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Investigative Study of Neutronic Safety Parameters of HPR and EPR Using the MCNP Code

R. G. Abrefah,^{a,b,*} P. M. Atsu,^a and R. B. M. Sogbadji^{a,c}

^aUniversity of Ghana, Graduate School of Nuclear and Allied Sciences, P.O. Box AE1, Atomic Energy, Accra, Ghana

^bGhana Nuclear Regulatory Authority, P.O. Box AE 50, Atomic-Kwabanya, Ghana

^cGhana Atomic Energy Commission, Nuclear Power Institute, P.O. Box LG80, Legon, Ghana

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Abstract — *As part of technology assessment of proposed commercial nuclear power reactor technologies for Ghana's Nuclear Power Programme, the neutronic safety parameters of the European Pressurized Reactor (EPR) and High Temperature Pressurized Reactor (HPR) reactor technologies are theoretically analyzed and compared. The MCNP neutronic code was employed as a computational tool to analyze the reactivity temperature coefficients, moderator void coefficient, criticality, and neutron behavior at various operating conditions. The HPR, which is still under construction and under theoretical safety analysis, showed good inherent safety features comparable to the already existing EPR technology.*

Keywords — *Reactivity temperature coefficients, moderator void coefficient, criticality, neutron behavior.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

Safety analysis of reactor systems continues to be crucial in reassuring the public of the safety and reliability of nuclear power plants. Safety analysis is also crucial in aiding countries to make decisions on the types of reactor systems to build and also to incorporate the lessons learned from nuclear accidents into safety analysis for added safety assurance.¹ To provide the safety analysis of the reactors under all operational conditions, knowledge of changes in reactivity caused by changes in void content and temperature are necessary. Reactivity coefficients are important for reactivity and power excursion transient analysis. The sign, rate of the change, response time, and magnitude of reactivity coefficients are of great importance.²

In nuclear power reactors, changes to these parameters and operating conditions mentioned earlier influence the reactivity of the reactor in different ways. The total reactivity coefficients consist of reactivity changes due to fuel

temperature changes, moderator temperature changes, and changes of void content. For a pressurized water reactor (PWR) working in steady state, the changes in reactivity due to temperature changes are the most significant. The changes in reactivity due to void content become very important when simulating loss-of-coolant accident situations.

II. DESCRIPTION AND COMPARISON OF REACTORS UNDER CONSIDERATION

The European Pressurized Reactor (EPR) and High Temperature Pressurized Reactor (HPR) are both Generation III four-loop PWRs. The EPR is a Generation III+ evolutionary four-loop PWR jointly developed by Framatome and Siemens through their subsidiary Nuclear Power International and received technical support from the French utility Electricité de France as well as from the German utilities who financed most of the development work.³ The HPR-1000 was designed by the China Zhongyuan

*E-mail: adibola2001@yahoo.com

Engineering Corporation under the supervision of the China National Nuclear Corporation.⁴ The main components of both designs, the loop configuration, and the primary system designs are not any different from operating PWR designs around the world.

The distribution and configuration of the fuel assemblies and fuel pins and their enrichment in the reactor core significantly affect neutronics and safety when operating the reactor. The EPR core is built up by 241 mechanically identically designed fuel assemblies. Each fuel assembly consists of 264 fuel rods and 25 guide tubes arranged in a 17×17 array with active height of 4.20 m. The fuel rods are made of Zircaloy tubing containing uranium dioxide ceramic pellets, the initial enrichment of which is below or equal to 5.0 wt% (Ref. 5). The core is designed for UO_2 fuel assemblies and incorporates the capability also to insert mixed-oxide fuel assemblies up to about 50%. The average linear heat generation rate is about 178.6 W/cm, giving the prospect of achieving average batch burnups of up to 65 GWd/tonne U (Ref. 5). Basic safety objectives are met by designing the core to have stabilizing reactivity coefficients under all operation conditions. Reactivity control is accomplished by changing the boron concentration in the primary coolant and by moving control assemblies. Slow reactivity changes caused by changes of xenon concentration and burnup are compensated by changes of the boron concentration, while fast reactivity changes for adaptation of the power level are compensated by control rod insertion or withdrawal. Some fuel assemblies contain burnable absorber (Gd_2O_3) to suppress high excess reactivity, especially in the first core.⁵

The HPR-1000 reactor core generates 3050 MW (thermal) of thermal power with an average linear power density of $173.8 \text{ W}\cdot\text{cm}^{-1}$ (Ref. 6). The reactor core is loaded with 177 China Fuel Series (CF3) fuel assemblies, ensuring sufficient thermal margin while increasing output power. The CF3 fuel assembly is composed of 264 fuel rods arranged within a 17×17 supporting structure. The fuel rods contain UO_2 pellets or Gd_2O_3 - UO_2 pellets.⁶ Zircaloy is used as cladding material for the fuel pins. The CF3 has excellent performance and is applicable for a long refueling cycle. Three independent means exist for core reactivity and power distribution control: burnable absorber of gadolinium (Gd_2O_3) poisons, rod cluster control assemblies (RCCAs), and soluble boron absorber. The RCCA comprises 24 control rods fastened to a spider connector. The absorber material used in the control rod is Ag-In-Cd alloy or stainless steel. The HPR-1000 is designed with a thermal margin greater than 15% to improve safety and operational performance.⁶

TABLE I
Fuel Assembly Design Specification

Parameter	Dimension (mm)	
	EPR	HPR
Fuel pin diameter	9.5	9.5
Cladding thickness	1.14	1.14
Gap	0.16	0.168
Fuel part diameter	8.2	8.192
Step between rods	12.6	12.6
Length of active height	4199.99	3658
Number of fuel rods	265	265

The difference in fuel and core design specifications is as shown in Table I.

III. THEORY

III.A. Void Coefficient of Reactivity

In water-moderated reactors, changes in moderator density significantly affect the reactivity. Changes in moderator density can be due to thermal expansion, void formation, or loss of coolant. A change in the moderator void content leads to a change in multiplication factor k and alters the reactivity of the system. The void coefficient of reactivity is therefore defined as the rate of change in the reactivity of a water-moderated reactor resulting from any modification of the moderator/coolant as the power level and temperature change. The principal effect is the loss of moderation that accompanies a decrease in moderator density and causes a corresponding increase in resonance.⁷ For PWRs, about 80% of neutron moderation occurs in the light water moderator.⁷

For a given value of k_∞ , the reactivity ρ in the core is determined by the expressions:

$$\rho = \frac{k_0 - k_\infty}{k_0} \quad (1)$$

and

$$\Delta\rho = \rho_0 - \rho_1, \quad (2)$$

where k_∞ is the multiplication factor at the present reactor operating conditions and k_0 is the multiplication factor at the normal reactor operation conditions. Similarly, ρ_0 is the reactivity at the normal operating conditions, and ρ_1 is

the reactivity at the present reactor operation conditions. From the definition of the void coefficient of reactivity above, the void coefficient of reactivity γ_ξ is mathematically given as

$$\gamma_\xi = \frac{\Delta\rho}{\Delta\xi} , \quad (3)$$

where ξ represents the reactor parameter affecting the reactivity and $\Delta\rho$ represents the corresponding change in reactivity. If ξ represents void, then the change in void is $\Delta\xi$, and the void coefficient of reactivity is defined by γ_ξ .

III.B. Temperature Coefficient of Reactivity, α

The influence of temperature on the neutron transport is caused by the thermal movement of nuclei influencing the scattering of thermal neutrons and the Doppler broadening of resonances, which is due to variation in neutron cross section with temperature and by thermal expansion of different materials within the core. There are two main temperature coefficients that are defined with respect to which temperature change is considered: fuel temperature reactivity coefficient and moderator temperature reactivity coefficient. Reactivity changes associated with a degree change in the moderator temperature are referred to as the moderator temperature coefficient of reactivity.⁸ The value of the temperature coefficient is determined from its definition by simply dividing the change in reactivity $\delta\rho$ due to the change in temperature by the corresponding change in temperature δT (Ref. 9):

$$\alpha = \frac{\delta\rho}{\delta T} . \quad (4)$$

The temperature coefficient of reactivity α has different effects on reactivity in the core:

1. a nuclear temperature coefficient arising from a change in cross section with changing neutron temperature
2. a density temperature coefficient arising from a change in temperature
3. a volume temperature coefficient arising from a change in geometric buckling when temperature changes.

The total temperature coefficient of reactivity is given by the sum of the moderator temperature coefficient and the fuel temperature coefficient:

$$\alpha = \left(\frac{\delta\rho}{\delta T} \right)_f + \left(\frac{\delta\rho}{\delta T} \right)_m \quad (5)$$

$$= \alpha_f + \alpha_m . \quad (6)$$

The reactivity change is given by Eqs. (1) and (2) in Sec. III.A.

IV. MODELING AND REACTIVITY COEFFICIENT CALCULATIONS

To calculate the reactivity coefficients of the core of the reactors, a detailed three-dimensional computational model of the HPR and EPR assemblies was developed using the Monte Carlo simulation code MCNP5. The assemblies of the reactor cores under investigation were modeled with reflective surfaces to mimic an infinite medium. A reflective modeling of an assembly enables the calculation of $k_{inf}(k_\infty)$ since the leakage term has been eliminated in the simulation. The code utilized ENDF/B-VII as the cross-section library for the materials in these computations. The fuel assemblies were modeled to include all 265 fuel pins in a square lattice and the moderator/coolant for each reactor core. Materials and their various compositions were specified in the code. The significant difference in the HPR and the EPR MCNP model is the moderator fuel ratio. The EPR has a larger moderator fuel ratio due to its smaller pin size. The MCNP plot of the HPR and EPR fuel assemblies is shown in Fig. 1.

In calculating the effects of void on the reactivity in the reactor core, the developed MCNP input model was modified to have different moderator densities to depict increasing void content while other conditions in the core were kept constant. The core temperature was also varied while other conditions were kept constant in order to determine the effect of changing temperature on the reactivity in the core of the reactors.

These calculations were carried out with a total number of 550 cycles of iteration on a source size of 500 000 particles per cycle. The first 50 cycles were skipped to decrease statistical errors in the estimates.

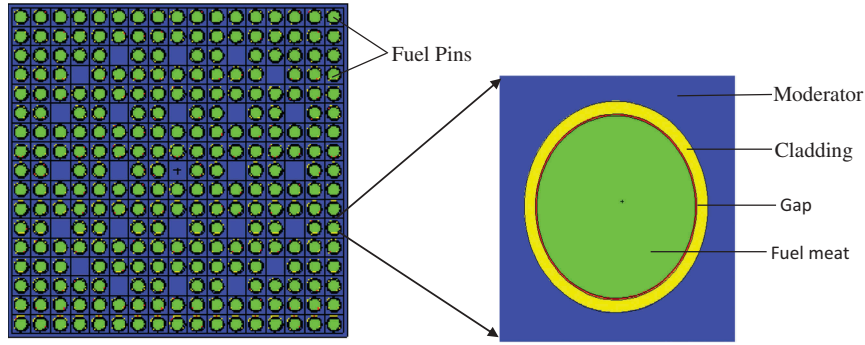


Fig. 1. MCNP plot of HPR and EPR fuel assemblies.

The infinite multiplication factor k_{∞} was obtained for each run from the respective output to calculate the corresponding change in reactivity:

$$\% \text{ Change in Criticality} = \left\{ \frac{(k_{\infty})_o - (k_{\infty})_i}{(k_{\infty})_o} \right\} \times 100\% , \quad (7)$$

where $(k_{\infty})_o$ is the criticality at no moderator void and $(k_{\infty})_i$ is the criticality at a specific moderator void i .

V. RESULTS AND DISCUSSION

When the quantity of void in the core is increased, the moderator expands creating more gaps between the water molecules and hence reducing the probability of thermalization of the fast fission neutrons. This effectively reduces the number of thermalized neutrons that can cause fission thereby reducing the reactivity in the core. From Fig. 2, both the HPR and the EPR cores show similar moderator void reactivity and have desirable moderator void reactivity at all stages of moderator void fraction. The calculated void coefficients of reactivity calculated at different void fractions for both reactors are given in Table II.

Figures 3 and 4 show the neutron behavior at various stages of varying void content of the HPR and EPR, respectively. The MCNP neutron energy spectrum was performed for 20 484 energy grids combined for all three categories of the energy distribution: thermal, slowing down, and fast.

The following energy bins were used in the MCNP tally for the various energy groups: 1.89×10^{-08} MeV energy bin for (0 to 6.25×10^{-07}) MeV thermal energy range, 1.89×10^{-03} energy bin for (0.821 to 6.94) MeV slowing-down energy range, and 1.89×10^{-03} MeV energy bin for (6.96 to 20) MeV fast energy range.

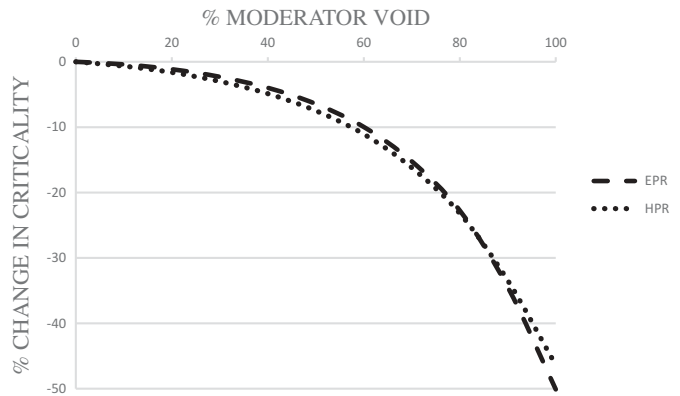


Fig. 2. Moderator void coefficients for HPR and EPR, comparable to Ref. 9.

TABLE II
Moderator Void Coefficients of Reactivity at Different Void Fractions

Void Range (%)	Moderator Coefficient of Reactivity, α_m	
	EPR	HPR
0 to 10	-0.04073501	-0.067942041
10 to 20	-0.07700235	-0.100794371
20 to 30	-0.12252846	-0.146902789
30 to 40	-0.18242773	-0.207326331
40 to 50	-0.27973985	-0.303010506
50 to 60	-0.42717926	-0.444101632
60 to 70	-0.70283506	-0.702349479
70 to 80	-1.17127094	-1.107637638
80 to 90	-2.36255624	-2.047544738
90 to 100	-5.42159943	-4.114301779

The spectrum graph in arbitrary neutron flux density refers to a normalization of flux density of 1.0. The relevance of the spectrum is to ascertain that the reactors are safety inherent. Hence, the real values of the flux do not change the shape of the spectrum.

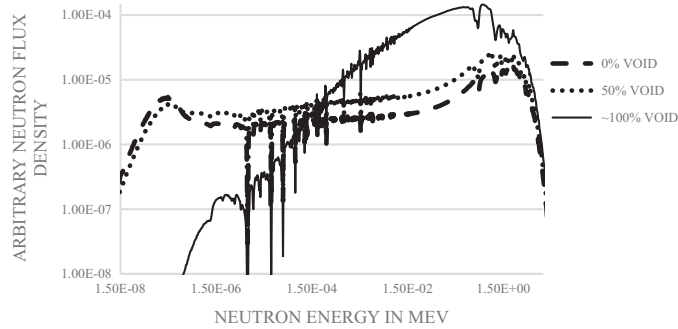


Fig. 3. Neutron spectra of different percentages of loss of coolant for HPR assembly, comparable to Ref. 10.

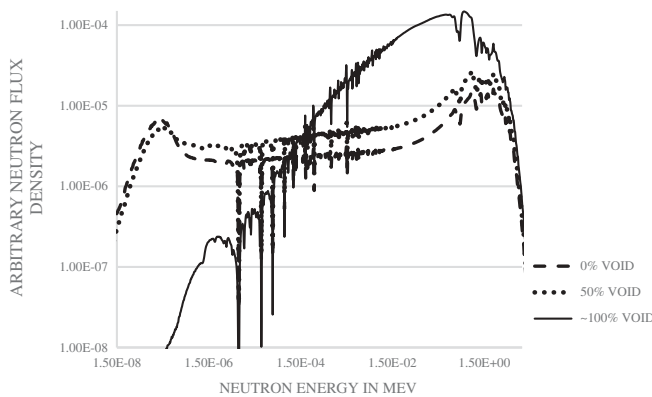


Fig. 4. Neutron spectra of different percentages of loss of coolant for the EPR assembly, comparable to Ref. 10.

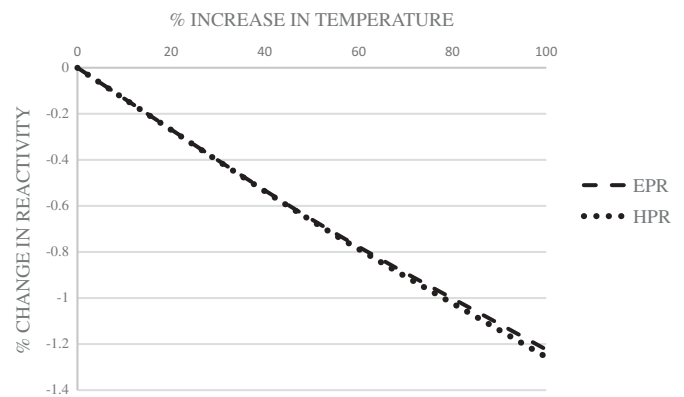


Fig. 5. Moderator temperature coefficients of EPR and HPR reactor assemblies, comparable to Ref. 10.

As shown in the spectra above, there is significant loss of thermal neutrons as the void content increases. The loss of thermal neutrons is associated with loss of light elements in the core as a result of loss of coolant, hence the loss of thermal neutrons to fission with the fissile isotopes to sustain the chain reaction. This puts the reactor in a subcritical condition, and it is an important safety feature in PWRs.

The moderator temperature coefficient of all the reactor assemblies considered is desirable since the reactivity decreases with increasing moderator temperature as shown in Fig. 5. An increase in the moderator temperature makes the core undermoderated due to the increase in energy of the lighter nuclides that can cause moderation. This results in negative reactivity being added to the core as shown in Table III. Nuclei cross sections are energy specific; hence, a change in the energy of the nuclei changes the probability of interaction as shown in Fig. 6. In this case, increasing temperature does not necessarily reduce the probability of nuclei interaction since the decrease in the resonance peak height is compensated for by the broadened

width. An undermoderated reactor gives a negative moderator temperature coefficient while an overmoderated reactor will give a positive moderator temperature coefficient.

VI. CONCLUSION

Both reactor designs are under consideration for Ghana’s nuclear power plant, and as such, this study was carried out as part of technology assessment for all reactor designs under consideration. The study focuses on neutronic parameters, which is an important indicator of safety in nuclear reactors. In this work, the criticality and the reactivity changes of the HPR and EPR at various operation conditions were analyzed and compared. The effects of increasing void fraction and increasing core temperature on the reactivity of the reactors and the associated coefficients of reactivity were calculated. Modeling and simulation of the fuel assemblies of the reactors were carried out using the MCNP5 neutronics code. The MCNP code was used to

TABLE III
Change in Criticality at Different Moderator Temperatures

Temperature (K)	Percent Temperature Change	EPR		HPR	
		k_{∞}	Percent Change in k_{∞} , $\left\{ \frac{(k_{\infty})_o - (k_{\infty})_i}{(k_{\infty})_o} \right\} \times 100\%$	k_{∞}	Percent Change in k_{∞} , $\left\{ \frac{(k_{\infty})_o - (k_{\infty})_i}{(k_{\infty})_o} \right\} \times 100\%$
600	0	1.42997	0	1.42668	0
900	50	1.42055	-0.65876	1.41719	-0.66518
1200	100	1.41247	-1.2238	.40875	-1.25676

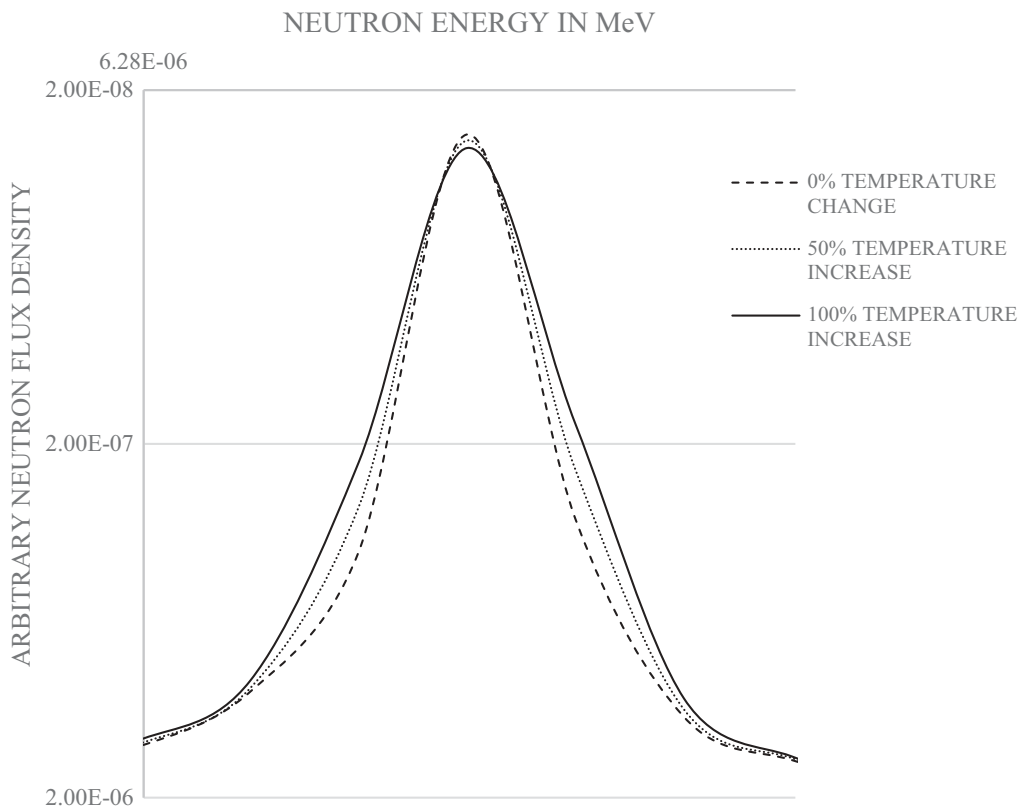


Fig. 6. Doppler broadening effect in EPR assembly (zooming on the first resonance peak).

determine the k_{eff} of the cores at the different operating conditions. Calculations and comparison of the void coefficient of reactivity and the moderator temperature coefficients of reactivity of the HPR-1000 and the EPR-1000 are the main tasks of this study. Both the HPR and the EPR showed good inherent safety of -0.0126 and -0.0122 for temperature, respectively, which is a desired design safety feature of PWRs.

References

1. O. MOTWENDI, "Neutronic Simulation of a European Pressurized Reactor," North-West University, Potchefstroom (2014).
2. F. FAGHIHI, A. FADAEI, and R. SAYAREH, "Reactivity Coefficient Simulation of the Iran VVER-1000 Nuclear Reactor Using WIMS and CITATION Codes," *Prog.*

- Nucl. Energy*, **49**, 1, 68 (2007); <https://doi.org/10.1016/j.pnucene.2006.09.003>.
3. “US EPR Brochure,” Framatone ANP/AREVA (2012).
 4. “HPR-1000 Technology,” China National Nuclear Corporation (2015).
 5. H. HERNANDEZ-LOPEZ and M. A. LUCATERO, “Linear Heat Generation Rate for Breaking Plastic Strain Limit of Cladding in a BWR Fuel Rod Type,” *Ann. Nucl. Energy*, **90**, 234 (2016); <https://doi.org/10.1016/j.anucene.2015.12.005>.
 6. J. XING, D. SONG, and Y. WU, “HPR1000: Advanced Pressurized Water Reactor with Active and Passive Safety,” *Engineering*, **2**, 79 (2016); <https://doi.org/10.1016/J.ENG.2016.01.017>.
 7. R. KHAN, M. VILLA, and H. BOCK, “Monte Carlo Modelling of Void Coefficient of Reactivity Experiment,” p. 1, Vienna University of Technology (2011).
 8. R. J. LAMARSH, Introduction to Nuclear Engineering, Addison-Wesley Publishing Company, New York (1982).
 9. O. SAFARZADEH, F. SAADATIAN-DERAKHSHANDEH, and A. SHIRANI, “Calculation of Reactivity Coefficients with Burn-Up Changes for VVER-1000 Reactor,” *Prog. Nucl. Energy*, **81**, 217 (2015).
 10. R. B. SOGBADJI, “Neutronic Study of Th Mono-Recycling of Americium in PWR and the Core Conversion in MNSR Using the MURE Code,” L’Université Paris-Sud (2012).