APPLICABILITY OF DIFFERENT FUEL TYPES IN THE ALLEGRO REACTOR

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MOMENTUM OF INNOVATION

Introduction-first core of ALLEGRO

ALLEGRO gas cooled fast reactor

pre-conceptual design phase

core design

fuel selection

MOX UOX

ESNII

European Sustainable Nuclear Industrial Initiative

a common MOX type fuel

ASTRID ALFRED MYRRHA ALLEGRO	for the ESNII demonstrators

The common development and sharing the qualification experience of fast reactor fuel could be a large benefit for the developers of the four reactor types.

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Oxide fuel in fast reactor

In the 1950s fuel of the first fast breeder reactors (US and UK) was: **metallic**.

EBR-I and EBR-II (Experimental Breeder Reactor): **95% uranium + 5% precious metal (Mo, Ru, Rh, Pa, Zr, Nb)**

By the 1960s, mixed uranium and plutonium oxide $(U,Pu)O_2$ was known to be highly radiation tolerant and began to be considered as a reference fuel for fast reactors.

Advantageous properties of oxide fuel:

- high melting point, with no allotropic changes,
- excellent stability,
- excellent behaviour under irradiation.

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Oxide fuel in fast reactor

In the 1960s, it was argued that the EBR-II reactor should use an oxide fuel, which could lead to higher burn-up, and the reactor could operate at higher temperatures.

Other reactors which used oxide fuel:

- EBR-II, FFTF and CRBR (Fast Flux Testing Facility and Clinch River Breeder Reactor);
- Japanese, French, English, Russian, and Indian Sodium Cooled Reactors:

significant development of the uranium and plutonium oxide fuel,

development of steel structural material

new alloys: swelling has decreased slightly, burn-up can be increased.

- and it is planned that the first zone of the gas-cooled ALLEGRO reactor will also start with oxide fuel.

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Main characteristics of fast reactor fuel

- **Fuel pin**: length: 2-3 m,
  diameter: 5-10 mm,
  outer cladding diameter: 6-10 mm,
  cladding thickness: 0,4–0,6 mm,
  material: steel,
  two gas plenums are located at both ends of the fuel pin, and the largest plenum is preferably located at the bottom end of the fuel pin.
- **Pellet:** sintered,
  outer diameter slightly smaller than the inner diameter of the clad,
  full pellets or annular pellets (with a central hole between 1.5 and 2 mm),
  mixed oxide $(U,Pu)O_2$ (with the exception of Russian reactors where they are made of uranium oxide) with a plutonium content between 15% and 30%.
- oxide fissile column ≈ 1 m,
- radial gap ≈ 100 mm
- UO_2 axial blankets (0.3–0.5 m long) made with natural or depleted uranium are placed at the lower and upper ends of the fissile column,
- initially pins contain helium gas under 1 atm,
- fuel subassembly: hexagonally,
- number of fuel pins :~100–300 pins in each subassembly.

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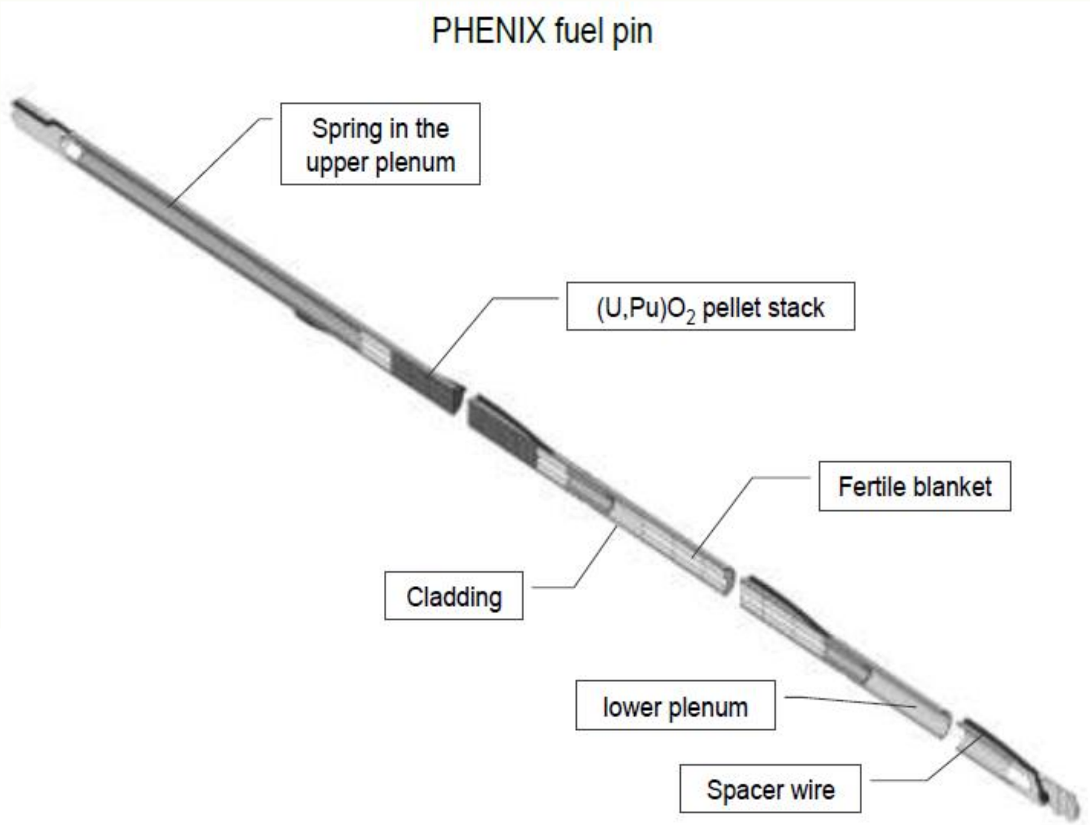
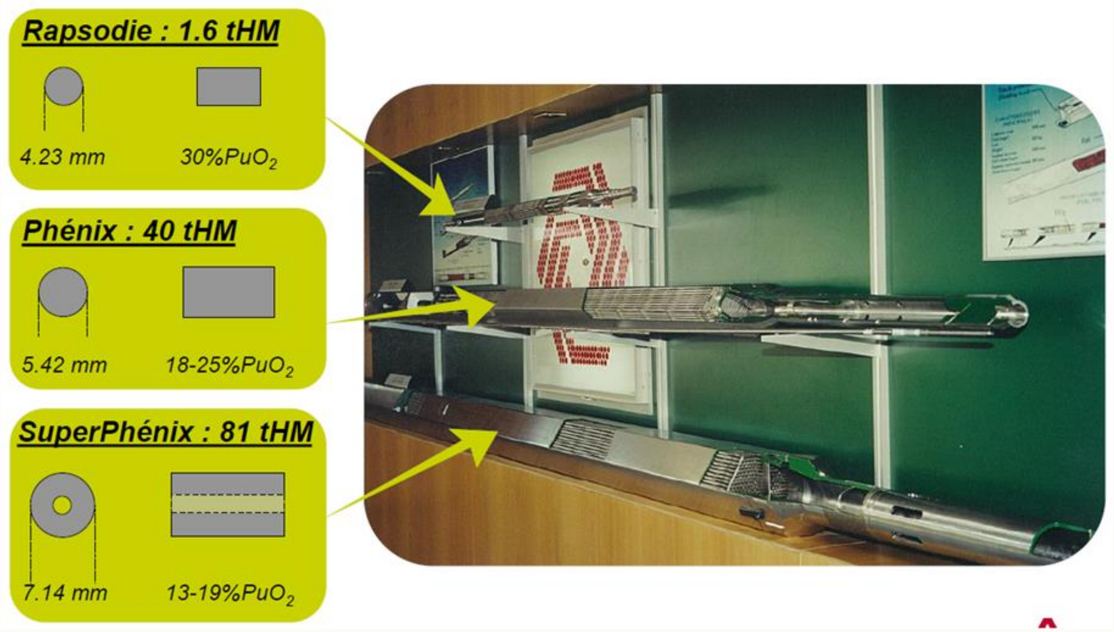
ASTRID type fuel for fast reactor

The ASTRID fuel is developed based on the French Phénix and Superphénix fuels.





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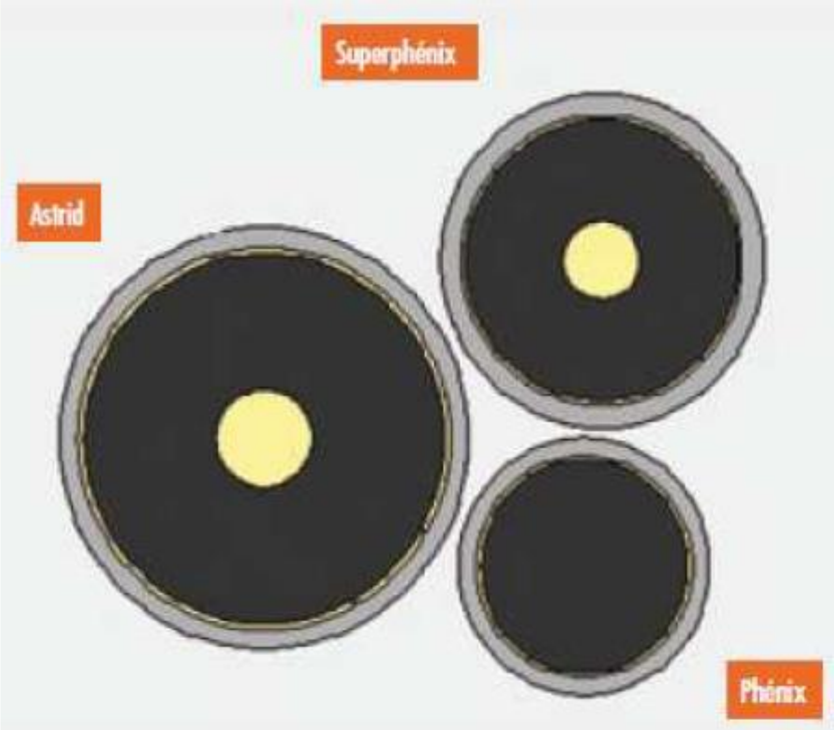
ASTRID type fuel

<u>Changes</u>:

- In comparison with the former Phénix, Superphénix or European Fast Reactor designs, the diameter of the pins of the new concepts is bigger with outside diameter values of approximately 9 to 10 mm (to be compared with 8.5 mm for Superphénix).



ASTRID type fuel

- The diameter of the helical wire wound around the fuel pins to separate them and make the passage of sodium between the pins easier, is reduced to 1 mm.

- For the first ASTRID cores, the cladding material of the fuel subassembly will be the 15-15 Ti work-hardened austenitic steel AIM1.

	C	Cr	Ni	Mo	Si	Mn	Ti
AIM1	0,1	15	15	1,2	0,8	1,5	0,4

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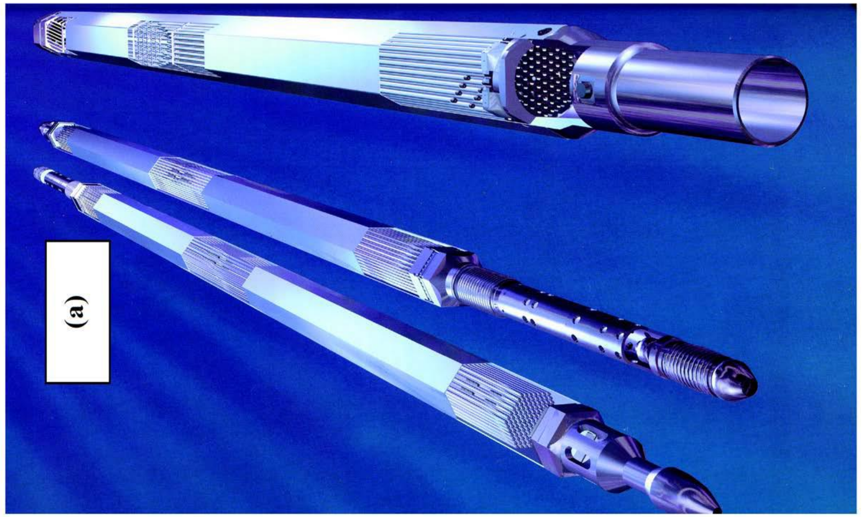
BN-600 type fuel

The fast sodium reactor fuel develops with the development of SFR reactor:

BR-10 (1959)
BOR-60 (1969)
→ BN-350 (1973)
→ BN-600 (1980)
BN-800 (2015)
→ BN-1800 (conceptual design)
BN-1200 (under development)

At present two reactors, BN-350 and BN-600 operate with a uranium oxide fuel, the MOX vibro-compact fuel is used in the BOR-60 reactor and uranium nitride is used as a fuel in the BR-10 reactor.



Fuel subassemblies for the fast reactor BN-600.

BN-600 type fuel

- The BN-350 and BN-600 reactors began producing electricity in **1973** and **1980**.
- The BN-350 used uranium in the range of **20%** enrichment in its core.
- The BN-600 used fuel with enrichments ranging from **17%** to **33%**.
- **1980-1986**: The enrichments of initial BN-600 fuel assemblies were **21%** and **33%**.
- In **1987** the core was contained with 3 different enrichment levels: **17%; 21%** and **26%**.

Modifications of the BN-600 core :

- M: 1987–1991
- M1: 1993–2004
- M2: from 2005.

Cladding material: ChS68 austenitic steel.

	C	Cr	Ni	Mo	Si	Mn	Ti	V	B	P
ChS68	0,05-0,08	15,5-17,0	14.0-15,5	1,9-2,5	0,3-0,6	1,3-2,0	0,2-0,5	0,1-0,3	0,002-0,005	<0,02

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Use of the ASTRID and BN-600 type fuels in NPP

- Both fuel types were successfully applied in NPPs with sodium cooled fast reactors for many years.

ASTRID type fuel

- Phénix reactor: 1974-1990
- Superphénix reactor: 1985-1998

BN-600 type fuel

- BN-350 reactor: 1973-1998
- BN-600 reactor: 1980-

- The long term operation was accompanied with important developments.
- Fuel failure experience was an important driver of those developments.
- The structural materials were optimised to withstand long irradiation periods without significant changes.

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Use of the ASTRID and BN-600 type fuels in NPP

Fuel failures

Phénix reactor

- During the 35 years of operation, occurred **15 fuel failures** with delayed neutron signals (8 of them on experimental sub-assemblies) and 11 gas leaks without neutron signal.

Superphénix reactor

- **101 anomalies** and incidents occurred and the **fuel failure is only 4%** of these events and cladding damage has not been identified.

BN-600 reactor

- From 1981 to 1987 of the BN-600 reactor was operated with failed fuel pin claddings. Approximate number of these **inhermetic fuel** pins is assessed to be **60**.
- As a result of examination of the irradiated fuel sub-assemblies it was found that the fuel failures had been generally caused by the strained operational fuel conditions as well as by poor fuel cladding structural materials.
- The shutdowns due to the loss of integrity of standard fuel have not occurred since 1999.

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Handling of spent fuel

- Handling of fuel subassemblies is a major challenge as it has a significant impact on the duration of the outage periods for reloading or rearrangement of the core, and therefore on the reactor's availability factor.
- The function of the fuel handling system is to transfer and manage the fuel within the nuclear island.

There are three main types of fuel handling:

1. In-vessel handling of the reactor fuel subassemblies
2. Loading/unloading system
3. Ex-vessel handling

- Spent fuel assemblies from all existing prototype and demonstration Liquid Metal Cooled Fast Breeder Reactors are taken out of the reactor and transferred to an intermediate fuel storage drum cooled by sodium.
- They remain there until their decay heat has sufficiently decreased for further cleaning and transport.
- In the BN-600 reactor after cleaning, the subassemblies are transferred to the water pool for hold-up for 3 years.
- Afterwards, the subassemblies are shipped to the reprocessing plant.
- The failed subassemblies are stored in the leak-tight flasks and the sound subassemblies are stored in the open flasks.

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Base irradiation of fuels

Fuel examination was carried out for both fuel types after irradiation in research reactors and NPPs. The main mechanisms leading to loss of fuel integrity have been identified. The obtained data has been used for the development of numerical models to predict fuel behaviour in reactor conditions.

Irradiation of ASTRID type fuel

- Osiris
- Rapsodie
- Phénix
- Superphénix

Irradiation of BN-600 type fuel

- BOR-60
- BN-350
- BN-600

PIE

- Fuel and fission product behaviour.
- Mechanical and chemical interaction between fuel and cladding (FCMI).
- Irradiation to study transmutation of minor actinides and long-lived fission products.

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Transient testing

Transient testing with MOX fuel has been performed in the past in the TREAT and BR2 reactors. The behaviour of French fast reactor MOX fuel was investigated in several experimental programmes in the Cabri reactor. Probably some tests were conducted with the Russian fast reactor UOX fuel, but the results are not available in the open literature.

Transient testing of ASTRID type fuel

The main objective of the Cabri test program was to arrive at a better phenomenological understanding of the fast transient domain of initiation phase scenarios considered in risk analysis studies.

Cabri test programme

Cabri-I: the first series (1973-1987), consisting of 32 tests:

- transient overpower (TOP),
- undercooling overpower (TUCOP), and
- unprotected loss-of-flow (ULOF) conditions in fast sodium-cooled breeder reactors.
- The test fuel consisted of pellets of 20%-enriched UO_2 for the fresh fuel tests, and 15.5% Pu/(Pu+U) in the form of mixed oxide $(U,Pu)O_2$ for tests with irradiated fuel.
- using test pins of single geometry, with low or medium burn-up, solid fuel.

Cabri-II: a new series, consisting of 12 in-pile tests:

- using different claddings and pellet (annular fuel) designs and higher burn-up, as well as
- slower ramp rates, has recently been conducted.

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Fuel fabrication

ASTRID type fuel

- The **MOX** fuel for French fast reactors was fabricated in the **Cadarache** plutonium workshop.
- In **2003** the commercial MOX fuel production ceased and the workshop is prepared for **final shutdown** and dismantling.

In order to produce fuel for the **ASTRID** reactor **new facilities** will be needed.

BN-600 fuel

The **Mashinostroitelny Zavod**, Electrostal, commenced serial production of BN-350 and BN-600 nuclear fuel 1970's (**UO_2 fuel**) and is still in operation todays.

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Comparison of ASTRID and BN-600 fuel

The on-going feasibility studies indicate that both MOX and UOX fuel could be used in the first core of the ALLEGRO reactor. Beyond the physical possibility several other aspects must be taken into account during the selection of fuel type.

	ASTRID	**BN-600**
Pellet	MOX	UOX
Cladding	AIM1	ChS-68CW
Operational experience	Decades of operation in Phénix and Superphénix reactors	Decades of operation in BN-350 and BN-600 reactors
Fuel examination after base irradiation	Irradiation in Osiris, Rapsodie, Phénix and Superphénix reactors	Irradiation in BOR-60, BN-350 and BN-600 reactors
Testing in transient conditions	Large number of tests in the Cabri reactor	No test data available in open literature
Fuel production	Not in production today	Continuous production in Electrostal

Second core of ALLEGRO

The high temperature operation of the second core of the ALLEGRO gas cooled fast reactor will result in specific requirements for the fuel. It is obvious that the fuel of the first core with stainless steel cladding and UOX or MOX pellets cannot be operated with 850 °C core outlet temperature, since the cladding creep would lead to large deformations and the pellets centreline temperature may reach the melting point of the ceramic material due to its low thermal conductivity. New pellet and cladding materials are needed for the second core of ALLEGRO.

There are several fuel types that theoretically could be applied in the second core of ALLEGRO and their advantages and limitations were reviewed in the present work:

- metallic,
- carbide,
- nitride,
- silicide,
- oxide,
- minor actinide-bearing (MA),
- coated particle type,
- inert matrix,
- CERMET and
- thorium fuel.

Metallic and oxide fuels cannot be considered in second core of the ALLEGRO, but most of the past experience has been accumulated on these fuel types in sodium cooled fast reactors.

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Second core of ALLEGRO

- Our review showed that metallic and silicide fuel could not be considered for ALLEGRO due to their material properties.
- Minor actinide-bearing, coated particle type, inert matrix, composite and thorium fuel can be taken into account in a longer perspective, but not in the first version of the second ALLEGRO core.
- The most suitable candidates for the second ALLEGRO core are the carbide and nitride fuel having the following important characteristics:

	Oxide	**Nitride**	**Carbide**
Melting temperature	high	high	high
Thermal conductivity	low	high	high
Swelling due to irradiation	moderate	high	high
Operational experience	wide	very moderate	moderate
Experimental testing	wide	very moderate	moderate
Fabrication experience	wide	very moderate	moderate
Reprocessing	PUREX	PUREX	PUREX non applicable

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Characteristics of nitride and carbide fuel

- Both materials have high **melting point** and good **thermal conductivity**. Compared to oxide fuel the temperature difference between the fuel centreline and surface is 6-7 times lower for nitride and carbide.
- The main disadvantages of nitride and carbide pellets are the high volumetric **swelling rate**, which can reach 10-30%.
- The **irradiation** experience in nuclear reactors is modest for both nitride (Russia, EU) and carbide (US, India).

- The number of irradiated carbide is more than that of nitride, but is not comparable to the oxide fuel irradiation experience.
- The qualification of nitride or carbide for ALLEGRO will need the execution of new irradiation programme.

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Characteristics of nitride and carbide fuel

- The **handling** and **production** of nitride and needs special treatment and equipment since both nitride and carbide are pyrophoric (especially the carbide).
- **Fabrication** capabilities are very limited today, serial production of nitride or carbide does not exist.
- In long term the **reprocessing** of spent GFR fuel should be taken into account:
    - the nitride fuel can be reprocessed using the standard PUREX technology,
    - for the carbide fuel new technology should be developed.

Recommended fuel for second core of ALLEGRO

- characteristic of fuels
- irradiated data
- larger fabrication experience

CARBIDE fuel → pellet: UC or (U,Pu)C
cladding: SiCf/SiC

During the design of fuel geometry relatively large gas gap must be considered for the following reasons:

- The cladding will not be able to withstand plastic deformation and so the pellet-cladding interaction must be avoided, the gap cannot be closed between pellet and cladding. The leak tightness problem of SiCf/SiC cladding is a crucial issue which should be handled together with a development of pellet materials.
- During the design of the fuel rod it should be considered that gap closing must be avoided in both normal operational and transient, accident situations.

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Recommended fuel for second core of ALLEGRO

- The irradiation swelling of carbide fuel is enhanced at high temperatures typical for ALLEGRO core and it can reach about 30%. Considering the fuel pellet with 5.42 mm diameter (such data is used in the ALLEGRO thermal hydraulic benchmark) the gap size should be about 0.2-0.3 mm. The design of fuel element must be supported with detailed fuel behaviour calculations.
- The design of fuel element must be supported with detailed fuel behaviour calculations.

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