Conceptual Design of Fuel and Radial Shielding Sub-Assemblies for ASTRID

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Abstract. The French 600 MWe Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) project reached in 2015 the end of its Conceptual Design phase. The core design studies are being conducted by the CEA with support from AREVA and EDF. Innovative design choices for the core have been made to comply with the GEN IV reactor objectives, marking a break with the former Phénix and SuperPhénix Sodium Fast Reactors. The CFV core of ASTRID demonstrates an intrinsically safe behaviour with a negative sodium void worth achieved thanks to a new fuel sub-assembly design. This one comprises $(U,Pu)O_2$ and UO_2 axially heterogeneous fuel pins, a large cladding versus small spacer wire bundle, a sodium plenum above the fuel pins, and an upper neutron shielding with both enriched and natural boron carbide. The upper shielding also maintains a low secondary sodium activity level and is made removable on-line through the sub-assembly head for washing compatibility. Calculations have been performed to increase the stiffness of the stamped spacer pads in order to analyse its effect on the core mechanical behaviour during hypothetical radial core compaction events. Concerning the radial shielding sub-assemblies surrounding the fuel core, heavy iterative studies have been performed in order to fulfill ASTRID requirements of minimising the secondary sodium activity level and maximising the in-core life-time. Evaluated options were reflectors sub-assemblies made of steel or MgO rods, and radial neutron shielding sub-assemblies made of B_4C or borated steel, with different configurations in the design and in the core layout. This paper describes the design of the fuel and radial shielding sub-assemblies for the ASTRID CFV v4 core at the end of the Conceptual Design phase. Focus is placed on innovations and specificities in the design compared with former French SFRs.

Key Words: ASTRID, fuel, radial shielding, reflector, sub-assemblies.

1. Introduction

The conceptual design phase (AVP2) for the ASTRID prototype reached its end in 2015 [1]. The aim of the past five years (2013-2015) of studies was to propose a consistent, innovative preliminary design to fulfill ASTRID requirements. The core design studies are being conducted by the CEA with support from AREVA and EDF. Innovative design choices for the core have been made to comply with the GEN IV reactor objectives, thereby marking a break with the former Phénix and SuperPhénix French Sodium Fast Reactors.

The main objective to improve safety levels compared with current GEN II or III reactors has led to a core design that shows intrinsically safe behaviour in transients of accident situations. A negative or null sodium void worth can be obtained by using a heterogeneous core (CFV). Other challenging objectives such as minimising the secondary sodium activity level, preventing reactivity insertions due to core compaction and applying cost-killing measures have led to rather new sub-assemblies designs compared with former SFRs.

This paper describes the design of the fuel and radial shielding sub-assemblies for the ASTRID CFV v4 core at the end of the Conceptual Design phase with focus on innovations, specificities and studies performed to optimise the performances.

2. CFV Core

The CFV v4 core architecture [2] consists of 180 inner fuel sub-assemblies and 108 outer fuel sub-assemblies, surrounded by 3 rows of reflector sub-assemblies with MgO, 3 rows of shielding sub-assemblies with B_4C, 3 rows of shielding sub-assemblies with MgO, and another 2 rows of shielding sub-assemblies with B_4C. The lateral reflector and shielding arrangement was optimised to provide enhanced radial neutron absorption (see Section 4). The core's absorber device relies on an innovative architecture composed of 9 control/shutdown rods and 9 diverse control/shutdown rods, which are used both to manage core reactivity during the cycle, and to shut down the reactor. The other main characteristics of the ASTRID core include the integration of Complementary Safety Devices for severe accident prevention and mitigation, and an internal storage and debugging positions in the radial shielding area.

3. Design of Fuel Sub-Assemblies

Although the overall design of the ASTRID fuel S/A mainly relies on feedback from French SFRs Phénix and SuperPhénix, the CFV core concept involves some appreciable differences: axially heterogeneous fuel pins with a thick internal fertile plate for the inner core, longer fissile zone for the outer core, large cladding/small spacer wire fuel bundle, sodium plenum above the fissile area, enriched B_4C zone at the bottom of upper neutron shielding.

A schematic diagram of the CFV v4 fuel S/A is given in *FIG. 1*. The main parts are detailed in the following paragraphs.

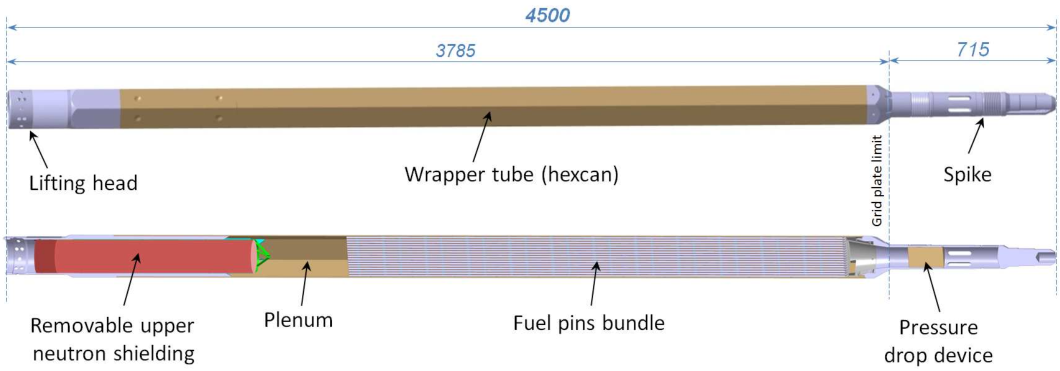


FIG. 1. ASTRID CFV v4 fuel sub-assembly.

The fuel S/A height is 4.50 m. It has been reduced by 0.9 m compared with SuperPhénix (5.40 m) thanks to removal of the upper axial blanket used in former SFRs, coupled with the choice of a compact pressure drop device.

3.1. Fuel Pins

One major innovation that has helped reducing the core's sodium void effect by about 5$ [3] compared with previous reactors is the wide fuel pin/small wire concept. The ASTRID design consists of 217 fuel pins in each S/A with a 9.70 mm outer diameter separated by a 1 mm-diameter spacer wire helically wound around the pins. The pin's lower plugs are mounted on the rails of a stainless-steel single-part grid to form a bundle that is vertically held within the hexagonal wrapper tube.

The inner and outer core fuels pins differ slightly by their composition and their length (*FIG. 2*). The CFV core is characterised by axially heterogeneous fuel pins, with a UO_2 fertile zone inside the (U,Pu)O_2 fissile area for the inner fuel sub-assemblies. A 10 cm-longer fissile zone for the outer fuel sub-assemblies provides a global asymmetrical, crucible-shaped core. Fuel pins in the core are provided with a lower fertile axial blanket of 30 cm. Fissile pellets have a central hole to improve margins with respect to the fuel melting temperature, while fertile pellets are solid to increase the breeding gain, to reduce the loss of reactivity and to make it possible to be differentiated from fissile pellets.

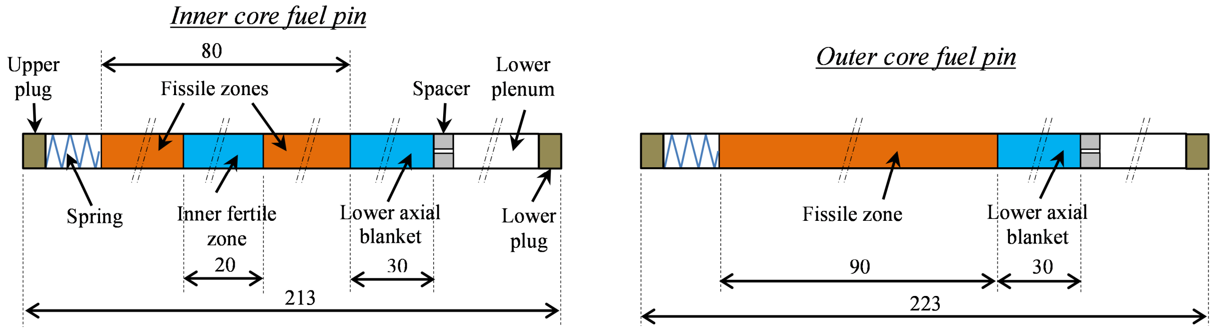


FIG. 2. Fuel pins composition (dimensions in cm).

In nominal conditions, the end-of-life pressure in the pins reaches about 40 bars. Thermomechanical analysis based on finite-element computations of the fuel pins during irradiation have started with the CEA code called GERMINAL V2 [4] within the PLEIADES fuel simulation platform [5]. Fuel melting probability has also been evaluated with a statistical approach taking into account uncertainties on many parameters such as pellet geometry and dimensions, irradiation conditions, materials behaviour laws [6]. The reference cladding material for ASTRID is composed of special 15%Cr-15%Ni austenitic steel – called AIM1 [7] – which was tested up to high dose rates in Phénix.

3.2. Wrapper Tube and Spacer Pads

The closed leaktight hexagonal wrapper tube (hexcan) is made of EM10 steel which has been extensively tested in Phénix. This martensitic 9%Cr-1%Mo steel exhibits excellent dimensional stability under irradiation [7]. A record dose of 155 dpa was reached on the EM10 hexcan during the BOITIX-9 experiment in Phénix. The external and internal widths of the ASTRID hexcan are respectively 168.7 mm and 161.5 mm.

To ensure suitable clearance between the sub-assemblies in the core, a rectangular spacer pad is stamped through each side of the hexcan just above the fuel pins. This pad design was originally used in Phénix and SuperPhénix. Not only do the pads play a central role in the mechanical equilibrium of the core, they also enhance safety by preventing reactivity increases due to a core compaction that follows a flowering, which may occur when core is subjected to dynamic stresses such as earthquake or internal pulse load release. Following the safety requirements for ASTRID wherein core compaction must be minimised, studies have been engaged to analyse the effect of a pad stiffness increase on the core mechanical behaviour. Finite element calculations [8] to increase the stiffness of the stamped pads have been done using the LICOS code [9] based on the CAST3M solver within the PLEIADES fuel simulation platform [5]. They showed that pad stiffness could be increased by a maximal ratio of ~7 compared with the previous design.

3.3. Upper Neutron Shielding

3.3.1. ASTRID Requirements

The CFV core's requirement for ASTRID makes it impossible to reuse the upper neutron shielding designs of Phénix and SuperPhénix. Indeed, the new requirement specifies a negative sodium void worth. For this reason, the bottom of upper neutron shielding, located just above the sodium plenum, should reflect neutrons as little as possible. This required choosing the best neutron absorber – boron carbide (B_4C) – which benefits from extensive feedback since it has been used in SFR absorber rods. Core studies [10] concluded that the optimum design for reaching the CFV effect would be a lower part made of 7.5 cm-thick B_4C enriched to 90% in ^{10}B for enhanced absorption, together with the smallest steel structure thickness to minimise neutrons reflection.

The second main function assigned to the upper neutron shielding is obviously to provide proper axial neutron shielding to minimise damage to reactor internals and reduce the activation of surrounding Na circuits. ASTRID core shielding studies finally led to the choice of natural B_4C (19.78% of ^{10}B) as material in the top part the upper neutron shielding.

3.3.2. Upper Neutron Shielding Design

After selection of B_4C as absorber in the upper neutron shielding, the other complex part of the task consisted in proposing a design compatible with the design criteria and operational constraints. The axially heterogeneous fuel pins provide a maximal flux shifted towards the top of the sub-assemblies compared with former SFRs. This means that the B_4C-enriched zone is subjected to a higher flux (3.0E+14 n/cm^2/s) which produces a large volume of helium and dissipates a great deal of power due to the neutron capture reactions. The latter aspect led to the choice of B_4C in the form of a 19-pin bundle to meet the maximal temperature criterion, together with a Na-bond to take advantage of the high thermal conductivity of sodium. Non-leaktight pins design also provided a solution to the problem of significant helium production by avoiding any excessive pressurisation.

But Na-bonded pins maybe raised the most important issue that needed to be resolved. It is well-known that the sodium inside pins cannot be totally removed after draining, because of the partially plugging of the microporous vents. Yet this non-negligible amount of Na trapped inside the pins engenders unacceptable safety risks during the fuel S/A washing or underwater storage due to possible uncontrolled sodium-water reactions The solution to this problem was found following a brainstorming and value engineering process [11]: the whole upper neutron shielding would be made removable on-line through the S/A head just before washing.

The main difficulty was then to find a design which enables the upper shielding extraction and that maximizes the B_4C volume. The removable upper neutron shielding consists of a 14.5 cm-diameter and 94 cm-long cylinder (*FIG. 3*).

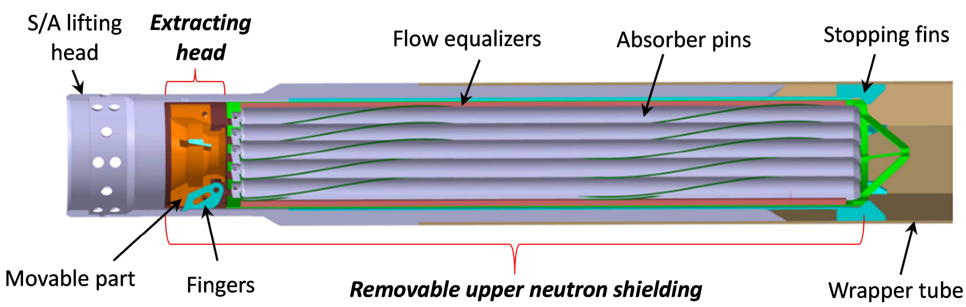


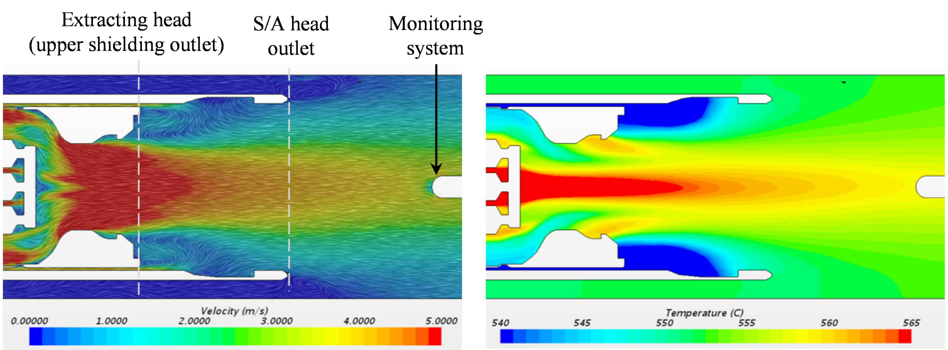
FIG. 3. Removable upper neutron shielding inside the S/A (section view).

The top part of upper neutron shielding is the extracting head. The upward axial locking is ensured by three lateral fingers spread out inside a groove in the S/A head and maintained thanks to the weight of a movable part. These fingers are withdrawn by the upward movement of the movable part thanks to a specific grapnel (not shown), allowing the upper shielding to be extracted in the washing pit through the S/A head. The lower part, in its first design, comprises the 19-pin bundle equipped with flow equalisers. From bottom to top inside the pins, there are 7.5 cm of B_4C enriched to 90% in ^{10}B, 69.5 cm of natural B_4C, and an upper plenum comprising a spring.

3.3.3. Thermal-Hydraulic CFD Computations

Preliminary thermal-hydraulic calculations on the first design (*FIG. 3*) were performed to evaluate the pressure drop and to characterise the sodium flow fields (velocity and temperature) above the S/A outlet at the core monitoring system level (thermocouples and flowmeter). CFD STAR-CCM+ simulations showed a heterogeneous flow in temperature with a gradient of 13°C between the center and the periphery in the monitoring section plane. The velocity field showed that the flow was disturbed as it passed through the extracting head of the upper shielding. Several separated flows were created which would alter the measure by the flowmeter. The total pressure drop in this upper shielding was ~0.6 bar.

A second design was then defined and evaluated. It showed a more symmetrical and flat response in temperature with a gradient of ~5°C between the center and the periphery. The velocity field was much more homogeneous and acceptable (*FIG. 4*). The total pressure drop in the upper shielding has been decreased to ~0.3 bar.



<u>*Na velocity field*</u> <u>*Na temperature field*</u>

FIG. 4. CFD computations at the S/A outlet (second design).

3.4. Lifting Head

The objective of maximising the B_4C volume in the removable upper neutron shielding led to an increase in the S/A head inner diameter (15 cm) through which it is extracted. The S/A lifting head is a ~1 meter-long single part made of AISI 316L stainless steel. The top part of the head is cylindrical and comprises an internal groove for S/A lifting using the handling grapnel. To prevent handling errors, a bar code is engraved on the edge of the head for sub-assembly identification by under-sodium ultrasonic readings. The bottom part of the head has an outer hexagonal section while retaining the inner cylindrical section. The hexagonal part of the head is inserted inside the wrapper tube and the binding is made by cold stamping.

3.5. Spike

All core sub-assemblies are supported vertically with the spike inserted in the shroud tubes provided in the grid plate. The shroud tubes comprise multiple holes to allow cold sodium from the grid plate to flow into the spike through oblong slots (*FIG. 5*).

Two helicoid labyrinths are provided on the spike with a reduced gap with the shroud tube. The function of the top labyrinth is to control the leakage flow of cold sodium into the hot pool in the interspace between S/A. The function of the bottom labyrinth is to ensure the minimum flow required for main vessel cooling. It also prevents the S/A from hydraulic lifting or floating by generating a hold-down force owing to the pressure drop developed here.

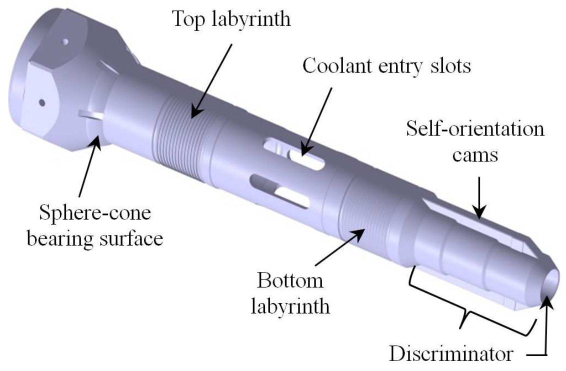


FIG. 5. Spike.

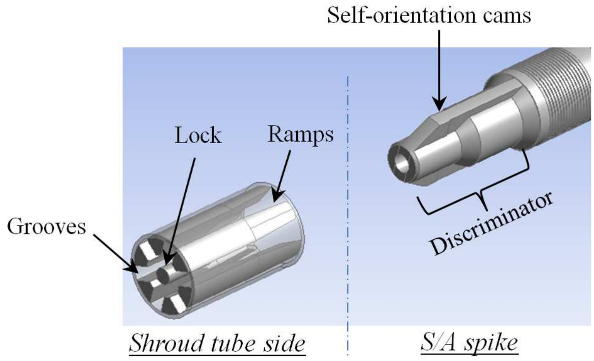


FIG. 6. Discriminatory and self-orientation features.

A compact pressure drop device is provided in the spike to achieve the required flow rate through the S/A (*FIG. 1*). This device consists of multiple aligned or crossed honeycomb orifice plates, within a total length of less than 15 cm.

Additional protection against handling errors during refueling is provided thanks to discriminatory features located at the end of the spike (*FIG. 6*). Those discriminators have a specific design for each flow zone in the core. They prevent a fuel S/A from being completely lowered into the wrong position in the core. A novelty for ASTRID is that self-orientation devices are provided in parallel to the discriminator at the end of the spike. It consists of two cams on the spike to be inserted into vertical grooves in the shroud tubes. The top of each groove is provided with conical ramps to rectify the angular mis-orientation when refueling.

4. Design of Radial Shielding Sub-Assemblies

In this section, the design studies on radial shielding sub-assemblies performed during the last five years to fulfill ASTRID requirement on the secondary sodium activity level are summarized [12].

4.1. Requirements

Fast neutron reactor cores are necessarily surrounded by thick shielding structures to minimise damage to reactor internals and activation of surrounding circuits. In the case of pool-type reactors, an important design issue is the activation of the secondary sodium in the Intermediate Heat eXchangers (IHX) which must be limited to an appropriate level. More specifically, these radial shields have to reduce the production of ^{24}Na by activation of ^{23}Na which is the main contributor to external exposure doses.

The specification for ASTRID is a ^{24}Na activity level below 20 Bq/cm^3 in the secondary sodium circuits. This value is equivalent to a maximum surface dose rate of 7.5 µSv/h at the contact of Na circuits thus ensuring that the ASTRID steam generator building remains classified as a non-controlled area. To take some margins regarding the uncertainties, the target value for ^{24}Na activity level in simulations has been set to $\leq$ 10 Bq/cm^3. Regarding the internal structures, the neutron damage must be limited to 1 dpa over the reactor life-time (60 years) to remain compliant with design and construction codes.

To summarize, radial shielding sub-assemblies must be designed to limit neutron flux at IHXs locations to an acceptable value consistent with the secondary sodium activity criterion. Different configurations in the core layout and in the radial S/A design have been considered since the beginning of the project; those are described in the following sections.

4.2. First Configuration

Initial design choices made at the beginning of the ASTRID project were based on former French SFRs feedback. During the pre-conceptual design phase (AVP1), the radial shielding of the CFV core designs v1 (version 1) and v2 was composed of 3 rows of reflectors S/A made of stainless-steel and 4 rows of neutron shielding S/A made of boron carbide (B_4C). Steel S/A were formerly used as reflectors and radial shields in PX and SPX1. SPX2 and EFR projects also considered reflectors made of steel, but studied radial shielding made of B_4C.

The reflectors S/A for ASTRID CFV v1 core are made of EM10 steel arranged in a bundle of 19 rods separated by a helical spacer wire. The expected life-time, defined with respect to the dose criteria, was 24 years for the first row of reflectors S/A and 60 years for the other rows.

The radial neutron shielding S/A comprise a bundle of 37 pins filled with natural B_4C. The B_4C pins are leaktight as a preliminary design. The helium produced due the neutron capture reactions leads to a pressure of 150 bars inside the pins over 20 years. The pins were designed with an upper and a lower plenum of 50 cm each.

Neutron transport simulations have been performed thanks to MCNP Monte-Carlo code for the CFV v1 core with a modelisation of the above core structure (ACS) lower plate, the diagrid and one IHX. The secondary sodium activity was calculated at ~1400 Bq/cm^3 which was significantly above the target value (10 Bq/cm^3). Simulations showed that 60% of this activity level was due to radial neutron leakages, 35% was due to lower axial neutron leakages while only 5% was due to upper axial neutron leakages. These results demonstrated the need to optimize the radial and lower shields of the core. Those parametric studies are described in the following section.

4.3. Parametric Studies

Preliminary simulations showed that lower axial leakages in the radial S/A area contributed for 1/3 in the secondary sodium activation. This was imputed to the presence of lower plenum inside the B_4C pins of radial S/A. Further optimisations should thus consist in minimising the plenums height. In the radial direction, where neutron leakages are preponderant, a thickness of about 2 m remains available from the core boundary to the diagrid outer periphery. Within such a distance four additional rows of radial S/A have been implemented without impact on the vessel size.

Parametric and iterative studies have been performed to identify an acceptable configuration [12]. Regarding materials provided in the radial sub-assemblies, many candidates have been evaluated to fulfill the different functions (reflection, moderation or absorption): SiC, MgO, $MgAl_2O_3$, $^{11}B4C$, B_4C (^{10}B enriched or not), vanadium alloy, Hf, $Hf^{11}B_2$ or borated steel. Besides neutron performances of each configuration, other criteria were consequently evaluated: behaviour under irradiation, life duration, chemical compatibility, safety, maturity levels, availability, washing and manufacturing capability, qualification needs.

B_4C is obviously the best neutron absorber and benefits from a large feedback in SFR. Despite an efficient reflector-moderator effect, $^{11}B_4C$ and $Hf^{11}B_2$ have been finally excluded due to prohibitive manufacturing costs regarding the enrichment in ^{11}B. Hafnium has been excluded due to a possible large-scale limited availability. Moderation hydride compounds (YH_2) have been excluded for safety consideration (hydrogen risk in accidental conditions).

The reflector function is only required in the radial direction at the fissile core boundary. For this purpose SiC, MgO and $MgAl_2O_3$ appeared as the best candidates and benefited from R&D done at CEA. Regarding the manufacturing and costs, MgO was finally preselected as the reflector S/A material in the first rows just surrounding the fissile core.

For the remaining regions, which are radially behind the MgO reflectors S/A, a large number of configurations have been calculated with the economic objective of limiting the amount of B_4C – especially enriched in ^{10}B – as much as possible. Providing the suppression of upper and lower plenums in the radial S/A to reduce axial leakages, MCNP calculations showed that the alternation of moderation and absorption zones provided satisfactory attenuation factors. Good results were obtained with successive alternations of 3 rows of MgO and 3 rows of natural B_4C. Calculations exhibited that radial neutron leakages were largely reduced but the secondary sodium activity reached ~70 Bq/cm^3, which represent a reduction by a factor 20 compared to first design, but was still significantly above the target value.

Since increasing the distance between the core and the IHXs was not an option regarding the vessel size, it has been proposed to implement neutron shields on the IHXs. Reactor and equipment designers proposed solutions to implement 25 to 50 mm-thick borated steel plates – grade 304B7 with 2% natural boron – to envelop the upper and the lower zones of the heat exchangers. MCNP calculations demonstrated that the IHX shielding decreased the secondary sodium activity by a factor ~10 which permitted to reach the objective of < 10 Bq/cm^3.

4.4. Second Configuration

Parametric studies on radial S/A materials and arrangement finally led to a core layout based on successive alternation of MgO and B_4C sub-assemblies providing enhanced radial neutron absorption, as described in section 2 for the CFV v3 and v4 core architectures.

The reflectors S/A in CFV v4 core comprise a bundle of 19 pins filled with MgO pellets at 96% relative density. The cladding, made of AIM1 steel, has an outer diameter of 33.6 mm and a thickness of 1 mm. Pins are separated by a 2 mm-diameter helical wire. The surface fraction of MgO is 52% inside the bundle. As MgO does not produce gas under irradiation, pins are leaktight and filled with helium at atmospheric pressure. The other parts of the reflector S/A, such as the wrapper tube, the lifting head and the spike, are similar to those of fuel S/A. Due to the high loading radial gradient in the first row adjacent to the fuel S/A, several management scenarios – to be undertaken during fuel reloading periods – have been studied to increase the reflectors life-time by limiting the damage on the cladding and the bowing of the hexagonal duct. Some of them, consisting in turning S/A of first row by 180 degrees and/or permuting S/A between different rows, can increase the lifetime between 10 and 30 years for the first row, and up to 60 years for the other rows. Preliminary thermal-hydraulic calculation have been performed with the STAR-CCM+ CFD code to verify the need of feeding the reflectors S/A with cold sodium. It has been found out that the first row would require a sodium flow of ~0.6 kg/s to respect thermal criteria on claddings, while the others rows could be cooled by natural convection.

The radial shielding S/A in CFV v4 core comprise a bundle of 19 pins filled with natural B_4C pellets. Parametric studies showed that neutron leakages could be reduced thanks to the suppression of upper and lower plenums in the radial S/A. For this purpose B_4C pins were made non-leaktight, thus avoiding any excessive pressurization – by allowing produced helium to be continuously released out of the pin – and taking advantage of the high thermal conductivity in the sodium bond. The height of the B_4C column is increased to ~3.3 m with this design, which is one meter more than for the previous CFV v2 leaktight-pin design.

Considering the above described radial shielding S/A and providing that the IHXs are shielded with borated steel sleeves, ^{24}Na activity in the secondary loops was calculated to ~7.7 Bq/cm^3 (best estimate) without an internal storage in the core, and ~9 Bq/cm^3 with an internal storage.

These results are compliant with the ASTRID specification but two issues have been raised concerning the radial shielding S/A composed of Na-bonded B_4C pins. As for upper neutron shielding of fuel S/A (see section 3.3), it is considered that Na-bonded pins are not washable due to risks of sodium-water reactions. Moreover, experimental feedback in Phénix for Na-bonded pins is limited to ~2 years, due to carburisation of cladding and shroud, which is far from minimum target lifetime of 20 years for these S/A in ASTRID.

Regarding these issues, the Conceptual Design phase ended in 2015 with the observation that the radial shielding S/A design for CFV v4 core was not acceptable. Additional studies are being pursued during Basic Design phase to make realistic the leaktight-B_4C-pins design. Two options currently under evaluation are based on either leaktight B_4C rings or borated steel shielding S/A behind 5 rows of reflectors S/A. Preliminary results are acceptable and some ways of optimization have been identified.

5. Conclusion

The design of the fuel and radial shielding sub-assemblies for the ASTRID CFV v4 core at the end of the Conceptual Design phase (AVP2) is described. Innovative design choices have been made to meet the ASTRID project requirements, marking a break with the former Phénix and SuperPhénix French SFRs.

Fuel sub-assemblies ensure a low sodium void worth (CFV core) thanks to axially heterogeneous fuel pins, a wide cladding/small spacer wire bundle, a sodium plenum above the fuel pins, and upper neutron shielding with B_4C sodium-bonded pins. The upper neutron shielding will help to reach a low secondary sodium activity level and will be made removable on-line through the assembly head so as to meet washing constraints. Studies have been performed to increase the stiffness of the stamped spacer pads on the wrapper tube in order to analyse its effect on the core mechanical behaviour during hypothetical radial core flowering and compaction events.

Since the beginning of ASTRID project, various reflectors and radial shielding S/A designs (steel, MgO, B_4C pins with helium or sodium bond) have been evaluated in different core layouts through many iterative studies. A strict value analysis process considering various criteria has been followed. ASTRID specification for ^{24}Na activity in secondary loops (10 Bq/cm^3 best estimate) now appears to be reachable. Some ways of optimization have been identified and are being studied during the Basic Design phase that has just started. This phase will also focus on consolidation of main design choices, performance levels confirmation, design optimization, as well as the start of qualification programme.

6. Acknowledgments

The authors wish to thank all the teams involved in design studies of fuel and radial shielding sub-assemblies at CEA and AREVA-NP for their support.

7. Nomenclature

ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
AVP1/2	Conceptual design phase #1/2
CFD	Computational Fluid Dynamics
CFV	Low void worth core
dpa	displacement per atom
EFR	European Fast Reactor
IHX	Intermediate Heat Exchanger
PX	Phénix reactor
S/A	Sub-Assembly(ies)
SFR	Sodium Fast Reactor
SPX	SuperPhénix reactor

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