback effects. In the cases of positive feedback effects and of significant Doppler-effect degradation, deterministic safety of critical systems is generally not achievable.

The subcritical regime (both ADS and DEN) improves the safety potential significantly, leading to the considerable increase of the grace time up to dozens of minutes in the case of "degraded" feedback effects and up to several hours in the case of the "standard" negative feedback effects. This effect is observed even for small subcriticality levels of 1-3 dollars.

One of the most important safety effects of subcritical systems is the suppression of power and fuel temperature oscillations during unprotected transients. This significant enhancement of safety could play an important role for long-lived waste transmutation (degraded safety properties).

The weakest point of ADS in respect to the deterministic safety is thermo-hydraulic unprotected transients which exhibit a continuous increase of temperatures despite favorable feedback effects. It means that ADS are unable to present the "unlimited" grace time.

Unlike ADS, DEN demonstrate acceptable behavior with respect to most unprotected transients (ULOF, ULOHS, UGOF, UOVC), while the TOP/TOC transients remain a point of concern. The DEN systems inherit the best safety features of both critical reactors and ADS. For that, the total feedback reactivity effect has to be negative.

As demonstrated, even a very small subcriticality level improves significantly the safety of the system. Therefore, the economics of subcritical systems should not be penalized significantly when compared with critical reactors due to the small consumption of energy for proton acceleration: neither powerful accelerators, nor large energy consumption are required for spallation. According to the pessimistic assessment, about 0.002-0.003 mA of proton current per MW(th) of system total power is expected to be sufficient.

The results that we presented above are a good illustration of the quite diverse inherent dynamic behavior of the critical- and different kinds of subcritical systems. In the next Chapter we will pass on from the direct problem formulation (how does dynamics change in presence of subcriticality) to the inverse one: i.e. how the core subcriticality may be used for safety improvement.

3. Core subcriticality as a tool of safety enhancement

Résumé *– Dans ce Chapitre nous montrerons que la sous-criticité du cœur peut jouer un rôle important pour l'amélioration de la sûreté d'un système nucléaire si c'est nécessaire. C'est le cas lors de la dégradation d'effets de contre-réaction ou/et de la réduction significative de la fraction des neutrons retardés causée par les actinide soumis à la transmutation. Deux types de réalisation d'un système sous-critique : l'ADS et l'DEN (système sous-critique avec couplage entre le taux de fission et l'intensité de la source externe de neutrons) sont étudié dans ce contexte. Il est montré que tous les deux peuvent améliorer la sûreté. Deux moyens : une optimisation thermo-hydraulique et la sous-criticité peuvent compenser la dégradation de l'effet Doppler et la réduction de la fraction des neutrons retardés. De plus, dans un DEN le couplage thermo-hydraulique entre un cœur sous-critique et une source de neutron de spallation produit un groupe complémentaire de neutrons retardés.*

3.1. Foreword

The results of the safety analysis, reported in the previous Chapter prove that a core subcriticality can play an important role if the safety enhancement of a nuclear system is necessary. In the present Chapter we demonstrate that core subcriticality together with a thermo-hydraulics optimization can compensate the possible degradation of the Doppler-effect and the reduction of the delayed neutron fraction. A number of quantitative examples are provided in this context. In this Chapter the problems are formulated in the way to demonstrate how the core subcriticality can "cure" these particular safety problems.

After comparison of the kinetic features of different kinds of subcritical systems (Chapter 2), it became obvious that the response of these systems on the external perturbation is intrinsically different. Moreover, as mentioned above, the control of these systems may differ as well. Consequently, it is imperative to treat DEN and ADS separately. Moreover, it seems that the *entire* safety analysis for each system has to be built in a different way.

3.2. Study of a DEN-system based on the AMSTER core

3.2.1. Introduction

In Chapter 2 we analyzed a number of "hypothetic" systems. Let us apply a similar approach to a reactor with somewhat more developed concept. This will allow us to reduce uncertainties and to carry out more detailed investigation in the different stages of study: from the analysis of phenomena causing reactivity variations to transient simulations. The molten salt AMSTER core is taken as a reference in our study.

The DEN system has demonstrated the more promising potential (when compared with ADS) on the way to reach deterministic safety. For that reason we concentrate our attention on the DEN systems. In contrast to Chapter 2, where the subcriticality level was fixed, in the present Section we will monitor the evolution of the system's behavior with increase of the subcriticality level.

This Section aims determining the safety potentials of the subcritical MSR coupled to the external neutron source. Critical and subcritical MSR with different subcriticality levels are examined in a comparative way. A set of unprotected transients as UTOP, ULOF, ULOHS, UTOC and their combinations are then evaluated in so-called "source dominated" (deeply sub-critical system) and "feedback dominated" (slightly sub-critical system) regime (see Ref. [5] for details). Finally, an inter-comparison between critical and coupled sub-critical systems is carried out with core sub-criticality level being a free parameter.

A simplified point model of the core kinetics is adopted for safety analysis. The mathematical model, analogous to this one applied in previous Chapter, includes also a description of the thermo-hydraulics of the circulated fuel as well as feedback effects in the core.

3.2.2. Major parameters of the reference molten salt cores

The thermal-spectrum molten salt reactor AMSTER (after Actinides Molten Salt TransmutER) is chosen as a reference MSR. Two different AMSTER cores (see Refs. Vergnes *et al.*, 2000; Lecarpentier, 2001) have been selected for our analysis. The first one is the **TRU-incinerator** core with support-uranium and the second one is the **self-generator** core with support-thorium. Table VI and Table VII summarize feedback effects and delayed neutron characteristics in both cases. We note the major difference between two systems considered is negative and slightly positive total feedback in the case of uranium and thorium based cores respectively (see Table VI).

Table VI. Feedback effects in the reference cores (from Ref.: Lecarpentier, 2001).

coefficient, pcm/°C	support-uranium	support-thorium
$\alpha_{Doppler}$	- 3.01	- 2.40
$\alpha_{density}$	+ 1.45	+ 1.95
α_{salt}	- 1.57	- 0.45
$\alpha_{graphite}$	- 1.20	+ 1.20
$\alpha_{feedback}$	- 2.45	+ 0.75

A discussed earlier, a particular feature of systems with circulating fuel is a partial loss of delayed neutrons during its circulation outside the core region. In the case of AMSTER-like configuration this delayed neutron decrease is as much as ~30% (compare β and β^* in Table VII). Indeed, in the support-uranium core the delayed neutron fraction falls from $\beta = 446\,\mathrm{pcm}$ down to $\beta^* = 336\,\mathrm{pcm}$, and for support-thorium core it decreases from $\beta = 350\,\mathrm{pcm}$ down to $\beta^* = 256\,\mathrm{pcm}$. It is interesting to note that delayed neutron fraction in AMSTER-like cores is considerably smaller if compared to industrial Pressurized Water Reactors - PWR (typically of the order 650 pcm), but comparable to the β of a fast breeder reactor. It should be also stressed that the delayed neutron precursors with the longest life-times ($\lambda_i < 1$ s^{-1}) decay more than the others out of core, what is not advantageous at all for the reactor control (compare β and β^* for all six groups in Table VII).

To quantify further the delayed neutron decrease we will estimate a so called one-group decay parameter in the case of mobile and immobile fuels. Generally, the one-group decay parameter is expressed as follows (Hetrick, 1971):

$$\lambda^{-1} = \sum_{i=1}^{6} \beta_i \lambda_i^{-1} \Big/ \sum_{i=1}^{6} \beta_i \ . \tag{3.1}$$

In the mobile fuel core we have to replace β_i by β_i^*. Thus, for the support-uranium core this parameter equals $\lambda^{-1} = 10.7\,\mathrm{s}$ (12.85 s for immobile salt), and for support-thorium core - $\lambda^{-1} = 14.8\,\mathrm{s}$ (16.42 s for immobile salt). Another important kinetic parameter, which is also used in our study, is the prompt neutron generation time Λ. Its value is of the order of $4 \cdot 10^{-4}\,\mathrm{s}$ for both the support-uranium core and for the support-thorium core (Lecarpentier, 2001).

Table VII. Delayed neutron fractions in the AMSTER cores with support-uranium and support-thorium (from Ref.: Lecarpentier, 2001). Also see the text.

Group	λ_i, s^{-1}	β_i, pcm		β_i^*, pcm	
		support-uranium	support-thorium	support-uranium	support-thorium
1	0.013	12	23	8	15
2	0.032	102	93	66	60
3	0.128	85	79	60	55
4	0.304	170	114	130	88
5	1.349	63	27	58	25
6	3.629	14	13	14	13
Total		446	350	336	256

3.2.3. Reactivity variations in the reference MSR

A complete safety analysis requires simulating the multitude of all accidental situations permitted by physical laws. In this Section we continue to analyze a limited set of unprotected accidents: Unprotected Transient Over Power (UTOP) – accidental insertion of all reactivity reserve or physical process leading to change of core reactivity; Unprotected Loss Of fuel Flow (ULOF) accident – failure of first loop pumps; Unprotected Loss Of Heat Sink (ULOHS) from the first loop. Another restriction of our approach is that only accidents in a nominal regime are considered.

To simulate accidental reactivity growth in the system and to give some recommendations on the choice of subcriticality level, let us analyze possible causes of reactivity insertion. The data collected in the Table VIII are evaluated-extracted from Refs. (Blinkin and

Novikov, 1978; Lecarpentier, 2001; Nuttin, 2002). In this table we distinguish two groups of reactivity change. The first one is so called “fast reactivities”, i.e. reactivities that can be inserted rather quickly (either by physical processes or control rods devoted for their compensation). The second group contains rather slow physical processes, what in principle can be improved by continuous fuel reprocessing (see Ref. [Nuttin, 2002] for details), and therefore no control rod reactivity reserve has to be anticipated for their compensation.

Table VIII. Physical processes leading to the reactivity variation and corresponding reactivity values in the reference cores (from Refs.: Blinkin and Novikov, 1978; Lecarpentier, 2001; Nuttin, 2002).

reason for reactivity variation	**support U**	**support Th**
reactivities, compensated by control rods (“fast reactivities”), pcm		
homogenous core heating		
450°C to 562°C	- 275[a]	+ 84[b]
562°C to 705°C	- 350[a]	+ 107
705°C to 1300°C	–	+ 446[a,b]
Total	- 625[a]	+191 (+ 637[b])
^{135}Xe poisoning (Blinkin and Novikov, 1978)	- 32	- 32
fuel stop	+ 110	+ 94
decompression (Blinkin and Novikov, 1978)	+ 100	+ 100
total (“fast reactivities”)		
nominal regime	242	333
start-up regime	409	226
maximum	867	943
reactivities, compensated by adjustment of fuel composition (“slow reactivities”), pcm		
$^{239}Np/^{233}Pa$ – effect (Nuttin, 2002; Adamov *et al.*, 2001)	+ 50	+ 1600 (2.5 pcm/h)
fluctuation of fissile isotopes concentration (Nuttin, 2002)	+ 400 (180 pcm/h)	+ 400 (180 pcm/h)
Total (“slow reactivities”)	450	2000
Total (“fast + slow”)	1317	2943

[a] Empty at nominal regime.

[b] Supplementary margin which may be introduced for account of positive feedback effect.

In our analysis we will search for the hardest conditions of accident development in terms of system’s safety. Technically this can be achieved by inserting all reactivity reserves simultaneously.

As is mentioned above, the ULOF accident is particularly dangerous in mobile fuel systems. It does not only fail to remove the heat from the core but it also causes reactivity growth by the value

$$\Delta\rho_{precursors} = \left(\beta - \beta^{*}\right)$$

due to all precursors decay in the core. In this study it is supposed that the fuel flow decreases by 90 % of its nominal value (the remaining 10% is believed to be assured by natural convection). In our simulation of the UTOP accident alone we do not include $\Delta\rho_{precursors}$ into

$\Delta\rho_{\mathrm{TOP}}$ (this is equivalent to assumption that no reactivity reserves are foreseen to compensate this effect). However, when we simulate superposition of UTOP and ULOF, both $\Delta\rho_{precursors}$ and $\Delta\rho_{\mathrm{TOP}}$ are taken into account, what corresponds to the situation when potentially maximal reactivity can be inserted. The particularity of the LOHS-accident in ACS (DEN) is that the loss of heat transfer to the energy generation device switches off external neutron source, what is favourable for system inherent safety in the case of this accident.

3.2.4. Simulation of Unprotected Transients

Here we provide just a brief description of the model. A complete description of the mathematical formulation used for accident simulations is nearly identical to this one utilized in Chapter 2. The only difference is that now 6 groups of delayed neutrons are introduced and the heat release in the graphite (about 8 % of the total thermal power; after Ref. Blinkin and Novikov, 1978) is taken into account. Neutronics of the nuclear system is described by the point-wise model of a core filled by homogeneous molten salt and graphite. The thermo-hydraulic model of the first cooling loop includes two spatially separated elements: the core and the heat-exchanger connected by tubes of finite dimensions. Our mathematical model contains a coupled system of point reactor kinetics equations with six groups of delayed neutrons, salt and graphite reactivity feedback effects, thermo-hydraulic equations and initial conditions (for nominal regime). The intensity of an external neutron source is proportional to the output energy. Since no parameters of the 2$^{\text{nd}}$ loop and energy production system are available by now in the AMSTER project (Lecarpentier, 2001), there are no 2$^{\text{nd}}$ and 3$^{\text{rd}}$ loops included in our model. It is supposed that electric energy is produced immediately after the 1$^{\text{st}}$ loop, what results in the underestimation of the subcritical system safety. An environment temperature of 450°C is chosen to avoid eventual salt solidification. Newton cooling model is used for description of heat exchange with this environment. It is supposed that the maximal acceptable temperature during accidents is the temperature of salt boiling (~1300°C) as it was proposed by Lecarpentier (2001).

In our work we carry out a parametric study of subcriticality impact on the MSR safety. Four different levels of subcriticality r_0 have been chosen for simulation:

(i) 100 pcm being the maximal limit for the industrial hybrid MSR in the case of electron accelerator driver (see discussions in Ref. (Bokov et al., 2003) as well as in Chapter 6 of this thesis work for further explanations);

(ii) 350 pcm being a level, corresponding to β-compensation up to the value typical for industrial nuclear reactors (600-700 pcm);

(ii) 700 pcm being the value comparable with the fraction of delayed neutrons in industrial nuclear reactors;

(iv) 1050 pcm being close to the maximal limit for the prototype hybrid MSR based on the electron accelerator driver (see Chapter 6).

In the **TRU incinerator system (support uranium)** maximal reactivity insertion $\Delta\rho_{ext} = 132$ pcm is simulated according to Table VIII. Obtained results are summarized in Table IX in terms of the maximal temperature increase and the corresponding time (in parenthesis) to reach this temperature. After detailed analysis of the obtained results we can formulate the following findings (also see Figure 26):

– all accidents in all systems (including critical one with $r_0 = 0\,\text{pcm}$) do not lead to dangerous temperature growth;

– nevertheless, added subcriticality improves the behavior of system during transient;

– this improvement is not only quantitative, but it is also qualitative: one can see that most of the transients become slower, smoother (no oscillations) and monotonous (asymptotic value is also the maximal value) with increasing subcriticality;

– artificial feedback caused by the core-accelerator coupling changes the behavior of system during complex accidents; e.g., contrary to a critical system, in the case of a subcritical system (1050 pcm) the superposition of ULOF and UTOP accidents is less dangerous than single UTOP.

We conclude that in the case of the TRU incinerator core (support uranium) the added subcriticality is not indispensable to enhance its safety, simply because the system has got its inherent safety potential from the very beginning (e.g., $\alpha_{feedback} = -2.45\,\text{pcm/°C}$ from Table VI). Nevertheless, it should be mentioned that qualitative improvements on the system's response to different unprotected transients are observed already at subcriticality level around 350 pcm.

Table IX. Maximal temperature reached in the support uranium TRU incinerator core, and the corresponding time needed to reach this value (given in parenthesis) for different unprotected transients.

r_0 , pcm	UTOP	ULOF	UTOP + ULOF	ULOHS
critical	771°C (16 s)	782°C (22 s)	**844°C** (18 s)	761°C (30 s)
100	759°C (28 s)	766°C (18 s)	**810°C** (19 s)	741°C (34 s)
350	751°C (145 s)	750°C (12 s)	**769°C** (15 s)	721°C (47 s)
700	751°C (350 s)	744°C (10 s)	**753°C** (11 s)	712°C (47 s)
1050	751°C (350 s)	742°C (10 s)	747°C (10 s)	710°C (52 s)

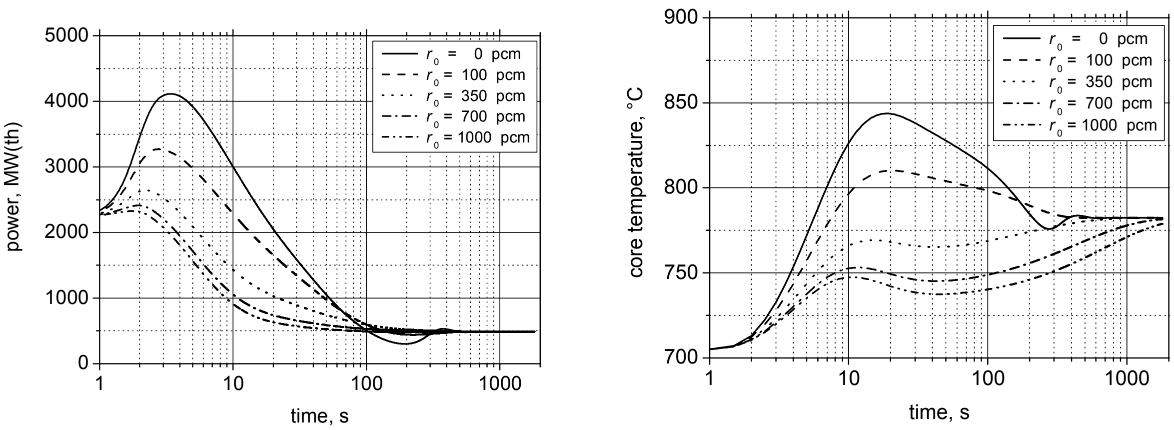


Figure 26. Simulation of a complex accident in the support-uranium TRU incinerator system: the UTOP accident (insertion of $\Delta\rho_{ext} = 132$ pcm) with the simultaneous ULOF accident (reduce of salt flow by 90 % in the period of 10 s).

A particularity of the **self generator system (support thorium)** is its negative salt feedback effect and a strong positive feedback effect of graphite (see Table VI for details) resulting in a slightly positive total feedback in the case of a homogeneous core heating.

The maximal TOP-reactivity insertion is $\Delta\rho_{ext} = 239$ pcm (see Table VIII), what is slightly smaller compared to $\beta^* = 256$ pcm for this system. Taking into account eventual reactivity growth due to the fuel stop (~94 pcm), a prompt criticality is probable. The positive total feedback effect, as it is described above, can not any longer prevent a possible core power excursion. Let us study a subcriticality role for safety enhancement of this particular system.

As in the previous case, we chose subcriticality levels of 100 pcm, 350 pcm, 700 pcm, and 1050 pcm for unprotected accident simulations. A supplementary level of 2000 pcm is also tried to study system behavior in a so-called "source dominated domain" or deep subcriticality regime. Results of our simulation are presented in Table X and Figure 27. Below we summarize our major findings:

- due to the positive total feedback unprotected transients in all cases lead to salt temperature raise up to the temperature $T^{\dagger} = 1300°\text{C}$ of salt boiling, what is considered as a disintegration criterion of the system (Lecarpentier, 2001);
- added subcriticality increases core vitality period (time from the beginning of an accident to salt boiling): approximately 10 s per 100 pcm in the "feedback dominated" region up to approximately 25 s per 100 pcm in the "source-dominated" region;
- similarly like for the support-uranium system, subcriticality level higher than ~100 pcm results in UTOP and ULOF accident superposition less dangerous than UTOP taken alone.

Table X. Time necessary to reach the boiling temperature (1300°C) of salt in the case of different unprotected transients in the support-thorium self generator system.

r_0, pcm	UTOP	ULOF	UTOP+ULOF	ULOHS
critical	10 s	67 s	7.5 s	112 s
100	20 s	137 s	17.5 s	457 s
350	59 s	707 s	82 s	6580 s
700	136 s	1964 s	389 s	> 2 h
1050	228 s	3324 s	853 s	> 2 h
2000	506 s	> 1 h	2232 s	> 2 h

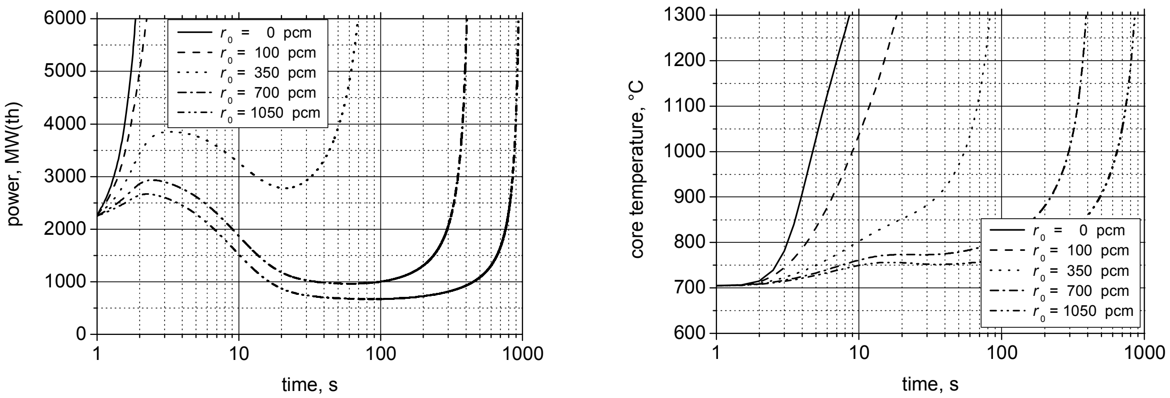


Figure 27. Simulation of complex accident in the support-thorium self generator system: the UTOP accident (insertion of $\Delta\rho = 239\,\mathrm{pcm}$) followed by the ULOF accident (reduce of salt flow by 90 % in the period of 10 s).

We found it interesting to compare directly the influence of subcriticality with an improved feedback effect in the case of critical system. The question can be formulated as follows: *what subcriticality level would give the same result for system vitality persistence as feedback effect improvement*? In order to answer this question a set of supplementary simulations were carried out.

We start our analysis by comparing two ways of feedback optimization:
(i) of graphite $\alpha_{graphite}$, and
(ii) of salt $\alpha_{salt} = \alpha_{expansion} + \alpha_{Doppler}$,
giving the same value of $\alpha_{feedback}$.

We note separately that in our study we use linear model of feedback, so the variation of either Doppler-effect or salt expansion effect gives the same final result. A purpose of this comparison is to verify which parameter is more favorable for safety enhancement. The simulation of transients in the critical system with modified (ameliorated) feedbacks showed that it is preferable to optimize α_{salt} because it is faster and, therefore, more effective than graphite feedback effect.

Afterwards, we carried out a parametrical study of sensibility of critical system behavior due to the variation of Doppler-effect. We simulated the increase of $\alpha_{Doppler}$ by 10 % and 20 % with respect to the initial reference value. In Table XI we present system vitality time for different unprotected transients. By comparing these results with the ones, presented in Table X, and with the help of some interpolations, we conclude that
(a) 10 % amelioration of the Doppler-effect would be comparable with approximately 150 pcm of the additional core subcriticality;
(b) 20 % improvement of the salt feedback effect would be comparable with approximately 320 pcm of the additional core subcriticality.

Table XI. Time necessary to reach the boiling temperature (1300°C) of salt in the case of different unprotected transients in the support-thorium self generator system as a function of different Doppler coefficients and for a critical system only.

Doppler, pcm/°C	UTOP	ULOF	UTOP+ ULOF	ULOHS
Ref. -2.40	10 s	67 s	7.5 s	112 s
-2.64	31 s	111 s	23 s	196 s
-2.88	71 s	195 s	53 s	425 s

3.2.5. Conclusions

This Section aimed in determining the safety potentials of the subcritical MSR (AMSTER-like core) coupled (DEN-type coupling) to the external neutron source. A direct comparison between critical and sub-critical systems was done by simulating a number of unprotected transients. A point kinetics model of the core was adopted for safety analysis. The mathematical model included a description of the thermo-hydraulics of the circulated fuel as well as feedback effects in the core.

Two different AMSTER-like systems were chosen for our analysis. The first one was the TRU incinerator core with support-uranium and the second one was the self generator core with support-thorium. The major difference between them was the negative and slightly positive total feedback effects respectively. Different levels of sub-criticality were tried in order to improve safety of the system and, at the same time, to define the required intensity of an external neutron source in each case.

The following conclusions can be drawn after our investigations on both systems mentioned above:

Support-uranium core. The added sub-criticality is not indispensable to enhance its safety, simply because the system seems to have its inherent safety potential from the very beginning. In other words, all simulated accidents did not lead to dangerous temperature growth. Nevertheless, qualitative improvements on the system response to different unprotected transients are observed already at sub-criticality level around 350 pcm: *most of the transients become slower, smoother and monotonous with increasing sub-criticality.*

Support-thorium core. The *positive* total feedback effect of the system in this case can not any longer prevent a possible core power excursion. As a result, unprotected transients in all cases led to salt temperature raise up to the boiling temperature $T^{\dagger} = 1300°\mathrm{C}$, what was considered as a disintegration criterion of the system. Added *subcriticality increased the grace time* from approximately 10 s per 100 pcm in the "feedback dominated" regime up to approximately 25 s per 100 pcm in the "source-dominated" regime. In terms of an absolute value, a considerable expansion of the grace time (by a factor of 10) was achieved even if a very low subcriticality level (350-700 pcm) is applied.

Summarizing above results, the following preliminary recommendations for subcriticality application (in the framework of the DEN concept) may be done:

– in cores with a “good” feedback an added subcriticality may be utilized to make unprotected transients smoother and to eliminate the potential overheating (above the asymptotic temperature level) of the core;

– In cases of the DEN-system with a “bad” feedback the deterministic safety is apparently not achievable. However, as in the case of ADS (see Ref. Schikorr, 2001 for detail), the large subcriticality level allows decrease the sensibility of the system on this unfavorable feedback. As demonstrated, it may lead to increase of the grace time up to several hours.

3.3. Kinetics of DEN-systems

3.3.1. Introduction

In some cases critical systems may suffer from the *decrease of the delayed neutron fraction* due to the specific fuel properties (e.g., in actinide transmuter cores, in systems with circulating fuels, etc.). It may lead to undesirable acceleration of transients, consequently to the necessity of the limitation of admissible reactivity variation and, finally, to the significant deterioration of safety. As already discussed in previous Sections, the fraction of delayed neutrons may be artificially increased with the help of the external neutron source, namely in the framework of the ACS- and DEN-concepts (Slessarev *et al.*, 1999; Gandini *et al.*, 2000; Bokov *et al.*, 2003; Slessarev and Bokov, 2004) or “beta-compensated” reactor concept (Bernardin *et al.*, 2001; Ridikas *et al.*, 2002).

Nevertheless, despite the “obvious” intuitive significance of the DEN-concept approach, a clear understanding of the kinetic and dynamic issues of such a coupling is not achieved yet. The studies by Slessarev *et al.* (1999), Gandini *et al.*, (2000), D’Angelo *et al.* (2001), Slessarev and Bokov (2004) were pioneer works in this way.

The objective of the study presented in this Section is to characterize the appearance of a supplementary group of delayed neutrons within the framework of the ACS (DEN) concept. Another goal is to characterize the kinetics of the DEN-system (in absence of in-core feedbacks) and, in particular, its response to variation of the efficiency of the external neutron source. The Accelerator Coupled System (ACS) is taken as example, although this analysis may be expanded to other types of the coupled subcritical systems. Within the framework of a simple mathematical model of coupled system, an interpretation of the external neutron source as a supplementary group of delayed neutrons is given. An auxiliary quantity – ‘*source reactivity*’ is introduced for convenience and a *modified inhour equation* for coupled systems is deduced. Analytical solution of the modified inhour equation is obtained in approximation of one group of delayed neutrons.

The principal conclusion resulting from this analysis is as follows: the response of the coupled system to “source reactivity” variation is intrinsically different from the response to core reactivity variation. Namely, there is no equivalent of prompt criticality (accompanied by drastic decrease of the reactor period) in the case of ‘source reactivity’ variation.

3.3.2. New kinetic features of coupled hybrid systems

Operating in the critical mode with the subcritical core and coupled in a way discussed above, the hybrid system becomes similar to the critical reactor from the point of view of its dynamics (Gandini *et al.*, 2000; D'Angelo *et al.*, 2001). In this context, the fraction f of produced energy, feeding the external neutron source, plays an essential role in the reactor kinetics: a proper choice of its nominal value f_0 guarantees the self-sustainability of the system with respect to the entire neutron balance (comprising fission neutrons and external neutrons). Any mismatch of this parameter to the value, necessary to maintain chain reaction in the core, would lead either to reactor runaway or to gradual attenuation of chain reaction in the absence of thermal feedbacks. That is why the parameter f may be considered as an analogue of reactivity $\rho = \left(k_{eff} - 1\right) / k_{eff}$ for the 'external' contribution to neutron balance in the core.

Gandini *et al.* (2000) assumed that parameter f is fixed at any time and can be slowly adjusted during burn-up to follow the subcriticality level evolution. One may anticipate that in practical situations f may vary from its nominal value. This can be due to uncertainties, technical failures, human errors, variation of energy production efficiency, etc. For this reason, and in view of the role played by f in the neutron balance, it becomes quite important to study kinetic or/and dynamic responses of a coupled hybrid system to fluctuations of f, and to compare the resulting response with this one due to reactivity fluctuations.

Nevertheless, one may expect that this response to an *equivalent*[e] perturbation of the parameter f would be *intrinsically different*. Preliminary considerations for such a statement are as follows. Indeed, the above analogy of f to reactivity is valid only in the case of the quasi-static variation of the reactor power. If the subcriticality level is chosen to be sufficiently large, the reactor core remains subcritical at any instant. This means that a self-sustaining nuclear reaction in the core remains impossible. On the other hand, any response of the external source on reactor power perturbation is delayed in time: the external neutron source "waits" for the arrival of fission energy. As a result, any perturbation of the fraction f results in a prompt reactor response only in the very beginning of the transient. After that a slow transient takes place, limited by the inertia of heat transfer in the reactor.

Summarizing the above considerations, one may conclude that the fraction f would be *analogous to* ρ *from the point of view of a quasi-static neutron multiplication factor*, but its role for the kinetics of coupled system would be *quantitatively and qualitatively different*.

Below we elucidate and qualify these differences.

3.3.3. On "artificial" group of delayed neutrons and its delay time

We introduce a simple mathematical model, describing a coupled hybrid system, what permits us to carry out a complete analytical study as well as to reveal and illustrate the most significant kinetic features of the system under consideration. Moreover, it permits us to realize a direct comparison with conventional point kinetics of a critical reactor.

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[e] See explanation below.

The equations of point kinetics for a coupled system can be presented in the "classical" form (Waltar, Reynolds, 1981):

$$\begin{cases} \dfrac{d}{dt} P(t) = \dfrac{\rho(t) - \beta}{\Lambda} P(t) + \sum\limits_{i=1}^{N_g} \lambda_i W_i(t) + S(t), \\ \dfrac{d}{dt} W_i(t) = \dfrac{\beta_i}{\Lambda} P(t) - \lambda_i W_i(t), \quad i = \overline{1, N_g}. \end{cases} \tag{3.2}$$

where $\rho(t) = -r_0 + \Delta\rho(t)$ is the core reactivity, $r_0 = \left(1 - k_{eff}^0\right) / k_{eff}^0$ is the nominal subcriticality level and $\Delta\rho(t)$ is the eventual reactivity variation; P is the reactor specific power; the term W_i describes the contribution of delayed neutron precursors of the i^{th}-group with the fraction β_i and the corresponding decay constant λ_i; $\beta = \sum\limits_{i=1}^{N_g} \beta_i$ is the total fraction of delayed neutrons; Λ is the neutron generation time; the term $S(t)$ describes an external source of neutrons.

It is generally assumed (see Refs. Slessarev *et al.*, 1999; Gandini *et al.*, 2000; D'Angelo *et al.*, 2001; Slessarev and Bokov, 2004) that the intensity of the external neutron source $S(t)$ is *proportional* to the output power $P^{out}(t)$ of the reactor in *coupled* hybrid systems. Let us denote by f the fraction of produced power feeding the external source. Then one may express the intensity of the external source: $S(t) = \psi_s f P^{out}(t)$, where ψ_s is the corresponding normalization coefficient. For neutron self-sustainability, in nominal conditions the external source has to be equal to $S_0 = r_0 P_0 / \Lambda$ (where $P_0^{out} = P_0$, i.e. in its steady state the system has to evacuate all generated heat). Therefore, we find that $\psi_s = r_0 / \left(f_0 \Lambda\right)$ and, therefore for the external neutron source:

$$S(t) = \left(\frac{f(t)}{f_0}\right) \frac{r_0}{\Lambda} P^{out}(t). \tag{3.3}$$

The nominal fraction f_0 of produced power, devoted to feed the external neutron source, depends on the peculiarities and the performance of the specific neutron production mechanism. In general, it may be deduced from neutron economy:

$$f_0 = \frac{\nu r_0 z_n}{\eta_e \varphi^* \epsilon_f}. \tag{3.4}$$

In this notation η_e is the reactor electric efficiency, ϵ_f is the energy released per fission, ν is the mean number of fission neutrons, φ^* is the importance of the source neutrons, c_n is the electric energy cost of neutron production, i.e. the electric energy consumed to generate one source neutron.

The following remark has to be made. From Eqs. (3.3)-(3.4) one finds that $S \propto \eta_e f / \left(z_n \varphi^*\right)$, i.e. the intensity of the external source in the coupled hybrid system depends also on the reactor electric efficiency (η_e) and on the neutron production performance ($z_n \varphi^*$) of the external source. In general, these factors may vary and their perturbation acts in a similar way to a perturbation of the fraction of produced power, feeding the external source

($\delta f = f - f_0$). In the present study we assume that $\delta\eta_e = \delta z_n = \delta\varphi^* = 0$. However, our approach may be easily generalized by the simple substitution:

$$f(t) \rightarrow \frac{\eta_e(t) f(t)}{z_n(t)\varphi^*(t)}. \tag{3.5}$$

For this reason, in our considerations below for the parameter f we utilize the term "*source efficiency*", assuming in such a way a broadened interpretation of this parameter as implicitly incorporating all the scope of above factors.

Let us introduce an auxiliary parameter – *source reactivity*:

$$r \equiv r_0 (f / f_0), \tag{3.6}$$

i.e. a value proportional to the normalized fraction f and to the nominal subcriticality level r_0. Introduced in such a way, the parameter r determines the steady-state neutron multiplication of the system in the same manner as the core reactivity $\rho = (k_{eff} - 1) / k_{eff}$. Indeed, one can demonstrate that steady states ($\bar{P}, \bar{W}_i$) of the coupled system meet the conditions

$$(\rho + r)\bar{P} = 0, \quad \bar{W}_i = \frac{\bar{P}\beta_i}{\lambda_i \Lambda}. \tag{3.7}$$

Introducing the "total" neutron multiplication factor of the system m_{eff} by analogy with core neutron multiplication factor k_{eff}

$$\frac{m_{eff} - 1}{m_{eff}} = \rho + r, \tag{3.8}$$

we may express similarly the condition of neutron self-sustainability (analogue of criticality with regard to the entire neutron balance) for a coupled hybrid system:

$$m_{eff} = 1 . \tag{3.9}$$

Furthermore, as will be demonstrated below, the parameter r determines the kinetic response of the coupled hybrid system to perturbation of the source efficiency like reactivity ρ determines the kinetic response of critical reactors on variation of the core neutron multiplication factor.

Now, let us return to Eqs. (3.2)-(3.3). To take into account the explicit dependence of P^{out} on t, one needs to describe the transfer of fission energy from the core to the electric energy producing mechanism. The simplest one-point thermo-hydraulic scheme of such a heat transfer can be presented by the Newton cooling model (Murray, 1957; Hetrick, 1971)

$$C_p \frac{dT(t)}{dt} = P(t) - P^{out}(t), \tag{3.10}$$

where T is the core temperature and C_p is the heat capacity of the core. The first term on the right-hand side of Eq. (3.10) describes the rate of energy production (i.e. thermal power of the core) and the second describes the rate of thermal energy evacuation. The latter is assumed to be proportional to the temperature drop ($\theta(t) \equiv T(t) - T_k$) to the surroundings (condenser), which may be considered as an infinite reservoir (Murray, 1957), i.e.

$$P^{out}(t) = H\theta(t), \tag{3.11}$$

while H is the corresponding coefficient.

Now, the source term in Eq. (3.2) can be rewritten as follows:

$$S(t) = \frac{r(t)}{\Lambda} H\theta(t). \tag{3.12}$$

The following remark has to be made with regard to Eq. (3.12). Within the framework of the above model there are, in principle, other ways to perturb the reactor equilibrium, related to the external neutron source. For example, *external* impact may manifest itself either in the form of a perturbation of the heat-transfer coefficient H or via perturbation of the environment (condenser) temperature T_k. Both these events are of interest as they lead to corresponding transients in the hybrid system, but, in the absence of in-core feedback effects, they do not corrupt the entire neutron multiplication factor of the system. Therefore, in the present work it is assumed that $\delta H = 0$ and $\delta T_k = 0$.

Let us denote the variable component of the source reactivity for convenience as

$$r(t) = r_0 + \delta r(t) = r_0\left(1 + \varepsilon^{(r)}(t)\right), \tag{3.13}$$

where $\varepsilon^{(r)} = \delta r / r_0$ is the source reactivity variation expressed in values of initial subcriticality level. With the notations

$$W^+ \equiv \frac{r_0 C_p \theta}{\Lambda}, \ \beta^+ \equiv r_0, \ \lambda^+ \equiv \frac{H}{C_p}, \tag{3.14}$$

and from Eqs. (3.2), (3.8), (3.12) and (3.13) one obtains the following system of coupled equations:

$$\begin{cases} \dfrac{d}{dt}P = \dfrac{\Delta\rho - \left(\beta + \beta^+\right)}{\Lambda}P + \displaystyle\sum_{i=1}^{N_g}\lambda_i W_i + \left(1 + \varepsilon^{(r)}\right)\lambda^+ W^+; \\ \dfrac{d}{dt}W_i = \dfrac{\beta_i}{\Lambda}P - \lambda_i W_i, \quad i = \overline{1, N_g}; \\ \dfrac{d}{dt}W^+ = \dfrac{\beta^+}{\Lambda}P - \lambda^+ W^+. \end{cases} \tag{3.15}$$

One may note that in Eqs. (3.15) the external source imitates the evolution of delayed neutron precursors, i.e. the external source plays the role of a supplementary (artificial) group of delayed neutrons (Refs. Gandini *et al.*, 2000, D'Angelo *et al.*, 2001; Slessarev and Bokov, 2004). The parameters W^+ and λ^+ may be interpreted as the effective concentration and the effective decay constant respectively. The subcriticality level plays the role of the fraction β^+ of the "artificial" delayed neutrons.

Despite the above analogy, the artificial group of delayed neutrons provides some unique properties from the point of view of reactor kinetics, namely:

- in contrast to the "natural" groups of delayed neutrons, there is an ability (to some extent) to "design" both the effective "decay constant" λ^+ and the fraction β^+;
- moreover, as discussed above, an external impact may perturb both λ^+ and W^+;
- a perturbation of the external source efficiency may lead to a mismatch between a decay of the "artificial" delayed neutron group [term $-\lambda^+ W^+$ in Eqs. (3.15)] and of its contribution to the neutron balance [term $\left(1+\varepsilon^{(r)}\right)\lambda^+ W^+$]. It should be stressed that a limited analogy to this phenomenon may be flow variation event in the circulating fuel systems;
- for this group of delayed neutrons there is no undesirable growth of reactivity during loss of flow events in systems with circulating fuels;
- penalties related to the supplementary neutron production can be relatively modest because a small level of core sub-criticality, consequently, a rather weak intensity of the external neutron source (when compared with conventional ADS) would be sufficient.

Finally, this consideration shows a *way to "repair" the degraded fraction of delayed neutrons*. In addition, the relations given by Eqs. (3.14) - (3.15) allow characterizing the supplementary artificial group of delayed neutrons in traditional terms of the effective concentration of delayed neutron precursors, their effective decay constant and the effective fraction.

Obviously, these relations were different if another model of heat transfer would be chosen. In the next subsection we will discuss this aspect in more detail.

3.3.4. "Heat removal" model versus "delayed argument" model

We note, that the interpretation of the coupled external source as a supplementary group of delayed neutrons is not novel. For example, in Refs. (Slessarev *et al.*, 1999; Gandini *et al.*, 2000) the authors discussed this problematic within the framework of the "delayed-argument" model, i.e., they supposed that $P^{out} = P\left(t - t_{SP}\right)$. In our work the more adequate "heat-removal model" is applied and, as a result, another interpretation for the parameters λ^+ and C^+ is given.

Our model simulates the heat transfer inertia due to thermal resistance of the heat transfer system. This effect leads to the non-simultaneity of the core power P (i.e., fission rate) and of the reactor output power P^{out}. Another physical mechanism leading to non-simultaneity, namely the time delay t_{SP}, arising physically from the transport of the coolant from the core to the turbine, is not included. Hence, our model is valid for the systems, where the thermal inertia prevails. The issues of the "delayed-argument" model (i.e. $P^{out} = P\left[t - t_{SP}\right]$) are discussed by

Gandini *et al.* (2000) and D'Angelo *et al.* (2001). This delay was equally taken into account (via parameters $\tau_{tube}, \tau_{c \to h}, \tau_{h \to c}$) in our simulations, presented, in Chapter 2.

These differences in the description of the coupling between the core power and the source intensity yield to the important issues for the core dynamic behavior. For example, the "delayed power model" by Gandini *et al.* (2000) may lead to oscillations of core power in the case of a sharp power increase due to reactivity insertion, while the "heat-removal model" predicts a smooth and monotonous behavior of the transient. Indeed, taking into account Eqs. (3.10)-(3.12), the explicit expression for the dependence of the external neutron source S on the core power can be written in the following way:

$$S(t) = \frac{\lambda^+ \beta^+}{\Lambda} \int_{-\infty}^{t} P(t') \exp\left[-\lambda^+ \left(t - t'\right)\right] dt' . \tag{3.16}$$

As a matter of fact, the artificial neutron production in the DEN-system depends on the thermal energy, accumulated in the core [time integral of core power as presented in Eq. (3.16)], as well as on particularities of the heat transfer in the system (parameter λ^+). In other words, the artificial neutron production, caused by a single fission event at any point in time t, will not be only delayed by some characteristic time t_{SP}, but it will be also distributed over the whole period following this event.

This detail was observed during the preliminary simulations in the framework of study presented in Chapter 2. It turned out that for the systems under consideration the variations of the time delay related to fuel circulation (parameters $\tau_{tube}, \tau_{c \to h}, \tau_{h \to c}$) do not affect significantly the transients. It appears that for these systems the heat transfer inertia due to thermal resistance of the heat transfer system is the most important factor.

In the next section we will use the mathematical model for the coupled hybrid system in the form of Eqs. (3.15) in order to explore its kinetic properties.

3.3.5. Kinetic response to variation of source efficiency in the one-group approximation

The objective of the present study is to demonstrate the difference between kinetic responses of the coupled hybrid system to variation of core reactivity $(\Delta\rho_{ext})$ and "source" reactivity (Δr), in particular, to inter-compare asymptotic reactor periods for these two cases. It is well known that an asymptotic period Θ of the reactor kinetic response to reactivity perturbation is described by the *characteristic equation*, called the *inhour equation* or Nordheim equation (Refs. Hetrick, 1971; Ash, 1979; Rozon, 1992). The inhour equation for the coupled hybrid systems has to be modified in order to take into account new features appearing in these systems. In Appendix we discuss in detail the characteristic equation which is derived from the model describing hybrid system kinetics in the form of Eqs. (3.15). There we demonstrate that the structure of obtained modified inhour equation differs qualitatively from this one of the "ordinary" inhour equation for a critical reactor.

The response to the reactivity insertion $(\Delta\rho_{ext})$ is well known in the literature (e.g. Refs. Hetrick, 1971; Ash, 1979; Rozon, 1992; Reuss, 2003) and may be directly used for the inter-comparison. Therefore, in the following analysis we assume no reactivity variation ($\Delta\rho_{ext} = 0$) and we focus our attention on the kinetic response of the coupled hybrid system to the source reactivity variation. Despite this simplification, the problem [in the form of Eqs. (3.15)] remains difficult to resolve and demands some cumbersome evaluations. For this reason we apply a so-called one-group approximation for delayed neutrons. In the case of coupled hybrid system this approximation (an usual procedure in kinetic analysis of critical reactors) is particularly justified since in most practical situations the artificial groups of delayed neutrons will prevail over natural groups (see discussion in Appendix).

In this one-group formulation, the problem may be easily solved. In Appendix an exact solution of one-group kinetic equation is obtained, applying the Laplace Transforms method. It is shown that a perturbation of the source reactivity $\tilde{\varepsilon}^{(r)} = \Delta r / \beta^{(\Sigma)}$ leads to the power transient

$$P(t) = P_0 \sum_{j=1}^{2} \Psi_j\left(u, \tilde{\varepsilon}^{(r)}\right) \exp\left(\omega_j^{(r)} t\right), \tag{3.17}$$

where the factors $\omega_j^{(r)}$ are the roots of the modified inhour equation (in one-group approximation), given by

$$\omega_{1,2}^{(r)} = \frac{\lambda^{(\Sigma)}}{2u}\left[-(1+u) \pm \sqrt{(1+u)^2 + 4\tilde{\varepsilon}^{(r)} u}\right] = \frac{\lambda^{(\Sigma)}}{2u}\left[-(1+u) \pm R\left(u, \tilde{\varepsilon}^{(r)}\right)\right]; \tag{3.18}$$

the coefficients $\Psi_{1,2}\left(u, \tilde{\varepsilon}^{(r)}\right)$ are assigned by the expressions:

$$\Psi_1\left(u, \tilde{\varepsilon}^{(r)}\right) = \frac{1}{2}\left(1 + \frac{1 + u + 2\tilde{\varepsilon}^{(r)}}{R\left(u, \tilde{\varepsilon}^{(r)}\right)}\right), \qquad \Psi_2\left(u, \tilde{\varepsilon}^{(r)}\right) = \frac{1}{2}\left(1 - \frac{1 + u + 2\tilde{\varepsilon}^{(r)}}{R\left(u, \tilde{\varepsilon}^{(r)}\right)}\right), \tag{3.19}$$

where the function $R\left(u, \tilde{\varepsilon}^{(r)}\right) = \sqrt{(1+u)^2 + 4\tilde{\varepsilon}^{(r)} u}$ is introduced for convenience and the superscript '(r)' over $\omega_{1,2}$ denotes the solution for source reactivity variation.

In Eqs. (3.18)-(3.19) $\beta^{(\Sigma)}$ and $\lambda^{(\Sigma)}$ are the fraction of delayed neutrons and the one-group decay constant for effective precursors of delayed neutrons, correspondingly. The parameter $u \equiv \Lambda\lambda^{(\Sigma)} / \beta^{(\Sigma)}$ is introduced for convenience. Our estimates show that in all cases, interesting for eventual applications, the factor u may be considered as small ($u \ll 1$). The prompt neutron life-time can vary by a few orders of magnitude: from $\Lambda \approx 10^{-7}$ s in fast-spectrum cores to $\Lambda \approx 10^{-3}$ s in thermal spectrum cores (Ref. Rozon 1992), i.e. one can suppose in further estimations that it does not exceed the value $\Lambda^{\max} \approx 10^{-3}$ s. The fraction of the supplementary group of delayed neutrons (subcriticality level) can vary from $\beta^{+} = 350$ pcm in the case of "beta-compensated" systems up to $\beta^{+} = 5000$ pcm or, eventually, greater. Hence, it would be meaningful to assume that $\beta^{(\Sigma)} \geq 700$ pcm. A value of λ^{+} is, as the subcriticality level β^{+}, an object of optimization and so it can be, to some extent, chosen arbitrarily. A reasonable assessment for λ^{+} and, therefore, for $\lambda^{(\Sigma)}$ would be $10^{-2} \div 10^{0}$ s^{-1}. Hence, we obtain the upper and the lower limits for the parameter u: $10^{-8} < u < 10^{-1}$, i.e. $u \ll 1$.

One may consider in accordance with above estimates, that conditions $u \ll 1$ and $2\tilde{\varepsilon}^{(r)} u \ll 1$ are fulfilled in any practical circumstance. In this situation the above solution for reactor power [Eqs. (21)-(22)] can be simplified. Thus, expanding $R\left(u,\tilde{\varepsilon}^{(r)}\right)$ in a Taylor series up to the 1st order in u:

$$R\left(u,\tilde{\varepsilon}^{(r)}\right) = 1 + \left(1 + 2\tilde{\varepsilon}^{(r)}\right)u + O\left(u^2\right) \tag{3.20}$$

one obtains for the roots of the characteristic equation

$$\begin{aligned}
\omega_1^{(r)} &= \tilde{\varepsilon}^{(r)}\lambda^{(\Sigma)} + O\left(u^2\right) \approx \Delta r\left(\frac{\lambda^{(\Sigma)}}{\beta^{(\Sigma)}}\right),\\
\omega_2^{(r)} &= -\lambda^{(\Sigma)}\left[u^{-1} + \left(1 + \tilde{\varepsilon}^{(r)}\right)\right] + O\left(u^2\right) \approx -\left[\frac{\beta^{(\Sigma)}}{\Lambda} + \lambda^{(\Sigma)} + \Delta r\left(\frac{\lambda^{(\Sigma)}}{\beta^{(\Sigma)}}\right)\right].
\end{aligned} \tag{3.21}$$

Similar simplification for the coefficients $\Psi_{1,2}\left(u,\tilde{\varepsilon}^{(r)}\right)$ yields:

$$\Psi_1\left(u,\tilde{\varepsilon}^{(r)}\right) \approx \left(1 + \tilde{\varepsilon}^{(r)}\right)\left(1 - 2\tilde{\varepsilon}^{(r)}u\right), \qquad \Psi_2\left(u,\tilde{\varepsilon}^{(r)}\right) \approx -\tilde{\varepsilon}^{(r)}\left(1 - 2u\left(1 + \tilde{\varepsilon}^{(r)}\right)\right). \tag{3.22}$$

Hence, we obtain the following approximate (at $u \to 0$) expression:

$$P(t) = P_0\left[\left(1 + \tilde{\varepsilon}^{(r)}\right)\exp\left(\tilde{\varepsilon}^{(r)}\lambda^{(\Sigma)}t\right) - \tilde{\varepsilon}^{(r)}\exp\left(-\frac{\beta^{(\Sigma)}}{\Lambda}t\right)\right]. \tag{3.23}$$

We complete these asymptotic analytical results with numerical illustrations calculated in accordance with Eqs. (3.18)-(3.19). Dependences of the dimensionless roots $\Omega_{1,2} \equiv \omega_{1,2}^{(r)} / \lambda^{(\Sigma)}$ and of the coefficients $\Psi_{1,2}$ on $\tilde{\varepsilon}^{(r)}$ for different values of parameter u are presented in Figure 28 and Figure 29. These calculations were performed for $-1 \le \tilde{\varepsilon}^{(r)} \le 10$ and cover the range of all possible values of the parameter u (see above estimates).

It follows from Figure 28 and from Eq. (3.21) that the first root $\omega_1^{(r)} > 0$ corresponds to a solution increasing with time (an asymptotic gradual growth), whereas the second root $\omega_2^{(r)} < 0$ describes a prompt jump of reactor power (a term, rapidly decreasing with time). As long as quantitative results for the dependences $\Omega_{1,2}\left(u,\tilde{\varepsilon}^{(r)}\right)$ are concerned, one may establish, that at $u \lesssim 10^{-3}$, all curves of Ω_1 collapse to only one, i.e. dependence on this parameter disappears when $u \to 0$. In this case (i.e. at $u \lesssim 10^{-3}$) the second root Ω_2 does not depend on $\tilde{\varepsilon}^{(r)}$ and may be approximately assumed to be $\Omega_2 = 1/u$. A particularity of the curves for $\Omega_1\left(u,\tilde{\varepsilon}^{(r)}\right)$, compared with similar calculation for the reactivity insertion in a critical reactor, is their monotony. Thus, for $\tilde{\varepsilon}^{(r)} = 1$, i.e. for the value which would lead to criticality on prompt neutrons if k_{eff} were modified, one obtains $\Omega_1 \approx 1$ (i.e. $\omega_1^{(r)} \approx \lambda^{(\Sigma)}$). Even if $\tilde{\varepsilon}^{(r)} = 10$, the parameter Ω_1 increases only by a factor of $6 \div 10$. Summarizing the above considerations, we conclude that if the source reactivity varies, *the asymptotic reactor period* $\Theta^{(r)}$ *will be of the order of the inverse effective decay constant* $\lambda^{(\Sigma)}$ in any circumstances.

Hence, if the total neutron multiplication factor of the system m_{eff} is modified by means of the source reactivity r, *there is no analogue to prompt criticality* (with consequent drastic decrease of the reactor period), typical for the conventional critical reactor when reactivity $\Delta\rho > \beta$ is inserted. In the next Section we discus this particularity in detail.

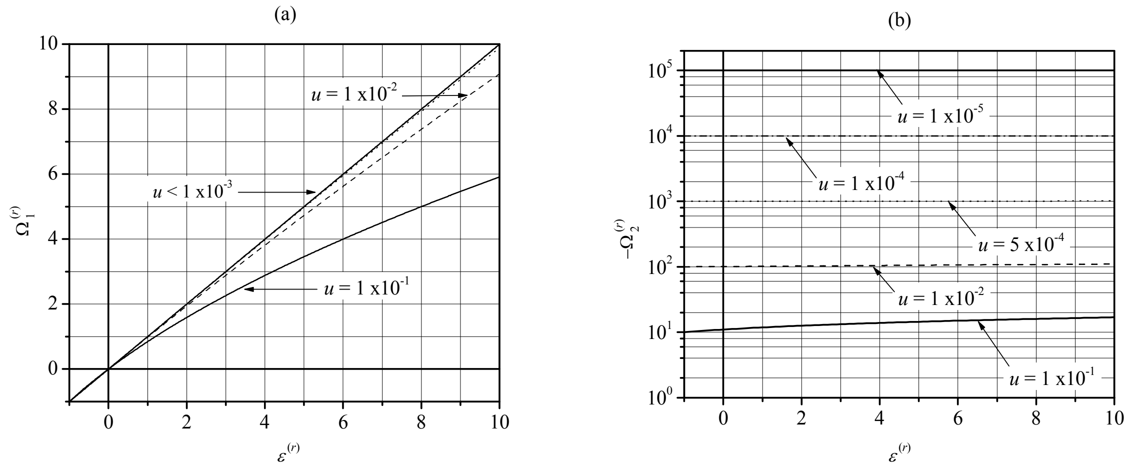

Figure 28. Dependence of the dimensionless roots Ω_1 (a) and Ω_2 (b) of the modified inhour equation on the source efficiency variation $\tilde{\varepsilon}^{(r)}$ at different values of parameter u.

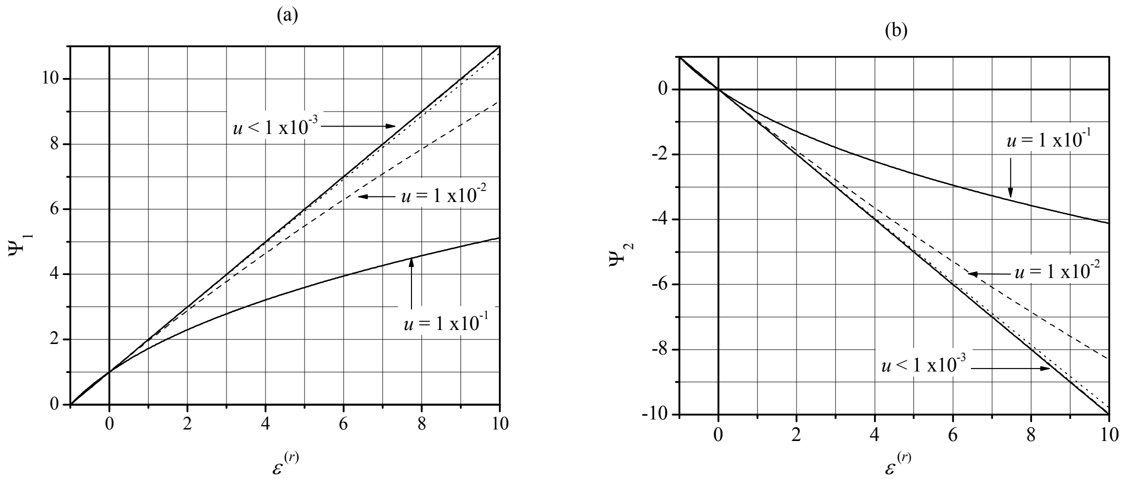

Figure 29. Dependence of the coefficients Ψ_1 (a) and Ψ_2 (b) on the source efficiency variation $\tilde{\varepsilon}^{(r)}$ for different values of the parameter u.

Now let us return to Eqs. (3.19), (3.23). Figure 29 demonstrates that coefficients $\Psi_{1,2}$ have nearly linear dependence on $\tilde{\varepsilon}^{(r)}$ when $u \to 0$. One can note that if $u \lesssim 10^{-3}$ they collapse, in accordance with Eq. (3.22), to asymptotes $\lim\limits_{u\to 0} \Psi_1\left(u, \tilde{\varepsilon}^{(r)}\right) = 1 + \tilde{\varepsilon}^{(r)}$ and $\lim\limits_{u\to 0} \Psi_2\left(u, \tilde{\varepsilon}^{(r)}\right) = -\tilde{\varepsilon}^{(r)}$, correspondingly. This permits us to estimate the magnitude of the prompt power jump (after the second term in Eq. (3.23) has disappeared):

$$\Delta P_{prompt}^{(r)} \approx P_0\left(\Psi_1 - 1\right) \approx \tilde{\varepsilon}^{(r)} P_0\,, \tag{3.24}$$

i.e. it is proportional to perturbation of the source efficiency.

3.3.6. Discussion. Comparison with the case of reactivity insertion

In contrast to conventional critical reactors, in coupled hybrid systems there are two ways to affect the total neutron multiplication factor: either by means of reactivity insertion $(\Delta\rho_{ext})$ or by means of modification of the source efficiency. Introduction of the source reactivity r gives us an easy-to-use tool making it possible to inter-compare these two modes, since this parameter has exactly the same meaning from the point of view of the steady neutron multiplication factor as the core reactivity $\rho = (k_{eff} - 1)/k_{eff}$. However, as was mentioned above, transients (as a response to an equivalent multiplication factor perturbation $\Delta r = \Delta\rho$) can also be different. To quantify this eventual difference let us compare the important kinetic characteristics: roots of the inhour equation in both cases. Particular attention is paid to the inter-comparison of the asymptotic periods $\Theta^{(r)}$ and $\Theta^{(\rho)}$ for these two cases. A solution in the case of r-variation was obtained in a previous Section [Eqs. (3.18)-(3.24)].

For case of step-wise reactivity insertion, there exists a vast literature, from which a solution could be taken (e.g. Refs. Hetrick, 1971; Ash, 1979; Rozon, 1992; Reuss, 2003). Thus, supposing $\Lambda \ll \beta^{(\Sigma)} / \lambda^{(\Sigma)}$ and having introduced parameter $\varepsilon^{(\rho)} \equiv \Delta\rho / \beta^{(\Sigma)}$ we can write the solution for this case in the following way:

$$P(t) = P_0 \left[\frac{1}{1-\varepsilon^{(\rho)}} \exp\left(\frac{\varepsilon^{(\rho)}}{1-\varepsilon^{(\rho)}} \lambda^{(\Sigma)} t \right) - \frac{\varepsilon^{(\rho)}}{1-\varepsilon^{(\rho)}} \exp\left(-\frac{\beta^{(\Sigma)}}{\Lambda} \left(1-\varepsilon^{(\rho)}\right) t \right) \right]. \tag{3.25}$$

Two ultimate limits are considered in our analysis:

1. Small reactivities: $\Delta\rho \ll \beta^{(\Sigma)}$.

In this case a reactivity insertion leads to the following roots of the characteristic equation:

$$\omega_1^{(\rho)} = \frac{\lambda^{(\Sigma)}\Delta\rho}{\beta^{(\Sigma)} - \Delta\rho} \approx \lambda^{(\Sigma)} \frac{\Delta\rho}{\beta^{(\Sigma)}} = \lambda^{(\Sigma)}\varepsilon^{(\rho)}, \quad \omega_2^{(\rho)} = \frac{\Delta\rho - \beta^{(\Sigma)}}{\Lambda} \approx -\frac{\beta^{(\Sigma)}}{\Lambda}, \tag{3.26}$$

where $\varepsilon^{(\rho)} \equiv \delta\rho / \beta^{(\Sigma)} \ll 1$. A comparison with the result for source variations [Eq. (3.21)] demonstrates that for small perturbations of neutron multiplication factor $\varepsilon = \varepsilon^{(\rho)} = \varepsilon^{(r)} \ll 1$ one finds: $\omega_1^{(\rho)} = \omega_1^{(r)}$ and $\omega_2^{(\rho)} = \omega_2^{(r)}$. Consequently, the asymptotic reactor period in both cases would be identical $\Theta^{(r)} = \Theta^{(\rho)} = \left(\varepsilon^{(\rho)}\lambda^{(\Sigma)}\right)^{-1}$. In addition, this period is rather large when compared with the effective generation time of precursors of delayed neutrons $\tau^{(\Sigma)} = \ln(2)/\lambda^{(\Sigma)}$.

We can also compare the prompt power jumps in these two cases. In the case of reactivity insertion we obtain from Eq. (3.25): $\Delta P_{prompt}^{(\rho)} = P_0\varepsilon^{(\rho)} / \left(1-\varepsilon^{(\rho)}\right)$. Hence, the ratio of prompt jumps in equivalent circumstances $\left(\varepsilon = \varepsilon^{(\rho)} = \varepsilon^{(r)}\right)$ is:

$$\Delta P_{prompt}^{(r)} / \Delta P_{prompt}^{(\rho)} = 1 - \varepsilon. \tag{3.27}$$

i.e. the case of the source reactivity variation is more advantageous when compared with core reactivity variation. Indeed, in the case of *positive* reactivity insertion $(\varepsilon > 0)$ it results in lesser prompt power jump.

2. Large reactivities: $\Delta\rho > 1.5\beta^{(\Sigma)}$ $\left(\varepsilon^{(\rho)} > 1.5\right)$

In this case the reactor becomes super critical on prompt neutrons and an analysis of the inhour equation yields in the well known result:

$$\omega_1^{(\rho)} = \frac{\Delta\rho - \beta^{(\Sigma)}}{\Lambda} \approx \frac{\beta^{(\Sigma)}}{\Lambda}\left(\varepsilon^{(\rho)} - 1\right) > 0\,, \quad \omega_2^{(\rho)} = \frac{\lambda^{(\Sigma)}\Delta\rho}{\beta^{(\Sigma)} - \Delta\rho} \approx -\frac{\lambda^{(\Sigma)}}{\beta^{(\Sigma)}}\left(\frac{\varepsilon^{(\rho)}}{\varepsilon^{(\rho)} - 1}\right) < 0\,. \tag{3.28}$$

As one may remark, the positive root $\omega_1^{(\rho)}$ of the characteristic equation, governing the rate of power growth, increases drastically when compared with Eq. (3.26), while Eqs. (3.21) remain valid for variation of the source reactivity. Let us assess the ratio $\omega_1^{(\rho)} / \omega_1^{(r)}$ for the equivalent perturbation of the neutron multiplication factor $\varepsilon = \varepsilon^{(\rho)} = \varepsilon^{(r)} > 1$:

$$\frac{\omega_1^{(\rho)}}{\omega_1^{(r)}} \approx \frac{\beta^{(\Sigma)}}{\Lambda\lambda^{(\Sigma)}} \frac{\left(\varepsilon^{(\rho)} - 1\right)}{\tilde{\varepsilon}^{(r)}} \approx \frac{\beta^{(\Sigma)}}{\Lambda\lambda^{(\Sigma)}} = \frac{1}{u} \gg 1\,. \tag{3.29}$$

The asymptotic period for the case of variation of the source efficiency becomes much greater than the asymptotic period in the case when the reactivity is inserted $\Theta^{(r)} / \Theta^{(\rho)} = u^{-1} \gg 1$. In addition, its value remains comparable with the effective generation time of precursors of delayed neutrons $\Theta^{(r)} \sim 1/\lambda^{(\Sigma)}$.

One can note that behavior of the coupled system is essentially different in the case of source reactivity variation. The explanation is rather simple: in this case the core remains subcritical and works in the mode of “energy amplifier”. If some perturbation of the external source effectiveness leads to a prompt change in the core power, development of the consequent transient will be limited by the rate of energy transfer from the core to the neutron production mechanism (e.g. proton accelerator).

The above result can be reformulated in the following way: the change in the neutron multiplication factor of the coupled hybrid system through the effectiveness of the external source does not affect essentially its asymptotic period. This result leads to an important conclusion concerning the operation of these systems: from the point of view of reactor kinetics, it is preferable to regulate the neutron multiplication factor by means of the source reactivity.

These results allow us to give another practical recommendation: *it is preferable to envisage reactivity reserves (e.g. devoted to compensate eventual reactivity swing) in the form of the source reactivity. In this case an erroneous insertion of these reserves will not lead to drastic decrease of the reactor period.*

3.3.7. Conclusions

In the present Section we have proposed an approach, which allows elucidating the role of the source efficiency in kinetics of the coupled system. The total neutron multiplication factor

m_{eff} and the source reactivity r are introduced by analogy with core neutron multiplication factor k_{eff} and core reactivity ρ. The source reactivity r becomes a valuable tool to compare variation of the source effectiveness with reactivity insertion.

With the support of a simple mathematical model, describing the coupling of the subcritical core and of the external neutron source, we demonstrate that the latter may be treated as a supplementary group of delayed neutrons. As was shown, this similarity between "natural" and "artificial" delayed neutrons is not absolute: some new opportunities arise and they have to be taken into account when the kinetics of the coupled hybrid system is considered.

The modified inhour equation, which takes into account the ability to modify source reactivity, is deduced and an analysis of its roots is performed. An asymptotic reactor period, in the case of source reactivity variation, is obtained as a solution of this modified inhour equation.

From the above analysis we conclude that the kinetic response of the coupled hybrid system to "source reactivity" variation is intrinsically different from that to core reactivity, in particular, when large (when compared with the effective fraction of delayed neutrons) reactivity is introduced. Namely, there is no equivalent of prompt criticality (accompanied by drastic decrease of the reactor period) for "source reactivity". These results allow us to give the following practical recommendation: it is preferable to have reactivity reserves in the form of the source reactivity from the point of view of reactor kinetics, since in this case an erroneous insertion of these reserves will not lead to drastic decrease of the reactor period.

3.4. On behavior of ADS when feedback effects are degraded

3.4.1. Subcriticality level, necessary to compensate feedback degradation

Let us imagine that a degradation of safety characteristics of a critical core leads to decrease (down to zero-level) of the negative Doppler-effect (or a similar rapid negative feedback effect) which usually plays the most important stabilization role in the standard safety-related situations. One could ask if a *reasonably chosen sub-criticality level in the case of ADS could compensate the degraded Doppler-effect so that the asymptotic power* $\overline{P}^{(ADS)}$ *after insertion of the positive reactivity* $\Delta\rho_{ext} > 0$ *would be equal to the asymptotic power* $\overline{P}^{(CRT)}$ *of the "non-degraded" critical core*.

The answer to the above question depends, in principle, on the particularities of a system. For our simplified analysis we may employ either the reactor model utilized in Chapter 2 [Eqs. (2.1), (2.11)-(2.12)] or the simplified model of the reactor as in previous Section [Eqs. (3.2) and (3.10)]. In both cases, the source term $S(t)$ is described by Eq. (2.5).

For a detailed analysis of the safety potential of any nuclear system, one is obliged to take into account all important feedbacks inherent to this system. However, for illustration of particularities of new feedbacks, it seems sufficient to use the model with only one "generalized" traditional feedback as it is presented below, characterized by the feedback coefficient $\alpha_{feedback} < 0$.

Let us compare the asymptotic power levels after reactivity $\Delta\rho_{ext}$ insertion in two systems: in the critical system with "normal" thermal feedback ($\alpha_{feedback}$) and in the ADS with degraded feedback ($\alpha'_{feedback}$), where $\left(\alpha_{feedback} < 0\right) \to \left(\alpha'_{feedback} \le 0\right)$, $\left|\alpha_{feedback}\right| > \left|\alpha'_{feedback}\right|$. Here a linear model of feedback effects is assumed, i.e. reactivity variations due to feedback in the standard and degraded core are given by:

$$\begin{Bmatrix} \Delta\rho_{feedback} \\ \Delta\rho'_{feedback} \end{Bmatrix} = \begin{Bmatrix} \alpha_{feedback} \\ \alpha'_{feedback} \end{Bmatrix} \Delta T \,. \tag{3.30}$$

The stationary solution for the reactor power can be obtained from Eqs. (2.1), (2.11)-(2.12) with zero time derivatives. In the case of ADS the following expression describes the asymptotic value of power variations:

$$\Delta P^{(ADS)} = \frac{P_0}{2\mathbb{A}'}\left[-\left(\mathbb{A}' + r_0 - \Delta\rho_{ext}\right) + \sqrt{\left(\mathbb{A}' + r_0 - \Delta\rho_{ext}\right)^2 + 4\Delta\rho_{ext}\mathbb{A}'}\right], \tag{3.31}$$

where the following notations are introduced:

$$\mathbb{A}' \equiv -\alpha'_{feedback} P_0 / H_{tot}; \qquad H_{tot}^{-1} \equiv H^{-1} + \left(\varrho c_p^{(s)} D_0\right)^{-1}. \tag{3.32}$$

In this context the physical meaning of above parameters is evident: the parameter H_{tot} describes the effective thermal resistance of the heat transfer system and the parameter $\mathbb{A}$ is the normalized power feedback coefficient taken with “minus” sign in order to obtain a positive-value parameter (we remind, that $\alpha'_{feedback}$ is assumed to be small but negative), given by:

$$\mathbb{A} = -P_0\left(\frac{d\rho_{feedback}\left(P_0\right)}{dP_0}\right). \tag{3.33}$$

In the case of a *critical* system with a “non-degraded” feedback ($\mathbb{A} = -\alpha_{feedback} P_0 / H_{tot}$) the asymptotic response of the system to a reactivity insertion is given by

$$\Delta P^{(CRT)} = \frac{P_0}{\mathbb{A}} \Delta\rho_{ext} \,. \tag{3.34}$$

Comparing Eq. (3.31) with Eq. (3.34) one can evaluate the subcriticality level, necessary to reach the same asymptotic power level in both cases:

$$r_0 = \left(1 - \frac{\mathbb{A}'}{\mathbb{A}}\right)\left(\mathbb{A} + \Delta\rho_{ext}\right), \tag{3.35}$$

which, in the case of a total loss of feedback effects ($\alpha'_{feedback} = \mathbb{A}' = 0$), becomes

$$r_0 = \mathbb{A} + \Delta\rho_{ext} \tag{3.36}$$

Therefore, the subcriticality level, required to compensate the degraded feedback effect is defined by Eq. (3.35). In the limit case of complete absence of the in-core feedback it consists of two parts [Eq. (3.36)]: the first one compensates the positive reactivity, which would appear due to the core cooling from the temperature T to T_k, and the second compensates the inserted reactivity $\Delta\rho_{ext}$.

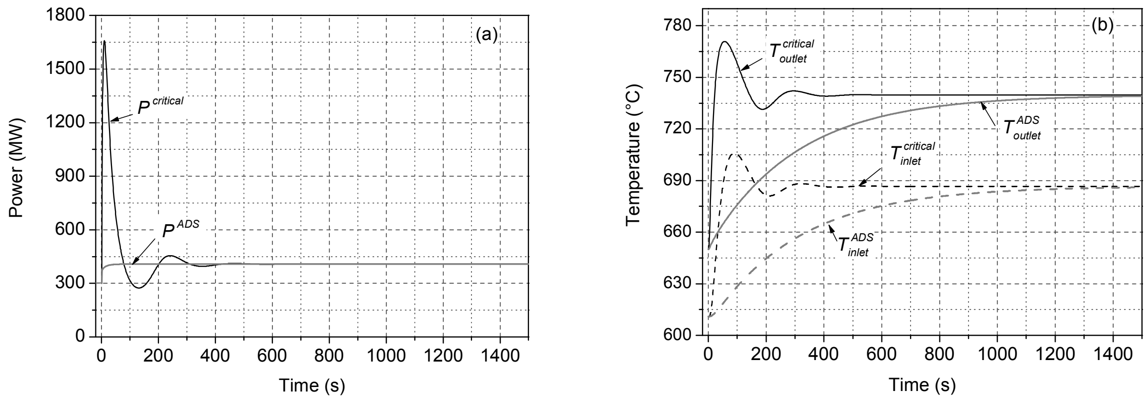

Figure 30. Unprotected TOP transients in the critical reactor with the standard feedback effect and in the corresponding subcritical system (ADS) with fully degraded feedback effect.

The following example illustrates these evaluations. Figure 30 presents transients of core power (a) and of core temperature (b) in the *critical* molten salt thermal reactor (analogous to AMSTER concept) with the "standard" feedback coefficient $\alpha_{feedback} = -1.95\,\text{pcm/°C}$ ($\mathbb{A} = 487.5$ pcm, $P_0 / H_{tot} = 250\,°\text{C}$), and the inserted reactivity $\Delta\rho_{ext} = 175$ pcm. This result can be compared with the corresponding "fully degraded" *subcritical* system ($\alpha'_{feedback} = \mathbb{A}' = 0$, $r_0 = 665\,\text{pcm}$), where subcriticality level was chosen in accordance with Eq. (3.36). Transients for the critical system as well as for ADS are calculated in one-group approximation for delayed neutrons ($\beta = 350$ pcm, $\lambda = 0.08$ s^{-1}).

In brief, both critical system with the standard feedback effect and subcritical system without feedback have similar asymptotic values with respect to the core power and temperature. However, transients are quite different: there are considerable power and temperature oscillations in critical reactors, while transient in subcritical systems is rather monotonic. This monotony of power and temperature curves during both reactivity and thermo-hydraulic transients has already been noted in Chapter 2. The observed monotony of transients in the case of subcritical system (ADS) permits to restrict our analysis by the comparison of only asymptotic values in both cases.

Therefore, Eq. (3.35) shows a *way to compensate the feedback degradation in terms of equal asymptotic power levels in a "non-degraded" critical- and the corresponding "degraded" subcritical system* after insertion of the maximal available positive reactivity.

3.4.2. On the grace time of ADS with a positive feedback

In Chapter 2 we have seen an important increase of the grace time in ADS when compared with corresponding critical system for the systems with the positive feedback ($\alpha_{feedback} > 0$). We found it interesting to characterize the dependence of the grace time as a function of the subcriticality level. The goal of this study is not to find the exact solution of the problem (this seems to require some cumbersome evaluation), but to estimate the lower limit of the grace time in the most severe conditions.

Let us consider the event, which consists in simultaneous insertion of all reserves of the reactivity ($\Delta\rho_{ext}$) or/and of the proton current (Δr_{ext}) under conditions where heat evacuation from core is completely halted and the external neutron source continues working (this situation is analogous to the UTOP/UTOC + ULOF/ULOHS complex accident). In this case core heating by the external neutron source is aggravated by continuous decrease of the subcriticality level caused by an unfavourable positive feedback. This decrease of the subcriticality level accelerates transient to core disruption, as it leads to a rise of core energy multiplication factor according to the well-known relationship $P \propto -S / \rho$.

It is evident, that neglecting delayed neutrons, we obtain the most pessimistic estimation for the grace time. In these conditions the core transient may be described by the following coupled system of equations (we utilize the same notation as in the previous Sections):

$$\begin{cases} \dfrac{d}{dt} P(t) = \left(\Delta\rho_{ext} + \Delta\rho_{feedback}(T) - r_0\right) \dfrac{P(t)}{\Lambda} + \left(r_0 + \Delta r_{ext}\right) \dfrac{P_0}{\Lambda}; \\ C_p \dfrac{d}{dt} T(t) = P(t). \end{cases} \tag{3.37}$$

This system of equation may be easily resolved in so-called quasi-static approximation, i.e. assuming, that core power is in a quasi-static equilibrium with the external neutron source. The physical basement for this assumption is that the characteristic time of the temperature change is much greater than the prompt neutron generation time[f]. In these conditions we may take approximately, that $dP(t)/dt = 0$ and therefore

$$P(t) = -\frac{\left(r_0 + \Delta r_{ext}\right) P_0}{\left(\Delta\rho_{ext} + \Delta\rho_{feedback}(T) - r_0\right)}. \tag{3.38}$$

Substituting Eq. (3.38) into Eq. (3.37) and after some rearrangements one obtains:

$$dt = \frac{C_p}{P_0} \frac{\left(r_0 - \Delta\rho_{ext} - \Delta\rho_{feedback}(T)\right)}{\left(r_0 + \Delta r_{ext}\right)} dT. \tag{3.39}$$

Integration of Eq. (3.39) yields:

$$\Delta t^{\dagger} = \int_{t_0}^{t^{\dagger}} dt = \frac{C_p}{\left(r_0 + \Delta r_{ext}\right) P_0} \left\{ \left(r_0 - \Delta\rho_{ext}\right)\left(T^{\dagger} - T_0\right) - \int_{T_0}^{T^{\dagger}} \Delta\rho_{feedback}(T)\, dT \right\}. \tag{3.40}$$

In principle, Eq. (3.40) is the solution of the considered problem. For further estimations one needs the explicit dependence of the core feedback on the core temperature as well as the core disruption temperature. Let us consider for simplicity the linearized feedback:

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[f] As stated by Reuss (2003) this approximation is valid if core is subcritical on prompt neutrons.

$$\Delta\rho_{feedback}\left(T\right)=\alpha_{feedback}\left(T-T_0\right). \tag{3.41}$$

Moreover, let us assume, that the prompt criticality corresponds to the disruption temperature (as it was considered in Chapter 2), i.e.,

$$\Delta\rho_{feedback}\left(T^{\dagger}\right)+\Delta\rho_{ext}-r_0=0\,. \tag{3.42}$$

In these conditions Eq. (3.40) reduces to

$$\Delta t^{\dagger}=\frac{C_p}{2\alpha_{feedback}P_0}\frac{\left(r_0-\Delta\rho_{ext}\right)^2}{\left(r_0+\Delta r_{ext}\right)}. \tag{3.43}$$

From Eq. (3.43) it follows, that the grace time for this ultimate event increases approximately as a *square of the margin to the prompt criticality* $\Delta\rho_m=\left(r_0-\Delta\rho_{ext}\right)$. Applying notations, introduced in Sections 3.3 and 3.4.1, we may rewrite Eq. (3.43) in the following way:

$$\Delta t^{\dagger}=\frac{1}{2\lambda^{+}}\frac{r_0}{\left|\mathbb{A}'\right|}\frac{\left(1-\Delta\rho_{ext}/r_0\right)^2}{\left(1+\Delta r_{ext}/r_0\right)}. \tag{3.44}$$

From this expression becomes obvious that in the case of a large subcriticality level, i.e., when $r_0\gg\left(\Delta\rho_{ext},\Delta r_{ext}\right)$, the grace time will be by a factor $0.5r_0/\left|\mathbb{A}'\right|$ greater than the characteristic heat transfer time in the system given by $\left(\lambda^{+}\right)^{-1}$.

All the methods, considered in this Chapter allow “patching up” some particular safety-related problems. Unfortunately, utilized one by one, they do not guarantee the intrinsic safety. In the next Chapter we will discuss a promising comprehensive approach, which permits to “repair” inherently both the degraded feedback and the fraction of delayed neutrons.

4. A hybrid system with intrinsic improvement of the delayed neutron fraction and of the power feedback: DENNY concept

Résumé *- Une des options de systèmes hybrides est le système couplé, où un couplage intrinsèque entre l'intensité de la source externe et la puissance du cœur est réalisé. Ce système peut être caractérisé comme un système fonctionnant en mode critique où la source externe de neutrons joue le rôle d'un groupe supplémentaire de neutrons retardés. Des études antérieures de la cinétique et de la dynamique des systèmes hybrides couplés ont démontré les avantages de tels systèmes en comparaison avec leurs homologues critiques en matière de sûreté. Pourtant, les systèmes couplés héritent de certains problèmes intrinsèques aux systèmes fonctionnant en mode critique. Notamment une dégradation des effets de contre-réaction (par exemple l'effet Doppler) prive le système de ses propriétés stabilisantes.*

Dans l'objectif de remédier à ce problème une nouvelle façon de réaliser un système couplé est proposée (concept DENNY). Dans le cadre de cette approche il est proposé, contrairement aux travaux antérieurs, d'utiliser l'énergie des protons incidents (à la place de l'intensité) en tant que paramètre de couplage, c'est à dire de changer le mode de couplage. Dans ce cas on peut profiter de certaines particularités de production de neutrons dans une cible de spallation en fonction de l'énergie des protons incidents (Y_n-effect) et de ce fait accentuer les propriétés stabilisatrices de la puissance du cœur lors des accidents de réactivité non protégés.

Une caractérisation générale du fonctionnement de DENNY est donnée. Lors de cette étude il a été démontré que l'influence bénéfique de l'effet Y_n-effect sur la dynamique des systèmes hybrides couplés peut être considérable (surtout en l'absence d'effet Doppler).

4.1. Introduction

In this Chapter we discus a new approach for the realization of ACS, where a significant improvement of the feedback effect is expected due to the modification of the accelerator-core coupling mode and due to the particularities of the neutron production in a spallation target. The goal of this new concept is to combine the intrinsic safety features of ADS and ACS with respect to their behavior during unprotected accidents in a single concept. This approach was initially proposed in Ref. (Bokov *et al.*, 2004b) and also addressed and refined in Refs. (Bokov *et al.*, 2004a, Bokov *et al.*, 2005).

4.2. Generalities of hybrid systems

As demonstrated in previous Chapters, the core subcriticality will improve the safety, in particularly, when feedback effects, the delayed neutron fraction or other safety related parameters are degraded, for example, due to presence of long-lived actinides subjected to transmutation. As already discussed above, there are at least two different ways of functioning of a subcritical core in combination with an external neutron source. In brief, this source can be independent on energy production in the cores (ADS), or it can depend on the energy production in the cores and in this way becomes "coupled" or "coordinated" by the core power level (ACS). Each combination opens some new opportunities related to the safety improvement.

In terms of safety ADS is *inherently* more favorable (compared with the similar critical reactor) regarding reactivity accidents, where the core subcriticality mitigates the consequences of the reactivity insertion (see for example Refs. Takahashi, 1995; Wade, 2000; Schikorr, 2001; Slessarev and Bokov, 2004). On the other hand, a system functioning in a critical regime (including ACS) is *intrinsically* safer in the case of thermo-hydraulic type of transients (under conditions that those in-core feedbacks are favorable).

In the case of ACS a "source of artificially delayed neutrons" allows increasing the fraction of delayed neutrons and therefore, the effective neutron lifetime artificially. If compared to the conventional critical reactors, this particular property of ACS can improve the reactor dynamics significantly. Moreover, ACS operates in a critical mode and, therefore, in contrast to ADS, takes advantages of favorable temperature feedbacks, which might exist in these systems. As we discussed in Chapters 2 and 3, the drawback of this system is that in the case of the unfavorable in-core feedbacks, the deterministic safety can not be achieved.

Therefore, it would be rather attractive to combine these inherent advantages of both ADS and ACS in a single installation. In other words, one needs to realize a system, for which during the *unprotected transients*

(i) the intensity of an external neutron source decreases with the decrease of core power,

(ii) the intensity of an external neutron source remains stable or even decreases with the increase of core power, and, finally

(iii) conditions i) and ii) have to be *intrinsic.*

One of possible solutions[g] to merge the above advantages is presented in the following Section. It is based on the physical processes taking place in the neutron production target, what makes our approach inherent.

4.3. Accelerator-core coupling modes in the case of ACS

Traditionally, it is assumed that in hybrid systems the current of a proton accelerator is the coupling parameter, which one can vary to change the neutron source intensity. In the case of ACS, at least two modes of coupling between external neutron source and core could be envisaged:

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[g] Other approaches to create inherent shut-down mechanisms are discussed, for example, by Eriksson and Cahalan (2001).

1. When it is supposed to modify the intensity of an external neutron source S by varying the proton beam current I_p at a fixed nominal value of the proton energy, namely

$$I_p = I_{p,0} \frac{P^{out}}{P_0^{out}}. \tag{4.1}$$

Here P^{out} is the output power of the installation (we assume for the simplicity, that $\eta_a = const$, $\eta_e = const$ and, therefore, the parameter P^{out} denotes either electric output power or thermal output power), and a subscript "0" denotes nominal values of the corresponding variables. Hereafter this method of the "accelerator-core" coupling is designated as "*I*-mode" coupling.

2. When any *change of the output power leads to a proportional change of the proton energy* ϵ_p at a fixed nominal value of the proton current, namely

$$\epsilon_p = \epsilon_{p,0} \frac{P^{out}}{P_0^{out}}. \tag{4.2}$$

This coupling method is denoted as "*E*-mode" coupling.

In this work we propose *to utilize the proton energy as coupling parameter* (*E*-mode ACS). The difference between the *E*- and *I*-modes, we would like to make use of, is based on a *non-linear behavior of the neutron yield* Y_n *with respect to the proton energy* ϵ_p *variation* (hereafter "Y_n*-effect*").

Indeed, as it is shown in a number of studies (see Refs.: Andriamonje *et al.*, 1995; Pankratov *et al.*, 1996; Letourneau *et al.*, 2000; Leray, 2000 and Refs. therein), when the energy of incident protons becomes higher than ~1 GeV, the neutron yield normalized per incident proton energy

$$y_n\left(\epsilon_p\right) = \frac{Y_n\left(\epsilon_p\right)}{\epsilon_p} \tag{4.3}$$

becomes nearly constant and even slightly decreases with proton energy. There are two major reasons for this decrease of neutron production efficiency with increase of the proton energy:
(i) opening of new reaction channels other than neutron production,
(ii) escape of high energy particles from the spallation target with finite geometry.

This dependence of neutron production is illustrated qualitatively in Figure 31. The neutron yield Y_n, after proton energy passed the reaction threshold [zone (1')], grows rather rapidly with energy [zone (1")]; above a certain value of ϵ_p, this dependence has a moderated quasi-linear behavior [zone (2)]. So, there is a value of proton energy $\epsilon_p^{optimum}$, which is optimal with respect to the neutron economy, i.e. the neutron yield (y_n) per one incident proton and per consumed energy reaches its maximal value. Therefore, it is generally considered that there is no sense to increase the energy of protons further than $\epsilon_p^{optimum}$ since the production of neutrons in the spallation target becomes less efficient if compared with the equivalent increase of the proton current (i.e., the accelerator power being constant).

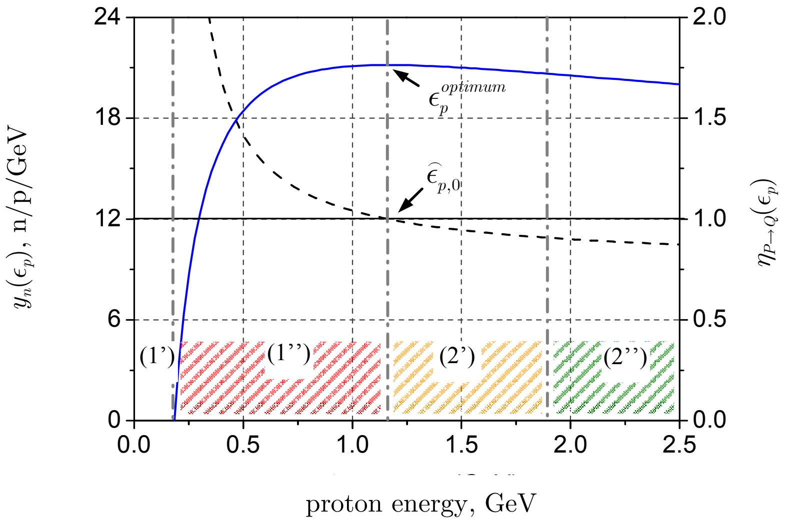

Figure 31. Dependence of the spallation neutron yield $y_n(\epsilon_p)$ (solid line) and that of the source effectiveness $\eta_{P\to Q}(\epsilon_p)$ (dashed line – to be defined by Eq. 4.13 below). Also see the text for details.

Quantitatively, the Y_n-effect as a function of proton energy can be described by an empirical formula proposed by Pankratov *et al.* (1996) in the units of neutron yield per one incident proton interacting with a thick heavy metal target:

$$Y_n\left(\epsilon_p\right) = -\gamma + \mu\left(\epsilon_p\right)^{\sigma}, \tag{4.4}$$

where the parameters $\gamma, \mu \geq 0,\ \ 0 \leq \sigma \leq 1$ can be fitted to the experimental data depending on the target geometry and materials. This particular dependence of the neutron yield on target geometry and material should not be neglected. Furthermore, one should make use of these particular situations. Indeed, our preliminary estimates have shown that some optimization on the geometry of the spallation target might strengthen further the Y_n-effect. More quantitative calculations in this context are needed.

4.4. Principle of the operation – DENNY concept

In this Section we propose to utilize this particularity of neutron production to form a quasi-linear dependence (the Y_n-effect) between energy production in the core and external neutron production in the spallation target aiming to get an auto-regulating behavior of the ensemble "accelerator – subcritical core". A proposed system (*E*-mode coupled ACS) would have the kinetics of a critical system with artificial group of delayed neutrons as in the case of the "standard" ACS. In addition, its external neutron production would contain the supplementary feedback, tending to stabilize the installation power in its nominal state.

To elucidate this statement, let us remind that the ACS may be considered as a *critical* system with two types of neutrons contributing to the *global* neutron balance: "core neutrons" and "source neutrons". Despite the fact that this separation of neutrons is relatively artificial, it reflects their origin and, therefore, corresponding neutron production feedbacks existing in each case. In the same context, the Y_n-effect can be compared to the Doppler feedback effect but for the external source neutrons. Similarly as the Doppler feedback effect, the Y_n-effect is intrinsic. It would be quite advantageous for the system safety to have this supplementary feedback acting on the entire neutron balance if the "standard" core feedbacks are degraded and can not play their stabilizing role indispensable for the inherent system safety.

The advantage of the above realization of a coupled hybrid system can be illustrated by the "neutron production versus core power" (Figure 32a) as well as by the "core power versus accelerator power" (Figure 32b) diagrams for the case of unprotected accidents. We note the equivalence between Figure 32a for the *E*-mode ACS and Figure 31 for the neutron yield Y_n-dependence, which is possible to make use of only in the case of *E*-coupling. According to the new concept (proposed ACS with the *E*-mode coupling), the power (and temperature) excursion would be less important than in the "standard" *I*-mode ACS, what is clearly seen from Figure 32b. The system with an accelerator coupled to the core in the *E*-mode will be named also "DENNY" (after Delayed Enhanced Neutronics with Non-linear neutron Yield) in the present work. Below we propose the principle of DENNY functioning.

Let us consider the *E*-mode ACS with a pre-defined subcriticality level $r_0 = \left(1 - k_{eff,0}\right) / k_{eff,0}$ and a fraction $f < 1$ of the produced core power, which is used to drive an external neutron source. External neutrons are created in the spallation target by incident protons accelerated up to the energy ϵ_p. It is preferable to choose the nominal proton energy $\epsilon_{p,0} > \epsilon_p^{optimum}$ in order to avoid an eventual instability of the DENNY power with respect to negative reactivity insertions (power decrease). Hence the proton energy has to be chosen as follows: $\epsilon_{p,0} = \epsilon^{optimum} + \Delta\epsilon_m$ (region (2") in Figure 31). Here the margin $\Delta\epsilon_m$ [zone (2') in Figure 31] makes the system more stable with respect to negative reactivity insertions. This is valid if during the system operation the proton energy remains above the optimal energy, i.e. the condition $\epsilon_p \geq \epsilon^{optimum}$ is fulfilled.

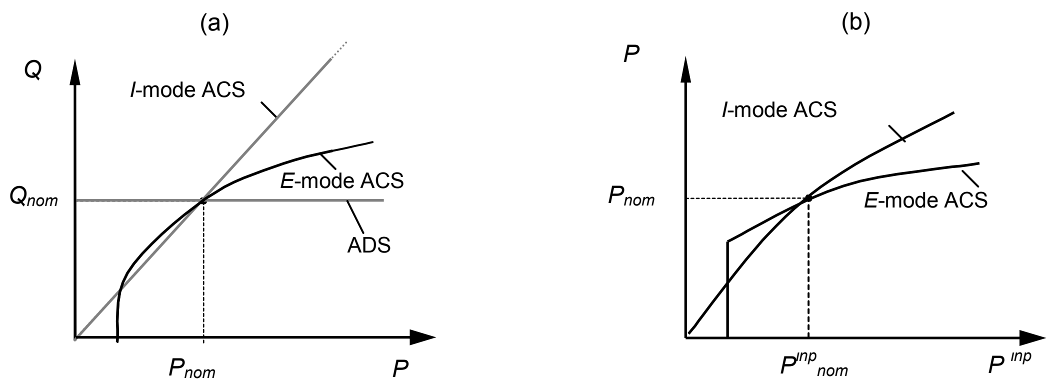

Figure 32. Diagrams of the intrinsic dependences of: (a) the external neutron production Q on the core power P, and (b) the equilibrium core power P on the accelerator power P^{inp} for different concepts of a coupled hybrid system.

The nominal values of proton current ($I_{p,0}$) as well as of the fraction of accelerator feed power (f_0) are chosen in the way to sustain the power level P_0 in a nominal state (Salvatores *et al.*, 1996):

$$I_{p,0} = \frac{r_0 \nu P_0}{\epsilon_f \varphi^* Y_n\left(\epsilon_{p,0}\right)}, \quad f_0 = \frac{r_0 \nu}{\epsilon_f \varphi^* y(\epsilon_{p,0}) \eta_a \eta_e}. \tag{4.5}$$

The value of the proton current is *fixed* over all period of the *E*-mode ACS functioning. On the contrary, the fraction f may be adjusted to compensate eventual reactivity swing (e.g., due to the burn-up). In other words, for the proton energy we write:

$$\epsilon_p = \epsilon_{p,0} \frac{fP}{f_0 P_0}. \tag{4.6}$$

Above we explained schematically the principle of the DENNY functioning, where some details are omitted with a view to simplify our description (for example, we suppose that accelerator efficiency is identical for all proton energies, importance of source neutrons does not depend on proton energy, etc.). However, in order to give some quantitative illustration of the main principle, a simplified model of the system operation with the *E*-coupling is presented below.

4.5. Results and discussion

Let us study the response of the *E*-mode ACS on an accidental reactivity insertion in order to describe qualitatively the influence of the Y_n-effect on its kinetics. A new equilibrium power level $\overline{P}$ of the system after insertion of the reactivity $\Delta\rho_{ext}$ can be found from generalized reactivity-power balance equation (Gandini *et al.*, 1999; Gandini *et al.*, 2000) following from the stationary kinetic equation:

$$\Delta\rho_{ext} - r_0 + \Delta\rho_{feedback}(\overline{P}) + r_0 Q(\overline{P}) / \overline{P} = 0, \tag{4.7}$$

where the term

$$Q(P) = P_0 \frac{Y_n\left(\epsilon_p(P)\right)}{Y_n\left(\epsilon_{p,0}\right)} \tag{4.8}$$

describes the external neutron source, while the proton energy ϵ_p was already defined in Eq. (4.6). In this context the last term in Eq. (4.7) may be considered as a "source (feedback) reactivity", i.e.

$$r = r_0 Q(\overline{P}) / \overline{P}. \tag{4.9}$$

Eq. (4.7) together with the feedback model and neutron yield dependence $[Y_n\left(\epsilon_p\right)]$ describes equilibrium states of the *E*-mode ACS after reactivity-insertion transients. In this case, a new power level $\overline{P}$ after the reactivity transients will be determined not only by the core feedback but also by the ability of the external source to produce sufficient neutrons to sustain this power.

Eq. (4.7) is non-linear with respect to the variable $\bar{P}$ and can be solved numerically. However, linearization of Eq. (4.7) allows us to characterize the Y_n-effect analytically with respect to the infinitesimal power fluctuation. Moreover, it permits to compare the in-core feedbacks with the Y_n-effect.

Introducing normalized power reactivity coefficients

$$\mathbb{A} \equiv -P_0 \left(\frac{d\rho_{feedback}(P_0)}{dP_0} \right) \quad \text{and} \quad \mathbb{B} \equiv -P_0 \left(\frac{dr(P_0)}{dP_0} \right) \tag{4.10}$$

we rewrite Eq. (4.7) in the linearized form:

$$\Delta\rho_{ext} - (\mathbb{A} + \mathbb{B}) \frac{\Delta \bar{P}}{P_0} = 0 \,. \tag{4.11}$$

Taking into account the initial condition $Q(P_0) = P_0$ and after some modifications, one obtains the following expression for the parameter $\mathbb{B}$:

$$\mathbb{B} = -r_0 P_0 \left[\frac{d}{dP_0} \left(\frac{Q(P_0)}{P_0} \right) \right] = r_0 \left(1 - \eta_{P \to Q}(P_0) \right) \tag{4.12}$$

with the function

$$\eta_{P \to Q}\left(P_0\right) \equiv \frac{dQ\left(P_0\right)}{dP_0} \tag{4.13}$$

being a measure of the local source effectiveness, i.e. a source response due to an infinitesimal power change in a nominal state (i.e. $P = P_0$). With respect to the global neutron balance in the *E*-mode ACS, Eq. (4.12) demonstrates that the parameter $\mathbb{B}$ may be considered as a coefficient, which is a measure of the supplementary neutron production feedback, arising in the system due to the Y_n-effect. As it follows from Eq. (4.12), the coefficient $\mathbb{B}$ is proportional to the nominal subcriticality level r_0 and depends on the $\eta_{P \to Q}(P_0)$ functional behavior.

A non-linear neutron production influences the equilibrium power level, and its effectiveness $\eta_{P \to Q}(P_0)$ will depend on the choice of the nominal proton energy $\epsilon_{p,0}$. We may incorporate the choice of the nominal energy to the source efficiency by introducing the function

$$\mathbb{H}_{P \to Q}\left(\epsilon_p, \epsilon_{p,0}\right) \equiv \frac{dQ}{dP} \,, \tag{4.14}$$

which describes the global effectiveness of the external neutron production at any proton energy ϵ_p with the nominal energy being $\epsilon_{p,0}$ (see Figure 31).

Using Eqs. (4.8) and after some simplifications, it becomes

$$\mathbb{H}_{P\to Q}\left(\epsilon_p,\epsilon_{p,0}\right)=\left(\frac{\epsilon_{p0}}{Y_n\left(\epsilon_{p0}\right)}\right)\left(\frac{dY_n\left(\epsilon_p\right)}{d\epsilon_p}\right). \tag{4.15}$$

The Y_n-effect increases the asymptotical power if $\epsilon_{p,0}<\epsilon_p^{optimum}$ [region (1) in Figure 31] and, contrary, it reduces the power growth if $\epsilon_{p,0}\geq\epsilon_p^{optimum}$ [region (2) in Figure 31]. In fact, we can see from Eq. (4.12) that, if the condition $\mathbb{H}_{P\to Q}\left(\epsilon_p,\epsilon_{p,0}\right)=\left(\delta Q/\delta P\right)<1$ is fulfilled, the external neutron source is not able to support the increasing power, what will limit the consequent power growth $\Delta\overline{P}=\overline{P}-P_0$.

Let us suppose for simplicity that $f=f_0$. In this case the function $\eta_{P\to Q}\left(\epsilon_p\right)$ can be expressed as follows:

$$\eta_{P\to Q}\left(\epsilon_{p,0}\right)=\left(\frac{\epsilon_{p,0}}{Y_n\left(\epsilon_{p,0}\right)}\right)\left(\frac{dY_n\left(\epsilon_{p,0}\right)}{d\epsilon_{p,0}}\right). \tag{4.16}$$

As it follows from Eq. (4.4) and Eq. (4.16) at

$$\widehat{\epsilon}_{p,0}=\left(\frac{\gamma}{(1-\sigma)\mu}\right)^{\frac{1}{\sigma}}, \tag{4.17}$$

the function $\eta_{P\to Q}(\widehat{\epsilon}_{p,0})=1$. This energy point defines the limit between the “destabilizing” area of the Y_n-effect (amplification of $\Delta\overline{P}$, similar to positive feedback) at $\epsilon_{p,0}<\widehat{\epsilon}_{p,0}$ and the “stabilizing” domain of the Y_n-effect (suppression of $\Delta\overline{P}$, similar to negative feedbacks) at $\epsilon_{p,0}\geq\widehat{\epsilon}_{p,0}$ (see Figure 31). It is important to note that in the present case $\widehat{\epsilon}_{p,0}$ is equal to the optimum energy $\epsilon_p^{optimum}$ with respect to the neutron economy: $\widehat{\epsilon}_{p,0}=\epsilon_p^{optimum}$.

In order to quantify the benefits of the proposed DENNY concept we perform a comparative analysis in the case of ACS with the *I*-mode coupling and *E*-mode coupling, resulting in a linear $Q(P)$ dependence and non-linear $Q(P)$ dependence correspondingly (see Figure 32a). The effectiveness of the Y_n-effect for the safety improvement can be described by the *transient suppression parameter* $\mathfrak{D}$. This parameter is defined as a ratio of *asymptotic* power values of the *E*-coupled and *I*-coupled systems after a certain reactivity insertion transient, namely

$$\mathfrak{D}=\frac{\overline{P}^{(E-mode)}}{\overline{P}^{(I-mode)}}. \tag{4.18}$$

If the condition $\mathfrak{D}<1$ is fulfilled, this signifies that the Y_n-effect *stabilizes* the system. Values of the parameter $\mathfrak{D}$ at different r_0 and $\Delta\rho_{ext}$ for the linear model of in-core feedback are presented in Figure 33a. For a quantitative comparison we had to define the parameters in Eq. (4.4), which we took from Ref. (Pankratov et al., 1996), namely $\gamma=8.2$, $\mu=29.3$ and $\sigma=0.75$. According to the discussion in the previous Section we choose the nominal energy value $\epsilon_{p,0}=$ 1.6 GeV, i.e. greater than $\widehat{\epsilon}_{p,0}=$ 1.16 GeV for our comparative analysis, from which the following conclusions are drawn:

– stabilizing role of the Y_n-effect increases when both r_0 and $\Delta\rho_{ext}$ increase. This effect can be quite significant (up to 27 % at $r_0 = 15\beta$) even in the case of a "good" in-core feedback ($\mathbb{A} = 488\,\text{pcm}$). A further growth of $\Delta\rho_{ext}$ leads to the saturation of such a tendency;

– the augmentation of the nominal proton energy $\epsilon_{p,0}$ enhances the stabilizing impact of Y_n-effect due to the reduction of the source effectiveness $\eta_{P\to Q}(\epsilon_{p,0})$.

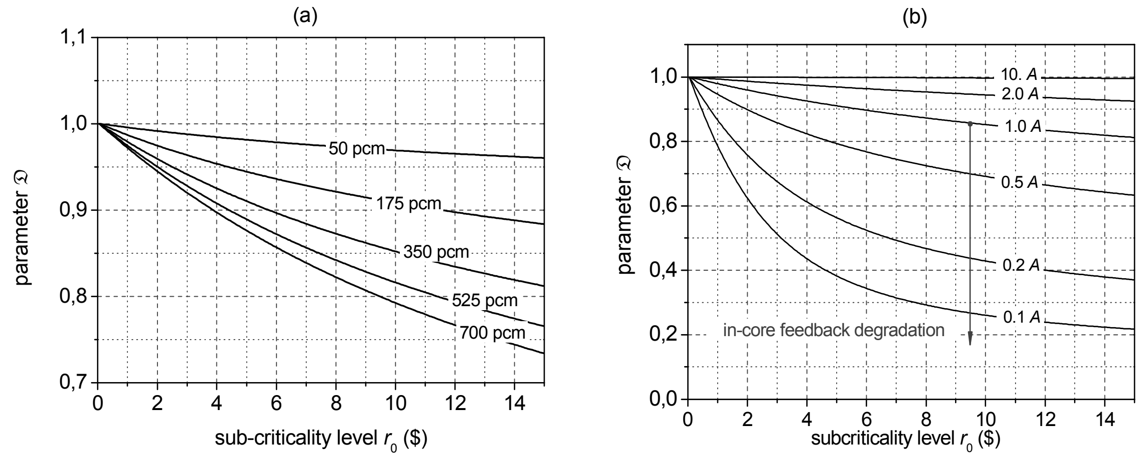

Figure 33. Transient suppression parameter $\mathfrak{D}$ as a function of the subcriticality level: (a) at different values of the inserted reactivity $\Delta\rho_{ext}$ (the power feedback coefficient $\mathbb{A} = 488$ pcm), (b) at different values of the parameter $\mathbb{A}$ (the inserted reactivity $\Delta\rho_{ext} = 350$ pcm).

Parameter $\mathfrak{D}$ depends also on the feedback coefficient $\mathbb{A}$, defined earlier in this work [Eq. (4.10)]. It should be reminded that this parameter reflects both the in-core feedback effects and thermo-hydraulics of the system. Figure 33b demonstrates that the impact of the Y_n-effect on power stabilization increases when the absolute value of the feedback coefficient $\mathbb{A}$ decreases. This dependence of the transient suppression parameter $\mathfrak{D}$ on the parameter $\mathbb{A}$ is expectable. Indeed, if $\mathbb{A} \to 0$, i.e. in-core feedback effects are absent, the Y_n*-effect becomes the only feedback effect, exiting in the system.*

A coupled hybrid system, as mentioned already in Section 4.4, may be considered as a *critical* system with two types of neutrons contributing to the neutron balance: "core neutrons" and "source neutrons". Though this separation of neutrons is artificial, it reflects their origin and, therefore, shows the corresponding feedbacks existing in each case. In the same sense, the Y_n-effect *together with core subcriticality* can be compared with the Doppler feedback effect with respect to the external neutron source. This last assumption needs more explanations. Indeed, the Doppler-effect is both *intrinsic* and *instantaneous* feedback effect, leading to a *limitation of both the asymptotic reactor power* and *asymptotic* temperatures.

As for the Y_n-effect, it is also based on the *intrinsic* physical phenomenon: dependence of the neutron production in a finite spallation target upon the energy of the incident particle. However, to guarantee that the Y_n-effect is intrinsic, at least two conditions have to be fulfilled:

(I) the system has to be engineered in such a way that it leads to an increase of the proton energy but not of the proton current (as described above);

(II) the system has to remain subcritical in any situation, i.e. subcriticality level has to be greater than the maximal value of the inserted reactivity.

For further clarification, let us consider time intervals shorter than the characteristic time of energy transfer from the core to the external neutron source $\Delta t \ll (\lambda^+)^{-1}$. During these short periods of time the intensity the external neutron source *remains unchanged*, i.e. the prompt kinetic response of the coupled system on the reactivity insertion is the same as for an ADS. If (II) is valid, then any prompt reactivity insertion will result (similarly as due to the Doppler-effect) in a *limited* prompt jump of the reactor power (as discussed in Section 3.4) with a magnitude depending only (without any in-core feedbacks) upon the ratio "inserted reactivity/"margin to core criticality". In *this* context, the Y_n-effect *together with core subcriticality* can be considered as an intrinsic and instantaneous feedback, i.e., analogues to the Doppler-effect.

In fact, the Y_n-effect leads to the moderation of the asymptotic reactor power, if Conditions (I) and (II) are fulfilled. Therefore, it would be quite advantageous for the system safety to have this complementary feedback when "standard" core feedbacks are degraded.

4.6. Conclusions

In the present Chapter a new approach for the realization of an Accelerator Coupled hybrid System (ACS) was proposed. The concept, nominated as the DENNY system (Delayed Enhanced Neutronics with Non-linear neutron Yield), is based on the particularity of the neutron production forming a quasi-linear dependence between energy production in the core (coupled to the proton accelerator via its energy) and the external neutron yield Y_n in the spallation target (Y_n-effect). This particular dependence provides an auto-regulating behavior of the ensemble "accelerator – subcritical core". A proposed system has the kinetics of a critical system with artificial group of delayed neutrons as in the case of the "standard" ACS. In addition, its external neutron production contains the supplementary feedback, able to stabilize the installation power in its nominal state.

We showed that a significant improvement of the feedback effect due to this particular coupling between the accelerator and subcritical core (denoted as *E*-mode coupling) could be achieved. The proposed Y_n-effect can be compared to the Doppler feedback effect but for the external source neutrons. Similarly as the Doppler-effect, the Y_n-effect is intrinsic. Finally, our qualitative estimates show that the implementation of this concept could compensate eventual feedback degradation in the cores dedicated to transmute nuclear waste. Further and more quantitative analysis in this context is urgently needed. These studies should equally include the feasibility estimates to answer if the *E*-mode ACS could be realized in practice.

5. Preliminary safety study of the REBUS-3700 concept

***Résumé** – Les études présentées dans ce Chapitre sont consacrées au concept de Réacteur à Neutrons Rapides à sel fondu, fonctionnant en cycle fermé, avec une alimentation en uranium naturel ou appauvri (concept REBUS-3700). Dans la première partie du Chapitre, nous formulons les exigences et les critères auxquels un système du futur doit répondre ; en particulier les aspects ressources naturelles, sûreté, économie et non-prolifération. La deuxième partie décrit brièvement des paramètres principaux de fonctionnement. Dans la troisième et principale partie de ce Chapitre une étude préliminaire des potentialités de la sûreté déterministe de REBUS-3700 a été réalisée. Ce réacteur, fondé sur le cycle U/Pu, est caractérisé par un coefficient très important de contre-réaction ce qui est le souci principal du point de vue de sa sûreté. L'objectif de l'étude est d'évaluer le potentiel de la sûreté déterministe du concept REBUS, ainsi que le rôle éventuel de la sous-criticité dans le renforcement de la sûreté. L'étude est base sur l'analyse du bilan quasi-statique de la réactivité. Cette étude a abouti à certaines recommandations concernant le fonctionnement du réacteur (par exemple : chauffage externe du sel lors du démarrage du réacteur, contrôle de la puissance du réacteur par le débit du sel, etc.), afin de diminuer les réserves de réactivité existant dans le réacteur. Elle a démontré le potentiel excellent du REBUS (configuration critique) en ce qui concerne la sûreté déterministe, si on évite la solidification du sel primaire.*

5.1. Introduction

This Chapter is devoted to a preliminary safety study of fast spectrum molten salt reactor REBUS-3700 (Mourogov and Bokov, 2004a,b). Before starting the safety analysis of this reactor we will characterize the system in detail. We will formulate goals of this concept, the principle characteristics, and its operation. It should be noted that some reactor characteristics and operation principles were not considered as a “framework” nor “initial conditions” for safety study. On the contrary, the goal of this work was not merely to characterize the safety potential of the reactor, but also to “condition” its design and operation principles in order to meet in maximal extent the inherent safety requirements. Fortunately, the molten salt technology offers a wide range of means, which help to approach the inherent safety. In this context the core subcriticality was considered as an extra tool to achieve this objective.

5.2. Goals of the concept

The REBUS concept aims to meet the common requirements (mentioned already in the first Chapter), which have to be applied to the future nuclear systems. In the case of REBUS concept these requirements are formulated as follows.

Sustainability. Utilisation of the entire resource potential. The reactor has to be characterized by breeding gain more or equal to zero. In this case, the available depleted uranium and plutonium stocks become sufficient to cover several hundred years of operation of the world nuclear park[h].

Safety. Reactor has to satisfy the requirements of deterministic safety to a maximal extent. This goal has to be ensured by the inherent features of reactor and not by using active control systems.

Economy. Taking into account that the economical assessment applied for the future system, being in the initial stage of its development (and, hence, having a great number of uncertainties), could not offer yet reliable results, it is considered, that a future nuclear system must have a reactor design and fuel cycle configuration as simple as possible.

Non-proliferation. An effort has to be realized to reduce proliferation risk, including three principal measures: (i) elimination of uranium-plutonium separation on any fuel cycle stage; (ii) utilisation only depleted or natural uranium as supplied material; (iii) exclusion of the transport stage for irradiated or fresh fuel.

5.3. Description of the System

As demonstrated in Ref. (Mourogov and Bokov, 2004a) the combination of fast neutron spectrum and of molten salt reactor technology allows designing a nuclear power installation (see Table XII) with very attractive characteristics in respect to the above requirements. REBUS operates at the nominal power of 3700 MW(th) in a closed U-Pu cycle and supposed to be used in prospective NP, as its principal component. The fuel is based on the uranium- and transuranics chlorides dissolved in the sodium chloride, i.e. (U,TRU)Cl_3+NaCl. As demonstrated by Taube (1978), this choice permits to obtain a sufficiently hard neutron spectrum.

The attractive breeding characteristics together with on-line reprocessing allow reactor feeding only by the natural uranium. In REBUS on-line reprocessing is organized in such manner that long cooling time for spent fuel is not necessary any more. The reprocessing consists in fission product replacement by depleted uranium without any correction of trans-uranium content in the fuel. A "rich" neutron budget in fast spectrum allows reducing the reprocessing rate of the fuel. In the proposed reactor, the total core mass inventory is reprocessed in 3000 days (approximately 73 kg/FPD). The core breeding ratio is approximately 1.0, what provides a small reactivity swings during the operation. As a result, the corresponding reactivity margin can be comparable with the delayed neutron fraction. The reactor breeding could be increased by using fertile blanket.

[h] In fact, in current situation the LWR reactors use less than 1 % of uranium energetic potential. Moreover, their operation inevitably reduces to a transformation of natural uranium into depleted one in addition to plutonium accumulation. One may suppose that this situation will probably last as long as possible. Therefore, an important inventory of depleted uranium (~10^7 t) and plutonium (~10^4 t) will be accumulated in the second half of 21 century in the world. Therefore, one may suppose that the next generation reactors could be operated in the closed U-Pu fuel cycle.

Table XII. Parameters of REBUS-3700 (from Mourogov and Bokov, 2004).

Parameters	Values
Thermal Power, MW(th)	3686
Diameter/Height, m	3.8/3.25
Specific power, kW/l	100
Fuel	45%(85.2%U+14.8%TRU)Cl_3 + 55%NaCl
T_{in} / T_{out}, °C	650/730
Salt velocity (core), m/s	1.117
Volume of salt in-core/out-of-core	2/1
Masse of salt in the system, t	221
Fuel flow, kg/s	$5.1 \cdot 10^4$
Matter of structure	Alloy TZM (99% Mo)

The so-called external cooling design was chosen for studies. We remind that the molten salt playing the role of the fuel and of the heat carrier circulates with the help of pumps from core to heat-exchanger. Advantage of this type of design is simplicity of exploitation and of maintenance. In addition, in absence of the structure materials there are no structure-related feedbacks. A drawback of this design is a decay of delayed neutron precursors out of core. We remind that this leads to a decrease of the delayed neutron fraction as well as to insertion of reactivity in the case of the fuel flow change.

Another concern of REBUS-3700 is eventual fuel freezing at temperatures lower than 600°C. Here we have to do the following remarks concerning this menace:
(i) as we will see below, reactor is temperature-controlled, so any temperature decrease will lead (via reactivity feedbacks) to power growth, tending to compensate this perturbation;
(ii) in present study we do not take into account residual power which is also subject to reduce the menace of the fuel solidification;
(ii) salt solidification is not a severe accident itself (under condition that the salt homogeneity is preserved), nevertheless it can lead to some economical penalties. Anyway, some special engineering measures have to be done in the future reactor design in order to prevent salt solidification in the reactor.

Another concern is positive reactivity insertion when fuel, overcooled in a heat-exchanger, enters into the core. It should lead to undesirable significant fluctuations of core power and core temperature.

The high salt boiling temperature of approximately 1400°C gives a comfortable margin to the nominal core temperature (730°C at the exit from the core). Therefore, we may suppose that like in the case of MSR analyzed in previous Chapters the disruption criterion is based on the salt boiling temperature, hence $T^{\dagger} = 1400°\text{C}$.

One more important characteristics of the reactor which we make us of in our analysis is the physics properties of the coolant in the secondary loop. It is assumed that a secondary cooling loop to be filled with the same molten salt as in the MSBR concept (92% NaBF4+8%NaF) with fusion temperature of 385°C (Novikov and Ignatiev, 1990).

5.4. Safety Analysis

The principal difficulty for a complete safety study of REBUS-3700 consists in a shortage of the detailed description of the reactor design, which is not available in this stage of reactor concept development. Therefore, the present study will be sometimes restricted to the considerations based on general description of system properties (core dimensions, thermal feedbacks, fuel flow, etc.).

5.4.1. Reactivity coefficients and reactivity effects

The neutronics studies demonstrated that the system is characterized by a strong negative temperature feedback effect due to the elevated thermal salt expansion coefficient and large leakage. As a result, the salt expansion feedback effect is two orders of magnitude greater then Doppler effect (see Table XIII). Nevertheless, it should be noted that there exist a significant uncertainty concerning the salt heat expansion coefficient, which leads to some complications in our analysis.

Table XIII. Reactivity coefficients and reactivity effects in the REBUS-3700 core (from Ref. Mourogov and Bokov, 2004).

Coefficients	Value
Salt thermal expansion, pcm/°C	-42
Doppler, pcm/°C	-0.39
Doppler constant K_D , pcm	-404
Effects	
Neptunium, pcm	52
Fuel mass fluctuation, pcm	116
Circulation stop, pcm	83
Total, pcm	251
Fraction of delayed neutrons β (β^*), pcm	346 (263)

Within the framework of deterministic approach to safety analysis we have to consider an event, consisting in simultaneous insertion of total excess reactivity existing in the system. For this purpose the analysis of phenomena, leading to reactivity variation, is carried out. An optimisation of core parameters and reactor operation principles is performed with the objective to minimize the excess reactivity. Effects contributing into excess reactivity of REBUS-3700 in equilibrium are summarized in Table XIII. Other reactivity effects play somewhat less important role because:

- analysis of neutronics at equilibrium demonstrated that reactivity swing due to fuel burn-up can be compensated by breeding of fissile materials and fuel continues reprocessing;
- core poisoning by fission products (essentially Xe and Sm) is negligible, since the neutron absorption by these nuclei is low in fast spectrum;

– core *power level can be governed by fuel flow* in the primary loop (see Section 5.4.4.2); *fuel can be externally heated* and instilled into core at nominal temperature. Hence, none reactivity reserve for core heating is necessary.

Finally, we note that the sum of all reactivity effects (excess reactivity) is 251 pcm and therefore does not exceed the effective fraction of delayed neutrons of 263 pcm.

5.4.2. Reactor model

The safety study was based on a simplified reactor model containing three distinct elements: (1) core, (2) primary and secondary (3) salt in the heat-exchanger (see Figure 34). Every element was characterized by set of physical parameters: the mass of fuel, the specific heat capacity, the density, input and output temperatures etc. Reactor neutronics (and, consequently, heat production in the core) is described by point kinetics model (see below). In-core and out-of-core precursors decay as well as core feedback effects (Doppler and salt expansion) are taken into consideration. Relatively rapid salt expansion feedback effect (our estimations show that the relaxation period may be of the order of a few tens of milliseconds) was assumed being instantaneous. Advective model of heat-transfer from core to heat-exchangers and Newton model of heat-transfer in the heat-exchanger is adopted. Complete mathematical description of the model will be presented in subsequent Section.

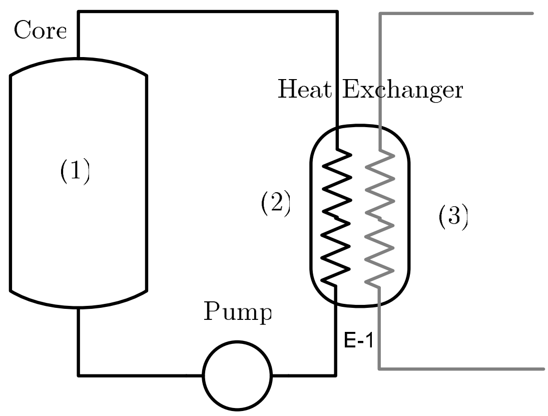

Figure 34. Schematic representation of the reactor model.

5.4.3. Reactivity balance equation for the circulating-fuel system

For the safety analysis of REBUS-3700 we will apply so-called reactivity balance method (Refs. Wade, 1986; Wade and Chang, 1987; Wade and Fujita; 1989, Gandini *et al.* 1999; Gandini *et al.* 2000). This approach consists in the analysis of a quasi-static balance of the core reactivity, assuming that the criticality has been reached asymptotically. It is implied that the system is in equilibrium state if the criticality condition ($k_{eff} = 1$) is fulfilled. Note that sometimes asymptotic states can not be reached or do not exist: these situations can be easily recognized, as in these cases some parameters have non-physical values (e.g. negative core power or negative core temperature).

For systems with circulating fuel the reactivity balance equation has to be slightly modified in order to take into account the effective reactivity appearing due to the variation of fuel flow. Point kinetics equations for the system with circulating fuel have to take into account the precursors decay beyond the active core region. In this Section we will describe dynamics of the considered reactor with the help of the following model:

$$\begin{cases} \dfrac{dP(t)}{dt} = \dfrac{\rho(t) - \beta^*}{\Lambda} P(t) + \sum_{i=1}^{N_g} \lambda_i W_i(t), \\ \dfrac{dW_i(t)}{dt} = \dfrac{\beta_i P(t)}{\Lambda} - \lambda W_i(t) + \left(\dfrac{W_i(t - \tau_{core}) \exp(-\lambda_i \tau_{out}) - W_i(t)}{\tau_{core}} \right), \quad i = \overline{1, N_g} \end{cases} \tag{5.1}$$

where ρ is the reactivity; P is the core power; W_i describes a contribution of delayed neutron precursors of i^{th}-group with the fraction β_i and the corresponding decay constant λ_i; $\beta = \sum_{i=1}^{N_g} \beta_i$ is the total fraction of delayed neutrons, β^* is its corrected value (see below); Λ is the prompt neutron generation time, τ_{core} and τ_{out} are time intervals when the fuel is "in core" and "out of core" correspondingly.

Steady-state limit of Eqs. (5.1) gives the following relationships for stationary power $(\overline{P})$ and stationary precursor concentrations $(\overline{W}_i)$ in the core:

$$\begin{cases} \dfrac{\rho - \beta^*}{\Lambda} \overline{P} + \sum_{i=1}^{N_g} \lambda_i \overline{W}_i = 0, \\ \lambda_i \overline{W}_i + \dfrac{D}{V_{core}} \left[1 - \exp\left(-\lambda_i \dfrac{V_{out}}{D} \right) \right] \overline{W}_i - \dfrac{\beta_i}{\Lambda} \overline{P} = 0, \quad i = \overline{1, N_g} \end{cases} \tag{5.2}$$

where parameter D is the salt flow; V_{core} and V_{out} are the volumes of salt in the core and outside the core correspondingly. Here it is applied that in steady state $\tau_{core} = V_{core} / D$ and $\tau_{out} = V_{out} / D$. Introducing the out-core decay factor

$$\vartheta_i \equiv \left\{ 1 + \frac{D}{\lambda_i V_{core}} \left[1 - \exp\left(-\lambda_i \frac{V_{out}}{D} \right) \right] \right\}^{-1}, \tag{5.3}$$

and after some transformations we obtain:

$$\frac{\rho - \beta^*}{\Lambda} \overline{P} + \sum_{i=1}^{N_g} \frac{\beta_i \vartheta_i}{\Lambda} \overline{P} = 0. \tag{5.4}$$

Supposing a non-trivial solution ($\overline{P} \neq 0$) and taking into account the initial condition:

$$\beta^* = \sum_{i=1}^{N_g} \beta_i \vartheta_{i,0}, \tag{5.5}$$

we can combine two terms of Eq. (5.4):

$$-\beta^* + \sum_{i=1}^{N_g} \beta_i \vartheta_i = \sum_{i=1}^{N_g} \beta_i \Delta\vartheta_i \ , \tag{5.6}$$

where $\Delta\vartheta_i = \vartheta_i - \vartheta_{i,0}$.

The effective fractions $\beta_i \vartheta_{i,0}$ and decay constants λ_i for six groups of delayed neutrons in the case of REBUS-3700 are summarized in Table XIV.

Table XIV. Parameters of delayed neurons for REBUS-3700 (from Mourogov and Bokov, 2004).

No	1	2	3	4	5	6	Total/effective
β_i (pcm)	8.14	68.25	57.13	125.83	63.52	22.96	346
$\beta_i \vartheta_{i,0}$ (pcm)	5.44	45.84	39.68	89.86	61.25	20.63	263
λ_i (s^{-1})	$1.332 \cdot 10^{-2}$	$3.049 \cdot 10^{-2}$	$1.178 \cdot 10^{-1}$	$3.131 \cdot 10^{-1}$	$9.133 \cdot 10^{-1}$	2.9711	0.108

The total reactivity of the core is the sum of the steady state (nominal) reactivity ρ_0 , of the "externally" inserted reactivity $\Delta\rho_{ext}$ and of the reactivity due to feedback effects $\Delta\rho_{feedback}$, namely $\rho = \rho_0 + \Delta\rho_{ext} + \Delta\rho_{feedback}$. In the nominal state the reactor is supposed to be critical, i.e. $\rho_0 = \left(k_{eff,0} - 1\right) / k_{eff,0} = 0$. Finally, we obtain the following reactivity balance equation for the core with circulating fuel :

$$\Delta\rho_{ext} + \Delta\rho_{precusors} + \Delta\rho_{feedback} = 0 \ , \tag{5.7}$$

where $\Delta\rho_{ext}$ is the external reactivity insertion; $\Delta\rho_{feedback}$ is the reactivity emerging due to in-core thermal feedback effects;

$$\Delta\rho_{precusors} = \sum_{i=1}^{N_g} \beta_i \Delta\vartheta_i \tag{5.8}$$

describes the effective reactivity, appearing in the case of the fuel flow change. Below we discuss every term of Eq. (5.7) in detail.

The external reactivity insertion

The external reactivity insertion term $\Delta\rho_{ext}$ in Eq. (5.7) represents either some specific in-core phenomena leading to the reactivity variation (e.g., introduction of fresh fuel from reprocessing unit) or erroneous run out of the control rods.

Reactivity, appearing due to fuel flow change

To simplify description of the effective reactivity, appearing in the case of fuel flow change ($\Delta\rho_{precusors}$), let us apply one-group approximation for delayed neutrons. In this case $\Delta\rho_{precusors} = \beta\Delta\vartheta$ with the term β being the fraction of delayed neutrons and the term $\Delta\vartheta$ being variation of the correcting factor expressed as follows

$$\vartheta(f) \equiv \left(1 + \frac{D_0 d}{\lambda V_{core}}\left[1 - \exp\left(-\frac{\lambda V_{out}}{D_0 d}\right)\right]\right)^{-1}, \tag{5.9}$$

and introduced to take into account the part of delayed neutrons decaying out of the active core region. The parameter D is the salt flow and $d \equiv D / D_0$ is the normalized salt flow (D_0 is the nominal value of the salt flow).

From Eq. (5.9) follows the upper and the lower limits for the factor ϑ: $\vartheta^{max} = \lim_{d\to 0} \vartheta(d) = 1$ and $\vartheta^{min} = \lim_{d\to\infty} \vartheta(d) = V_{core} / (V_{core} + V_{out})$. As a result, the maximal reactivity insertion due to the fuel stop $\Delta\rho_{precusors}^{max}$ will never exceed the value $\Delta\rho_{precusors}^{max} \le \beta V_{out} / (V_{core} + V_{out})$.

Reactivity due to in-core thermal feedback

The term $\Delta\rho_{feedback}$ in Eq. (5.7) is the reactivity emerging due to the in-core thermal feedback effects. In order to describe this phenomenon, one has to introduce some model of the heat transfer in the reactor. In the steady-state conditions the heat energy balance yields the following set of equations:

$$\begin{cases} \varrho c_p^{(s)} D\left(\bar{T}_{out} - \bar{T}_{in}\right) = \bar{P}, \\ \varrho c_p^{(s)} D\left(\bar{T}_{in} - \bar{T}_{out}\right) = -H\left(\bar{T}_{in} - \bar{T}_k\right). \end{cases} \tag{5.10}$$

In Eq. (5.10) $\bar{P}$ is the reactor power; $\bar{T}_{out}$, $\bar{T}_{in}$ are the primary salt mean temperatures in the core and in the heat-exchanger correspondingly; T_k is the temperature of secondary salt in the heat-exchanger; H is the heat-transfer coefficient; ϱ is the salt density; $c_p^{(s)}$ is the salt specific heat capacity. (Note that here and in further sections we use a dash over variable to denote its asymptotic value.) From Eq. (5.10) the equilibrium values of the core input and of the core output temperatures are

$$\bar{T}_{out} = \bar{T}_k + \bar{P}\left[H^{-1} + \left(\varrho c_p^{(s)} D\right)^{-1}\right], \quad \bar{T}_{in} = \bar{T}_k + \bar{P} / H. \tag{5.11}$$

A “complete” thermal feedback effect should be the sum of two terms, corresponding to the Doppler feedback effect and the salt thermal expansion feedback effect, i.e.

$$\Delta\rho_{feedback}(\bar{T}) = K_D \ln\left(\bar{T} / \bar{T}_0\right) + \alpha_{expansion}\left(\bar{T} - \bar{T}_0\right), \tag{5.12}$$

where $\bar{T} = \left(\bar{T}_{out} + \bar{T}_{in}\right)/2$ is the mean core temperature. Here, as for other fast neutron spectrum cores, a conventional logarithmic dependence is applied. In our simulation the feedback due to the fuel expansion is supposed to be *instantaneous*. For the purpose of simplicity, leading to the analytical expressions, in this section we apply a linear model for the in-core feedback, i.e.:

$$\Delta\rho_{feedback} = \alpha_{feedback}\Delta\bar{T}. \tag{5.13}$$

Here we take into account that for the system under consideration the salt expansion-related feedback coefficient $\alpha_{expansion}$ is by two orders of magnitude (see also Table XIII) stronger than the Doppler coefficient $\alpha_{Doppler}$. Therefore this latter can be neglected to a first approximation, i.e. $\alpha_{feedback} = \alpha_{Doppler} + \alpha_{expansion} \approx \alpha_{expansion}$.

Introducing for convenience the following dimensionless parameters (here subscript "0" designates a nominal value of the corresponding parameter):

$$A = -\frac{\alpha_{feedback} P_0}{H_0}, \quad B = -\frac{\alpha_{feedback} P_0}{2\varrho c_p^{(s)} D_0}, \quad h = \frac{H}{H_0}, \quad p = \frac{P}{P_0}, \tag{5.14}$$

as well as the parameter $C = -\alpha_{feedback}$, we can express the feedback term in Eq. (5.7) as follows:

$$\Delta\rho_{feedback} = -\left[C\Delta T_k + A\left(\frac{p}{h} - 1\right) + B\left(\frac{p}{d} - 1\right)\right]. \tag{5.15}$$

The reactivity balance equation (5.7) becomes

$$\Delta\rho_{ext} - C\left(T_k - T_{k,0}\right) - A\left(\frac{p}{h} - 1\right) - B\left(\frac{p}{d} - 1\right) + \beta\left(\vartheta(d) - \vartheta_0\right) = 0. \tag{5.16}$$

The reactivity balance equation in the form of Eq. (5.16) will play the major role in our analytical studies, as it describes implicitly the dependence of the equilibrium reactor power on external impacts. These external effects are introduced in our model by means of the parameters f, h, ΔT_k and $\Delta\rho_{ext}$.

Resolving Eq. (5.16) together with Eqs. (5.11) one obtains the expressions for asymptotic states of the system after some reactor parameters have been changed. In our analysis the main attention will be paid to the asymptotic core output temperature, expressed as follows:

$$\overline{T}_{out} = T_{out,0} - \Delta T_k \frac{Bh}{Ad + Bh} + \frac{AB(h-d) + (Ad + 2Bh)(\beta\Delta\vartheta + \delta\rho_{ext})}{C(Ad + Bh)}, \tag{5.17}$$

which is expected to be maximal. The equilibrium core input temperature (what corresponds to the output temperature of the heat-exchanger) is described by the following expression:

$$\overline{T}_{in} = T_{in,0} + \Delta T_k \frac{Bh}{Ad + Bh} + \frac{AB(d-h) + Ad(\beta\Delta\vartheta + \delta\rho_{ext})}{C(Ad + Bh)}. \tag{5.18}$$

We can also obtain the equilibrium value of the core power:

$$\overline{P} = P_0 \frac{\left[(A + B) + \beta\Delta\vartheta - C\Delta T_k + \Delta\rho_{ext}\right]}{Ad + Bh} hd. \tag{5.19}$$

An asymptotic mean temperature of core is determined by the values of externally inserted reactivity and of the total feedback coefficient:

$$\bar{T} = T_0 + \frac{\left(\beta\Delta\vartheta + \Delta\rho_{ext}\right)}{C}, \tag{5.20}$$

i.e. the reactor mean temperature is feedback controlled.

Table XV. Parameters of the main configuration of REBUS-3700 utilized for transient analysis (from Mourogov and Bokov, 2004).

Parameter	Value
core power P_0	3686 MW(th)
core output temperature $T_{out,0}$	730°C
core input temperature $T_{in,0}$	650°C
heat-sink temperature T_k	630°C
fuel flow D_0	12.48 m^3/s
thermal feedback coefficient C	42 pcm/°C
power feedback coefficient $A = C\left(T_{in,0} - T_{k,0}\right)$	848 pcm
power feedback coefficient $B = C\left(T_{out,0} - T_{in,0}\right)/2$	1695 pcm
correcting factor ϑ_0	0.76
heat-transfer coefficient $H_0 = P_0 / \left(T_{in,0} - T_{k,0}\right)$	184.3 MW/°C

In next Sections we carry out the quasi-static analysis of the following unprotected transients: Unprotected Transient Over-Power (UTOP), Unprotected Loss of Flow (ULOF), Unprotected Loss of Heat Sink (ULOHS) etc. The parameters of REBUS-3700 utilized for safety analysis are summarized in Table XV.

5.4.4. Quasi-static Analysis of Unprotected Transients

5.4.4.1. Unprotected Transient Over Power

The asymptotic core output ($\bar{T}_{out}$) and input ($\bar{T}_{in}$) temperature, as well as the core power ($\bar{P}$) after an accidental reactivity insertion $\delta\rho_{ext}$ are described by the expressions, following from Eqs. (5.17)-(5.20) with $\Delta T_k = 0$ and $d = h = 1$:

$$\bar{T}_{out} = T_{out,0} + \frac{(A+2B)}{C(A+B)}\Delta\rho_{ext}, \quad \bar{T}_{in} = T_{in,0} + \frac{A}{C(A+B)}\Delta\rho_{ext},$$

$$\bar{P} = P_0\left(1 + \frac{\Delta\rho_{ext}}{A+B}\right), \quad \bar{T} = \bar{T}_0 + \frac{\Delta\rho_{ext}}{C}. \tag{5.21}$$

Note that there are significant uncertainties with respect to some parameters, namely there is a significant uncertainty with respect to the salt heat expansion coefficient, and consequently, to the corresponding feedback coefficient $\alpha_{feedback}$. In order to examine the sensibility of after-transient equilibrium states on this parameter, let us replace in Eq. (5.16) the parameters $A,\ B,\ C$ by the corresponding perturbed parameters $A' = \xi A,\ B' = \xi B,\ C' = \xi C$, where the parameter ξ is a correction factor. As it follows from Eq. (5.16), this procedure is equivalent to rescaling of non-feedback terms $\Delta\rho_{ext} \to \Delta\rho_{ext}/\xi$ and $\beta \to \beta/\xi$ with parameters A, B, C being non-perturbed. In other words, to estimate the sensibility of obtained results on $\alpha_{feedback}$, it is sufficient to replace in Eqs. (5.17)-(5.20) the terms $\Delta\rho_{ext}$ and $\beta\Delta\vartheta$ by their rescaled values $\Delta\rho_{ext}/\xi$ and $(\beta\Delta\vartheta)/\xi$ correspondingly. Note that the last statement is valid if the term $\Delta\rho_{ext}$ does not depend on the parameter $\alpha_{feedback}$. It is the case of REBUS-3700 since it is supposed that there are no reactivity reserves devoted to the core heating.

Figure 35 presents dependences [Eqs. (5.21)] of the asymptotic core power and of the asymptotic core input- and output temperatures as a functions of the inserted reactivity $\Delta\rho_{ext}$. This result demonstrates that even all available excess reactivity is inserted, the maximal salt temperature does not exceed 740°C. The reactivity $\Delta\rho_{ext}$, necessary to reach the upper temperature limit, has to be nearly *two orders of magnitude greater* than the total excess reactivity available at nominal regime (see also Table XIII). Therefore, even if the reactivity feedback coefficient $\alpha_{feedback}$ is somewhat overrated and/or the excess reactivity $\Delta\rho_{ext}$ is underestimated, the system has a sufficiently great margin to the eventual core disruption temperature.

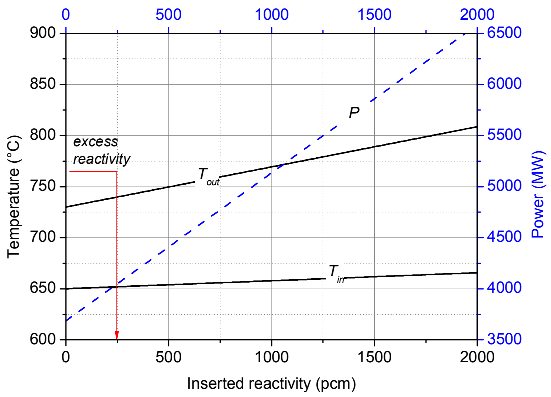

Figure 35. Equilibrium core power (P), input (T_{in}) and output (T_{out}) temperatures as a function of the inserted reactivity. The value of the excess reactivity is also indicated.

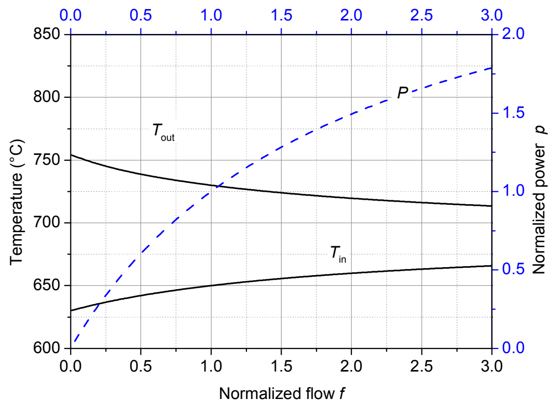

Figure 36. Equilibrium values of a normalized core power (p), input (T_{in}) and output (T_{out}) temperatures as a function of the normalized fuel flow.

5.4.4.2. Unprotected Loss Of Flow

In the case of the Unprotected Loss Of Flow (ULOF) event it is assumed that some pumps fail, leading to the decreased fuel flow. This event is twofold dangerous in the case of circulating fuel systems: a reduction of the heat evacuation would be accompanied by a certain reactivity insertion due to the increased fraction of delayed neutron precursors decaying in the core. Equally, an increase of the pump strength and, consequently, of the fuel flow may also lead to some undesirable consequences (e.g. positive reactivity insertion due to core overcooling), so this event has to be analyzed too.

The ultimate ULOF event would be a simultaneous failure of all pumps. In this case the salt flow would fall to the natural circulation value. Neither the natural convection limit of the salt flow, nor the maximal pump strength, nor the characteristic times of these events are available. Therefore, a parametrical study is carried out for the fuel flow varying in a "reasonable" range, i.e. from zero $(d = 0)$ to three nominal values $(d = 3)$. The following expressions describe the asymptotic states of the system when the fuel flow is changed:

$$\bar{T}_{out} = T_{out,0} + \frac{AB(1-d) + (Ad + 2B)\beta\Delta\vartheta(d)}{C(Ad + B)}, \quad \bar{T}_{in} = T_{in,0} + \frac{AB(d-1) + Ad\beta\Delta\vartheta(d)}{C(Ad + B)},$$
$$\bar{P} = P_0 \frac{(A + B) + \beta\Delta\vartheta(d)}{Ad + B} d\,, \ \bar{T} = T_0 + \frac{\beta\Delta\vartheta(d)}{C}. \tag{5.22}$$

Note, that a halt of the fuel circulation $(d \to 0)$ leads to the limited values of the core power and, equally of the core temperature:

$$\lim_{d\to 0} \bar{T}_{out} = T_{out,0} + \frac{A + 2\beta(1 - \vartheta_0)}{C} = 754°C\,, \quad \lim_{d\to 0} \bar{T}_{in} = T_k^0 = 630°C\,, \quad \lim_{d\to 0} P = 0\,. \tag{5.23}$$

The case of the infinite flow $(d \to \infty)$, even if this limit has no physical meaning, supplies an upper asymptote for the after-transient reactor power. In this case the reactor power is limited to: $\lim_{d\to\infty} \bar{P} = P_0 \left\{(1 + B/A) + \beta\left[V_{core}/(V_{core} + V_{out}) - \vartheta_0\right]/A\right\} \approx 3P_0$. The limit for the core temperatures is $\lim_{d\to\infty} \bar{T}_{out} = \lim_{d\to\infty} \bar{T}_{in} = 689°C$.

Figure 36 illustrates the above results. One can observe that a modification of the fuel flow leads to a monotonous change of the core input and core output temperatures, as well as of the core power. These results show that neither flow slowing down nor flow acceleration in the case of a single ULOF event may lead to unacceptable values of the core temperature.

5.4.4.3. Unprotected Loss Of Heat Sink

In this scenario it is supposed that the heat evacuation from the primary loop decreases drastically due to the decreased parameter h. In the present theoretical study we do not reveal the specific mechanism leading to this heat-transfer decrease (it can be, for example, an accidental draining of the secondary cooling loop) – it is assumed that our model describes an ensemble of such phenomena. The reactivity balance analysis gives the following dependences of the reactor temperatures $\bar{T}_{out}$, $\bar{T}_{in}$ and of the core power on the parameter h:

$$\bar{T}_{out} = T_{out,0} - \frac{AB(1-h)}{C(A + Bh)}; \quad \bar{T}_{in} = T_{in,0} + \frac{AB(1-h)}{C(A + Bh)}; \quad \bar{T} = T_0; \quad \bar{P} = P_0 \frac{A + B}{A + Bh} h\,. \tag{5.24}$$

This event is also the case, where only parametric study could be realized because of the lack of required data. Figure 37 illustrates the dependences related to Eqs. (5.24) for REBUS-3700. Moreover, the above analytical results allow us to assess the upper and lower limits for the core variables. Hence, if the heat evacuation from primary loop decreases down to zero-level $(h \to +0)$, it leads to the following values of the above parameters:

$$\lim_{h\to+0}\overline{T}_{out} = \left(T_{out,0} - \frac{B}{C}\right) = \lim_{h\to+0}\overline{T}_{in} = \left(T_{in,0} + \frac{B}{C}\right) = T_0 = 690°C \quad \text{and} \quad \lim_{h\to+0}\overline{P} = 0. \qquad (5.25)$$

Note that this case has to be treated with some precautions, since the reactivity balance equations (5.2) and (5.16) assumes a non-trivial solution (non-zero reactor power – see Section 5.3.1 for detail). So, strictly speaking, the solutions given by Eqs. (5.25) are not compatible with the initial assumption. Nevertheless, if one supposes a small but finite value of h, the results given by Eqs. (5.25) remain valid.

If $h \to \infty$, then the parameters $\overline{P}$, $\overline{T}_{out}$ and $\overline{T}_{in}$ have the following limits

$$\lim_{h\to\infty}\overline{P} = P_0\left(\frac{A+B}{A}\right) = 3P_0,\ \lim_{h\to\infty}\overline{T}_{out} = \overline{T}_{out,0} + \frac{A}{C} = 750°C \ \text{and} \ \lim_{h\to\infty}\overline{T}_{in} = \overline{T}_{in,0} - \frac{A}{C} = 630°C. \ (4.25)$$

Our estimates of the equilibrium power and temperature, presented in Figure 37, demonstrate that in the reasonable range of the variation of parameter h $(0 < h < 3)$, the reactor temperatures remain acceptable. This result let us conclude that ULOHS is “harmless” for the safety of REBUS-3700.

It has to be emphasized that the above analysis of the ULOHS events did not take into account the residual heat. Anyway, some adequate measures in the design are mandatory to guarantee a reliable residual heat evacuation. A potential solution is mentioned by Gat (1997), where the author recommends draining the fuel, by gravity, into dump tanks that are assured to retain subcriticality and have sufficient natural cooling to assure cooling of the fuel.

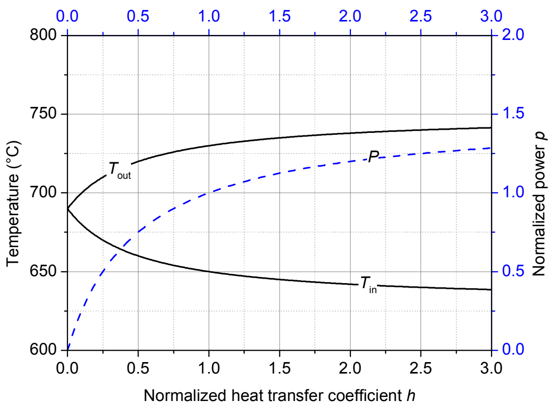

Figure 37. Equilibrium values of the normalized core power (p), input (T_{in}) and output (T_{out}) temperatures as a function of normalized heat transfer coefficient.

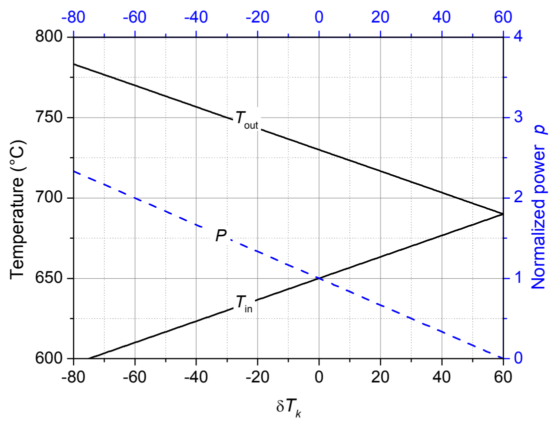

Figure 38. Reactivity balance analysis of the UOVC-event. Normalized core power (p), input (T_{in}) and output (T_{out}) temperatures are presented as a function of ΔT_k.

5.4.4.4. *Unprotected Over-Cooling*

Over-Cooling (UOVC) represents an event where the heat-sink temperature decreases radically, resulting in the decreased temperature of the salt entering the core. This event contains at least two menaces: (i) the salt solidification in the heat-exchanger and (ii) the insertion of an important positive reactivity when the overcooled fuel enters the core. The primary salt solidification would be quite undesirable event, as it may bring to loss of salt homogeneity, leading to a number of complex physicochemical phenomena.

In the case of the REBUS-3700 system it is supposed that the initiator of the overcooling event is a decrease of the secondary salt temperature (T_k) in the heat-exchanger. The magnitude of this eventual temperature drop depends on the particularities of the reactor design. However, it may be supposed that the solidification temperature of the secondary salt of 385°C (for details see Ref. Mourogov and Bokov, 2004) would be the lower limit for T_k-reduction. Indeed, one can presume that if the secondary salt is frozen, the heat transfer in the secondary loop falls drastically, what "stabilizes" the temperature of the secondary salt in the heat-exchanger. Moreover, our model is not valid if solidification of the primary or secondary salt occurs. Therefore, in our further considerations those domains of parameters, where the primary or/and the secondary salt is frozen, are "forbidden".

Let us start our quantitative analysis with the reactivity balance method. The reactivity balance equation predicts a linear dependence of the reactor temperatures and of the reactor power on T_k (see Figure 38):

$$\bar{T}_{out} = T_{out,0} - \frac{B}{A+B}\Delta T_k\,, \quad \bar{T}_{in} = T_{in,0} + \frac{B}{A+B}\Delta T_k\,, \quad \bar{T} = T_0\,, \quad \bar{P} = P_0\left(1 - \frac{C}{A+B}\Delta T_k\right). \tag{5.26}$$

As one can see from Figure 38, the solidification of the primary salt in the heat-exchanger happens if the temperature of the secondary salt falls from $T_{k,0} = 630\,°\mathrm{C}$ down to $T_k \approx 550\,°\mathrm{C}$. In this case the core output temperature still remains acceptable ($T_{out} \leq 780°\mathrm{C}$).

The analysis of further overcooling requires a much more powerful model taking into account other in-reactor phenomena. On the other hand, we may suppose that the salt solidification in the heat exchanger would lead to a halt of the salt circulation in the core. The mean core temperature being "fixed" by the thermal feedback effects, the maximal core temperature would not exceed 780°C. Nevertheless this scenario is based on a very rough vision of the in-core phenomena and should be considered with a precaution.

Note that if the secondary salt temperature is allowed to decrease below 550°C, the *solidification of the primary salt becomes possible* (i.e. reactor has no intrinsic capability to resist this menace). Anyway, a special engineering effort has to be done in order to prevent the overcooling of the secondary salt (or radically decrease the probability of this event). However, as it was already stated, our model does not include the residual heat generation. This latter could play an important role in preventing the salt solidification hazard.

Another serious menace, which attracted our attention, is a risk of the prompt criticality during the reactor start-up (i.e. at "zero"-power level) due to the fuel overcooling. Indeed, since the reactivity feedback effect is very strong, an overcooling of the in-core salt by 7°C below the nominal value would lead to prompt criticality. On the other hand, if the reactor has non-zero power, one may expect that it is capable to compensate the reactivity growth. Indeed, the mean (effective) neutron life time

$$\ell_{eff} = \ell\left(1-\beta\vartheta_0\right) + \sum_{i=1}^{6}\beta_i\vartheta_{i,0}\left(\ell+\lambda_i^{-1}\right) = 2.24\times10^{-2}\mathrm{s}$$

is smaller than the characteristic time of the heat transfer in the primary loop (equal to ~5 s, according to our estimations). Therefore an insertion of some positive reactivity would soon lead to the power and, consequently, to the temperature growth, introducing a negative reactivity (characteristic time is of the order of tens of milliseconds), which tends to compensate the initial perturbation. Numerous simulations of the UOVC-event, carried out for REBUS-3700 in nominal state (supposing the feedback being prompt), confirm this hypothesis.

At the start-up, when the core power is nearly zero and the reactor is critical with account of delayed neutrons (core is filled in with the molten salt heated up to the nominal temperature of 690°C) there is a risk to achieve a prompt criticality before the compensation mechanism, described above, starts working. So, this scenario, i.e. the UOVC-event at "zero-power" start-up state may be considered as the most dangerous one.

An ultimate overcooling event would be inflow of the chilly salt, previously cooled in the heat-exchanger to ~600°C, into the core being initially at zero-power. Unfortunately, this transient can not be simulated within the framework of the model, which we make use of. Nevertheless, it seems that this event has to be a principal concern for future safety studies.

5.5. Conclusions and outlook

In this Chapter the preliminary safety analysis of REBUS-3700 was carried out. Principal concerns for the reactor safety are indicated and an exhaustive examination of phenomena, leading to the reactivity variation in the reactor core, was done. It was demonstrated that at the equilibrium the magnitude of the excess reactivity may be decreased down to the value, comparable with the effective fraction of delayed neutrons.

The safety analysis was performed in the framework of the so-called reactivity balance method, i.e. of the quasi-static analysis of dependences of core power and core temperature on external parameters (e.g. inserted reactivity, fuel flow etc.) during unprotected transients: Unprotected Transient Over Power (UTOP), Unprotected Loss of Heat Sink (ULOHS), Unprotected Loss Of fuel Flow (ULOF), Unprotected Overcooling (UOVC), as well as of their combinations).

Above studies demonstrate an excellent resistance of REBUS-3700 to majority of single and combined unprotected events, except unprotected overcooling. This latter can result in solidification of the primary salt, if the temperature of secondary salt falls below 550°C. This accident has to be considered as the most menacing and for that reason some special measures are necessary in the reactor design to prevent the overcooling of the primary salt.

Finally, in the context of results of this preliminary safety study we may conclude that REBUS-3700 does not require any support of the external neutron source to improve its safety characteristics.

We add that the reactivity balance method is quite effective for preliminary safety study. Nevertheless, even if quasi-static analysis predicts admissible values of asymptotic core power and core temperature, it does not guarantee that they remain within the viability domain during transients. Therefore, simulations of transients, and, in particular, unprotected transients, remain the mandatory work in future safety study.

We note also that some improvements of our model are indispensable for future safety studies. Namely, more detailed reactor design would allow diminishing most of the design-related uncertainties. In order to properly describe some space-time effects, the point reactor model has to be replaced by 2D or 3D description. A coupled model of neutron, thermo-hydraulic and chemical phenomena would be quite helpful. It would allow, for example, a more adequate description of the feedback effect due to salt expansion. Moreover, more precise data, concerning physical properties of fuel are needed. The most important of them (with respect to their impact on reactor safety) are: the salt thermal expansion coefficient and the sound velocity in the salt. Finally, extra studies, in order to determine the source term and the residual heat generation, are also necessary.

6. External Neutron Sources: revision of candidates

Résumé *– Dans ce Chapitre nous réexaminerons le potentiel des sources externes de neutrons basées sur un des réactions nucléaires autres que la spallation, notamment la réaction photo-nucléaire et la fusion thermonucléaire, en prenant en compte leurs performances, leurs aspects économiques et leurs réalisabilités technique.*

Des études antérieures ont montré que des accélérateurs d'électrons sont d'une part moins pénalisants économiquement mais, d'autre part, moins performants pour la production de neutrons (en comparaison avec des accélérateurs de protons). Des estimations faites lors notre étude montrent que, si les performances des accélérateurs d'électrons en tant que source externe de neutrons ne sont pas suffisantes pour des applications industrielles, cela reste une option intéressante pour la réalisation d'un prototype (faible puissance) de système hybride.

Le concept WISE, basé sur l'utilisation consécutive de combustible uranium naturel et thorium, est particulièrement intéressant pour la production d'énergie d'origine nucléaire. Cependant, l'utilisation de Th naturel dans un cycle « once - through » conduit à des performances, du point de vue de la production de neutrons, relativement faibles et nécessite une source de neutron considérable afin de maintenir la puissance du cœur sous-critique. Traditionnellement, des réactions de spallation ont été considérées comme source possible pour un ADS, bien qu'une fraction importante de la puissance doive être dépensée pour alimenter les accélérateurs. Des sources externes de type fusion relativement petites et économiquement peu pénalisantes, peuvent permettre d'obtenir les nouveaux avantages du concept WISE, et constituer, probablement, une alternative réelle aux sources de neutrons de spallation.

6.1. Realization restrictions for hybrid systems

In previous Chapters we discussed various ARTEN-concepts, foreseen for different objectives and thus demanding different characteristics with respect to the parameters and performance of the external neutron source. Therefore, a realization of the coupled subcritical system with the only goal to compensate eventual decrease of delayed neutron fraction would require a low subcriticality level, and consequently, a rather weak external neutron source. Indeed, we showed that in some cases even small subcriticality levels, say, of the order of 300 pcm may improve the safety of ACS significantly. In this context one may look for less performing external neutron sources (in terms of neutron production) but with more attractive economics and "easier" feasibility.

We remind briefly that in order to build a hybrid system one should take into account all possible constraints which may have either physical or economical or technical realization nature.

Physical constraint is evident and consists of a physical possibility to create a sufficient number of external neutrons that nuclear waste transmutation or/and energy production is feasible in terms of neutron balance, incineration efficiency, safety requirements, etc.

Economy of the entire nuclear cycle and of the installation, in particular, should be in a good shape as well. In the case of energy production it is clear that the total installation efficiency should not be too much penalized by the operation of the external neutron source.

Finally, **technical realization feasibility** has to be taken into account as well. For example, one cannot preview the operation power of particle accelerator much higher than it is today technically possible. Similarly, the energy deposition by the incident particle beam in the neutron production target will also impose comparable limitations.

Traditionally, spallation reactions have been considered to design such a potential neutron source in the case of Accelerator Driven Systems (ADS). Typical values found in the literature are as follows: high-energy high-intensity proton beam (1 GeV protons of a few tens of mA, i.e. a few tens of MW of the primary beam power) interacts with the liquid PbBi target, resulting in $Y_n \cong 20 \div 25$ neutrons produced per incident proton. More recent design studies show that both economic and technologic problems still exist in this particular case (high costs of accelerator, resistance of the target window, corrosion of construction materials in the liquid Pb-Bi environment, an important fraction of the elaborated power has to be spent to feed a powerful accelerator, feasibility of coupling of accelerator and core environments, etc.). Therefore, our goal was to re-examine other potential neutrons sources, namely based on photonuclear reactions and thermonuclear fusion.

6.2. Photoneutron source as a candidate for applications in subcritical systems: comparison with spallation reactions

6.2.1. Introduction

Spallation neutron sources, though very effective in neutron production, are large, expensive and presently would involve certain difficulties in their routine operation (e.g., beam trips). Contrary, an electron driver, although much less effective in neutron production, is rather cheap and compact machine, which would also bring advantages in terms of reliability (Bernardin *et al.*, 2001; Ridikas *et al.*, 2002; Ridikas *et al.*, 2003).

6.2.2. Analysis. Inter-comparison: photoneutron sources versus spallation sources

Let us consider charged electrons or protons as potential incident particles to produce external neutrons. For quantitative evaluations of the above restrictions in the case of a coupled hybrid system one needs to introduce a fraction of produced electric energy f necessary to run an accelerator providing with an external neutron source:

$$f = \frac{\epsilon_e^{inp}}{\epsilon_e^{out}}, \tag{6.1}$$

where ϵ_e^{inp} is the electric energy consumed to produce and to accelerate one incident particle, ϵ_e^{out} being the mean electric energy produced by the system per incident particle. Both ϵ_e^{out} and ϵ_e^{inp} can be expressed in terms of energy (per incident particle) deposited in the system (core and neutron production target taken together), i.e. with $\epsilon_e^{inp} = \epsilon_p / \eta_a$ and $\epsilon_e^{out} = G_b \eta_e \epsilon_p$ we obtain

$$f = \frac{1}{\eta_e \eta_a G_b} . \qquad (6.2)$$

In this notation ϵ_p is the incident particle energy, η_e is the reactor electric efficiency; η_a is the accelerator efficiency, G_b is the energy multiplication coefficient of a subcritical core, given by (Salvatores *et al.*, 1996)

$$G_b = \left[\varkappa + \frac{k_{eff} \varphi^* Y_n}{\left(1 - k_{eff}\right) \nu} \frac{\epsilon_f}{\epsilon_p} \right], \qquad (6.3)$$

where $\varkappa$ is the fraction of the particle energy deposited in the system, ϵ_f is the energy released per fission, ν is the mean number of fission neutrons, φ^* is the importance of source neutrons, Y_n is the mean neutron yield per incident particle, and k_{eff} is the multiplication factor of the system.

Finally, in the coupled hybrid system the intensity of the external neutron source, required to sustain the reactor power P_{th} can be expressed as follows:

$$I_n = \frac{P_{th}}{\epsilon_{th}^{out}} = \frac{P_{th}}{G_b \epsilon_p} . \qquad (6.4)$$

For qualitative estimates let us choose a molten salt AMSTER-like core (see Chapter 3 for details). The following parameters were employed for the analysis: $P = 2500\,\mathrm{MW(th)}$, $\eta_e = 0.44$ (Lecarpentier, 2001), $\varphi^* = 1.2$, $\epsilon_f = 200\,\mathrm{MeV}$, and $\varkappa = 1$. A number of fission neutrons were evaluated from Ref. (Blinkin and Novikov, 1978), namely $\nu = 2.505$. We suppose that the efficiency for both electron and proton accelerator is $\eta_a = 0.5$. It is suitable to use the ratio

$$y_n\left(\epsilon_p\right) = Y_n\left(\epsilon_p\right) / \epsilon_p , \qquad (6.5)$$

which one can consider approximately constant for incident particle energies higher beyond some threshold value (Letourneau, 2000). Based on the simulations using the multi-particle transport code MCNPX (Waters, 2003), for electrons we obtain

$y_n^{(e)} = 6 \cdot 10^{-4}$ neutron/MeV/electron

for a thick photonuclear target (^{238}U surrounded by ^{nat}Pb). Indeed, this value remains nearly constant for electron energies $\epsilon_p^{(e)} > 150$ MeV. For a thick spallation target built of liquid Pb-Bi we obtain

$y_n^{(p)} = 2.5 \cdot 10^{-2}$ neutron/MeV/proton.

Similarly like for electrons, this value is not changing for proton energies $\epsilon_p^{(p)} > 1000$ MeV. Consequently, for the parameter G_b we obtain

$$G_b^{(e)} = 1 + \frac{5.760 \cdot 10^{-2}}{r_0} \quad \text{and} \quad G_b^{(p)} = 1 + \frac{1.92}{r_0}.$$

We note that two different scenarios were considered (see Ref. Bokov *et al.*, 2003):
(i) the concept of an industrial hybrid system, based on sub-critical core with the ~2500 MW(th) power and being a net energy producer, and
(ii) the concept of a prototype hybrid system, based on subcritical core with the ~200 MW(th) power, and not necessarily net energy producer.

The results of the above formulation are summarized in Figure 39, where the use of proton and electron accelerators is quantitatively compared. Protons, being more efficient in neutron production than electrons, would use ~40 times smaller fraction of available energy f for a chosen subcriticality level (Figure 39a). In both cases nearly the same external neutron source intensity will be needed to produce the same output energy P_{th} (Figure 39b). The small difference (compare solid and dashed curves) is due to different beam power deposited in the production target, which is much higher for electrons. Therefore, to obtain the same total outlet power, which also includes the beam power deposited in the neutron production target, with electrons one would need smaller neutron source intensity (by ~15 %). In the same Figure 39b we distinguished the industrial and prototype system requirements since different realization constraints might be applied for these two cases.

In the case of the hybrid MSR, compared to the critical MSR, the total system efficiency decreases from $\eta_e^{(MSR)} = 44$ % down to the value: $\eta_e^{(hybrid)} = \eta_e^{(MSR)}(1-f)$. Let us suppose that the minimal acceptable efficiency of hybrid MSR is the actual value of the efficiency of present Pressurized Water Reactors (PWR): $\eta_e^{(PWR)} \approx 33\,\%$. Therefore, in terms of economical restrictions the maximal available fraction of the total produced energy by hybrid MSR is $f_{max} = 1 - \eta_e^{(PWR)} / \eta_e^{(MSR)} = 25\,\%$. This is valid for the industrial solution, while a prototype-demonstrator system does not necessarily need to be a net energy producer. In other words, f_{max} can be as high as 100 %.

As it was discussed by Ridikas *et al.* (2003), with today's **electron** machine and production target technology one could possibly reach neutron source intensities up to $I_n^{(e)} \simeq 2 \cdot 10^{17}$ n/s, e.g. 150 MeV electrons at 50 MW beam power. In the case of 1 GeV **protons**, 50 MW beam would result in $I_n^{(p)} \simeq 8 \cdot 10^{18}$ n/s. Higher continuous neutron source intensities will be hard to reach even in the near future. For the above values, the electron machine would be nearly by a factor of 10 cheaper (Bernardin *et al.*, 2001; Ridikas *et al.*, 2003), more compact, more reliable (e.g. beam trips) and easier to realize in terms of radioprotection requirements (e.g. shielding against high energy proton and neutron fluxes).

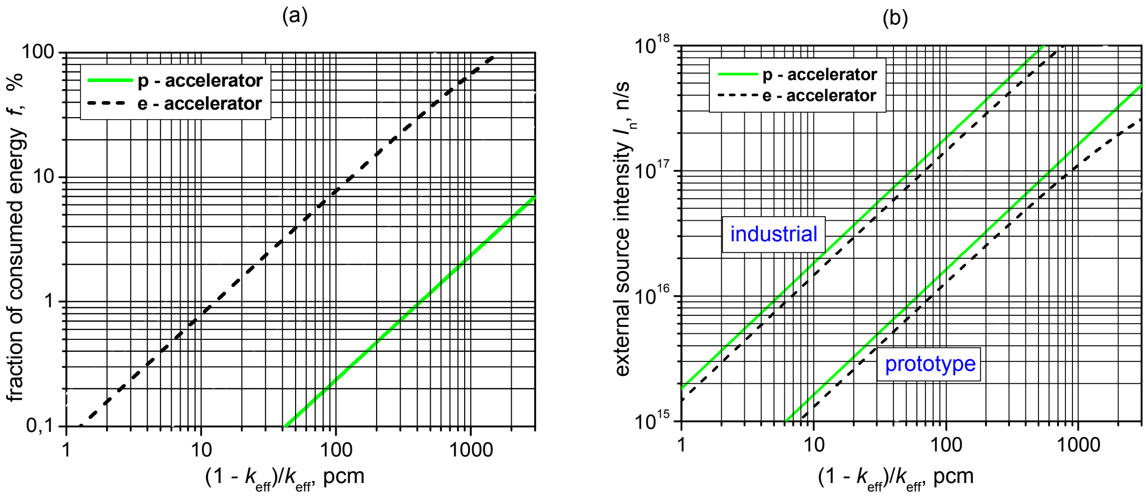

Figure 39. Fraction (f) of energy, consumed by accelerator (a), and the intensity of external neutron source (b) as a function of the subcriticality level. Proton and electron accelerator options are presented as solid and dashed lines correspondingly. Industrial and prototype options stand for 2500 MW(th) and for 200 MW(th) AMSTER-type subcritical cores respectively.

6.2.3. Results and conclusions

According to the technical realization and economical criteria as discussed above we may draw the following conclusions:

with the **electron** machines one could reach the subcriticality level of

- approximately 100 pcm in the case of the *industrial* hybrid reactor and
- approximately 2000 pcm in the case of the *prototype* hybrid reactor considered in this study (see Figure 39 for details).

The use of **proton accelerator**, being more efficient in neutron production, is much more flexible in this respect. For example, the proton accelerator would use less than 10 % of the available produced energy to reach subcriticality level as high as 3000 pcm.

Nevertheless, we note that the electron-option should not be neglected in the case when a design of a hybrid-demonstrator is planned. This solution would be certainly a cheaper option if compared to a proton driven external neutron source. On the other hand, it is clear that for industrial hybrids much more external neutrons are needed than it is achieved using electron accelerators.

6.3. Thermo-nuclear fusion as a candidate for external neutron source in hybrid systems

6.3.1. Introduction

Fission-fusion hybrid systems have been intensively studied a long time ago with orientation on different goals: utilization and multiplication of the fusion energy, breeding of fission materials for conventional fission reactors, enhanced breeding of tritium, etc. In these concepts, fusion was called to play the leading role in energy production, while fission played the subsidiary role.

Unfortunately, today the high costs of fusion energy, as well as its cumbersome designs still do not allow using them in practice. In addition, its practical economics still remains rather questionable. Meanwhile, some innovative hybrid concepts like the Energy Amplifier by Prof. C. Rubbia (1995), or the "mobile" fuelled WISE (Slessarev *et al.*, 2001), etc. have an extraordinary potential to satisfy current requirements of nuclear energy production for long term if reasonable neutron sources are found. The WISE (Waste-free, Intrinsically Safe, and Efficient) concept, based on the sequential application of, first, natural uranium and, afterward, thorium fuel (in the form of molten salt, liquid metals, etc.), seems to be particularly suited for the future NP for several reasons: there is no longer the necessity of fuel enrichment and of irradiated fuel massive reprocessing, a considerable reduction of long-lived toxic wastes, significant protection against weapons material proliferation, enormous fuel reserves, etc. However, the use of "poor" natural Th-fuel as well as of the "once-through" fuel cycle makes neutronics of the WISE-core particularly weak and it requires a considerable external neutron source to support subcritical cores.

As discussed above, traditionally spallation reactions have been considered as candidates for a realization of a potential neutron source in Accelerator-Driven Systems (ADS), although in the case of the WISE concept an important fraction of elaborated power should be spent to feed the corresponding powerful accelerators. Spallation by protons is evidently not the single way to support this innovative fuel cycle. For example, a small-size current fusion designs (in which unsatisfactory elevated cost of energy production or a negative net energy yield do not allow utilize it in practice) may be used for the external neutron production in hybrid systems. The appearance of WISE stimulates to revise the attitude to such "non-economical" fusion design, which would play now a subsidiary but still very important role of the external neutron source. In this context, the realization of WISE with all their advantages could compensate the economic penalty for their utilization.

6.3.2. Fusion reactions

Let us consider the potential of thermo-nuclear fusion to supply a subcritical hybrid core with a sufficient amount of neutrons. One notes that neutron production is also accompanied by energy production. To assess the effectiveness of this neutron production by fusion, a special parameter will be used, namely $\tilde{\epsilon}^{fusion}$ being the *energy released per fusion neutron produced in a fusion reaction*. In addition, one has to take into account the fusion neutron importance $\tilde{\varphi}^*$ (which reflects the difference between released fusion neutrons and the "averaged" neutron worth in a fission blanket) as well as the "parasitic" neutron captures in a wall, which separates the fusion reaction domain and the subcritical blanket. For the corresponding correction, one can use the parameter a_w as the coefficient showing the neutron loss in the walls: it is the ratio of all produced fusion neutrons to all neutrons entering the fission blanket.

There are several schemes of fusion reactions (Harms and Heindler, 1982) which can be applied.

I. "Pure D-D" fusion reactions present altogether (after summing all two channels):

$4\text{D} \rightarrow {}^3\text{He} + \text{T} + \text{n} + \text{p} + 7.3\ \text{MeV}$.

These reactions produce one fast neutron per fusion of about "fission energy" (with energy 2.4 MeV). After transportation through the "first wall", which separates fusion and fission domains, a fraction of such neutrons (about 20 %) is lost (Shmelev, 2000). Once in the hybrid core, their importance $\tilde{\varphi}^*$ does not exceed 1.2. Hence: $\tilde{\epsilon}^{fusion} = 7.3\,\text{MeV/neutron}$,

II. "SCAT-D" multi-channel reactions:

$5\,\text{D} \rightarrow {}^3\text{He} + 2\text{n} + \alpha + \text{p} + 24.9\ \text{MeV}$.

These reactions produce *one hard neutron* (14.1 MeV) with its assessed importance $\tilde{\varphi}^*_{fusion} = 1.8$ (Shmelev, 2000) and a *second neutron* similar to those released in D-D reactions. On average, for both neutrons: $\tilde{\varphi}^* = 1.5$; $\tilde{\epsilon}^{fusion} = 12.45\,\text{MeV/neutron}$.

III. "D-T" reaction (T breeding is required as one of fuel components):

$\text{D} + \text{T} \rightarrow \text{n} + \alpha + 17.6\ \text{MeV}$.

This reaction produces one hard neutron (14.1 MeV) with expected importance $\tilde{\varphi}^* = 1.8$ (i.e. the number of all incoming neutrons can be multiplied by this factor to take into account all secondary reactions as (n,2n); (n,3n), etc. (Shmelev, 2000). The neutron balance can be considered in two ways discussed below.

- Approximately one neutron is consumed for tritium (T) breeding: T is produced by exothermic (4.8 MeV) reaction on ^{6}Li. However, breeding requires one thermal neutron with importance close to unity ($\varphi^*_b = 1$). Thus, one can evaluate the neutron "effective" importance as $\tilde{\varphi}^* = \varphi^* - \varphi^*_b = 0.8$. Hence, $\tilde{\epsilon}^{fusion} = (17.6 + 4.8) = 22.4$ MeV/neutron .
- In the case, where ^{7}Li is used for T-breeding (Harms and Heindler, 1982), no neutron consumption is foreseen for breeding: ${}^7\text{Li} + \text{n}_\text{f} \rightarrow \text{n}' + \alpha + \text{T}$, where n_f and n' are a fast and thermalized neutrons respectively. The neutron importance is close to 1: $\tilde{\varphi}^* = 1$. In this case the breeding reaction is endothermic, therefore $\tilde{\epsilon}^{fusion} = 17.6 - 2.5 = 15.1$ MeV/neutron.

6.3.3. Neutron analysis. Inter-comparison: ADS versus a fission-fusion hybrid

Installations, using all mentioned above fusion reactions, produce in practice less electric energy than they consume, although are approaching gradually to the "break-even-point" due to enormous international efforts of scientists. Nevertheless, even with such a "negative" energy balance, these schemes can already be rather effective to replace, for example, proton sources in hybrids, when external sources have to be very powerful, in particular, for natural Th fuelled WISE system (Slessarev *et al.*, 2001).

Below we introduce a general energy balance scheme in the case of fusion-fission hybrid systems. A fusion device consumes some electrical energy for its needs ($\tilde{E}^{inp}_e$) and produces output energy $\tilde{E}^{fusion}$ in the form of kinetic energy of charged and neutral particles (see Figure 40).

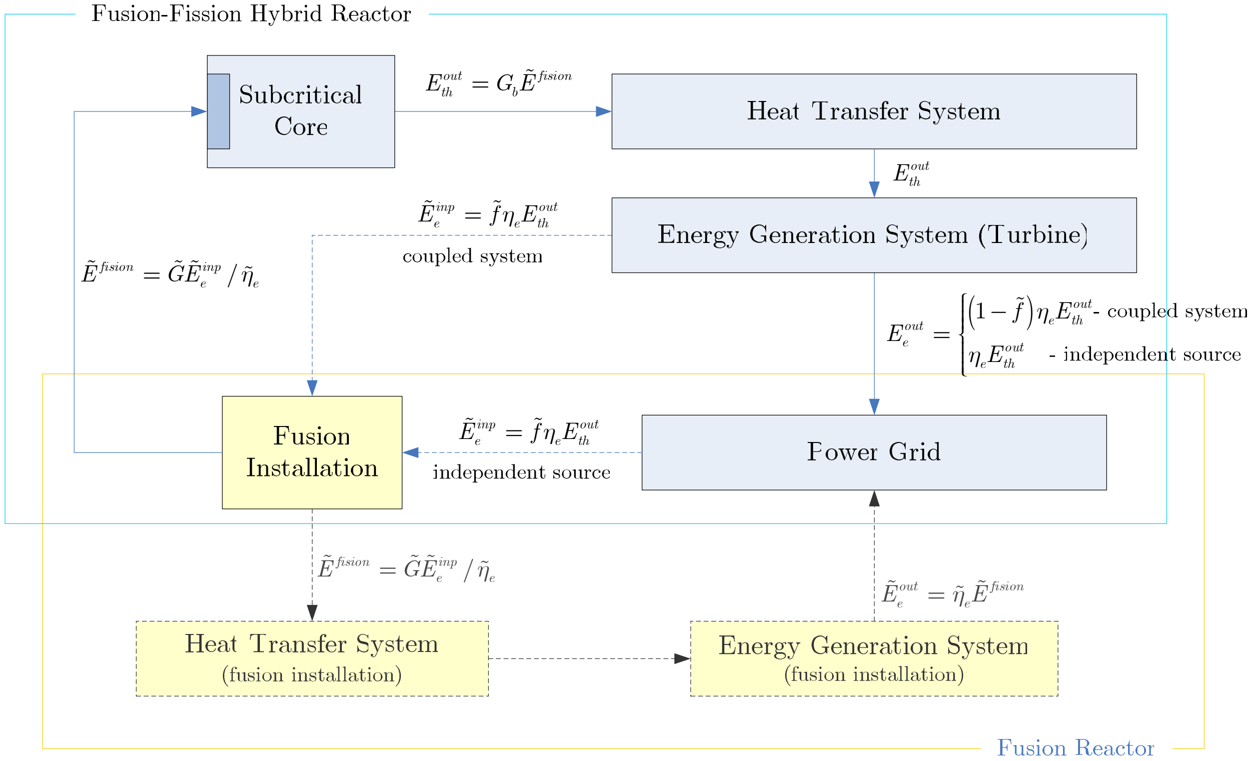

Figure 40. Energy transfer diagram in the Fusion-Fission Hybrid reactor.

Let us denote $\tilde{G}$ the ratio of the total output electric energy to the total consumed electric energy. The parameter $\tilde{G}$ plays a role of the power gain in the fusion installation. Denoting $\tilde{E}_e^{out}$ the total output electric energy of fusion reaction and $\tilde{\eta}_e = \tilde{E}_e^{out} / \tilde{E}^{fusion}$ the efficiency of its transformation to the electric energy, one writes:

$$\tilde{G} = \tilde{\eta}_e \frac{\tilde{E}^{fusion}}{\tilde{E}_e^{inp}}, \tag{6.6}$$

where $\tilde{G} \geq 1$ corresponds to the "positive" and $\tilde{G} < 1$ corresponds to the "negative" energy balance of a fusion installation. Referring to one produced "fusion" neutron, one can express $\tilde{G}$ in the following way:

$$\tilde{G} = \tilde{\eta}_e \frac{\tilde{\epsilon}^{fusion}}{\tilde{\epsilon}_e^{inp}}. \tag{6.7}$$

Note, that with $\tilde{G} \leq 1$, there is no practical sense to use "pure" fusion for energy production. However, this situation could still have some significance if the fusion device is used as a supplementary neutron source for a hybrid system, if such a system opens an attractive perspective for NP.

The thermal energy, which is produced in a subcritical core (blanket) when this core has received one fusion neutron, consists of the energy released from fusion reactions plus the total energy released from fission reactions in the core:

$$\epsilon_{th} = \tilde{\epsilon}^{fusion} + \frac{k_{eff}}{\nu\left(1-k_{eff}\right)}\frac{\tilde{\varphi}^*}{a_w}\epsilon_f\,, \tag{6.8}$$

where k_{eff} is the core multiplication coefficient, ν is the average number of neutrons released per fission; $\tilde{\varphi}^*$ is the importance of fusion neutrons; ϵ_f is the fission energy; a_w is the ratio of all produced fusion neutrons to all neutrons entering in the blanket. Denoting K_A the coefficient of multiplication (amplification) of fission energy in the subcritical blanket:

$$K_A = \frac{k_{eff}}{\left(1-k_{eff}\right)\nu} \tag{6.9}$$

and taking into account the efficiency of transformation of core thermal energy to electrical energy $\epsilon_e = \epsilon_{th}\eta_e$ one obtains that the fraction of total electrical energy, which has to be spent for the reproduction of one fusion neutron and to sustain the energy production, can be assessed now as

$$\tilde{f} = \frac{\tilde{\epsilon}_e^{inp}}{\epsilon_e^{out}} = \frac{\tilde{\eta}_e}{\tilde{G}\eta_e\left(1+\frac{\epsilon_f}{\tilde{\epsilon}^{fusion}}K_A\frac{\tilde{\varphi}^*}{a_w}\right)}\,. \tag{6.10}$$

Estimations show that for “WISE”-like cores (with $k_{eff} \approx 0.9$) one can neglect the first term in the denominator, so one gets:

$$\tilde{f} = \frac{\tilde{\epsilon}_e^{inp}}{\epsilon_e^{out}} = \frac{\tilde{\eta}_e}{\eta_e}\frac{1}{\epsilon_f K_A}\frac{\tilde{\epsilon}^{fusion}a_w}{\tilde{G}\tilde{\varphi}^*} = \frac{1}{\eta_e\epsilon_f K_A\tilde{z}_n}\,, \tag{6.11}$$

where $\tilde{z}_n$ is the effective fusion neutron yield in a blanket per consumed electric energy or electric energy cost of neutron production, based on a fusion reaction. In this notation $\tilde{z}_n$ is defined as follows:

$$\tilde{z}_n = \frac{\tilde{G}\tilde{\varphi}^*}{\tilde{\eta}_e\tilde{\epsilon}^{fusion}a_w}\,. \tag{6.12}$$

This means that the fraction of power consumed by the supplementary source is inversely proportional to the effective neutron yield of this source and, equally, to the coefficient of power amplification of the blanket.

Now we are ready to establish a direct comparison for the energy consumption of the supplementary neutron sources, based on fusion reactions with well-known ADS systems, where neutrons are produced by proton beam due to spallation reaction. One can carry out similar assessment of energy, which is required for production of neutrons via spallation. As a result, the fraction of total electrical energy, which has to be spent to sustain energy production in ADS, can be evaluated (see previous Section) as follows:

$$f = \frac{\epsilon_e^{inp}}{\epsilon_e^{out}} = \frac{1}{\eta_e \eta_a} \frac{1}{\left[\varkappa + K_A Y_n \varphi^* \frac{\epsilon_f}{\epsilon_p}\right]} . \tag{6.13}$$

After neglecting the first term in the denominator, one obtains, similarly to the case of the fusion source:

$$f = \frac{\epsilon_e^{inp}}{\epsilon_e^{out}} = \frac{1}{\eta_e \eta_a} \frac{\epsilon_p}{K_A \epsilon_f Y_n \varphi^*} = \frac{1}{\eta_e K_A \epsilon_f z_n} , \tag{6.14}$$

where the "effective" spallation neutron yield z_n per consumed electric energy is defined as

$$z_n = \eta_a y_n \varphi^* , \text{ with } y_n = Y_n / \epsilon_p . \tag{6.15}$$

Table XVI shows the fractions of energy consumption in the case of neutron sources based either on fusion or spallation reactions. It is evident that fusion produces many more neutrons per unit power than spallation. So, one can obtain the simple formula for the inter-comparison of the spallation/fusion fractions of energy in a hybrid required to support energy production by fission:

$$\frac{f}{\tilde{f}} = \frac{\tilde{z}_n}{z_n} = \frac{\tilde{G}}{\tilde{\eta}_e a_w \eta_a y_n \tilde{\epsilon}^{fusion}} \frac{\tilde{\varphi}^*}{\varphi^*} . \tag{6.16}$$

The inter-comparison of z_n-values demonstrates that the effectiveness of fusion for neutron production (when $\tilde{G} \to 1$) is significantly higher when compared with spallation. In addition, it rises when the $\tilde{G}$ increases (see Table XVI), as a lower fraction $\tilde{f}$ is necessary to feed the external neutron source (fusion installation).

Table XVI. Required supplementary energy consumptions ($\tilde{f}$, %) in subcritical hybrids supplied with fusion or spallation reactions. ($k_{eff} = 0.9$, $\eta_e = \tilde{\eta}_e = \eta_a = 0.45$, $\nu = 2.5$)

Sources of supplementary neutrons in different Hybrid Systems	Effective neutron yields z_n (MeV)$^{-1}$	$\tilde{f}$, %,		
FUSION ($a_w = 1.2$)	$\tilde{G} = 1$	$\tilde{G} = 1$	$\tilde{G} = 1/3$	$\tilde{G} = 1/10$
D-D (WISE-Fusion)	0.30	1.0	3.0	10
SCAT-D (WISE-Fusion)	0.22	1.4	4.2	14
D-T, breeding on Li-6 (WISE-Fusion)	0.07	4.1	8.2	41
D-T, breeding on Li-7 (WISE-Fusion)	0.12	2.5	7.5	25
SPALLATION				
Spallation by proton $\epsilon_p = 1$ GeV, lead target, $Y_n = 20$, $\varphi^* = 1.3$ (Slessarev *et al.*, 2001)	0.0117	$f = 26\%$		

It is important to note that the consumed power of the fusion sources is surprisingly small in most of the cases. So, for D-D reactions, the proportion between power of fusion source and the core of WISE is expected to be of order of 1 % if $\tilde{G} = 1$. Certainly, the fraction of the fusion part will grow if less effective fusion reactions or less beneficial economy of fusion sources ($\tilde{G} \leq 1/3$) are used.

6.3.4. Conclusion

The neutron abundance of fusion reactions (particularly D-D reactions) per consumed energy unit could make such sources more attractive when compared with the spallation neutron sources. For example, fusion sources are preferable (when compared with the spallation source) if their electric energy consumption does not exceed the total thermal energy production by a factor of about $\tilde{G} \geq 0.1$.

The closer the point $\tilde{G} = 1$ the more beneficial fusion sources become. Even for hybrids with significant core subcriticality (e.g., the WISE concept with Th-fuel), the required power (for the external neutron production) can be assessed as small as 3 % of the total blanket power if the D-D fusion source with $\tilde{G} = 1/3$ is employed. The weakest potential is expected in the case of the D-T reaction (with the T-breeding on Li-6).

It seems that the external neutron sources, based on fusion reactions, offer a possibility to profit from the new advantages of hybrid concepts. However, these neutron sources should be relatively small that the hybrids are not too much penalized economically. We conclude that fusion reactions could be considered as a promising alternative to the spallation-type neutron sources taking into account the technical feasibility and economics of the near future fusion installations with $\tilde{G} \leq 1$.

7. Conclusions

Des systèmes nucléaires innovants peuvent, dans certains cas, souffrir d'une dégradation de leurs propriétés importantes pour leur sûreté, notamment la diminution de la fraction de neutrons retardés et la dégradation des effets de contre-réaction. De ce fait, il devient crucial pour la réalisation de tels systèmes de garantir leur fonctionnement sûr.

Les systèmes hybrides (comprenant le cœur sous-critique et la source externe de neutrons) sont étudiés pour remédier à ces problèmes de sûreté. Pour cela, la source de neutrons doit être suffisamment performante et puissante, ce qui entraîne de considérables contraintes, à la fois économiques et technologiques. Des études antérieures ont démontré que si la sous-criticité est un moyen très efficace d'améliorer la sûreté, son rôle reste annexe. Ainsi, pour répondre aux exigences de la sûreté déterministe, certaines démarches d'optimisation des propriétés du cœur doivent également être effectuées (par exemple : la diminution des réserves de réactivité). Dans le cadre d'une approche déterministe pour l'analyse de la sûreté (utilisée dans nos études), tous les accidents potentiels causés par des impacts internes ou externes dus à des erreurs humaines ou à des défaillances techniques doivent être protégés en utilisant des propriétés intrinsèques des composants du réacteur. Il faut que le système ait la possibilité intrinsèque d'autoprotection, c'est à dire qu'aucun de ces événements ne conduise à des conséquences graves (comme le rejet de radioactivité dans l'environnement nécessitant une évacuation de la population). Les réacteurs à sel fondu sont connus pour être particulièrement propices à la sûreté déterministe, où le retraitement en ligne ainsi que d'autres propriétés intrinsèques diminuent considérablement les initiateurs d'accidents graves. C'est pourquoi, des réacteurs hybrides à sel fondu (WISE, AMSTER, RSF, REBUS) sont pris comme des systèmes de référence dans nos études.

Une réflexion qui concerne la place de la sous-criticité dans l'amélioration de la sûreté est l'idée clé de toutes les études. Ce rôle étant fortement dépendent du mode de fonctionnement du système (critique ou sous-critique, couplé ou découplé, etc.), du pilotage (par l'accélérateur ou par des barres de contrôle), et de l'approche (déterministe ou probabiliste). Une prise en compte de tous ces aspects parait nécessaire pour donner une réponse exhaustive à ce problème. Une tentative préliminaire de synthèse est effectuée. L'étude a inclus aussi bien des estimations analytiques que des simulations numériques de la cinétique et de la dynamique des systèmes. La réponse dynamique des systèmes aux perturbations externes (transitoires) non protégées prend une place essentielle lors de nos études.

Ces études ont démontré que la sous-criticité permet d'adoucir les transitoires, d'augmenter le temps de grâce, même si le coefficient de contre-réaction est défavorable. Les résultats montrent que même un petit niveau de sous-criticité (2-3 dollars) peut significativement améliorer la sûreté. Les deux types de réalisation de système hybride : l'ADS (source externe de neutrons est indépendante de la puissance du coeur) et l'ACS (système sous-critique avec couplage « taux de fission – intensité de la source externe de neutrons») peuvent améliorer la sûreté. Le couplage thermo-hydraulique entre un cœur sous-critique et une source de neutron de spallation produit un groupe supplémentaire de neutrons retardés. L'étude a aussi montré que deux moyens : une optimisation thermo-hydraulique et la sous-criticité peuvent compenser la dégradation de l'effet Doppler et la réduction de la fraction des neutrons retardés. De plus, il est démontré qu'on peut profiter de certaines particularités de production de neutrons dans une cible de spallation en fonction de l'énergie des protons incidents (Yn-effect) et de ce fait accentuer les propriétés stabilisatrices de la puissance du cœur lors des accidents de *réactivité non protégés (concept DENNY). Une caractérisation générale du fonctionnement de DENNY est donnée. Lors de cette étude il a été démontré que l'influence bénéfique de l'effet Y_n-effect sur la dynamique des systèmes hybrides couplés peut être considérable (surtout en l'absence d'effet Doppler).*

Une étude préliminaire des potentialités de la sûreté déterministe du Réacteur à Neutrons Rapides (RNR) à sel fondu (concept REBUS-3700) a été réalisée dans le cadre de ce travail de thèse. L'objectif de l'étude était d'évaluer le potentiel de la sûreté déterministe du concept REBUS, ainsi que le rôle éventuel de la sous-criticité dans le renforcement de la sûreté. L'étude est basée sur l'analyse du bilan quasi-statique de la réactivité. Cette étude a abouti à certaines recommandations concernant le fonctionnement du réacteur (par exemple : chauffage externe du sel lors démarrage du réacteur, contrôle de la puissance du réacteur par le débit du sel, etc.), afin de diminuer les réserves de réactivité existant dans le réacteur. Elle a démontré le potentiel excellent du REBUS (même dans la configuration critique) en ce qui concerne la sûreté déterministe, si on évite la solidification du sel primaire.

Enfin, le potentiel des sources externes de neutrons basées sur les réactions nucléaires autres que la spallation, notamment la réaction photo-nucléaire et la fusion thermonucléaire, a été réexaminé en prenant en compte des aspects tels que leurs performances, leurs aspects économiques et leurs réalisabilités technique. Des estimations faites lors de notre nouvelle étude montrent que, si les performances des accélérateurs d'électrons en tant que source externe de neutrons ne sont pas suffisantes pour des applications industrielles, cela reste une option intéressante pour la réalisation d'un prototype (faible puissance) de système hybride. Des sources externes de fusion relativement petites et économiquement peu pénalisantes, peuvent permettre d'obtenir les nouveaux avantages de nouveaux concepts, notamment du concept WISE, et constituer, probablement, une alternative aux sources de neutrons de spallation.

The safe operation of nuclear installations is one of crucial conditions for the future of Nuclear Power. The safety is a complex problem, which is interconnected with other aspects of the Nuclear Power such as economical competitiveness, waste minimization, long term sustainability, etc. Therefore, the safety issues have to be considered from the very beginning of the development of nuclear reactor concepts, and the corresponding measures have to be anticipated in the design. The inherent safety approach (which supposes that safety functions have to be achieved by use of inherent, based on nature laws, properties of reactor components) is one of the most radical means to meet this condition. Among others, the molten salt reactors are known to be particularly suitable for inherent safety due to its intrinsic properties, namely low internal pressure, low fuel-coolant inflammability, small reactivity margin due to on-line reprocessing, etc. However, on the way to the intrinsic safety other inherent properties as positive thermal feedback effect due to graphite-moderator can bring some unacceptable problems. A novel option aiming to handle the above problems may be the artificial enhancement of system neutronics: an external neutron source added to the core permits the system to operate with subcritical core (so-called hybrid systems). In the context of inherent safety approach two types of the hybrid system realization may be distinguished: in the first one the intensity of the external neutron source may be *independent* of core power (independent-source systems, e.g. Accelerator-Driven Systems - ADS), and in the second one the external neutron source depends intrinsically on core power (coupled-source systems, e.g. Accelerator-Coupled Systems - ACS).

The **main goal of this thesis work** was to investigate the role of core subcriticality for safety enhancement of advanced nuclear systems, in particular, molten salt reactors, devoted to both energy production and waste incineration/transmutation. The inherent safety is considered as ultimate goal of this safety enhancement.

An attempt to apply a systematic approach for the analysis of the subcriticality contribution to inherent properties of hybrid system was performed in this thesis. Starting with simulation of the kinetics and the dynamics of subcritical systems during unprotected (by the active safety tools) transients the obtained results were able to characterize the inherent behavior of the systems considered in this work. These results were used as a staring point for proceeding towards the above main goal. Some general relationships, connecting the thermo-hydraulic characteristics of the reactor, the temperature and power feedback coefficients, the subcriticality level, excess reactivity/current, etc., were looked for. These relationships provided a combination of above parameters, which could guide towards the acceptable/improved behavior of nuclear systems under consideration during unprotected transients. The results of this research, *listed below*, prove that in many cases the subcriticality may improve radically the safety characteristics of nuclear reactors, and in some configurations it helps to reach the "absolute" intrinsic safety.

I. In the present work a comparative analysis of the dynamics of different types (independent-source systems and coupled-source systems) of hybrid systems was carried out for the first time. The general conclusion is that the subcritical regime improves the safety potential significantly, leading to the *considerable increase of the grace time* up to several hours. This effect was observed for some examined systems even for small subcriticality levels of 1-3 dollars. In any case, a proper choice of subcriticality level makes *all* analyzed transients considerably *slower*, and *monotonic*, i.e. the maximal temperature/power values are achieved asymptotically. It was also shown that the weakest point of the **independent-source systems** with respect to the deterministic safety is *thermo-hydraulic unprotected transients*, while in the case of the **coupled-source systems** the *excess reactivity/current insertion events* remain a point of concern.

II. In the case of the **independent-source systems** (e.g. ADS) it was demonstrated that the added subcriticality may compensate the degradation of the core feedback effects. Thus, in the case of systems characterized by a small (when compared with conventional critical reactors) but negative feedback coefficient an appropriate choice of subcriticality level permits to *compensate this feedback degradation in terms of the asymptotic power level*. In the case of the positive feedback coefficient, subcriticality allows improving the safety characteristics in terms of *significant expansion of the grace time*. The corresponding relationships were obtained and provided in this thesis work.

III. In the present thesis work the safety potential of **coupled hybrid systems** (DEN concept) was investigated in detail. Our analysis demonstrated that this concept was able to improve considerably the inherent safety potential of a nuclear reactor under condition that the system has favorable negative feedbacks. With the help of a simple mathematical model, describing the coupling of the subcritical core and of the external neutron source, we demonstrated that the latter may be treated as a supplementary group of delayed neutrons. As shown, this similarity between "natural" and "artificial" delayed neutrons is not absolute: some new opportunities arise and they have to be taken into account when the kinetics of the coupled hybrid system is considered. The modified inhour equation, which takes into account the ability to modify source reactivity, was deduced and an analysis of its roots was performed. An asymptotic reactor period, in the case of source performance variation, was obtained as a solution of this modified inhour equation. The above analysis allows one to conclude that the kinetic response of the coupled hybrid system to a variation of source performance is intrinsically different if compared to a variation of core reactivity. This is true in particular when large (if compared with the effective fraction of delayed neutrons) source equivalent of reactivity is introduced. Namely, there is no analogue of prompt criticality, accompanied by drastic decrease of the reactor period. These results allow us to give a practical recommendation: it is preferable (from the point of view of reactor kinetics) to have reactivity reserves in the form of the source reactivity. In this case an erroneous insertion of these reserves will not lead to drastic decrease of the reactor period.

IV. The synthesis of above results yielded in a new concept of coupled "accelerator - subcritical blanket" hybrid, incorporating the observed intrinsic safety-related advantages of both independent-source (ADS) and coupled-source (DEN) systems. This concept, named as the **DENNY system** (Delayed Enhanced Neutronics with Non-linear neutron Yield), is based on the particularity of the neutron production, forming a quasi-linear dependence between energy production in the core and the external neutron yield in the spallation target. This particular dependence provides an auto-regulating behavior of the ensemble "accelerator – subcritical core". The proposed system has the kinetics of a critical system with artificial group of delayed neutrons. In addition, its external neutron production contains the supplementary feedback, stabilizing the installation power in its nominal state. This effect can be compared to the Doppler feedback effect but for the external source neutrons. Similarly as the Doppler-effect, it is intrinsic. Finally, our qualitative estimates showed that the implementation of this concept could compensate eventual feedback degradation.

V. The preliminary safety analysis of the innovative molten salt fast reactor (REBUS-3700) was carried out. Principal concerns for the reactor safety were indicated and an exhaustive analysis of phenomena, leading to the reactivity variation in the reactor core, was done. It was demonstrated that at the equilibrium the magnitude of the excess reactivity may be decreased down to the value, comparable with the effective fraction of delayed neutrons. The safety analysis was performed in the framework of the so-called reactivity balance method, i.e. the quasi-static analysis of dependences of core power and core temperature on external parameters (e.g. inserted reactivity, fuel flow etc.) during unprotected transients. These studies demonstrated an excellent resistance of REBUS-3700 to majority of single and combined unprotected events, except unprotected overcooling. This latter can result in solidification of the primary salt, and has to be considered as the most menacing; for that reason some special measures are necessary in the reactor design to prevent the overcooling of the primary salt. In the context of obtained results it was concluded that this reactor does not require any support of the external neutron source to improve its safety characteristics.

VI. The results of our analysis have demonstrated that in many cases the requirements concerning parameters of the external neutron source may be modified when compared to these ones in the “traditional” ADS concepts. This opens a perspective to alternative solutions for external neutron source realization. In this thesis work the potential of two candidates: (i) thermonuclear installation and (ii) photoneutron source were re-examined. As demonstrated, the photoneutron-option should not be neglected in the case when a design of a hybrid-demonstrator is planned. This solution would be certainly a cheaper and more reliable option if compared to a proton driven external neutron source. On the other hand, it is clear that for industrial hybrids much more external neutrons are needed than it is achieved using electron accelerators. The external neutron sources, based on fusion reactions, could be considered as a promising alternative to the spallation-type neutron sources taking into account the technical feasibility and economics of the near future fusion installations.

Finally we add that in general, the subcriticality, when applied in a proper way, becomes a valuable tool for safety enhancement. At the same time, our studies also demonstrated that a more general approach still needs to be developed. This global approach must explore fundamentally a new feature of subcritical systems - the possibility to control reactor power by means of the external neutron source, including all issues following this possibility. Indeed, one needs to demonstrate not only what kind of control, i.e. the traditional (via reactivity) or novel (via intensity of the external neutron source), is more advantageous from the safety point of view; the economics and feasibility should be equally analysed in this respect (e.g., decreased requirements for the external neutron source intensity). In addition, it is obvious, that this strategy will depend on the type of hybrid system considered (coupled source or independent source). In this context, our preliminary considerations permit to expect that DENNY-concept provides some quite promising features in this regard, which certainly should be explored in more detail. The second direction of the future studies might be the identification of the innovative cores, permitting to reach, with a help of subcriticality, the inherent safety.

Summary of principal findings of this thesis work:

- For the first time an inter-comparison of the deterministic safety potential of coupled- (DEN) and independent-source (ADS) systems, as well as of the corresponding critical reactors, is carried out; the intrinsic differences of system behaviour during unprotected events are demonstrated.
- It is shown that in the case of the independent-source system (e.g., ADS) the subcriticality may compensate to some extent the degradation of the core feedback effects in terms of asymptotic power after excess reactivity insertion or/and in terms of the significantly expanded grace time.
- The dynamics and the kinetics of the coupled subcritical systems (DEN) are investigated in detail. It is demonstrated that this concept has many novel attractive features, which allows improving considerably the nuclear reactor safety.
- In order to overcome the identified inherent drawbacks of the ADS and ACS concepts a new promising approach to realize a coupled hybrid system with a manifest improvement of the inherent safety potential (DENNY concept) is proposed and analyzed.

– A preliminary deterministic safety study of the novel fast reactor concept (REBUS-3700) is carried out with a goal not only to characterize its safety potential, but to “condition” from the very beginning its design and operation principles in order to meet the inherent safety requirements. It is shown that REBUS-3700 has an excellent resistance to majority of single and combined unprotected events.

The most important results presented in this thesis work have appeared in the following publications:

I. Bokov, P.M., Source reactivity as an extra kinetic characteristic of coupled-source subcritical systems. Annals of Nuclear Energy, 32(8) PP. 795-811 (2005).

II. Bokov, P.M., Ridikas, D., Slessarev, I.S., Köberl, O., On Core Subcriticality as a Tool of Safety Enhancement. Accepted for publication in Nucl. Sci. Eng (to appear in 2005).

III. Slessarev, I., Bokov P., Deterministic Safety Study of Advanced Molten Salt Nuclear Systems for Prospective Nuclear Power // Nucl. Eng. Des. 231(1) pp. 67-88 (2004).

IV. Bokov, P., Slessarev, I., Ridikas D., Procédé d'amélioration de la sûreté des systèmes nucléaires hybrides couples, et dispositif mettant en œuvre ce procédé. France. Brevet français. FR 2 856 837. 2004 -12- 31 (BOPI 2004 - 53).

V. Bokov, P.M., Ridikas, D. Slessarev, I.S., “On the supplementary feedback effect specific for accelerator coupled systems (ACS)”. Proc. of the 4th OECD/NEA Workshop on the Utilization and Reliability of High Power Proton Accelerators, KAERI, Daejeon, Korea, 16-19 May 2004.

VI. Bokov, P., Ridikas, D., Slessarev, I., Subcriticality Role for Safety Enhancement of Advanced Molten Salt Systems. Proc. of the Int. Conf. GLOBAL’03, ANS, New Orleans, US (Nov. 16-20, 2003).

VII. Slessarev I.S., Bokov P.M., On potential of thermo-nuclear fusion as a candidate for external neutron source in hybrid systems (applied to the “WISE” concept) // Annals of Nuclear Energy, 30, pp. 1691-1698 (2003).

VIII. Bokov, P.M., Slessarev, I.S., Core subcriticality as effective means of safety enhancement: implementation for the molten salt hybrid systems. Proc. of the n_TOF Winter School on Astrophysics, ADS, and First Results, 24-28 February 2003, Les Houches (France) p.125-127.

IX. Mourogov, A., Bokov, P.M., « Potentialités du Réacteur à Neutrons Rapides Fonctionnant avec un Combustible aux Sels Fondus : REBUS-3700 ». Note Technique, HI-27/04/012/A, EDF, Clamart (2004).

X. Ridikas, D., Bokov, P., Giacri, M.-L., “Potential Applications of Photonuclear Processes: Renewed Interest in Electron Driven Systems”. – Proc. of the Int. Conf. on AccApp’03, 1-5 June 2003, San Diego, California, USA.

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9. Appendix. Solution of the kinetic equation for coupled hybrid system in one-group approximation

9.1. Inhour equation for the coupled hybrid system

As well known, an asymptotic period Θ of the reactor kinetic response to reactivity perturbation is described by the characteristic equation, called the inhour equation or Nordheim equation (Refs. Hetrick, 1971; Ash, 1979; Rozon, 1992). For the mathematical model of the coupled hybrid system in the form of Eqs. (3.15), the modified inhour equation may be written as a characteristic equation with respect to the variable s (Bokov, 2005):

$$\Delta\rho_{ext} + \frac{\varepsilon^{(r)}\lambda^{+}\beta^{+}}{\left(s+\lambda^{+}\right)} = s\left[\Lambda + \sum_{i=1}^{N_g}\frac{\beta_i}{\left(s+\lambda_i\right)} + \frac{\beta^{+}}{\left(s+\lambda^{+}\right)}\right]. \tag{A.1}$$

It differs from the “ordinary” inhour equation for a critical reactor by two extra terms (the last terms on the left- and on the right-hand sides of the equation). Moreover, the structure of the equation has also changed. The last term on the right-hand side of this equation does not depend on $\varepsilon^{(r)}$. It represents, as expected, the supplementary group of delayed neutrons. In other words, at $\varepsilon^{(r)} = 0$, Eq. (A.1) collapses to an eventual inhour equation but with an extra group of delayed neutrons. On the contrary, in the “ordinary” inhour equation there is no analogue for the term $\varepsilon^{(r)}\lambda^{+}\beta^{+}/\left(s+\lambda^{+}\right)$. Hence, one can suppose, that its solution will differ not only quantitatively but also qualitatively.

An analytical study of this modified inhour equation would be relatively cumbersome. Nevertheless some preliminary remarks concerning its solution may be made. Rearrangement of Eq. (14a) in order to bring it into a form more appropriated for subsequent analysis gives:

$$s\left[\Lambda + \sum_{i=1}^{N_g}\frac{\beta_i}{\left(s+\lambda_i\right)} + \frac{\beta^{+}}{\left(s+\lambda^{+}\right)}\left(1+\varepsilon^{(r)}\right)\right] - \left(\Delta\rho_{ext} + \varepsilon^{(r)}\beta^{+}\right) = 0\,. \tag{A.2}$$

The signs of the roots of the above characteristic equation give important information about the asymptotic behavior of the solution of Eq. (3.15). From the Descartes Rule of Signs (Meserve, 1982), it follows that if condition $\left(\Delta\rho + \varepsilon^{(r)}\beta^{+}\right) < 0$ is fulfilled, then the characteristic equation [(A.1)-(A.2)] has no *positive* real roots. This result is expected, as it follows from the physical meaning of the parameter $\varepsilon^{(r)}$, namely perturbation of the source reactivity. Hence the condition $\left(\Delta\rho + \varepsilon^{(r)}\beta^{+}\right) < 0$ signifies that in accordance with Eq. (3.8) one obtains that $m_{eff} < 1$ and the chain reaction in the system has to decay with time.

9.2. Approximation of one group of delayed neutrons

9.2.1. The one-group kinetic equation for coupled hybrid system

A simple analytical solution of Eq. (3.15) may be obtained in the one-group approximation for delayed neutrons. Moving to the effective concentration W of an "average" emitter of delayed neutrons with total fraction $\beta^{(\Sigma)}$ and effective decay constant $\lambda^{(\Sigma)}$, one obtains the coupled system of kinetic equations:

$$\begin{cases} \dfrac{d}{dt}P = \dfrac{\Delta\rho - \beta^{(\Sigma)}}{\Lambda}P + \left(1 + \tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}W; \\ \dfrac{d}{dt}W = \dfrac{\beta^{(\Sigma)}}{\Lambda}P - \lambda^{(\Sigma)}W. \end{cases} \tag{A.3}$$

Note that in Eqs. (A.3) the source variation term is renormalized: $\tilde{\varepsilon}^{(r)} = \varepsilon^{(r)}\left(\beta^{+}/\overline{\beta}\right)$. As was noted in Section 3.3, β^{+} and λ^{+} are free parameters and their values may be optimized with the objective of:

(i) increasing the margin to prompt criticality;

(ii) slowing down eventual transients by increasing the mean neutron life time

$$\ell_{mean} = \ell\left(1 - \beta^{+} - \sum_{i=1}^{N_g}\beta_i\right) + \frac{\beta^{+}}{\lambda^{+}} + \sum_{i=1}^{N_g}\frac{\beta_i}{\lambda_i}. \tag{A.4}$$

For these purposes β^{+} and $\left(\lambda^{+}\right)^{-1}$ have to be increased as much as possible, and in most practical situations the artificial group (term β^{+}/λ^{+}) prevail over natural groups of delayed neutrons. Consequently, in this context one may neglect the contribution of the natural delayed neutrons and one may take:

$$\beta^{(\Sigma)} = \beta^{+} + \sum_{i=1}^{N_g}\beta_i \approx \beta^{+},\ \lambda^{(\Sigma)} = \beta^{(\Sigma)}\left[\frac{\beta^{+}}{\lambda^{+}} + \sum_{i=1}^{N_g}\frac{\beta_i}{\lambda_i^{-1}}\right]^{-1} \approx \lambda^{+},\ W \approx W^{+} \text{ and } \tilde{\varepsilon}^{(r)} \approx \varepsilon^{(r)}. \tag{A.5}$$

Under these conditions the modified inhour formula Eq. (A.2) reduces after some rearrangement to the quadratic equation:

$$\Lambda s^2 + \left(\Lambda\lambda^{(\Sigma)} + \beta^{(\Sigma)} - \delta\rho\right)s - \lambda^{(\Sigma)}\left(\delta\rho + \tilde{\varepsilon}^{(r)}\beta^{(\Sigma)}\right) = 0 \tag{A.6}$$

9.2.2. Properties of the roots of the modified inhour equation in the one-group approximation

Before writing the solution of Eq. (A.6), we analyze in detail the properties of its roots. Representing for convenience Eq. (A.6) in the form: $a_2 s^2 + a_1 s + a_0 = 0$ and taking into account that $a_2 = \Lambda > 0$ one obtains according to the Descartes sign rule:

(i) If $\mathrm{sgn}(a_1) = 1$ and $\mathrm{sgn}(a_0) = 1$, the above equation has no *positive* roots. The physical meaning of these conditions is as follows: the core is subcritical on prompt neutrons

($\Delta\rho < \beta^{(\Sigma)} + \Lambda\lambda^{(\Sigma)}$) and the total neutron multiplication factor is less than "one" (i.e. $m_{eff} = \left(1 - \delta\rho - \tilde{\varepsilon}^{(r)}\beta^{(\Sigma)}\right)^{-1} < 1$) correspondingly.

(ii) If $\mathrm{sgn}(a_1) = 1$ and $\mathrm{sgn}(a_0) = -1$ then Eq. (A.6) has one *positive* root and one *negative* root. In this case the reactor core remains subcritical on prompt neutrons, but the overall neutron multiplication factor m_{eff} is greater than "one".

(iii) If $\mathrm{sgn}(a_1) = -1$ then the condition $\mathrm{sgn}(a_0) = -1$ is fulfilled automatically, and Eq. (A.6) has one *positive* root and one *negative* root. Indeed, as follows from the definition of $\varepsilon^{(r)}$: $-1 \le \varepsilon^{(r)} \le 1 - f_0^{-1}$ and $\tilde{\varepsilon}^{(r)}\beta^{(\Sigma)} \ge -\beta^{+}$. Therefore condition $\mathrm{sgn}(a_1) = -1$ signifies that $\Delta\rho > \beta^{(\Sigma)} + \Lambda\lambda^{(\Sigma)}$, what leads to $\Delta\rho + \tilde{\varepsilon}^{(r)}\beta^{(\Sigma)} > \beta^{(\Sigma)}\left(1 + \tilde{\varepsilon}^{(r)}\right) + \Lambda\lambda^{(\Sigma)} > 0$, i.e. to the following condition $\mathrm{sgn}(a_0) = -1$.

A straightforward calculation of the determinant $\mathcal{D} = a_1^2 - 4a_2a_0$ of Eq. (A.6) proves that $\mathcal{D} > 0$ if $\tilde{\varepsilon}^{(r)} \ge -1$, i.e. for the all possible values of parameter $\varepsilon^{(r)}$. Consequently, Eq. (A.6) has neither complex, nor double roots.

In brief, case (i) corresponds to reactor power decreasing with time, whereas cases (ii) and (iii) correspond to solutions increasing with time.

9.2.3. The solution of kinetic equation for coupled hybrid system in one-group approximation

Let us use the following notation for convenience:

$$q \equiv \beta^{(\Sigma)} / \Lambda . \tag{A.7}$$

With this notation and supposing that there is no in-core reactivity variation (i.e. $\Delta\rho = 0$) Eqs. (A.3) may be rewritten in the following way:

$$\begin{cases} \dfrac{d}{dt} P(t) = -qP(t) + \left(1 + \tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)} W(t), \\ \dfrac{d}{dt} W(t) = qP(t) - \lambda^{(\Sigma)} W(t). \end{cases} \tag{A.8}$$

To solve the above system of equations it is useful to apply the Laplace Transformation method. We will denote the Laplace Transform for some arbitrary function of time $F(t)$ as follows: $\hat{F}(s) = \mathcal{L}[F(t)]$. Then, the Laplace transform of Eq. (A.8) yields in

$$\begin{cases} s\hat{P}(s) - P_0 = -q\hat{P}(s) + \left(1 + \tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}\hat{W}(s), \\ s\hat{W}(s) - W_0 = q\hat{P}(s) - \lambda^{(\Sigma)}\hat{W}(s), \end{cases} \tag{A.9}$$

where $P_0 = P(t = 0)$ and $W_0 = W(t = t_0)$. These equations may be solved algebraically for $\hat{P}(s)$ with some rearrangement

$$\hat{P}(s) = \frac{\left(s + \lambda^{(\Sigma)}\right)P_0 + \left(1 + \tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}W_0}{\left(s + \lambda^{(\Sigma)}\right)(s + q) - \left(1 + \tilde{\varepsilon}^{(r)}\right)q\lambda^{(\Sigma)}} = \frac{\left(s + \lambda^{(\Sigma)}\right)P_0 + \left(1 + \tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}W_0}{\left(s - \omega_1\right)\left(s - \omega_2\right)}, \tag{A.10}$$

where

$$\omega_{1,2} = \frac{-\left(q+\lambda^{(\Sigma)}\right) \pm \sqrt{\left(q+\lambda^{(\Sigma)}\right)^2 + 4\tilde{\varepsilon}^{(r)}\lambda^{(\Sigma)}q}}{2} \tag{A.11}$$

are the roots of the characteristic equation (A.6), which, applying the notations Eq. (A.7), may be rewritten in the following way

$$\left(s+\lambda^{(\Sigma)}\right)(s+q) - \left(1+\tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}q = 0\,. \tag{A.12}$$

Making use of the notation $u = \lambda^{(\Sigma)}/q$ and $R\left(u,\tilde{\varepsilon}^{(r)}\right) = \sqrt{(1+u)^2 + 4\tilde{\varepsilon}^{(r)}u}$, one can rewrite above expression Eq. (A.11) in the following way

$$\omega_{1,2} = \frac{\lambda^{(\Sigma)}}{2u}\left[-(1+u) \pm R\left(u,\tilde{\varepsilon}^{(r)}\right)\right]. \tag{A.13}$$

Applying the inverse Laplace Transformation $P(t) = \mathcal{L}^{-1}\left[\hat{P}(s)\right]$ to Eq. (A.9) one obtains the solution of Eqs. (A.8):

$$P(t) = \frac{P_0}{\left(\omega_1 - \omega_2\right)}\sum_{j=1}^{2}(-1)^{j-1}\left[\left(\omega_j + \lambda^{(\Sigma)}\right) + \left(1+\tilde{\varepsilon}^{(r)}\right)\lambda^{(\Sigma)}\frac{W_0}{P_0}\right]\exp\left(\omega_j t\right), \tag{A.14}$$

where ω_j are given by expression Eq. (A.13).

If at $t=0$ the system was in equilibrium, then the initial condition yields in $W_0 = P_0/u$. After some rearrangement, the solution of Eqs. (A.8) can be written in the form:

$$P(t) = P_0\sum_{j=1}^{2}\Psi_j\left(u,\tilde{\varepsilon}^{(r)}\right)\exp\left(\omega_j t\right), \tag{A.15}$$

where the coefficients Ψ_j are given by

$$\Psi_j\left(u,\tilde{\varepsilon}^{(r)}\right) = \frac{1}{2}\sum_{j=1}^{2}(-1)^{j-1}\left[(-1)^{j-1} + \frac{1+u+2\tilde{\varepsilon}^{(r)}}{R\left(u,\tilde{\varepsilon}^{(r)}\right)}\right]. \tag{A.16}$$

Analyse Comparative du Fonctionnement et de la Sûreté de Systèmes Sous-critiques et de Réacteurs Critiques Innovants

L'objectif de ce travail de thèse est d'examiner le rôle de la sous-criticité du cœur, en tant que moyen pour améliorer la sûreté des systèmes nucléaires innovants, notamment des réacteurs à sel fondu, dédiés à la production d'énergie et/ou à la transmutation/incinération des déchets nucléaires. La sûreté intrinsèque est considérée comme l'objectif ultime de cette amélioration. Une tentative d'appliquer une approche systématisée pour l'analyse de la contribution de la sous-criticitité au comportement intrinsèque des systèmes hybrides est effectuée. Les résultats de cette étude prouvent que la sous-criticité améliore bien la sûreté des réacteurs nucléaires, et même, dans certaines configurations, permet d'attendre la sûreté intrinsèque. Dans tous les cas, un choix approprié du niveau de sous-criticité rend les transitoires plus lents et monotones. Il est montré que le point faible pour des systèmes hybrides avec une source indépendante de neutrons sont les transitoires thermo-hydrauliques non protégés tandis que pour des hybrides avec des sources couplées ce sont les transitoires de réactivité. Pour surmonter les inconvénients intrinsèques à ces deux types de systèmes hybrides, un nouveau principe de réalisation des systèmes hybrides couplés est proposé (concept DENNY). De plus, des approches, qui permettent de remédier à certains problèmes de sûreté, sont proposées. Une analyse préliminaire du potentiel de sûreté intrinsèque pour un réacteur à sel fondu avec spectre rapide (concept REBUS) est effectuée. Enfin, le potentiel des sources alternatives de neutrons basées sur des réactions thermonucléaires et photo-nucléaires est examiné.

Mots clés : réacteur nucléaire, sûreté, systèmes hybrides, dynamique, cinétique, sel fondu, spallation, photo-nucléaire, thermonucléaire, feedback, neutrons retardés.

Comparative Analysis of Operation and Safety of Subcritical Nuclear Systems and Innovative Critical Reactors

The main goal of this thesis work is to investigate the role of core subcriticality for safety enhancement of advanced nuclear systems, in particular, molten salt reactors, devoted to both energy production and waste incineration/transmutation. The inherent safety is considered as ultimate goal of this safety improvement. An attempt to apply a systematic approach for the analysis of the subcriticality contribution to inherent properties of hybrid system was performed. The results of this research prove that in many cases the subcriticality may improve radically the safety characteristics of nuclear reactors, and in some configurations it helps to reach the "absolute" intrinsic safety. In any case, a proper choice of subcriticality level makes all analyzed transients considerably slower and monotonic. It was shown that the weakest point of the independent-source systems with respect to the intrinsic safety is thermo-hydraulic unprotected transients, while in the case of the coupled-source systems the excess reactivity/current insertion events remain a matter of concern. To overcome these inherent drawbacks a new principle of realization of a coupled sub-critical system (DENNY concept) is proposed. In addition, the ways to remedy some particular safety-related problems with the help of the core sub-criticality are demonstrated. A preliminary safety analysis of the fast-spectrum molten salt reactor (REBUS concept) is also carried out in this thesis work. Finally, the potential of the alternative (to spallation) neutron sources for application in hybrid systems is examined.

Key words: nuclear reactor, safety, hybrid systems, dynamics, kinetics, molten salt, spallation, photonuclear, thermonuclear, feedback, delayed neutrons.