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ANALYSIS OF HYPOTHETICAL UNPROTECTED LOSS OF FLOW IN SUPERPHÉNIX START-UP CORE: SENSITIVITY TO MODELING DETAILS

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SFR, Superphénix, ULOF, multi-channel model, TRACE

ABSTRACT

The Superphénix (SPX) Sodium Fast Reactor (SFR) core behaviour in a hypothetical Unprotected Loss Of Flow (ULOF) transient scenario is simulated and the impact of the core modelling options on the results are assessed. The analysis has been performed using the TRACE system code modified for the fast reactor applications (Mikityuk et al., 2005) using a point kinetics neutronics model and 1D multi-channel thermal-hydraulic model. The SPX start-up core benchmark specification (Ponomarev, et al., 2018) was used for calculation of neutron physics data employed as input in the TRACE simulations. The analysis covered transient simulation before sodium boiling onset. The influence of control rod drive lines (CRDL) expansion reactivity effect on the grace time before boiling was found to be significant. Series of calculations have been performed in order to assess sensitivity of the results to the core specification data and modelling details in both thermal hydraulics and neutronics inputs. In particular, two different models representing fuel subassemblies (three-channel and multichannel) have been employed. In addition, the influence of reactivity effects uncertainty on transient behaviour has been assessed, as well as sensitivity of the results to account for the spatial distribution of reactivity effects in the core. The results of this study serve further as a basis for the transient benchmark exercise proposed by Ponomarev, et al., 2018 within the framework of the Horizon-2020 ESFR-SMART project (Mikityuk, et al., 2017 and 2019).

1. INTRODUCTION

One of the most advanced and developed Generation-IV technologies (GIF, 2017) is Sodium-cooled Fast Reactor (SFR) technology, accumulated worldwide operational experience of more than 300 reactor-years, including experimental facilities and few industrial prototype reactors located in various countries (IAEA, 2012). High safety standards applied to such systems set specific requirements to validation of codes used for evaluation of their performance and behaviour during normal and abnormal operation.

The Horizon-2020 ESFR-SMART EU project was launched in 2017 in order to further develop the concept of the European Sodium Fast Reactor (ESFR) and, in particular, introduce and assess a number of new safety measures for this system (Guidez, et al., 2018). One of the particular goals of the project is to take an accurate look back to the legacy of SFR developments and experimental data available. Practically, it expected to benefit from the analysis of the largest ever-built SFR – French Superphénix (SPX) reactor. Renewed interest to the safety analysis of this reactor is raised due considerable amount of experimental data which constitutes a consistent basis for the codes validation for static and transient analysis. The benchmark study has been proposed within the ESFR-SMART project consisting of two phases: static neutronics and transient analysis (Ponomarev, et al., 2018).

Current study is linked to the benchmark by using its data for qualitative analysis of modelling options, applied for particular transient conditions, and contributes to better understanding of the SFR core behaviour in general. The focus of the current study is set to the sensitivity of the transient results to modelling details including two aspects: 1) details of representation of the core in the thermal hydraulic model and 2) uncertainties due to accounting for specific reactivity feedbacks. Thus it aims to contribute to preparation for the second (transient) part of the benchmark by assessing the options for modelling using the results obtained in the first (static) part of the benchmark.

In the current paper two SPX core models (multichannel and three-channel) were established and tested against each other in order to evaluate their consistency. A sensitivity study was conducted to help assessing the influence of the various reactivity feedbacks on the results.

At the next phases of the study, both the three- and multi-channel models will be used for the benchmark study of SPX start-up transients (Gourdon and Mesnage, 1990) using both point- and spatial-kinetics options.

The paper is structured as follows. After the given introduction the selected parameters of the SPX reactor and core designs are briefly provided in Section 2. Next, transient specification, codes and models are introduced in Section 3, while Section 4 presents the main results and discussion. Section 5 concludes the paper by summarizing the main findings.

2. SUPERPHÉNIX REACTOR

The SPX reactor with its about 360 fissile subassemblies (SAs) and about 5.7 tons of plutonium was the largest ever constructed liquid metal cooled fast breeder reactor in the history of nuclear energy production. Selected parameters of the reactor at its start-up configuration are given in Table 1.

Table 1. Selected parameters of SPX reactor (IAEA, 2012)

Parameter	Value
Thermal / electric power	2990 / 1242 MW
Average fissile / fertile fuel temperature	1227 / 627 °C
Primary sodium inlet / outlet temperature	395 / 545 °C
Primary sodium core flowrate	16400 kg/s
Fissile/fertile fuel	$(U,Pu)O_2$ / UO_2
Plutonium content in inner / outer core zones	16.0 / 19.7%
Total mass of plutonium in the fissile core	5780 kg
Volume of the fissile core	10.75 m3
Equivalent diameter of the fissile core	3.70 m
Height of the fissile pellet stack	1.00 m
Height of the lower/upper breeder blankets	0.30 / 0.30 m
Height of the radial blanket fertile pellet stack	1.60 m
Subassembly pitch	179.0 mm
Number of SAs in inner zone / outer zone / radial blanket	190/168/225
Number of control rods (CSD/DSD)	21 / 3

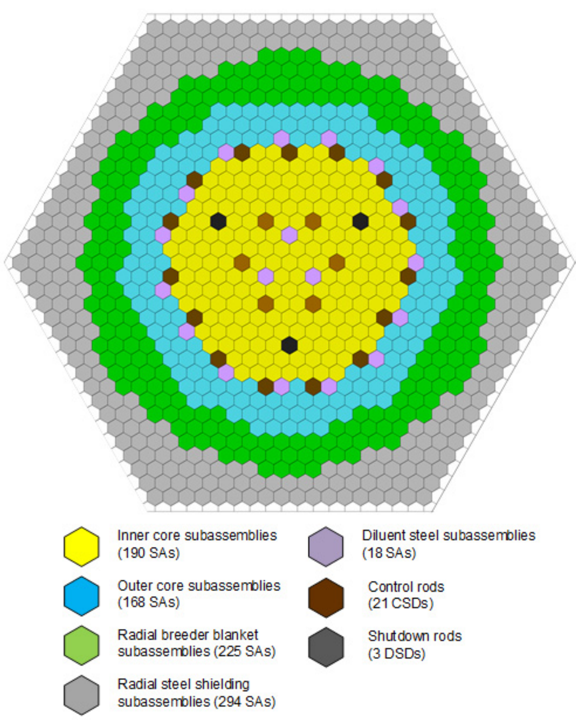

Fig. 1. SPX start-up core layout

The fuel zone of the core (Fig. 1) consist of two fissile zones with different Pu content for flattening the power radial profile: inner zone (IZ), outer zone (OZ), and radial breeder blanket (RB) with fertile fuel. Fissile SAs of IZ and OZ include upper and lower fertile blankets.

The core model was created using available open literature sources. Main parameters of the core, such as criticality level at different thermal states and selected reactivity feedbacks were found to be in good agreement with the experimentally measured values (Ponomarev, et al., 2018).

3. MODELLING

3.1. Unprotected Loss Of Flow transient specification

A primary aim of this study is to test different models by assessing the reactor core response to the hypothetical Unprotected Loss Of Flow (ULOF) transient assuming different levels of modelling details.

The transient study is simplified by considering only a part of the primary system. The core inlet sodium temperature is considered to be constant and corresponding to the nominal operating conditions (Table 1). This is an assumption simplifying the study considerably, justified by the limited duration of the transient. As result, a number of reactivity feedbacks related to differential expansion of primary system components (the vessel, diagrid plate and strongback) is neglected assuming no change of the inlet sodium temperature which is a driver parameter. This assumption is generally valid only for the initial period of the transient (about one minute) which is consistent with the simulation time considered in the study.

The ULOF scenario is defined by the mass flow rate variation in time. This change of the mass flow is dependent on the pump design and characterized by the mass flow halving time constant. The SPX reactor intentionally was equipped with a specific pump design that incorporated a specific flywheel connected to the pump driving motor shaft guarantying a slow deceleration, about 50 s for the reduction of half the rotation speed in the event of loss of the main power supply (Guidez and Prêle, 2017). This design allows to reach large grace time to boiling onset (few hundreds second). Nevertheless, in the given analysis the mass flow halving time constant was set to 10 s to be consistent with the modern ESFR design (Guidez, et al., 2018).

3.2. Codes used for simulations

The US NRC TRACE code modified at PSI in order to model fast reactor specific features is used in the study. The modifications include two-phase sodium flow option, fast reactor specific reactivity effects, new fuel performance model, *etc.* (Mikityuk et al., 2005). The TRACE code has been applied for analysis of different SFR designs in the past, see *e.g.* Lázaro et al., 2014 and Bubelis et al., 2017.

The neutronics parameters used for the transient calculations were obtained with Serpent 2, 3D continuous-energy Monte Carlo neutron transport code for reactor physics application being developed in VTT Technical Research Centre of Finland since 2004 by Leppänen et al., 2015. The JEFF311 nuclear data library was used in these calculations.

3.3. Thermal hydraulic core model

A simplified primary system model describes the core represented by multiple channels and employs specific boundary conditions at core inlet and outlet (Fig. 2).

In the multichannel model all 583 fuel subassemblies (190 SAs in IZ, 168 SAs in OZ and 225 SAs in RB) are represented by individual parallel channels as the most detailed simulation case.

A simplified model with three channels includes one individual channel for each of zones: IZ, OZ and RB. The channel represents a whole SA height of about 4300 mm from the inlet at diagrid plate up to the top of SA with the outlet shielding sleeve. The following axial sections of SA are modelled:

- the inlet section (empty hexcan);
- pin bundle represented by a single heat structure which corresponds to the whole pin length of about 2700 mm;
- upper transition section (empty hexcan);
- outlet shielding section.

Pin parameters of fissile and fertile SAs are given in Table 2. The channels are connected to the inlet and outlet plena. The sodium flow in the inter-subassembly gap is neglected. Inlet boundary condition is defined by providing the inlet sodium temperature and mass flow rate. Outlet boundary condition is defined assuming constant pressure in the outlet plenum.

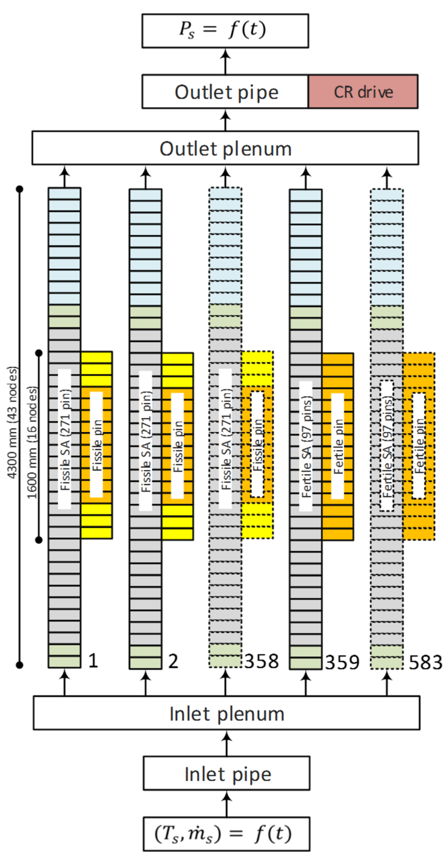

Fig. 2. Scheme of multi-channel core model in TRACE.

A simplified SA flow gaging scheme is established assuming three cooling groups which correspond to three core zones (IZ, OZ and RB). The cooling group mass flow rates are adjusted to achieve similar average sodium heat-up of about 145 K in IZ and OZ, while the sodium heat-up in RB subassemblies is set to somewhat lower value (up to 70 K in most powerful SAs). Additional heat structure is introduced in order to model the transient reactivity feedback related to thermal expansion of the control rod drive lines (CRDL). The main thermal hydraulic parameters of the model are given in Table 3.

Table 2. Fissile core and radial blanket pins parameters

Parameter	Fissile	Radial breeder
Hexcan outer flat-to-flat size, mm	173.0	173.0
Hexcan wall thickness, mm	4.5	4.5
Number of pins	271	91
Pin pitch, mm	9.8	16.9
Pin cladding outer diameter, mm	8.50	15.80
Pin cladding thickness, mm	0.565	0.57
Fissile fuel pellet diameter, mm	7.14	-
Fissile fuel pellet inner hole diameter, mm	2.0	-
Fertile pellet diameter, mm	7.07	14.36

Table 3. Main thermal hydraulic parameters of the core

Parameter	Value
Inlet temperature, K	673
Pressure drop on the core, bar	~4.5
Total sodium mass flow, kg/s	16400
Relative flow in cooling groups (IZ/OZ/RB), %	56 / 41 / 3
Core-average sodium heat-up, K	~145

In order to treat reasonably fuel-clad gap conductance the following correlation has been utilized:

$$h_{gap} = \min\left(3\left(10^3 - q_l + \left(\frac{q_l}{10}\right)^2 + \left(\frac{q_l}{10^2}\right)^3\right); 2.3 \cdot 10^4\right)$$

where h_{gap} is the gas gap heat conductance (W/m$^2\cdot$K) and q_l is the local linear heat generation rate (W/cm). The correlation was proposed by Lázaro et al., 2014 as result of a parametric study performed using the FRED fuel performance code (Mikityuk and Shestopalov, 2011). This correlation was obtained for the fresh-fuelled ESFR core from the CP ESFR project (Fiorini and Vasile, 2011) similar to the SPX core. The relevance of this correlation in ULOF transient conditions must be justified more accurately in future studies.

3.4. Point kinetics neutronics model

A point kinetics (PK) model of TRACE is utilized for the calculation of transient reactor power. To specify power distribution, reactivity coefficients and kinetics parameters for the PK model the static neutron physics simulation have been performed with the Serpent 2 Monte Carlo code employing the benchmark model, which includes detailed 3D description of the pin and subassembly geometry and composition (Ponomarev, et al., 2018).

The spatial power distribution is specified SA-wise in the multichannel model and for the three core zones in the 3-channel model for 16 axial nodes corresponding to the fuel height.

For calculating the specific reactivity coefficients the core configuration at hot zero power (HZP) conditions (at 673 K) is taken as reference for branch calculations considering different perturbed core configurations. The reference configuration is characterized by control rods (CRs) at the critical position.

The following reactivity feedbacks are of importance and have been considered in simulations (Table 4):

- Fuel Doppler effect;

- Sodium density (void) effect;
- Fuel axial expansion effect;
- Pin cladding expansion effect;
- Control rod differential position effect.

As it has been stated above, the reactivity effect related to differential thermal expansion of the structures of vessel and strongback and the diagrid radial expansion effect are not considered. Another specific effect, the subassembly pad effect, *i.e.* reactivity effect due to the thermal expansion of the pads between the hexcans leading to increase of the core diameter, is also not considered due to uncertainties associated to the modelling despite it may appear as a very efficient effect in case of a ULOF.

The Doppler effect is modelled considering the fuel isotopes temperature increase up to 1500 K. Individual contributions (5 in total) are derived for the fissile zones of IZ and OZ, lower and upper fertile blankets and RB.

Sodium density effect is modelled by variation of sodium density within the hexcan within the fuel height. Axial shape of the effect is obtained individually for IZ and OZ.

Fuel expansion is modelled assuming the fuel heat-up and corresponding "free" pellet stack elongation within the pin cladding (open gap regime as assumption for non-irradiated fuel). The fuel expansion coefficient is found using the average fissile fuel temperature.

The cladding expansion effect is found using the average cladding temperature and modelled assuming clad heat-up and thermal expansion, resulting in both axial elongation and increase of the outer pin radius.

The control rod differential position is defined in this work in a simplified manner by CRDL thermal expansion linked to the outlet plenum temperature (which strongly varies in transient) and fuel pellet stuck axial expansion introducing relative displacement of the fuel with respect to control rods.

4. RESULTS

4.1. Steady state core characterization

Power spatial distribution obtained by Serpent 2 calculation (Fig. 3) is supplied to the PK model of TRACE. The power shape accounts for CRs partially inserted into the core (by 40 cm at HZP). The detailed subassembly wise distribution is used in multichannel model while cumulative power of IZ, OZ and RB defines the powers of three-channel model. Peak values of SA power, fuel temperature and sodium heat-up characterizing two different models are collected in Table 5.

4.2. Reference transient simulations

The transient simulations have been considered with and without account for the differential insertion of CRs into the core. Two models are assumed to be consistent from viewpoint of representation of reactivity feedbacks. They use distributed reactivity coefficients (zone-wise, see Table 4), while axial profile for the sodium density effect individually for IZ and OZ zones (calculated with Serpent 2) is also considered. The results of the ULOF transient analysis are presented in Figs 4 to 6. According to the selected methodology the curves are presented up to the boiling onset.

Considerable influence of account for the CR-related feedback effect on power evolution is demonstrated for both models (Fig. 4). Fuel thermal expansion upward and CRDL thermal expansion downward cause insertion of an absorber and corresponding strong negative reactivity, resulting in a considerable increase of the grace time before boiling onset. Accordingly, due to different power-to-flow conditions the fuel temperature evolution results in the fuel Doppler reactivity (Fig. 5), which is becoming positive after 30 s of transient, as result of the temperature decrease. This sign change of the Doppler reactivity contribution has been observed for the CR differential worth value larger than about 3 pcm/mm (30% of the reference one), while for smaller values the fuel temperature constantly growing.

Table 4. Reactivity feedback parameters of the core used in PK model.

Reactivity feedback parameter	Value
Doppler constant (IZ/OZ//LAB*/UAB*/RB), pcm	-1135 (-757/-257/-54/-19/-28)
Sodium density coefficient (IZ/OZ), pcm/(kg/m^3)	0.992 (0.904/0.096)
Fuel expansion coefficient (IZ/OZ), pcm/C	-0.192 (-0.108/-0.084)
Clad expansion coefficient, pcm/C	0.05
CR differential worth, pcm/mm	10.0

*LAB/UAB – lower / upper axial blankets

Table 5. Parameters of the core at steady state calculated with two different models.

Parameter	Value	
	Multichannel model	Three-channel model
Peak SA power, MW	10.16 (IZ)	9.21 (IZ)
Peak fuel temperature, K	2560 (IZ)	2430 (IZ)
Peak sodium heat-up, K	185 (OZ)	150 (IZ)

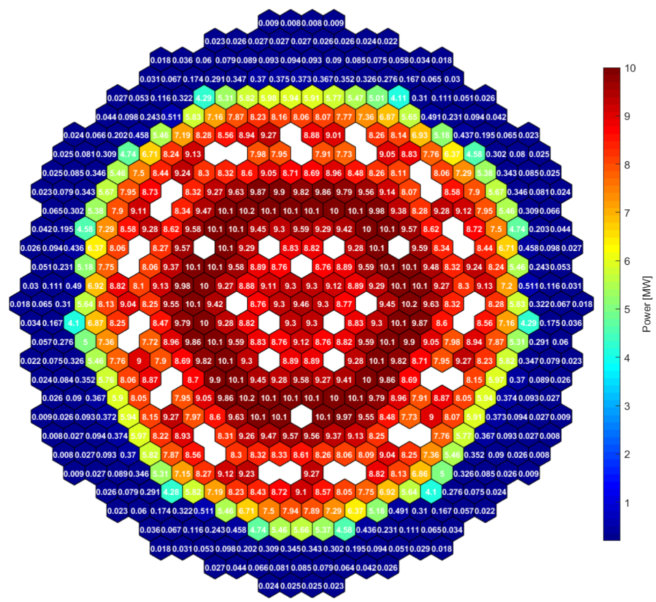

Fig. 3. SA power map of SPX start-up core as calculated with Serpent 2

As result of detailed power distribution and higher maximal power-to-flow ratio in the multichannel model the boiling occurs considerably earlier in transient as compared to three-channel model. This indirectly shows the importance of the multi-channel model for simulation of the sodium boiling stage of ULOF. Nevertheless, the overall evolution of transient power and reactivity for the single-phase stage of ULOF are consistent for two models (Fig. 5 and Fig. 6).

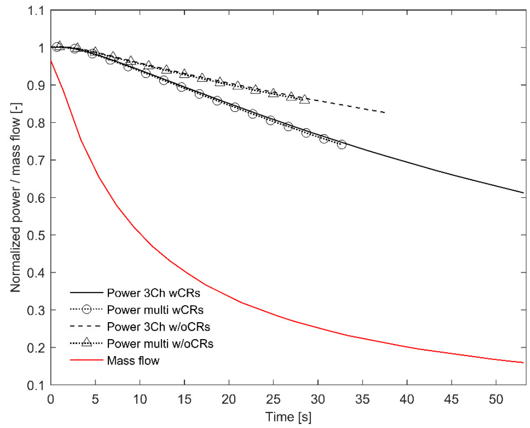

Fig. 4. Transient evolution of normalized power and mass flow for three- and multichannel models with and without account for CR-related feedback

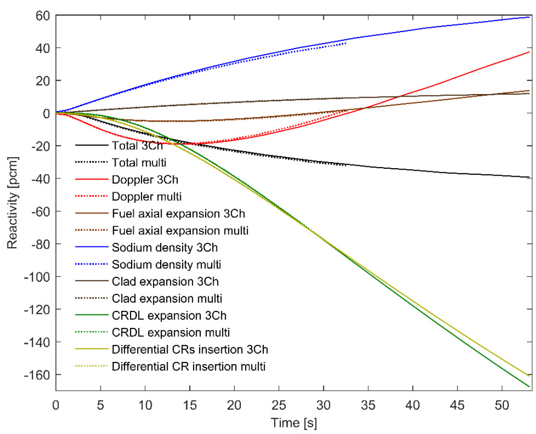

Fig. 5. Transient evolution of total reactivity and its components for three- and multichannel models with account for CR-related feedback

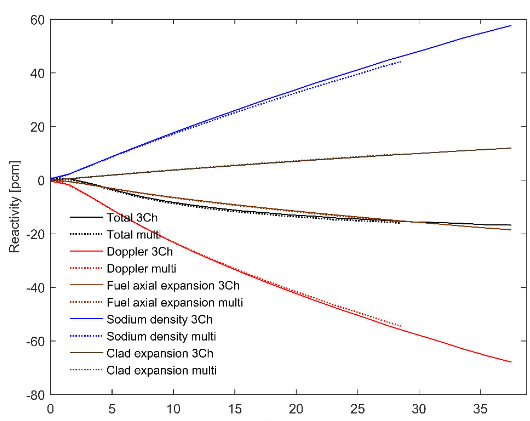

Fig. 6. Transient evolution of total reactivity and its components for three- and multichannel models without account for CR-related feedbacks

Practically it implies that the 3-channel model represent quite accurately the core behaviour assuming quite large variation range of state parameters, such as fuel and sodium temperature, for the transients with single-phase flow. In addition, it implies on good correspondence between two models from viewpoint of representation of reactivity feedbacks. Thus three-channel model may potentially serve as an appropriate basis for the transient analysis proposed in the benchmark study (Ponomarev, et al., 2018).

4.3. Sensitivity analysis for three-channel case results

The sensitivity analysis addresses two main aspects.

Firstly, the uncertainty in the results is assessed from viewpoint of the accuracy of evaluation of reactivity feedback coefficients. Prediction of reactivity coefficients depends on the neutron physics core model used in static neutronics simulations and since this model have been created on the basis of open literature sources, it may incorporate deviations from the realistic core design. Moreover, the results of evaluation strongly depend on the calculational methods and uncertainties in the used neutron cross section data.

Secondly, this sort of analysis may help to identify the potential differences in results related to simplification of the reactivity feedbacks representation in system codes, in practice, in those ones which utilize only one global value for each feedback coefficient being not capable to treat the spatial distribution of the effect in the core.

The results are basically evaluated by parameter of grace time to sodium boiling onset, which is of importance for the unprotected transient scenario.

The reference model is equipped with simplified representation of reactivity coefficients given as one global value for the whole core using core-averaged state parameters. The range from 80 to 120% was assumed as an ultimate uncertainty on evaluation of reactivity feedbacks and the following sensitivity cases have been calculated:

- Distributed sodium density coefficient;
- Sodium density effect decreased by 20%;
- Sodium density effect increased by 20%;
- Doppler constant decreased by 20%;
- Doppler constant increased by 20%;
- Fuel effect decreased by 20%;
- Fuel effect increased by 20%;
- CR differential position effect decreased by 20%;
- CR differential position effect increased by 20%.

The results are summarized in Fig. 7. The reactivity effect from differential insertion of CRs into the core (~16 mm in the considered cases at moment of boiling onset) has the strongest negative contribution into the total reactivity before boiling and is found to have the strongest impact on the boiling onset time. The ±20% variation of the corresponding coefficient results in about 10 s difference in the sodium boiling onset time. This effect depends on initial position of CRs which are considered to be inserted by about 30 cm at considered Hot Full Power (HFP) configuration at transient start. Variation of its worth within range from 80 to 120% correspond to CRs initial position from 20 to 40 cm. This effect should play larger role in transients where differential expansions of primary system components (CDRL, vessel, strongback and diagrid) are not mutually compensated and lead to a considerable change of the CR position with respect to the initial position in the core, in particular, in scenarios with nearly constant inlet temperature, as in the studied ULOF case.

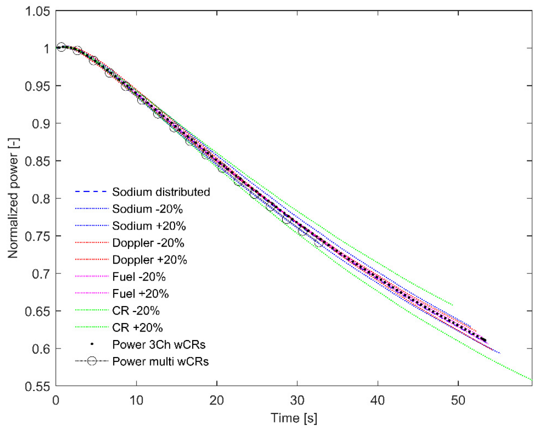

Fig. 7. Transient evolution of normalized power for three-channel model with account for CR-related feedbacks for various sensitivity cases

The uncertainty of sodium, Doppler and fuel expansion reactivity effects individually demonstrates a considerably smaller influence on the ULOF transient results (spread of boiling onset time is less than 5 s), while the power evolution stays close to the reference case. The smallest influence is observed when the distributed sodium density coefficient is considered, providing error of less than 0.5 s. Practically it implies that for considered transient framework one global (core-average) value for the sodium density reactivity coefficient is appropriate for reasonably accurate analysis of the transient behaviour. These considerations provide basis for definition of the benchmark transient exercise, implying that this simplified representation of the core reactivity feedbacks is reasonable. Practically, the inaccuracies of pre-calculated reactivity effects result in similar uncertainty of power evolution, as it has been observed for comparison with multichannel model simulations. In addition, cumulative uncertainty of few effects would result in much stronger difference in power evolution, than one observed between results of three- and multichannel cases.

5. CONCLUSIONS

The sensitivity to the modeling option was studied for the SPX SFR core behaviour in the single-phase stage of the ULOF transient using the SFR-specific version of the TRACE code. The modeling options considered are: 1) multi-channel representation of the core versus three-channel representation; 2) variation of the reactivity coefficients within the assumed uncertainty (±20%). The main findings of this analysis related to the single-phase stage of the ULOF transient are that the use of the three-channel model and of the single-value coolant density reactivity coefficients was found to be acceptable for the analysis. These findings will be used in the second phase of the SPX start-up test benchmark.

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