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Sub-criticality Role for Safety Enhancement of Advanced Molten Salt Systems

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ABSTRACT – *In this work we investigate the influence of sub-criticality on the safety of molten salt reactors. A direct comparison between critical and sub-critical systems is done by simulating a number of unprotected transients. For this purpose different levels of sub-criticality are tried to improve system's safety and, at the same time, to define the intensity of the external neutron source. It is shown that a considerable expansion of the time interval of core vitality can be achieved even if a very low sub-criticality level (350-700 pcm) is applied. Consequently, the requirements for the external neutron source intensity are considerably decreased when compared to deeply sub-critical systems. In this context, the use of an electron accelerator instead of proton machine is investigated in detail including economical, physical and technical realization constraints.*

I. INTRODUCTION

In general, molten salt reactors (MSR) have an appropriate basis to achieve deterministic safety level in terms of low internal pressure, low fuel-coolant inflammability, small reactivity margin, 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 [1, 2]. Besides, a number of other physical parameters, 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 freezing, may raise some problems on the way to the deterministic safety. The insertion of some quantities of minor actinides (MA) in the core definitely would degrade the situation further.

In order to overcome the above difficulties one could possibly use sub-critical molten salt reactors. It is known that core sub-criticality can be helpful at least in two cases: either for neutronics enhancement of cores when neutron balance is too tight or for safety improvement purposes when feed-back effects or other physical parameters are degraded. There are multiple regimes of sub-critical system functioning: an external neutron source is independent on core power (so called Accelerator Driven System – ADS [3]), and an external neutron source is dependent (coupled to) on the neutron production (the power) in the core (so called Accelerator Coupled System – ACS [4] or more generally Delayed Enhanced Neutronics – DEN [5]). As it was shown in [5], DEN seems to be more flexible than ADS in the case of molten salt systems as long as their inherent safety is concerned.

This work aims determining the safety potentials of the sub-critical MSR coupled (DEN-type coupling) to the external neutron source (Hybrid Molten Salt Reactor – HMSR). A sensitivity of unprotected transients in the case of critical and sub-critical MSR is examined in a comparative way. Different levels of sub-criticality are tried in order to improve system's safety parameters and at the same time to define the intensity of the external neutron source needed to drive the system. Economical, physical and technical realization constraints are also discussed. The molten salt AMSTER-like core, proposed and analyzed in [1], is taken as a reference in our study.

II. DESCRIPTION OF THE ACTUAL WORK

In this work major neutronic and thermo-hydraulic parameters are derived from the reference critical AMSTER core, developed at the Electricité de France (EdF) in the framework of the Generation IV program [1]. The MSR core under consideration can be either critical or sub-critical. In the latter case a fixed fraction of the produced energy is used to feed the accelerator providing an external neutron source (due to DEN-type coupling [5] we will call the system HMSR-DEN).

A simplified point model of the core kinetics is adopted for safety analysis. The mathematical model includes also a description of the thermo-hydraulics of the circulated fuel as well as feedback effects in the core. Our model (with respect to previous studies on accelerator coupled system kinetics [4,5]) is completed by an improved description of coupling of the external neutron source with the sub-critical core.

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.

II.A. Major Parameters of the AMSTER-like-Core

Two different AMSTER-like systems have been chosen 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**. Tables I and II summarize feedback effects and delayed neutron characteristics in both cases. We note the major difference between two systems considered – negative and slightly positive total feedback in the case of uranium and thorium based cores respectively (see Table I).

TABLE I. Feedback effects in the reference cores [1].

coefficient, pcm/°C	support uranium	Support thorium
$\alpha_{Doppler}$	-3.01	- 2.4
$\alpha_{density}$	+ 1.45	+ 1.95
α_{salt}	**- 1.565**	**- 0.45**
$\alpha_{graphite}$	**- 1.2**	**+ 1.2**
α_{total}	**- 2.45**	**+ 0.75**

TABLE II. Delayed neutron fractions in the AMSTER-like cores with support-uranium (U) and support-thorium (Th) [1].

Group	λ_i, s^{-1}	β_i, pcm		β_i^*, pcm	
		U	Th	U	Th
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**

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

To quantify further the delayed neutrons 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 [6]:

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

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

II.B. Appearance of "Artificially Delayed Neutrons" in the Case of HMSR-DEN

As we have discussed above the delayed neutron fraction may be considerably decreased for MSR due to: a) fuel circulation, b) fuel load based on thorium-support, c) fuel load with MAs [1]. Below we explain how this degradation could be compensated in the case of a sub-critical MSR when its external neutron source is coupled directly to the core energy production (HMSR-DEN).

The way to improve the above parameter has been initially proposed by Gandini et al. [4]. In their work reactor core operates in sub-critical regime and a fraction f of total produced energy by the same installation is used to run an external neutron source. This fraction is chosen to produce enough neutrons to sustain critical state of neutron production in the core. As a result, the external neutron production in the core will be delayed by the time required for the "transportation" of fission energy (heat transfer) to a chosen neutron production mechanism (e.g., spallation process). Physical background is rather simple: an intermediate heat transfer process "hides" neutrons temporarily to recover them later. This allows increasing the neutron life-time artificially with a final goal to slow down dangerous transients. As a result, a new "artificial" group of delayed neutrons appears (with its fraction β^+ equal to the sub-criticality level of the system).

Gandini et al. [4] investigated different aspects of the ACS safety using "balance of reactivity" method proposed initially by Wade [7]. They described external source Q by a term, proportional to the reactor power P and delayed by a characteristic time τ, necessary for the heat transfer from reactor's core to the external neutron production mechanism, i.e. $Q(t) \sim P(t-\tau)$. It has to be noted, that temporal delay of neutron production (with respect to neutron multiplication in the core) is not the only advantage of such a system. As a matter of fact, *the artificial neutron production, caused by a single fission* *event at any time t, will be not only delayed by a characteristic time τ, but also distributed over time following this reaction*. In other words, the "delayed-argument" model by Gandini et al. [4] neglects the fact that the fission energy (i.e., time integral of power) is released and transferred from core to electric energy generator in the form of heat. This also means that the electrical energy production at any time (and consequently the intensity of an external neutron source) depends on the interior power history as well as on particularities of the heat transfer in the system which might be important for reactor kinetics.

In Appendix A we analyze this dependence in more detail by including explicitly the energy transfer model. For this purpose a simplified Newton model of the reactor heat-transfer [6] is employed. Below we summarize only the major findings related to the safety issues.

In brief, the artificial group of delayed neutrons mentioned above may provide some unique properties with respect to the safety potential enhancement:

- due to the "integral nature" of the heat dissipation (external source intensity depends on thermal energy accumulated in the core), even "peaked" significant perturbations of reactivity or thermo-hydraulic parameters do not cause neither power nor temperature dangerous oscillations [5];
- it places an ability controlling of time characteristics of transients by choosing-changing both effective "decay constant" and fraction of the supplementary group of delayed neutrons, i.e. the sub-criticality level of the system;
- it does not lead to an undesirable growth of reactivity during loss of flow events in systems with circulating fuels [5];
- it can be realized with relatively low economic penalties because it requires a small level of core sub-criticality, consequently rather weak intensity of the external neutron source.

Let us now estimate the order of magnitude of the effective "decay constant" λ^+ for the artificial group of delayed neutrons in the case of AMSTER-like system. Unfortunately, no parameters of the 2nd and 3rd circuit are provided for this system in Ref. [1]. Nevertheless, based on the 1st loop thermo-hydraulic parameters we can estimate the lower limit for λ^+. There are three the most important process of heat transfer in the 1st loop: molten salt circulation (~14 s), change of the graphite temperature (~10 s), and heat transfer in the heat exchanger (~6 s). Hence, the lower estimate for characteristic decay time $\left(\lambda^+\right)^{-1}$ is of order of ~15-20 s. This value is similar to the one group decay constant for precursors of delayed neutrons (see Section II.A.).

Despite the above advantages, a hybrid system might be difficult to construct due to a number of other reasons. Below we study in more detail some potential HMSR-DEN realization constraints.

III. REALIZATION RESTRICTIONS

In order to build a proposed hybrid system one should take into account all possible constraints which may have either scientific, or economical or technical realization nature. *Physical (scientific) constraint* is evident and consists of a physical possibility to create 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 a 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.

Below we will consider electrons or protons as potential incident particles to produce external neutrons. For more 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:

$$f = \varepsilon_{el}^{inp} / \varepsilon_{el}^{out} ,$$

where ε_{el}^{inp} is the electric energy consumed to produce and accelerate one incident particle, ε_{el}^{out} being the mean electric energy produced by the system per incident particle. Note, that both ε_{el}^{out} and ε_{el}^{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 $\varepsilon_{el}^{inp} = \varepsilon / \eta_a$ and $\varepsilon_{el}^{out} = \mu \eta_{el} \varepsilon$ we obtain

$$f = \frac{1}{\mu \eta_{el} \eta_a} .$$

In this notation ε is the incident particle energy, η_{el} - reactor electric efficiency; η_a - accelerator efficiency, μ - energy multiplication coefficient of a sub-critical core, given by

$$\mu = \left[\kappa + \frac{k_{eff}\varphi^* Y}{\left(1-k_{eff}\right)\nu} \frac{\varepsilon_{fis}}{\varepsilon} \right],$$

where κ is the fraction of the particle energy deposited in the system, ε_{fis} - energy released per fission, ν - mean number of fission neutrons, φ^* - source neutron importance, Y - mean neutron yield per incident particle, and k_{eff} - 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:

$$I_n^{ext} = \frac{P_{th}}{\varepsilon_{th}^{out}} = \frac{P_{th} Y}{\mu \varepsilon}.$$

In the case of the AMSTER-like core the following parameters were employed for the analysis: $P = 2500$ MW$_{th}$, $\eta_{el} = 0.44$ [1], $\varphi^* = 1.2$, $\varepsilon_{fis} = 200$ MeV, and $\kappa = 1$. A number of fission neutrons were evaluated from Ref. [8], 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 $\left(Y/\varepsilon\right)$ which one can consider approximately constant for incident particle energies higher beyond some threshold value [9]. Based on the simulations using the multi-particle transport code MCNPX [10], for electrons we obtain $\left(Y/\varepsilon\right)^{(e)} = 6\cdot 10^{-4}$ neutron/MeV/electron for a thick photonuclear target (^{238}U surrounded by ^{nat}Pb). Indeed, this value remains nearly constant for electron energies $\varepsilon^{(e)} > 150$ MeV. For a thick spallation target built of liquid Pb-Bi we obtain $\left(Y/\varepsilon\right)^{(p)} = 2.5\cdot 10^{-2}$ neutron/MeV/proton. Similarly like for electrons, this value is not changing for proton energies $\varepsilon^{(p)} > 1000$ MeV. Consequently, for the parameter μ we obtain $\mu^{(e)} = 1 + 5.760\cdot 10^{-2}/r_0$ and $\mu^{(p)} = 1 + 1.92/r_0$, where $r_0 \equiv \left(1-k_{eff}\right)/k_{eff}$ is the sub-criticality level.

The results of the above formulation are summarized in Fig. 1, 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 sub-criticality level (Fig. 1a).

On the other hand, in both cases nearly the same external neutron source intensity will be needed to produce the same output energy P_{th} (Fig. 1b). The small difference (compare solid and dashed curves) is due to different particle beam power deposited in the production target, which is much higher for electrons. Therefore, with electrons one would need smaller neutron source intensity (by ~15 %) to obtain the same total outlet power (which also includes the beam power deposited in the neutron production target). In the same Fig. 1b we also distinguished the industrial and prototype system requirements since different realization constraints might be applied for these two cases.

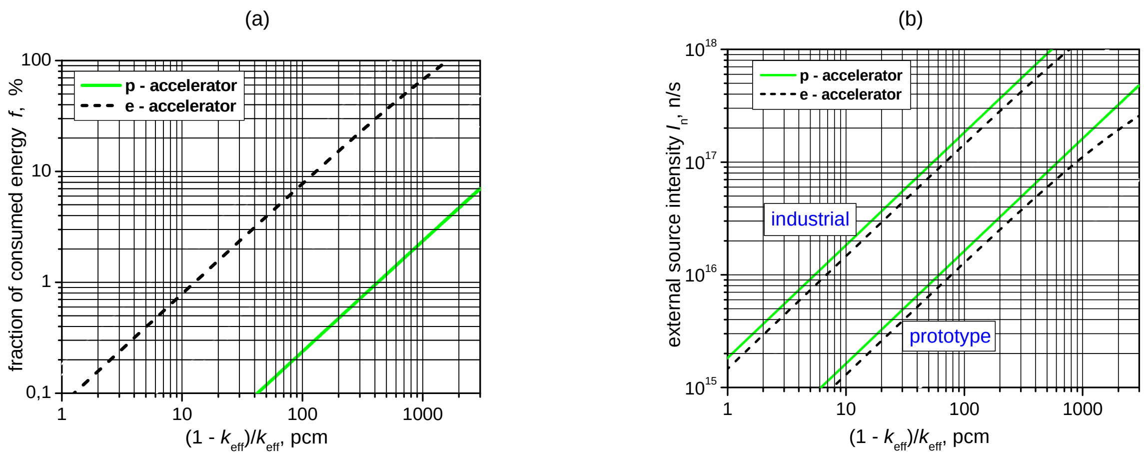


Fig. 1. Fraction of energy *f*, consumed by accelerator (a) and the intensity of external neutron source I_n (b) as a function of sub-criticality level. Proton and electron accelerator options are presented as solid and dashed lines correspondingly. Industrial and prototype options stand for 2500 MW$_{th}$ and for 200 MW$_{th}$ AMSTER-type sub-critical cores respectively.

In the case of the Hybrid Molten Salt Reactor (HMSR), compared to the critical MSR, the total system efficiency decreases from 44% down to the value: $\eta_{el}^{HMSR} = \eta_{el}^{MSR}\left(1-f\right)$. Let us suppose that the minimal acceptable efficiency of HMSR is the actual value of present PWR efficiency, i.e. ~33%. Therefore, in terms of economical restrictions the maximal available fraction of the total produced energy by HMSR is $f_{\max} = 1-\eta_{el}^{PWR}/\eta_{el}^{MSR} = 25\%$. 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 in Ref. [11], with today's electron machine and production target technology one could possibly reach neutron source intensities up to ~$2{\cdot}10^{17}$ 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 ~$8{\cdot}10^{18}$ 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 [10], 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).

According to the technical realization and economical criteria as discussed above, for example, with electrons one could reach the sub-criticality level of ~100 pcm in the case of the industrial HMSR and ~2000 pcm in the case of the prototype HMSR considered in this study (see Fig. 1 for details). The use of protons, 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 sub-criticality level as high as 3000 pcm. So how much sub-criticality one actually needs that system's safety is considerably enhanced against unprotected transients? The following section will address this question in detail.

IV. RESULTS

IV.A. Reactivity Variations in the Reference MSR

A complete safety analysis requires simulating the multitude of all accidental situations permitted by physical laws. In this paper we 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 sub-criticality level, let us analyze possible causes of reactivity insertion. The data collected in the Table III are evaluated-extracted from Refs. [1, 2, 8]. 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. [2] for details), and therefore no control rod reactivity reserve has to be anticipated for their compensation.

TABLE III. Physical processes leading to the reactivity variation and corresponding reactivity values in the reference cores [1, 2, 8].

reason for reactivity variation	**support U**	**support Th**
reactivities, compensated by control rods ("fast reactivities"), pcm		
homogenous core heating 450°C to 562°C 562°C to 705°C 705°C to 1300°C	 - 275[a] - 350 [a] –	 + 84[b] **+ 107** + 446[a,b]
Total	- 625 [a]	+191 (+ 637[b])
^{135}Xe poisoning [8]	**- 32**	**- 32**
fuel stop	**+ 110**	**+ 94**
decompression [8]	**+ 100**	**+ 100**
total ("fast reactivities") nominal regime start-up regime maximum	 242 409 867	 333 226 943
reactivities, compensated by adjustment of fuel composition ("slow reactivities"), pcm		
$^{239}Np/^{233}Pa$ – effect [2,5]	+ 50	+ 1600 (2.5 pcm/h)
fluctuation of fissile isotopes concentration [8]	+ 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 can be introduced for account of positive feed-back 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.

It is known, 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 $\Delta\rho_{LOF} = \left(\beta - \beta^{*}\right)$ due to all precursors decay in the core. In this work it was 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 UTOP accident alone we do not include $\Delta\rho_{LOF}$ into $\Delta\rho_{TOP}$. However, when we simulate superposition of UTOP and ULOF, both $\Delta\rho_{LOF}$ and $\Delta\rho_{TOP}$ are taken into account, what corresponds to the situation when potentially maximal reactivity can be inserted. The particularity of LOHS-accident in ACS (DEN) is that the loss of heat transfer to the energy generation device switches off external neutron source, what is favorable for system's inherent safety in the case of this accident.

IV.B. Simulation of Unprotected Transients

Here we provide just a brief description of the model. A complete description of our mathematical formulation used for accident simulations can be found in Ref. [5]. 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 2nd loop and energy production system are defined by now in the AMSTER project [1], there are no 2nd and 3rd loops included in our model. It is supposed that electric energy is produced immediately after the 1st loop, what results in the underestimation of the sub-critical system safety. An environment temperature of 450°C is chosen to avoid eventual salt freezing. 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 in Ref. [1].

In our work we carry out a parametric study of sub-criticality impact on the MSR safety. Four different levels of sub-criticality r_0 have been chosen for simulation: (1) 100 pcm being the maximal limit for industrial HMSR in the case of electron accelerator; (2) 350 pcm – a level, corresponding to β-compensation up to the value typical for industrial nuclear reactors (600-700 pcm); (3) 700 pcm; (4) 1050 pcm - being close to the maximal limit for prototype HMSR in the case of electron accelerator.

In the **TRU incinerator system (support uranium)** maximal reactivity insertion $\Delta\rho_{TOP} = 132$ pcm is simulated according to Table III. Obtained results are summarized in Table IV 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:

(a) all accidents in all systems (including critical one with $r_0 = 0$ pcm) do not lead to dangerous temperature growth;
(b) nevertheless, added sub-criticality improves the behavior of system during transient;
(c) 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 sub-criticality (e.g., see Fig.2);
(d) artificial feedback caused by the core-accelerator coupling changes system's behavior during complex accidents; e.g., contrary to a critical system, in the case of a sub-critical system (1050 pcm) the superposition of ULOF and UTOP accidents is less dangerous than single UTOP.

TABLE IV. Maximal temperature reached in the support uraniumTRU 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)

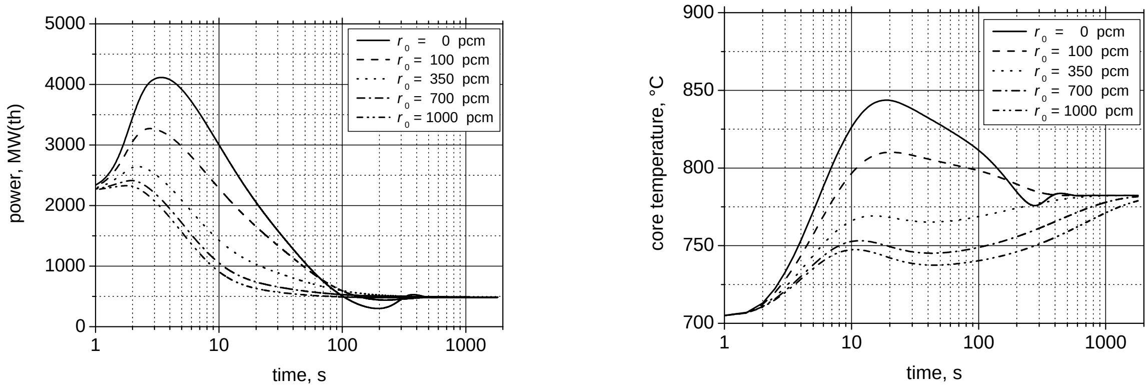


Fig. 2. Simulation of complex accident in the support-uranium TRU incinerator system: the UTOP accident (insertion of $\Delta\rho = 132\,\text{pcm}$) with the simultaneous ULOF accident (reduce of salt flow by 90 % in 10 s).

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

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

The maximal TOP-reactivity insertion is $\Delta\rho = 239$ pcm (see Table III), 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 sub-criticality role for safety enhancement of this particular system.

As in the previous case, we chose sub-criticality 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 behaviour in a so-called "source dominated domain" or deep sub-criticality regime. Results of our simulation are presented in Table V and Fig. 3. Below we summarize our major findings:

(a) due to the positive total feedback unprotected transients in all cases lead to salt temperature raise up to the boiling temperature T = 1300°C, what is considered as a disintegration criterion of the system [1];

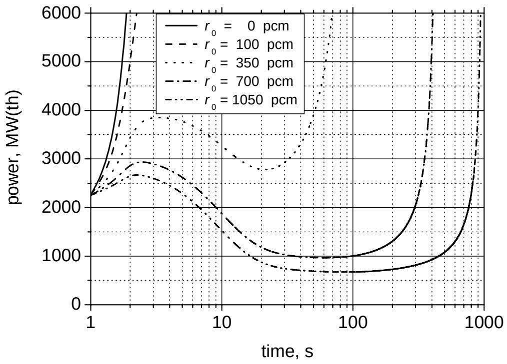
(b) added sub-criticality increases core vitality period (time from the beginning of an accident to salt boiling): ~10 s per 100 pcm in the "feedback dominated" region up to ~25 s per 100 pcm in the "source-dominated" region;

(c) similarly like for the support-uranium system, sub-criticality level higher than ~100 pcm results in UTOP and ULOF accident superposition less dangerous than UTOP taken alone.

TABLE V. Time necessary to reach the boiling temperature (T_c = 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

We found it interesting to compare directly the influence of sub-criticality with an improved feedback effect in the case of critical system. The question can be formulated as follows: what sub-criticality level would give the same result for system vitality persistence as feedback effect improvement. For this reason a set of supplementary simulation were carried out.



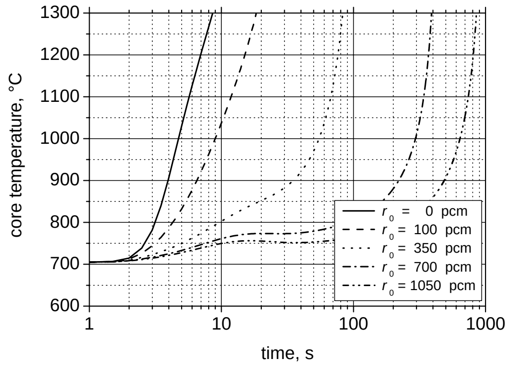


Fig. 3. Simulation of complex accident in the support-thorium self generator system: the UTOP accident (insertion of $\Delta\rho = 239\,\text{pcm}$) followed by the ULOF accident (reduce of salt flow by 90 % in 10 s).

We start our analysis by comparing two ways of feedback optimisation, i.e. of (a) graphite $\alpha_{graphite}$, and of (b) salt α_{salt}, giving the same α_{total}. We note separately that in our study we use linear model of feed-back, 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 favourable for safety enhancement. The simulation of transients in critical system with modified (ameliorated) feedbacks showed that it is preferable to optimise 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 behaviour 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 VI we present system vitality time for different unprotected transients. By comparing these results with the ones, presented in Table V, and with the help of some interpolations, we conclude that a) 10%-Doppler effect amelioration would be comparable with ~150 pcm core sub-criticality; b) 20% salt feedback effect improvement would be comparable with ~320 pcm core sub-criticality.

TABLE VI. Same as Table V but 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

III. CONCLUSIONS

This work aimed in determining the safety potentials of the sub-critical MSR (AMSTER-like core) coupled (DEN-type coupling) to the external neutron source (Hybrid Molten Salt Reactor – HMSR). 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, with a major difference between them being 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 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 has got 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 = 1300°C, what was considered as a disintegration criterion of the system. Added sub-criticality increased core vitality period from ~10 s per 100 pcm in the “feedback dominated” region up to ~25 s per 100 pcm in the “source-dominated” region. In terms of absolute value, a considerable expansion of the time interval (by a factor 10) of core vitality was achieved even if a very low sub-criticality level (350-700 pcm) is applied.

The above conclusions suggest that the requirements for the external neutron source intensity are considerably decreased when compared to deeply sub-critical systems. In this context, the use of an electron accelerator instead of proton machine was investigated in detail including economical, physical and technical realization constraints. We found that with electrons one could reach the sub-criticality level of ~100 pcm in the case of an industrial HMSR and ~2000 pcm in the case of a prototype HMSR considered in this study. The latter result should not be neglected in the case when a design of a demonstrator HMSR is planned. This solution would be certainly a cheaper option if compared to a proton driven external neutron source.

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APPENDIX A

Let us consider the simplest point model of reactor kinetics. Equations of point-wise kinetics can be presented in the classical form:

$$\begin{aligned}
\frac{dP(t)}{dt} &= \frac{\rho(t)-\beta}{\Lambda}P(t)+\sum_{i=1}^{6}\lambda_i C_i(t)+Q(t),\\
\frac{dC_i(t)}{dt} &= \frac{\beta_i}{\Lambda}P(t)-\lambda_i C_i(t).
\end{aligned} \qquad \text{(A.1)}$$

where $\rho=(k_{\text{eff}}-1)/k_{\text{eff}}$ is the reactivity of core, P is the power, C_i is the concentration of delayed neutron precursors of i^{th}-group with the fraction β_i and the decay constant λ_i ; $\beta=\sum_{i=1}^{6}\beta_i$ is the total fraction of delayed neutrons; Λ is the neutron life time; term $Q(t)$ describes external source of neutrons.

Let us consider that intensity of the external source is proportional to the output of the system energy. To take it into account, one needs to describe the heat energy “dissipation” (sink). The simplest one-point heat transfer scheme of this sink can be presented via Newton cooling [6]:

$$C\frac{dT_c(t)}{dt}=P(t)-K\theta(t), \qquad \text{(A.2)}$$

where T_{c} the average core temperature, C is the core heat capacity. The second term in the right part of (A.2) presents energy which the core looses per time unit and K is the corresponding coefficient, while $\theta\equiv T_{\text{c}}-T_{\text{k}}$ is the “core heating” or the difference between core and environment temperatures.

We suppose that core output energy serves immediately for energy production and, consequently, for feeding external source. For neutron self-consistency, i.e. compatibility with a core "criticality", the external neutron source has to be equal to

$$Q(t) = r_0 K \theta(t) / \Lambda ,$$

where value $r_0 = \left|\rho^{\text{nominal}}\right|$ is the nominal sub-criticality level.

With the notations: $C^+ \equiv r_0 c_p M \theta / \Lambda$, $r_0 \equiv \beta^+$, $\lambda^+ \equiv K / C$, one can get the following system of equations:

$$\begin{cases} \dfrac{dP}{dt} = -\dfrac{\left(\beta + \beta^+\right)}{\Lambda} P + \displaystyle\sum_{i=1}^{6} \lambda_i C_i + \lambda^+ C^+, \\ \dfrac{dC_i}{dt} = \dfrac{\beta_i}{\Lambda} P - \lambda_i C_i, \\ \dfrac{dC^+}{dt} = \dfrac{\beta^+}{\Lambda} P - \lambda^+ C^+. \end{cases} \qquad \text{(A.3)}$$

Comparing (A.3) with (A.1), one notes that the "coupled" neutron source leads to the appearance of the complementary group of delayed neutrons with the precursor concentration C^+ and with the "decay" constant λ^+ playing the role of the parameter which describes the rate of heat "dissipation". Besides, in absolute value the fraction β^+ is equal to the sub-criticality level: $\beta^+ = r_0$. It means that the system will have somewhat larger total fraction of all delayed neutrons $\beta_{\text{eff}} = \beta + \beta^+$ which, together with λ_i and λ^+ , defines now the "characteristic" transient time. The effective neutron life time Λ_{eff} is defined now by the expression:

$$\Lambda_{\text{eff}} = \Lambda + \frac{\beta^+}{\lambda^+} + \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} \approx \frac{\beta^+}{\lambda^+} + \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} . \qquad \text{(A.4)}$$

This resemblance between the "artificial" and natural groups of delayed neutrons is not casual – the system is forethought in such a manner to simulate the concentration evolution of delayed neutron precursors. In reality λ^+ will be difficult to assess analytically. Moreover, the model presented above is not adequate for circulating-fuel system. However, λ^+ should be *close to the inverse characteristic time of heat exchange* in a reactor.