# **Target Accuracy Requirements and an evidence-based background for MSFR safety assessment**

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**Abstract**

From the very beginning of nuclear era the mission of fast neutron reactors (FRs) has been conceived in a double-functionality to combine the generation of power and new fuel trying to make nuclear power resources practically inexhaustible.

Since then, due to conjuncture changing, the fast reactor became to be considered as just an element of the global fuel cycle playing a traditional role – being a robust source of energy and of artificial fissile materials – as well as complementary ones – burning of minor actinides and recycled plutonium.

In the majority of the fast reactor concepts these two functions – power generation and fuel treatment – are separated except for the fueling-salt reactors where fuel treatment might be fully immersed in the reactor systems. It allows a reduction of waste and increasing the flexibility of the fuel management.

Such enhanced flexibility, though, should be supported by their extraordinary safety potential. Note, though, that as in any innovative technology the comprehensive predictive simulations would play a leading role in the design and assessment of molten salt reactors.

Of course, the simulations, models, libraries and computational tools, should be somehow experimentally validated ensuring their predictive capability maturity. Unfortunately, no one experimental case can be considered as a fully representative one for the phenomena or processes of interest. This requires users to select indirect experiment-based benchmarks extrapolating, then, the knowledge from experimental to application domain.

At current early stage of conceptual design development, we can 1) focus on the phenomena essential for reactor control in normal and accidental conditions, 2) combining nuclear-driven and non-nuclear experimental data in a validation of relevant codes and models, and 3) establishing realistic target accuracy requirements for best-estimate and penalizing models.

The paper briefly discusses some ideas on how and which legacy data derived from other programs could be useful to provide an evidence-based background for the validation of the simulations of molten salt systems.

## Introduction

From the very beginning of nuclear era the niche of the Fast neutron Reactors (FRs) has been foreseen [1] in a double-functionality of power and fuel generation, the last one converting fertile materials (238U and 232Th) into artificial fissile ones (239Pu and 233U, correspondently) to make nuclear power resources practically inexhaustible. Since then, the list of potential missions has been complemented by, among other, a utilization of plutonium recycled from the light water reactors and a minor actinide transmutation.

While traditional schemes presume separated power generation and fuel reprocessing functions in the circulating fuel salt Molten Salt Reactor (MSR) they could be combined [2]. However, an investment in the research and development of such integrated paradigm-shift nuclear-chemical system makes sense if one could address one or another widely recognized the following so-called High Impact Problems [7] (HIPs): 1) safe and reliable source of heat suitable for electric and non-electric power generation, including contribution in a hydrogen economy; 2) step toward long-lived waste-free nuclear power industry; and 3) flexible facility capable to incinerate minor actinides or multi-recycle plutonium.

Indeed, molten salt as high-temperature chemically inert and transparent coolant could facilitate the heat removal and exchange to meet the general requirements of a highly efficient power conversion process. Then, existing chemically and structurally stable thorium-based salts would allow involving available thorium resources in the nuclear power industry and so on. Furthermore, continuous or batch-like fuel-salt treatment allows improving reactor physics performance achieving such level of neutron potential to make it suitable as for wastes’ burning as for breeding.

However, the understanding of the coupled physics and chemistry in MSRs seems to be still limited, potentially resulting in some complementary demands to prove its safety performance [4]. Then, due to a limited operational background in case of Molten Salt Fast Reactors (MSFRs) one could rely largely on comprehensive simulations than on pure expert judgement. Of course, these simulations, as well as models and relevant tools, should be somehow experimentally validated ensuring enough Predictive Capability Maturity (PCM) [7].

Because of many reasons, no one single experiment covers entirely the required state-space of the phenomena and processes essential for normal operation and accident states of MSRs. Therefore, one should combine many cases establishing dedicated criteria of similarity between them and applications [7].

## major Specificities of Molten salt Reactors designs

The term Molten Salt Reactor denotes the broad range of concepts where liquid salt carries out one or more the following functions: 1) a high-temperature coolant [2]; 2) stable liquid fuel [2], and 3) combined coolant and fuel minimizing in-core heat exchange constraints[3], [4]. In the two last cases one considers several options of the fueling-salt treatment in a continuous or periodic mode [2], [5].

The variety of the design concepts, though, is not unlimited because of HIPs that narrow the degree of freedom for designers. Here, we are considering, though, only a circulating fueling-salt fast concept which always comprises:

* primarily circuit that contains an active core and circulating loops with mechanical pumps, etc.;
* secondary (technological) circuits with heat exchangers and other auxiliary systems;
* emergency draining tank and core-to-tank interconnectors, including frozen salt valves and so on;
* salt conditioning and refreshing facilities and so on

The physical phenomena to be studied and their links to the relevant fields of expertise are roughly presented in TABLE 1.

TABLE 1: The matrix of traditional fields of expertise associated with a circulating fuelled salt MSR

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | Reactor Physics | Criticality | Thermal Hydraulics | Salt and radiochemistry | Mechanics |
| Fuel circuit |[x] [x] [x]  [x] , [ ]  | [x] , [ ]  |
| Draining Tank |[ ] [x] [x] [ ] [ ]
| Fuel Treatment Unit |[ ] [x] [x] [x] [ ]
| Containment  |[ ] [ ] [x] [x] [x]

It is easy to see that major part of fields of expertise looks similar to ones for other types of nuclear reactors (see TABLE 2). The parts of chemistry and mechanics have double-filled markers being linked somehow with thermal hydraulics and reactor physics, e.g. considering corrosion-triggered mushy zones and thermomechanical effects, respectively. In the future, along with the concepts’ and technology elaboration, one will have to consider each a very detail way while nowadays we could focus on the only the major ones. As said, it is reasonable to award priority to the safety assessment, however, not entirely, but such issues that if not addressed could jeopardize the project.

In the case of the fuelling-salt system we have to take into account that heat generation and exchange in-core are of volumetric type while the heat removal in out-of-core heat exchangers is conducted through the surface. It is easy to show some non-linear phenomena essential only for the MSRs where the rate of salt circulation determines in-core temperature and, therefore, the feedback reactivity. The changes of reactivity results in the variations of the thermal power and, in its order, the outlet temperature, the heat exchange capabilities, and, then, circulation rate again leading to a feedback loop. Then, a common issue for such reactors would be in some reductions and fluctuations of delayed neutron effective fraction [2].

TABLE 2 Macro-categories specific to a fuelling-salt Fast Spectrum MSR by their components - Neutronics, Thermal-hydraulics and Nuclear chemistry and their combinations

| Figure-of-Merit  | Relevance to MSR safety and control  |
| --- | --- |
| Reactor physics |  |
| FoM1.1: Reactivity  | Determines in-core power and criticality safety of Emergency Draining Tank  |
| FoM1.2: Power Level  | Relates to entire reactor control, transient behavior and residual heating  |
| FoM1.3: Kinetics Parameters  | Reactivity feedback coefficients and inherent safety performance  |
| Thermal-hydraulics |  |
| FoM2.1: Temperature and Power window  | Determines the design and safety margins  |
| FoM2.2: Power and velocity  | Forms the topology of velocity and turbulent fields  |
| FoM2.3: Liquid phase composition  | Being time-dependent it would be essential for fuel evacuation and storage  |
| FoM2.4: Solid Phase Composition  | Essential for transients, fuel evacuation and core commissioning processes |
| FoM2.5: Gas Transport  | Challenges pumping and, therefore, forced circulations  |
| Salt chemistry  |  |
| FoM3.1: Nuclides’ inventory  | Essential for source term and for corrosion and plate out  |
| FoM3.2: Margins to decomposing  | Target parameter/value in the reactor control and safety  |
| FoM3.3: R/a materials retention  | Solubility of the major r/a elements taking into account a radioactive decay  |
| Coupled phenomena  |  |
| FoM1-2: Turbulence mapping | Determines the link between mezzo- and macro-scale fluid motions  |
| FoM2-3: Nuclide and chemical kinetics  | Nuclide inventory taking into account elements’ extraction and transport  |
| FoM3-1: Fluid saturation in r/a retention | Conditions for re-criticality and dissolved gases release  |

Such phenomena, of course, might not be fully reproduced in any facility, but only in real power reactors. This is why, only partially representative data and their combinations might be available at this stage of development.

Next phenomenon unique for fueling-salt MSR is a mapping of turbulence modes while, intuitively, one considers such reactors as homogeneous ones with uniform or smoothly variable in-core parameters – fuel density, temperature, fluid velocity and neutron spectra, power and flux distributions etc. Reality, though, might be rather far from this imagination because of appearance some meta-stable turbulent modes in the molten pool.

Semi-qualitatively, the scale of such formations might be estimated by Kolmogorov theory of turbulence where the characteristic dimension of these meta-stable modes depends on the energy as [10]

|  |  |
| --- | --- |
| $$E=C∙L^{^{2}/\_{3}}$$ | (1) |

where $E$, $L$ and $C$ are energy deposited in the vortex, characteristic dimension (vortex diameter in our case) and empirical constant. In the continuum mechanics in the case of dominance of the forces of inertia we could link circulating velocity $u$ an energy as $E\~u^{2}$. In case of heat generating fluid $E$ includes internal sources as well. So, the scale of meta-stable formation would be determined by the power density.

The sizes of formations would be in the range from centimeters to, even, decimeters – higher, though, the fuel channels pitch in, for example, Molten Salt Reactor Experiment (MSRE) [3] making its experience non-representative for new concepts studies. This phenomenon never could be observed in loop-type experiments. It does mean that we need in specific experiments dedicated to the mechanics of heat-generating fluids convection and stratification. The first candidates might be experiments initially intended to the severe accidents’ experimental studies.

Although their geometry and topology would be far from ones in the molten salt systems they might become suitable for validation needs if establish the similitude upon traditional dimensionless numbers.

An assessment of the fueled circuit integrity will require quantifying the synthetic values of shock wave ($V\_{wave}$) and sound ($V\_{salt}$) velocities ($V\_{wave}>V\_{salt}$). The first component is given as follows:

|  |  |
| --- | --- |
| $V\_{wave}=\frac{dP}{ρ∙c} = \frac{1}{ρ c} dP=\frac{1}{ρ c} \left|\frac{α}{β}\right|τ\frac{dT}{dt} \~\cdots \frac{1}{ρ c}\cdots \left|\frac{α}{β}\right|τ\cdots \frac{dQ(t)}{dt} $ ,  | (2) |

where $V\_{wave}$, $dP$, $ρ$ and $c$ are velocity of waves propagation, pressure jump, fluid density and acoustic wave speed, respectively, and on the right hand side of the equation $τ$, $α$, $β$ , $Q(t)$ and $\frac{dQ(t)}{dt}$ are pseudo relaxation time constant (depended on core size and aspect), the salt dilatability, the compressibility, total reactor power and its time derivation, respectively.

It is clear that power peak depends on the reactivity feedback due to the salt dilatation - the higher salt dilatation the higher will be negative reactivity feedback and, in its order, the lower will be the temperature rise. The pressure jump, meanwhile, rises together with the salt dilatation. Thus, first, we can see that the transients are driven by the small value resulting from the controversial summation of complex large functions - pressure jump and power rush. Such fine balance of two large values increases, in its order, the requirements to an accuracy of simulations.

Then, we should consider the specificity of such reactors’ control and all the issues of its controllability. Similarly, the circulating salt carries away long-life precursors of delay neutrons returning them back with some delay making variable, in such a way, the effective delayed neutron fraction and their effective decay constants. Qualitatively the phenomena might be illustrated with point-wise models where instead of the following traditional form:

|  |  |
| --- | --- |
| $$\frac{dQ}{dt}=\frac{ρ\left(t,T\right)-β\_{eff}}{Λ\_{eff}}∙Q+λ\_{eff}∙C\left(t\right)\~\left\{in MSR\right\} \~\frac{ρ\left(t,T\right)-β\_{eff}}{Λ\_{eff}}∙Q+\tilde{λ}\_{eff}(Q, u)∙ω( u)∙C\left(t\right)… $$ | (3) |

where in the first equation $C(t)$, $β\_{eff}$, $λ\_{eff}$,$ Λ\_{eff}$ and $ρ\left(t,Q,T,…\right)$ are an amount of precursors, effective delayed neutron fraction, effective delay constant, neutron generation lifetime and reactivity as function of power, temperature ($T$) and time ($t$), respectively, while on the right hand side $ω,$ $ u$ and $\tilde{λ}\_{eff}$ are circulation velocity, correction factor on precursors relocation and corrected decay constant.

Although the development of the control strategy and Master Equations are not yet complete we might state that the Target Accuracy Requirements (TARs) will be much stricter than in case of traditional solid-fuel concepts. For example, in-depth discussion in the frame of several international projects initiated in the OECD-NEA allows estimating them by an expert-based panel studies (see [6] and personal communications). Thus, TARs for fast reactors are of ~ 0.3÷0.5% for the multiplication factor (equivalent to one-half of fuel sub-assembly worth), ~ 5 % and 10% for rods’ efficiency, reactivity coefficients and kinetic parameters and so on [6]. It is easy to show that the analogous requirements for MSR would be ~ 1% and 0.5% for multiplication factors in core and in Emergency Draining Tank, respectively, ~ 10% for the thermal reactivity feedbacks and ~ 1% of uncertainties for effective delayed neutron fractions but divided on two – fast and extremely slow – components.

Reactor physics modelling being taken alone can be calibrated against precise and high-fidelity tools (like continuous energy Monte-Carlo ones and so on) where remaining uncertainty will depend only on nuclear data uncertainties. It allows reducing the validation process to the only nuclear data libraries. In this case we are selecting benchmarks for whatever functionals – independently for criticality, reaction rates, reactivity studies, depletion and so on - by an existence of nuclides of interest, spectra and of low experimental uncertainty, i.e. high-fidelity ones.

Unfortunately, the experiments that contain the materials of our interests and, among them, with fast and epithermal spectra are very rare, if even exist. For example, the handbooks of high-fidelity experimental benchmarks like ICSBEP [9] among the fast or intermediate spectra cases (~ 3000 configurations) contains only a few ones with fluorine and no one fluoride diluted fast neutron case with 233U (~ 30 intermediated and ~150 thermal neutron configurations).

The only way would be to use pre-calibrated nuclear data for all impactive components of no interest (see [6], [8]). In our case the main nuclear data to be discriminated by pre-calibration are iron, chromium, nickel, hydrogen, oxygen, carbon, beryllium etc. in thermal spectra constructing a chain of the sets of integral experiments (a validation chain) to apply an iterative (progressive [6]) Data Assimilation approach adjusting step-by-step nuclear data from ones of no-interest to ones of low-interest and, then, to the particular interest of given MSRs.

$\frac{\left\{ND\right\}}{\left\{IE\right\}\_{1}}$ $\rightarrow \frac{\left\{ND\_{1},ND\_{RES}\right\}}{\left\{IE\right\}\_{2}}$ $\rightarrow \frac{\left\{ND\_{1},ND\_{2},ND\_{RES}\right\}}{\left\{IE\right\}\_{3}}$ $\rightarrow $ $…\rightarrow \frac{\left\{ND\_{1},…,ND\_{K-1},ND\_{RES}\right\}}{\left\{IE\right\}\_{K}}\rightarrow \frac{\left\{ND\_{1},…,ND\_{K-1},ND\_{K},ND\_{RES}\right\}}{}$

Fig. 1. Iterative Data Assimilation approach using a chain of the sets of experimental cases.

In the figure above the members like $\left\{ND\right\}$ and $\left\{IE\right\}$ mean given sets of nuclear data and integral experiments, the stroke indicates an adjustment of the nuclear data basing on the integral experiments, while the indices correspond to the iteration of the adjustment and RES denotes residual/non-adjusted data.

Finally, the process of nuclear data qualification and, therefore, of reactor physics codes’ validation will involve in a validation turnover all available and dedicated integral experiments (~ three-five thousand cases [9]).

## Regulatory environment, safety principles, guidance and standards

MSR combines together many nuclear technologies making operator and assessor implement safety standards related to the different fields. For example, because of on-line fuel treatment and conditioning we might not separate reactor itself and its fuel cycle. Therefore, the standards relevant the safety of reactor operation would include as traditionally applied ones – GSR n°4 – and standards related to the transport of r/a and fuel, to the safety of fuel cycle facilities.

 “3.24. The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control.

A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out in a particular State for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity”.

According to this paradigm the level of activity corresponds to the needed level of detail matters. In the case of the feasibility study one focusing the very principle aspects of design concept could note the leading role of comprehensive predictive modelling at all its stages as mentioned below. At the same time, it should be admitted the notable and numerous uncertainties as in the simulations as in the performance of the planned activities and facilities. Despite that these uncertainties might be fully accepted at the feasibility study level it seems better to quantify them according to the following recommendation.

“3.4. Other relevant factors, such as the maturity or complexity of the facility or activity, are also to be taken into account in a graded approach to safety assessment. The consideration of maturity relates to the use of proven practices and procedures, proven designs, data on operational performance of similar facilities or activities, uncertainties in the performance of the facility or activity, and the continuing and future availability of experienced manufacturers and constructors”.

“4.59. Uncertainties in the safety analysis have to be characterized with respect to their source, nature and degree, using quantitative methods, professional judgement or both”.

“Requirement 18: Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation”.

Then, according to the meaning of the IAEA standards one may state that even at the very early stage of conceptual design and analysis one recommends envisaging somehow a testing background and facilities to address all further demands on the validation of the simulations and the nuclear design, as such.

## An available background on development, studies and operation

The stringent requirements mentioned above make us to complement the available validation matrices by non-fully representative experimental data dedicated to the simulant materials tests and bulk-type thermal hydraulics studies for heat-generating fluids including, for example, such experimental facilities as BALI, COPO, ACOPO, RASPLAV-Salt, LIVE and so on [13], [14], [15], [16], [17].

RASPAV and, especially, RASPLAV-salt are semi-spherical 2D-slice experimental configuration with similar to molten core and molten salt two-sides heated fluids intended initially to study, among others, the relevant dimensionless numbers and thermal-hydraulic correlations as well as thermal flux studies. The experiments included, among others, mushy zone disappear, miscibility and immiscibility of different strata in the fluid. The last one might be used to validate the models to be developed coupling inherent heat generating fields and free convection of this fluid in bed-kind geometry. In other words, the mechanistic phenomena studied in the past in the series of numerous RASPLAV and RASPLAV-Salt experiments cold provide inputs for MSR coupled in-core modeling. In addition, their results – both convection modeling and heat flux direction - might be essential to perform a validation studies for salt for the modeling of dilution and cooling in an Emergency Draining Tank.

BALI hemi-cylinder 2D slice experimental facility with water as a simulant of molten fluid that was initially intended to study convection and heat fluxes in a homogeneous heat generating liquid in order to investigate accidental processes in Light Water Reactors. In a context of MSRs studies the date of this legacy experiments could, nevertheless, provide some inputs to verify the macroscopic invariants in a motion of an idealized homogeneously heat heated fluid.

The LIVE 3D test vessel simulates the hemispherical lower plenum of reactor pressurized vessel of a PWR in 1:5 scale [12, 13]. The melt surface can be either free surface by covering the test vessel with an insulation lid [14, 15] or cooled with a water-cooled lid.

TABLE 3 Legacy experimental data and their representativity to selected phenomena in MSR

|  | Initial intension  | Value for validation of MSR simulation  |
| --- | --- | --- |
| RASPLAV, and RASPLAV-salt | Heat transfer in the molten core of light water systems, including in some cases mushy zone, mixing and saturation  | Could provide inputs for MSR coupled in-core modeling. In addition, their results – both convection modeling and heat flux direction - might be essential to perform validation studies for salt for the modeling of dilution and cooling in an Emergency Draining Tank |
| BALI | 2D-slice hemi-cylinder convection and heat transfer in the low head of accidental reactor | Could give some inputs to verify the macroscopic invariants in a motion of an idealized homogeneously heat heated fluid |
| LIVE | 3D-hemi-sphere experimental modeling of molten core behavior using different homogeneous simulant fluids  | Free convection of heat-generating fluid, heat exchange and heat flux studies in several molten salt configurations  |
| COPO | Scaled 2D-slice experiments to study free convection and heat fluxes using water salt solutions to simulate heat-generating fluids  | Volumetric heat-generating fluid experimental modeling could be useful to clarify the mapping of convective streams  |
| ACOPO | Scaled 3D hemi-spherical configurations to reproduce free convection at the RPV bottom  | Might be useful to deeper investigate the differences between symmetric and asymmetric motions of the free convection modes  |
| SIMECO  | 2D-slice semi-spherical experimental modeling of heated 50/50 NaNO3 and KNO3 convection  | Continuum mechanics volumetric convection experimental studies for the fluid with similar to MSR viscosity and rheology  |
| COPRA  | 2D-slice semi-spherical experimental modeling of heated non-eutectic 20/80 NaNO3 and KNO3 convection | Complement to the SIMECO data to better understand the geometry factors in heat leakage and in convection  |
| DYNASTY | Loop-kind experimental facility with circulating pseudo-heat generating molten salt  | Constitutive relations and nodal schemes calibration for variable viscosity molten salt fluid imitating the internal heat generation  |

Legacy experiments (non-related to molten salt) COPO and ACOPO might be also considered as a suitable tool to optimize nodal schemes for cooling and convection in a pool filled by heat generating fluid. The value of these experimental configuration is that they represented 3D- experiments on heat transfer in liquid with internal heat generation at very high internal Rayleigh numbers typical for natural convection in a core melt. Of course, since the experiments used water as a simulant fluid the data are of limited representativity. However, they could be applied even for MSR studies using relevant dimensionless criteria as it is typical in the continuum mechanics.

SIMECO 50/50 NaNO3 and KNO3 mixture 2D experimental studies (KTH,). For convection law studies and, especially, for calculational methodology validation it does not matter which salt or another fluid has been used. The more important aspect would be to conserve a given interval of dimensionless numbers which encompasses the working regimes of interest.

COPRA (COrium Pool Research Apparatus) experiments initially intended to an experimental modelling of the natural convection and heat transfer in an internally heated melt pool using non-eutectic 20/80 NaNO3 and KNO3 fluid mixture [17].

In such paradigm the experimental data like mentioned above might be suitable adding value to understanding of the assessments inherent to the applied mathematical models and calculational algorithms.

Special importance is given to DYNASTY molten-salt loop as an experimental tool to provide users with wide range of data concerning viscous fluid natural circulation in closed circuit [18]. It could help in an estimation of the major correlations and, even, basic physics parameters relevant to the hydraulic phenomena in MSRs. The only deficiency of its value would be that the facility will not simulate in-bulk convections. This is why it seems rational to fill the gap of a volumetric motions by the legacy experiments planned for severe accidents studies.

## Conclusions

It seems a consensual position to require any statement or engineering solution to be proven on a solid basis of objective observations or representative sets of experimental data. It, first of all, is crucial for the domain of safety, especially, in case of direct involving predictive simulations in an assessment process.

In our case – MSRs characterized by limited operational experience – the design works and safety studies necessarily include an analysis and an assessment of the codes, models and data libraries inherently involved in these design and safety studies.

All MSR and, especially, the circulating fuel-salt fast-neutron reactors need multi-physics models and simulations. Unfortunately, the only single- and few-physics integral experiments - both separate and integral effect tests - are available to build-up an evidence-based background for global validation process. At the same time, the suitable data might be extracted from the experimental programs that were initially intended to other functionals than what would be needed for MSRs.

The only requirement is that the experimental data should be somehow assessed and peer reviewed translating them in a form of experiment-based benchmarks. And, of course, the standards to experiment-based benchmarks and to extrapolation beyond the experimental domain are not fully available and have to be developed.

The consideration of experimental validation over here has been done at rather general level without going in a very detail matter to the nuclear reactor concept. This is why, in the analysis we relied on the only basic principles of the fuel-salt fast neutron reactors, including its main specific features like unconventional reactor control strategy, circulating fuel compositions and main aspects of neutron economy, presence on site a fuel treatment unit, hydrodynamics of heat generating and non-homogeneous fluids and reduced barriers, and so on.

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## References

1. E. Fermi, The Future of Atomic Energy, United States, (1946), [www.osti.gov/accomplishments/documents/fullText/ACC0043pdf](http://www.osti.gov/accomplishments/documents/fullText/ACC0043pdf)
2. J. Serp *et al.*, “The molten salt reactor (MSR) in generation IV: Overview and perspectives,” *Prog. Nucl. Energy*, vol. 77, pp. 308–319, 2014
3. Murray W. Rosenthal, “AN ACCOUNT OF OAK RIDGE NATIONAL LABORATORY’S THIRTEEN NUCLEAR REACTORS”, ORNL/TM-2009/181
4. M. Allibert, et al, 7 - Molten salt fast reactors, Editor(s): Igor L. Pioro, In Woodhead Publishing Series in Energy, Handbook of Generation IV Nuclear Reactors, Woodhead Publishing, 2016
5. Ignatiev, V.V., Feynberg, O.S., Zagnitko, A.V. et al. Molten-salt reactors: new possibilities, problems and solutions. At Energy 112, 157–165 (2012)
6. Giuseppe Palmiotti, Massimo Salvatores, The role of experiments and of sensitivity analysis in simulation validation strategies with emphasis on reactor physics, Annals of Nuclear Energy, Volume 52, 2013
7. E. Ivanov, J. Baccou, B. Rearden, A. Boulore, K. Velkov, Role of a phenomenological validation and integral experiments for maturing the predictive simulations, Nuclear Engineering and Design, Volume 362, 2020
8. T. Ivanova, E. Ivanov and I. Hill, “Methodology and issues of integral experiments selection for nuclear data validation”, *EPJ Web Conf.*, 146 (2017)
9. J.B. Briggs, J.D. Bess, J. Gulliford, “Integral Benchmark Data for Nuclear Data Testing Through the ICSBEP & IRPhEP”, Nuclear Data Sheets, Volume 118, 2014
10. A. Kolmogorov, A refinement of previous hypothesis concerning the local structure of turbulence in a viscous incompressible fluid at high Reynolds number. Journal of Fluid Mechanics 13: 82-85, 1962.
11. IAEA Standard Safety Assessment for Facilities and Activities General Safety Requirements Part 4, GSR Part 4
12. GIF/RSWG/2010/002/Rev.1 An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems (June 2011)
13. V.G. Asmolov et al, Corium Electric Conductivity Measurement. In: RASPLAV Final Report, Attachment C, Properties Studies: Methodology and Results, OECD RASPLAV Project, Moscow, RRC KI, 2000, p. 50
14. Bonnet J.M., Seiler J.M. Thermal hydraulic phenomena in corium pools: the BALI experiment // Proceedings of the 7th International conference on nuclear engineering, Tokyo, Japan, April 19-23, 1999, ICONE-7057
15. A. Miassoedov, T. Cron, J. Foit, X. Gaus-Liu, S. Schmidt-Stiefel and T. Wenz, "LIVE experiments on melt behavior in the RPV lower head.," in Proceedings ICONE-16, Orlando, 2008
16. B. Sehgal et al, SIMECO Experiments on In-Vessel Melt Pool Formation and Heat Transfer with and without a Metallic Layer (NEA-CSNI-R--1998-18). Nuclear Energy Agency of the OECD (NEA) (Feb 1999).
17. L.Zhang et al, COPRA experiments on natural convection heat transfer with high Rayleigh numbers, NURETH-16, Chicago, IL, August 30-September 4, 2015
18. A. Pini, A. Cammi, M. Cauzzi, F. Fanale, L. Luzzi, An Experimental Facility to Investigate the Natural Circulation Dynamics in Presence of Distributed Heat Sources, Energy Procedia, Volume 101, 2016, pp 10-17