# Effect of strong N-TH coupling On core DesIGN calculations based on A TYPICAl 100 MWe integral PWR design

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**Abstract**

New generations of nuclear reactors including Small Modular Reactors (SMRs) have received a lot of positive attention. Due to the unique features (improvement in economic and safety features) of SMRs, these reactors are under different stages of design and construction all over the world. A typical 100 MWe integral PWR design reactor has been selected as the base case of the calculations in the present study. Different Thermo-Neutronic aspects of the reactor have been investigated using coupling of nuclear codes. DRAGON/DONJON codes have been used for neutronic cell and core calculation and COBRA code has been used for thermal-hydraulic calculations. Different parameters of the reactor core such as axial and radial PPFs, critical heat flux (CHF), minimum departure from nucleate boiling ratio (MDNBR), axial average coolant temperature in hot channel and core, and the maximum fuel temperature have been investigated using the strong coupling system. Some of the results have been checked by code-to-code verification which shows good agreement. Also, the coupling results demonstrate the non-negligible effects of the coupling in comparison to the stand-alone code modeling.

## INTRODUCTION

The majority of reactor core design relies on simulation techniques to avoid the significant cost and time required for other approaches based on laboratory and experimental processes. Therefore, the application of core simulation techniques is one of the most important factors in ensuring nuclear reactor safety and increasing their efficiency. A broad range of software tools should be used to model the complex, multi-physics, and multidimensional aspects of a reactor core. However, many of these software tools (codes) are interdependent, which means that the result of one code may be used as the input of another. The neutronic calculation is one of the basic calculations used in the nuclear reactor core design. The objective of a reactor's neutronic core design is to accurately predict the core behaviour during operation. Neutronic calculations rely on the nuclear data (very fine and detailed data about the probability of neutron reaction with the target nucleus: cross-section). The cross-section in nuclear data has a very small energy resolution, which is enough to produce complex behaviours such as resonance. Furthermore, a thorough analysis of thermohydraulic parameters is necessary for the safe functioning of the nuclear reactors. In addition to other accessible parameters, several core thermohydraulic characteristics, such as maximum fuel temperature, critical heat flux (CHF), and minimum deviation from nucleate boiling (MDNBR), can be employed as objective functions to optimize the reactor core and improve safety margins. On the other hand, only when the appropriate distribution of temperature and density of coolant, cladding, and fuel is provided, accurate neutronic computations of nuclear reactor cores could be performed.

Only thermohydraulic or neutronic codes have been utilized independently in various computations and studies, and only their average impacts have been seen while carrying out calculations. However, simultaneous investigations and studies of thermohydraulic and neutronic effects have been carried out in several previous simulations [1-2]. The primary objective of the current study is to compare the results of standalone neutronic or thermohydraulic calculations based on simulation to those of interconnected and simultaneous calculations utilizing the DRAGON/DONJON/COBRA programs [3].

Small modular reactors are categorized as the new-generation reactors, which are in various stages of design and construction in many countries due to their advantages [4]. The SMRs have unique design features that are under investigation and development as a new topic in the field of nuclear reactors. A typical 100 MWe integral PWR design has been used as a case basis to perform these calculations [5]. In line with this reactor simulation, many computations have been used for optimization, accident simulation, use of thorium fuels, etc., but in the present research, code coupling has been used to perform calculations. In the continuation of this article, at first, the specifications and features of the reactor core are reviewed, and then the codes used to perform various cell, core, and thermohydraulic calculations, as well as how to couple them are presented. In the end, the calculation results are presented separately and compared with each other.

## METHODS and materials

### Reactor characteristics

It is assumed that, the core of a typical 100 MWe integral PWR design as an integrated small modular reactor produces 0.36 GW thermal energy. The primary objective in developing such a reactor is to enhance safety and optimize economic efficiency. Passive methods are employed to enhance the natural safety features. The financial effectiveness has been enhanced by the reduced complexity of the system, a modular approach of the sub-systems, reduction of building time, and improved accessibility of the unit. The reactor possesses a unified design, with all of its key parts housed within a pressure vessel, as depicted in Fig. 1. Utilizing the unified system obviates the necessity for extensive pipelines, hence enhancing the reliability of the power facility [4]. It could possess the necessary capacities to fulfil the requirements for electrical energy production, heating, and desalination of water in small, isolated regions or those disconnected from the national power grid, with a population of approximately 100 thousand individuals. The projected lifespan of this reactor is 60 years, with a fuel replenishment interval of three years [5].



*FIG. 1. Main components of a typical 100 MWe integral SMR.*

### N-TH calculations

Various neutronic and thermal hydraulic codes are utilized for the purpose of design and evaluation, as previously mentioned. The DRAGON code is employed in this study for the purpose of conducting neutronic cell calculations. This tool carries out the computation of transport and diffusion equations in order to derive equivalent cross sections. The present paper employs the DONJON code's approach to solve the diffusion equation, employing various methods. Next, the DONJON code is applied to simulate a typical 100 MWe integral PWR reactor core consisting of 57 fuel assemblies. The DONJON input utilizes homogeneous cross-sections generated from the DRAGON code, which accounts for various fuel assemblies and reflectors. In this research, the thermal-hydraulic calculations were conducted with the COBRA code, which is a customized flow channel technique specifically intended for doing computations within nuclear reactor cores. A number of thermal-hydraulic factors associated with the core of the reactor have been gathered and utilized as input for the COBRA [3]. These variables comprise of the linear power calculated from the DONJON code output, channel number, control volume number in axial direction, the rate of cross-flow between flow channels, the wetted surface, and etc.

### N-TH simple and strong coupling

In order to gain a comprehensive understanding of the nuclear reactor core's behaviour, it is necessary to perform thermal-hydraulic and neutronic calculations concurrently. This is because the behaviour of these two domains is intricately interconnected and dependent on each other. The nuclear cross-sections of reactions are strongly influenced by temperature and density. These cross-sections play a crucial role in determining power, which subsequently affects the temperature and density fields. Consequently, calculations in both sectors are interdependent, and doing separate calculations in each domain results in calculation inaccuracies. In this paper, the mentioned codes have been merged to achieve accurate thermal properties fields for extracting cross-section and utilizing them in core calculations. Both simple one-way calculation and strong coupled calculation are performed in this study to reveal the effect of strong coupling on core design and evaluation parameters. The sequential procedure for merging these codes is as follows:

1. extraction of material properties and geometry dimension of the reactor core
2. Preparation of code input files.
3. Flow and temperature filed initial guess is made.
4. Transfer initial guess to be used in DRAGON code.
5. Macroscopic sections are extracted from the DRAGON code
6. Input of the DONJON code is established.
7. Neutronic core cell calculations are performed by DONJON.
8. Maximum axial and radial power peaking factors are extracted.
9. the linear power is extracted and mapped as the input of COBRA code.
10. Thermal-hydraulic calculations are performed by COBRA.
11. The convergence is checked
12. Above stages are iterated to reach convergence criteria.

## Result

The thermal-hydraulic and neutronic fields have been studied and evaluated independently to each other in various investigations. The current study assesses these solutions for the a typical 100 MWe integral PWR core through simple one-way coupling and strong coupling schemes to emphases the necessity and importance of using strong coupled methods.

### Validation

Based on the computational tools used in this study, this part explains the validation procedure of the coupling approach to perform thermal-hydraulic and neutronic calculation of the reactor core. Before doing any additional calculations, it is crucial to confirm that the geometry, input data, and other models used for the reactor core computations have been produced correctly. In this part, power peaking factor extracted from DRAGON/DONJON calculation as a deterministic tool is compared with result obtained from the OpenMC probabilistic tools. In these two distinguished codes, the comparison indicates an acceptable maximum relative difference of 2.4% and an average relative difference of 1.5% for PPF values. This difference is caused due to the various computation techniques employed in the probabilistic and deterministic codes, the slightly different cross-sectional libraries, and the inability of completely accurate reflector modelling in the deterministic codes. The comparison shows that for PPF values, there is an allowed maximum relative gap of 2.4% and an average relative gap of 1.5% in these two distinct approaches. This discrepancy results from the differing calculation strategies utilized in the deterministic and probabilistic codes, the different cross-sectional libraries, and the deterministic systems' incapacity to simulate reflectors with perfect accuracy. The outcomes of the thermal-hydraulics calculations performed with the COBRA code is compared with PARCS code's thermal-hydraulics module [6]. Fig. 2 compares the axial temperatures profile of fuel obtained using these two codes. it demonstrates that the thermal-hydraulics results obtained through these two techniques are rather identical. The differences in the average fuel temperature between these two codes are reflected in the different averaging procedures employed in each. The coupled technique yielded an average temperature of 324 °C for the core of the reactor. Comparing this finding to the 323 °C stated in the reactor's SAR, the relative error is in acceptable range.

*FIG. 2. Fuel temperature profile obtained from COBRA and PARCS TH modules*

### 3.2. Comparison of simple and strong coupling methods

To promote faster convergence, the average temperature of the reactor core is used as an initially estimate in the strong coupling calculations. The SAR of this reactor determined the effective multiplication factor at the BOC equal to 1.005724, without using the coupling process. Using strong coupled calculation this value decreases to 1.000593. The importance of strong coupling is highlighted by this noteworthy variance of 513 pcm. The PPF values at BOC for cases with simple one-way calculation and strong coupling calculation is presented in Table 1. The difference between PPF value of both methods is illustrated in Fig. 3. The axial PPF profile is depicted in Fig. 4. Based on coupling estimates, the highest variation in PPF between the two scenarios is 2.2%. The critical heat flux (CHF), a measure that sets temperature limits, is crucial to nuclear reactor design. The CHF value and CHF temperature obtained from simple one-way calculation and strong coupled calculation is compared in illustrated in Fig. 5. It depicts that the largest CHF difference between simple and strong result is 84.8 kW/m2. On the other hand, it is obvious that the highest CHF temperature difference is around 0.25 K. As a result, there is a 15.9% relative error in the MDNBR value for simple and strong coupling calculations.

TABLE 1. Power peaking factor for different assembly of a typical 100 MWe integral PWR reactor.

|  |  |  |  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
|  | Simple one-way coupling | | | | | Strong coupling | | | | |
|  | 1 | 2 | 3 | 4 | 5 | 1 | 2 | 3 | 4 | 5 |
| 1 | 0.9695 | 0.9394 | 0.9486 | 1.1303 | 1.0661 | 0.9911 | 0.9603 | 0.9666 | 1.1165 | 1.0474 |
| 2 | 0.9394 | 1.0004 | 0.9921 | 1.1153 | 0.9129 | 0.9603 | 1.017 | 0.9994 | 1.1027 | 0.9061 |
| 3 | 0.9486 | 0.9921 | 1.0872 | 0.8975 |  | 0.9666 | 0.9994 | 1.0823 | 0.8979 |  |
| 4 | 1.1303 | 1.1153 | 0.8975 |  |  | 1.1165 | 1.1027 | 0.8979 |  |  |
| 5 | 1.0661 | 0.9129 |  |  |  | 1.0474 | 0.9061 |  |  |  |



*FIG. 3. Difference between* *simple one-way coupling and strong coupling radial PPFs.*

*FIG. 4. comparison between* *simple one-way coupling and strong coupling axial PPFs.*

*FIG. 5. comparison between* *simple one-way coupling and strong coupling TH results. Right: temperature profile. Left: CHF profile*

## Conclusion

The calculations of the core of a typical 100 MWe integral PWR reactor have been examined in this study. First, a simple one-way thermal-hydraulic and neutronic calculation is performed with DRAGON/DONJON/COBRA. Next same calculation with strong iterative coupling method is conducted. Neutronic result is verified against probabilistic approaches and TH result is verified against PARCS code outcome. Different parameters have been computed and thoroughly compared for both approaches. Despite the fact that both approaches yielded acceptable average results, the comparison shows that using codes with simple coupling procedure in certain parameters (such MDNBR) resulted in values that significantly differ from the extracted values using the strong coupled method. Put another way, in terms of the overall distribution of various parameters inside the core, data obtained with a simple coupling scheme are not very trustworthy. For instance, the result indicates that there is a 15.9% relative error in the MDNBR value for simple and strong coupling calculations.

## References

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