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DEVELOPMENT AND VERIFICATION OF DYNAMICS CODE FOR MOLTEN SALT REACTORS

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ABSTRACT

To perform transient analysis for Molten Salt Reactors (MSR), the reactor dynamics code DYN3D developed in FZR was modified for MSR applications. The MSR as a liquid fuel system can serve as a thorium breeder and also as an actinide burner. The specifics of the reactor dynamics of MSR consist in the fact, that there is direct influence of the fuel velocity to the reactivity, which is caused by the delayed neutrons precursors drift. This drift causes the spread of delayed neutrons distribution to the non-core parts of primary circuit. This leads to a reactivity loss due to the fuel flow acceleration or to the reactivity increase in the case of deceleration.

For the first analyses, a 1D modified version DYN1D-MSR of the code has been developed. By means of the DYN1D-MSR, several transients typical for the liquid fuel system were analyzed. Transients due to the overcooling of fuel at the core inlet, due to the reactivity insertion, and the fuel pump trip have been considered. The results of all transient studies have shown that the dynamic behavior of MSR is stable when the coefficients of thermal feedback are negative.

For studying space-dependent effects like e.g. local blockages of fuel channels, a 3D code version DYN3D-MSR will be developed. The nodal expansion method used in DYN3D for hexagonal fuel element geometry of VVER can be applied considering MSR design with hexagonal graphite channels.

NOMENCLATURE

C_i	precursors of *i*-th Delayed neutrons group
z	axial coordinate position
t	time
v	fuel velocity
k_{eff}	effective multiplication coefficient
β_i	fraction of *i*-th delayed neutrons group
ν_g	the mean number of neutrons produced by fission caused by a neutron group *g*.
$\Sigma_{f,g}$	macroscopic effective cross-section of fission for the neutron group *g*.
$\Phi_g(z,t)$	neutron flux of group *g*.
λ_i	decay constant of *i*-th group of precursors
ρ	fuel salt density
$h(z,t)$	enthalpy
$Q_{fission}$	heat source; the fission energy directly released to the fuel salt
Q_{decay}	heat source; the fission energy released with delay through decay heat
$Q_{graphit}$	describes the heat exchange between the fuel salt and graphite block
FZR	Forschungszentrum Rossendorf
PWR	Pressurized Water Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
MSBR	Molten Salt Breeder Reactor

INTRODUCTION

The three-dimensional code DYN3D [1] for the calculation of steady states and transients in PWR reactors has been developed in FZR. The neutron kinetics of DYN3D is based on the nodal expansion method applied to Cartesian or hexagonal fuel assembly geometry. It is coupled with a two-phase thermo-hydraulic model and with a fuel rod model. The subject of this paper is the development and verification of modified code DYN1D-MSR, which is designed for the Molten Salt Reactors. The MSR is a project [2], which is presently revisited, because of its safety advantages, the possibility to use the uranium-thorium fuel cycle, and the potential to transmute actinides. The fuel of this reactor is liquid in the form of a molten salt mixture, which also acts as a coolant.

The 1D version is the first step of the DYN3D-MSR development. It is based on the numerical methods of DYN3D, which have been developed for the neutron flux expansion in axial direction and it includes new models describing the motion of liquid fuel. The code is designed to treat channel type reactors, e.g. the MSR design with fuel flowing in cylindrical channels inside a hexagonal graphite structure (Fig. 3).

Due to the liquid fuel the thermo-hydraulic and neutron kinetic characteristics of MSR differ significantly from that of a classical PWR. The core consists mostly of graphite blocks, which form the channels for the flowing fuel. The motion of fuel influences the distribution of delayed neutrons precursors. The precursors can be drifted with the flow in time period between their origin and decay. Overall changes in the distribution are dependent on the flow velocity, on the recirculation time, and on the decay constants of each precursors group.

The thermal-hydraulic model is determined by the fact that the energy from fission is predominantly released into the fuel salt and is directly moved out of the core. However, there is a small fraction of energy, which is released through the gamma radiation in the surrounding graphite and an additional part is released with delay as decay heat.

DELAYED NEUTRONS MODEL

It is a special feature of the MSR kinetics that the precursors of delayed neutrons (DN) are partly drifted with the flow between the origin and decay. Therefore, the distribution of released DN differs from the static distribution and is no longer proportional to the fission neutrons distribution. The level of perturbation depends mainly on the fuel flow velocity and on the fuel volumetric ratio in core and primary circuit.

The phenomenon mentioned above is taken into account in the DYN1D-MSR. The code is based on neutron kinetics routines of DYN3D and includes new model of the DN production. This model is based on the solution of 1D precursors equations in the axial direction and both core and primary circuit are accounted. The equations for these two parts differ only in the source terms.

The distribution of precursors source in the core is determined by the neutron kinetics routines of DYN3D. In the DYN3D code, the nodal expansion method [3] is used. The source distribution has a polynomial character in the axial direction and an exponential character in the time.

The drift of the DN precursors is represented in the equations by a convective term. Thus, the equations describing the distribution of the i-th group of precursors are the following:

$$\frac{\partial}{\partial t}C_i(z,t)+\frac{\partial}{\partial z}\left(vC_i(z,t)\right)=\frac{1}{k_{eff}}\sum_g \beta_i v_g \Sigma_{f,g}\Phi_g(z,t)-\lambda_i C_i(z,t)$$

in the core and

$$\frac{\partial}{\partial t}C_i(z,t)+\frac{\partial}{\partial z}\left(vC_i(z,t)\right)=-\lambda_i C_i(z,t)$$

in other parts of primary circuit. Standard notation is used.

The method of characteristics is used to find the solution of these equations. For this, the entire primary circuit and core are divided into nodes. It is supposed, that the velocity is constant within one node during one time step. So the partial differential equation can be easily transformed to an ordinary differential equation and the character of source term allows to find the analytical solution within one node.

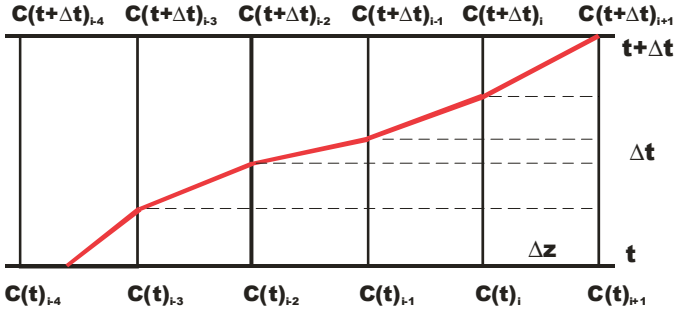

Fig. 1: Path of fictitious fluid particle during one time step Δt through several nodes with different fuel velocity to the edge of the i-th node.

Using these nodal solutions, the precursors concentration on the edge of each node can be found (Fig. 1). The space distribution of the precursors is in each node determined by the linear interpolation between these edge points. Fig. 2 shows the distributions of the relative power and of the delayed neutrons for a steady state condition with given fuel mass flow.

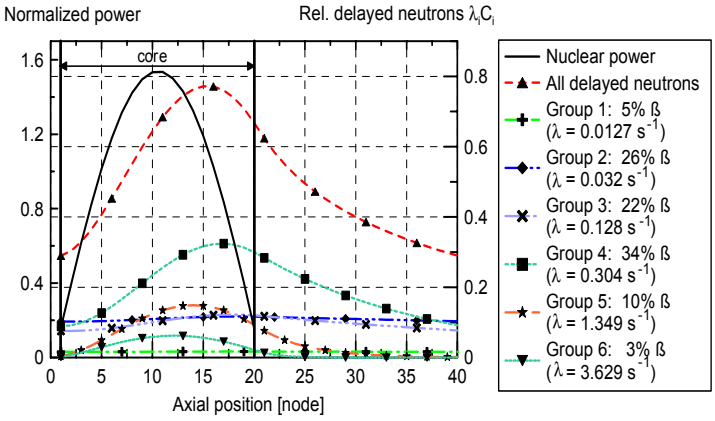

Fig. 2: Distribution (normalized to 1) of power (left scale) and of 6 groups of delayed neutrons (right scale) for flowing fuel reactor v ≈1m/s.

THERMO-HYDRAULIC (TH) MODEL

It is an specific feature of the MSR reactors that the liquid fuel is flowing through the core and whole primary circuit and is acting also like a coolant. Because of the geometry and the presence of graphite blocks, the fission reaction is taking part only in core. The graphite blocks form the channels for the fuel. The geometry of these channels can be various. However, in the DYN1D-MSR code it is supposed that the fuel flows through hexagonal graphite blocks in the core. In the TH model a cylinder with an inner channel of flowing fuel approximates this hexagonal graphite block. This configuration can be seen from Fig. 3.

Because of the MSR nature the energy from fission is predominantly released in the fuel and is directly convected by the flow out of the core. The graphite channels are heated-up in the same time by radiation. Therefore, in most cases the temperature of graphite is higher than that of fuel. Heat conduction in the graphite in the axial direction can be neglected due to the poor graphite conductivity and small temperature gradient. Naturally, there is a heat exchange at the fuel-salt interface. The graphite is in steady state practically cooled-down by the fuel. The above described heat exchange can be compared with the heat exchange on fuel pin surface in the PWR. However, in the case of the MSRs, the heat exchange is much smaller.

A special numerical model was developed for the dynamics simulation of the temperature distribution in the graphite. For this purpose the hexagonal graphite block was approximated by cylinder with an inner channel of flowing fuel. Furthermore, the cylinder was divided into an arbitrary number of coaxial segments (Fig. 3). The method of the effective heat transfer coefficients was used in the numerical model. These coefficients are derived for each segment from an analytical solution of the steady state heat conduction equation, assuming the piecewise constant heat conductivity. The boundary conditions of no heat transfer to the next graphite block at the outer boundary and negligible axial heat conduction are assumed. Effective heat transfer coefficients represent the heat exchange between two neighboring segments and determine the coefficients of a linear algebraic equations set. This set of equations is solved for every time step to obtain the temperature distribution with the heat exchange boundary condition on the surface of graphite block.

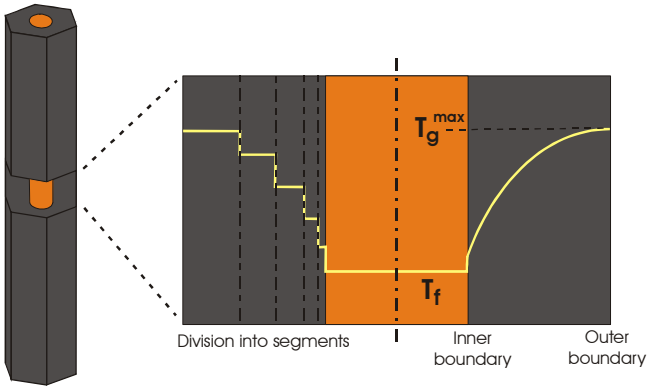

Fig. 3: Modeling of the temperature distribution in a graphite channel by 5 segments with effective heat transfers coefficients. Comparison of the calculated steady state distribution (on the left) with analytical solution (on the right).

So there are three major ways of heat production in the core. The most important and fastest way is the direct heat release to the fuel salt. The second way is the radiation heat-up of graphite and the third is the delayed release through decay heat. For the modeling of the decay heat it was supposed in DYN1D-MSR, that it is a small portion of steady state power, which is released during the transient uniformly in the core and primary circuit.

In the applied model, it was also assumed that the pressure remains constant, the fuel density depends on temperature and that the velocity for each node is obtained from the continuity equation. Assuming also the absence of either boiling or solidification of fuel salt, a single phase TH model can be used. In this case the energy equation will have the following form:

$$\frac{\partial}{\partial t}\left(\rho h(z,t)\right)+\frac{\partial}{\partial z}\left(\mathrm{v}\rho h(z,t)\right)=Q_{fission}+Q_{decay}+Q_{graphit}$$

Standard notation is used. The sources on the right hand side of the equation are the above mentioned: direct heat release $Q_{fission}$, decay heat Q_{decay} , and heat exchange between the graphite and salt $Q_{graphit}$.

As mentioned, $Q_{fission}$ is the heat directly released from fission to the fuel. This source is proportional to the neutron flux, it is located only in the core and represents around 90% of all energy. The second source Q_{decay} represents the decay heat. It is proportional to the steady state power level and assumed to be constant during the transients. The third source $Q_{graphit}$ encompasses the heat exchange between the graphite and the fuel [4].

The temperature distribution in the fuel and in the graphite can be obtained from the energy balance equation using the method of characteristics [5]. This method involves, similarly as in the DN model, integration along the path of a fictitious fluid particle moving with the flow. The axial temperature distributions in a hexagonal fuel element divided into 20 axial nodes are shown in Fig. 4.

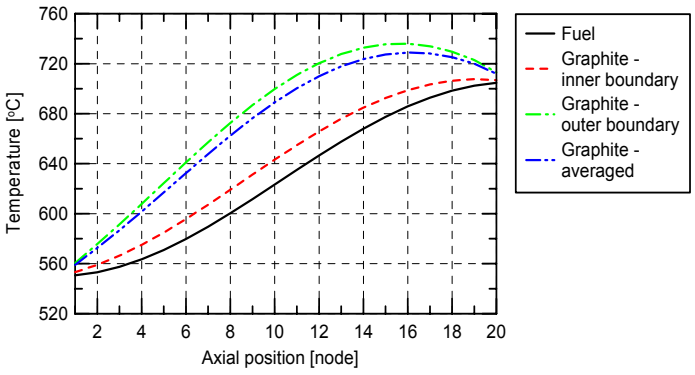

Fig. 4: Typical axial distributions of temperature in the steady state for the fuel channel of MSR The fuel temperature at the core inlet is 550°C. Flowing through the core; it is heated up by 150°C approximately.

VALIDATION OF 1D CODE KINETICS

The data measured during the Molten Salt Reactor Experiment in the Oak Ridge National Laboratory (ORNL) [6], [7] were used to validate the DYN1D-MSR kinetics, especially the delayed neutrons model. The MSRE was based on an 8 MW_{th} thermal reactor, with molten salt fuel and moderated by graphite. It was constructed to prove the ability of the technology for a civilian energy production. The next step of the development was planned to be 1000 MW_e Molten Salt Breeder Reactor. The MSRE was very successful and many experiments have been performed.

From these ORNL MSRE experiments, the measured data of one natural circulation experiment and three zero-power experiments were considered and used for the validation:

- Effective loss of DN in steady state operation
- Protected fuel pump start-up
- Protected fuel pump trip
- Natural circulation transient

These four experiments were chosen as a benchmark to be calculated by MOST project participants [8]. The MOST project 'Review of Molten Salt Technology' is focused to the review of MSR advantages and difficulties. The benchmarks were calculated using several codes in more than five institutions. The first benchmark was focused on the delayed neutrons leakage in the steady state operation due to the fuel circulation. The measured results in ORNL are represented in the terms of reactivity loss. The MSRE effective reactivity loss for U235 fuel was found to be 212 pcm and for the U233/U235/Pu239 fuel mixture 100±15 pcm. The reactivity loss values calculated by DYN1D-MSR are 253.2 pcm and 121.2 pcm for the U235 fuel and the U233/U235/Pu239 fuel mixture, respectively. The two groups cross-section libraries used for these calculations have been prepared in the frame of MOST project [8] by EDF with Apollo code [9] using the JEF 2.2 library. Two sets of DN data were used in the calculation: calculated JEF DN data and the original MSRE DN data [6]. The agreement with the results obtained by the other codes is good. However, the results of most numerical codes differ from the experiment. The differences between experiment and calculations can be caused by use of the simplified core geometry. Considering the 1D nature of some codes and to get comparable results, the elliptical upper plenum of MSRE reactor core was modeled as a flat area with constant height. The results for original ORNL DN data can be seen in Tab. 1.

Table 1: Loss of reactivity due to the fuel flow (Static benchmark calculated by MOST project participants).

Code used	Reactivity loss ^{235}U fuel [pcm]	Reactivity loss $^{233}U/^{235}U/^{239}Pu$ [pcm]
EXPERIMENT (ORNL)	**212**	**100±15**
PoliTO [1)]	260.9	119.2
DYN1D-MSR [2)]	253.2	121.2
Cinsf1D [3)]	228.8	107.8
SIMMER 3D [4)]	262.2	125.0
SimADS [4)]	212.2	105.2

[1)] *Politecnico di Torino* [2)] *FZ Rossendorf*
[3)] *Electricité de France* [4)] *FZ Karlsruhe*

It should be mentioned here that the β_{static} (effective DN ratio for non circulating fuel) was 666 pcm for the U235 fuel and 289 pcm for the U233/U235/Pu239 fuel mixture. So the relative loss of DN is 38% β_{static} and 42% β_{static}, respectively.

The second and the third experiment were focused on the protected fuel pump start-up and fuel pump trip transients. During these two transients a constant power was maintained. The change in the delayed neutron production (calculations for both DN data were made) was compensated by the use of control rods and the inserted compensative reactivity was the variable to be compared with experiment (Fig. 5 and 6).

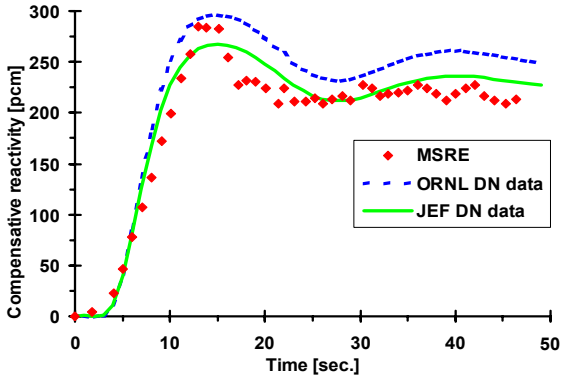

Fig. 5: Compensative reactivity inserted by the fuel pump start-up transient (constant power is maintained). Original ORNL DN data and calculated JEF DN data were used.

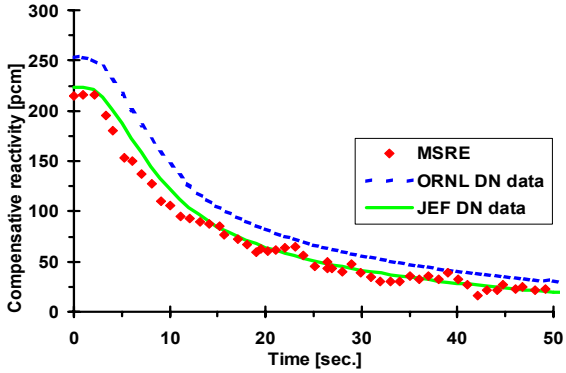

Fig. 6: Compensative reactivity inserted by the fuel pump trip transient (constant power is maintained). Original ORNL DN data and calculated JEF DN data were used.

In the case of fuel pump start-up transient, the velocity of fuel is changing from zero to the nominal value in 10 seconds. The delayed neutron precursors start to be drifted out of the core and the rods are withdrawn to compensate the loss. In the MSRE experiment there was a limitation of control rods withdraw speed, which was not applied in DYN1D-MSR calculation. The difference caused by this can be seen in the first 15 seconds of transient from Fig. 5, where the calculated reactivity is higher. The oscillations of inserted reactivity are caused by the changing level of precursors recirculation to the core during the transient. The differences in the reactivity starting level by the fuel pump trip transient are based on the differences in reactivity loss in steady state. The agreement in the inserted reactivity is in both cases very good.

The fourth benchmark was focused on natural circulation experiment performed in ORNL. On the beginning of this transient the reactor power is stabilized on the minimum level

and the fuel does not circulate. Then the secondary cooling circuit is put in the operation and so the primary fuel salt is started to be cooled-down. This causes the origin of natural circulation in primary circuit. Positive reactivity is inserted through the thermal feedback when the colder fuel reaches the core. The power starts to grow, and so does the temperature difference. The velocity of natural circulation is increasing.

In the numerical simulation, the secondary circuit and heat exchanger were not modeled. The fuel inlet temperature and velocity were given as a forcing function and the power level was the variable to be compared with experiment (Fig. 7).

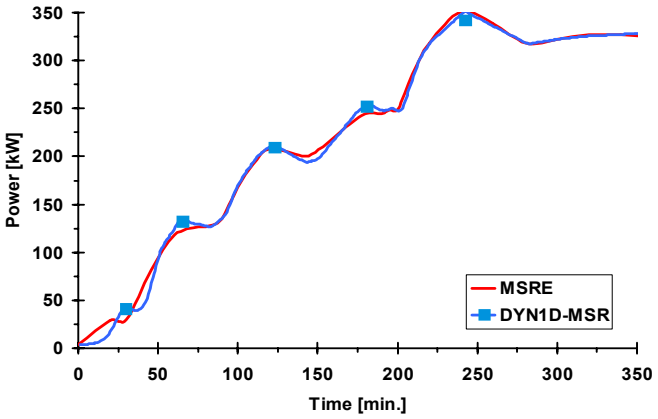

Fig. 7: The power history during the 350 minutes of the natural circulation experiment (U233/U235/Pu239 fuel).

There is small discrepancy at the beginning of transient. It seems that the experimental data are not starting from a steady state and so the power is already growing in *t=0*. However, the numerical simulation starts from the steady state power.

MSBR TRANSIENT CALCULATION

After the successful validation of DYN1D-MSR code it was applied to the analysis of several hypothetical transients in Molten Salt Breeder Reactor. The geometric data used as the input deck for the calculations with thermal feedback were taken from the MSBR project description [10]. The original core geometry was simplified for 1D purpose. Four cross-section libraries used for the calculation have been prepared for the participants of MOST project by EDF with Apollo code [9]. The first two are for the beginning of load (BOL) and second two for the equilibrium state (EQ) of operation. The difference in each two groups was the presence of Erbium (Er). These libraries are:

- BOL with Er
- BOL without Er
- EQ with Er
- EQ without Er

The erbium was added to graphite to compensate its positive thermal feedback coefficient, because the absorption cross-section of erbium is sensitive to the spectra change caused by graphite warm-up. Using these libraries and simplified 1D geometry, four tasks were calculated:

- Effective loss of DN in steady state operation
- Unprotected fuel pump trip
- Unprotected insertion of 300 pcm reactivity
- Unprotected overcooling of the fuel at core inlet

The first results were obtained for steady state operation. It is mainly the effective loss of DN and the thermal feedback coefficients. It was quite interesting to compare the DN loss with MSRE calculation. The β_{static} (effective DN ratio for non circulating fuel) was for MSBR fuel 330 pcm and the effective loss of DN was found to be 255 pcm. Therefore, the relative loss of DN in MSBR is 77%β_{static}. It means that in the MSBR relatively twice more DN are lost than in the MSRE case. That is caused by the higher speed of fuel circulation and also by the longer recirculation time. The second results obtained for steady state are the feedback coefficients. From the Tab. 2 it can be seen that the graphite thermal feedback coefficient is really positive for the libraries without erbium.

Table 2: Thermal feedback coefficients for MSBR.

Thermal feedback of MSBR PCM/°C				
Temperature 650-750[°C]	With Er 167		Without Er 167	
	BOL	EQ	BOL	EQ
Graphite	-0.605	-0.237	2.365	2.172
Salt	-3.233	-2.392	-2.646	-1.924
Overall	-3.881	-2.681	-0.251	0.263

The second transient studied was the fuel pump trip. Two antagonistic effects influence reactor power during this type of a transient. Together with the fuel velocity, also the heat removal is slowed–down. The increasing fuel temperature results in a negative (or positive) reactivity insertion by virtue of the thermal feedback coefficient being negative (or positive). At the same time, the change of delayed neutron distribution introduces positive reactivity.

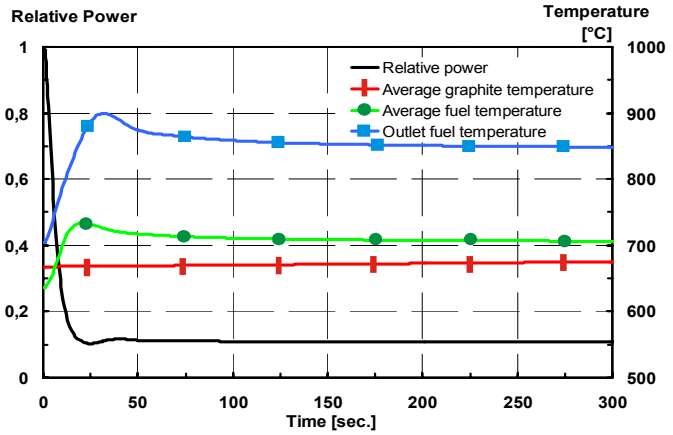

Fig. 8: MSBR response to the unprotected fuel pump trip. The power is sharply decreased because of the strong negative feedback coefficient of the fuel salt. (used library: EQ with Er)

In the case of 300pcm reactivity insertion the power increases rapidly in the very first moment. However, the increase is slowed-down soon and the power is again decreased by the introduction of negative reactivity. This reactivity is based on the rising temperature of the fuel, which has negative feedback coefficient. The long-term behavior of power is dependent on the graphite thermal feedback coefficient. The temperature of graphite follows the temperature of salt with a delay. If this warming-up introduces positive reactivity the power can grow up. On the other hand the power will be stable if the graphite feedback will be negative (Fig 9). The thermal feedback of graphite can be influenced by the erbium addition to the fuel salt. The EQ libraries with and without Er were used.

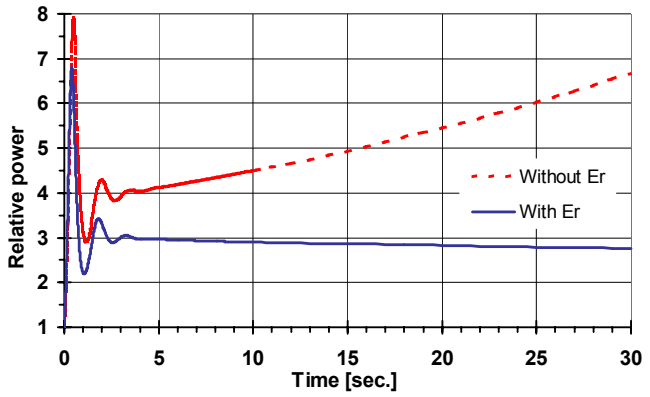

Fig. 9: MSBR response to the 300 pcm reactivity insertion.

The last transient considered was the overcooling of the fuel at the core inlet. The fuel temperature was decreased in this case by 50°C in 50 seconds. As mentioned above, the thermal feedback coefficient of the fuel salt is negative and so the decrease of fuel temperature introduces a positive reactivity and thus the power increases. It is interesting that the fuel outlet temperature is only increasing during all the transient. The changes of average fuel temperature are small and dependent on the graphite thermal feedback. Because the power grows during the overcooling transient, the radiation heat-up of the graphite also increases. The overall response of the reactor is dependent on the graphite feedback coefficient (Fig. 10).

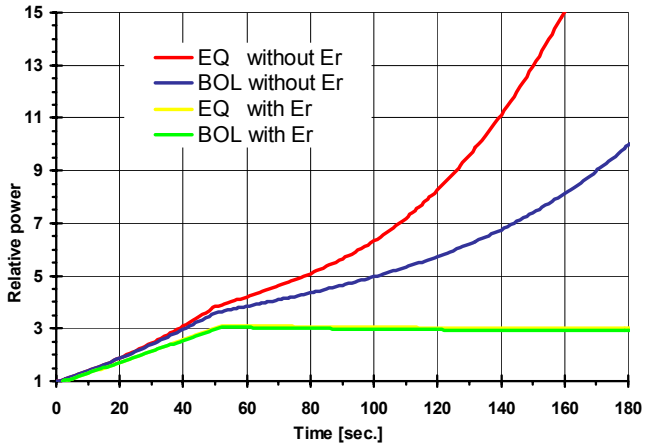

Fig. 10: MSBR response to the fuel overcooling by 50°C at the core inlet for four different cross-sections libraries.

CONCLUSION

DYN1D-MSR, the first step in a 3D version development, has been shown to be already an effective tool for MSR transient analysis. The new DN model, together with the TH model, was integrated into the nodal expansion method of DYN3D code neutron kinetics. The results of MSRE zero-power benchmark validated the applied models and proved the ability of the code to perform analysis for the liquid fuel reactor.

Three transients in MSBR were studied. The results of all transient studies have shown that the dynamic behavior of MSR is stable when the coefficients of thermal feedback are negative. However, the thermal feedback coefficient of graphite can be positive in some cases. In those transients the power is incessantly increasing. The addition of Erbium can help to eliminate the positive feedback coefficient of graphite.

The model developed for DYN1D-MSR will be in future combined with the full 3D neutron kinetics of the DYN3D code. The extension to 3D geometry involves the new model for delayed neutron mixing in primary circuit and the new thermal-hydraulic model. The new code DYN3D-MSR will be capable to cover the majority of all possible reactivity-initiated transients in MSR.

ACKNOWLEDGMENTS

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