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ESFR SMART a European Sodium Fast Reactor Concept including the European Feedback Experience and the new Safety Commitments following Fukushima Accident

Mr. Joel Guidez

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Meet the Presenter

Mr. Joël Guidez, a 1973 graduate from the Ecole Centrale de Paris, focused his career on the field of nuclear reactors in France and in Europe. After working for Superphénix, then for Phénix as head of tests, he led the thermohydraulic laboratory in the CEA center of Cadarache. In 1992 he became head of the Osiris reactor at Saclay ,near Paris. In 1997 he was seconded to the European Commission in Petten, the Netherlands where he took responsibility for the European Commission reactor: the High Flux Reactor. From 2002 to the end of 2007, he was director of the Phénix nuclear power plant in the CEA center of Marcoule. He continued in 2008 and 2009 as director of industrial nuclear support, at Saclay. His first European experience was followed by a second one in 2010, where he became nuclear representant at the French Embassy in Berlin. In 2011, he returned to Saclay in the director of the CEA nuclear energy division, as an international expert. At the end of 2012, he published a book on the experience feedback from the 35 years of operation of the reactor Phénix, translated into English in 2013 and republished in 2013, at edp / sciences. In 2015 he wrote a book on the technical and scientific achievements of Superphénix, which were published in 2016, republished in 2017 and were translated and published in English by Springer editions in 2017. In 2019, he was a member of the operational committee of the office of the High Commissioner. / Honorary President of the ST7 SFEN section / Representative of France at the RSWG of GIF / President of the GCFS - French safety advisory group - tripartite CEA / EDF / AREVA / Scientific manager of the GEN IV segment at CEA. Mr. Guidez retired in March 2020 while remaining a scientific advisor to the CEA, working on the ESFR SMART European project and writing a new book entitled *Fast Reactors: A solution to Avoid Global Warming?*

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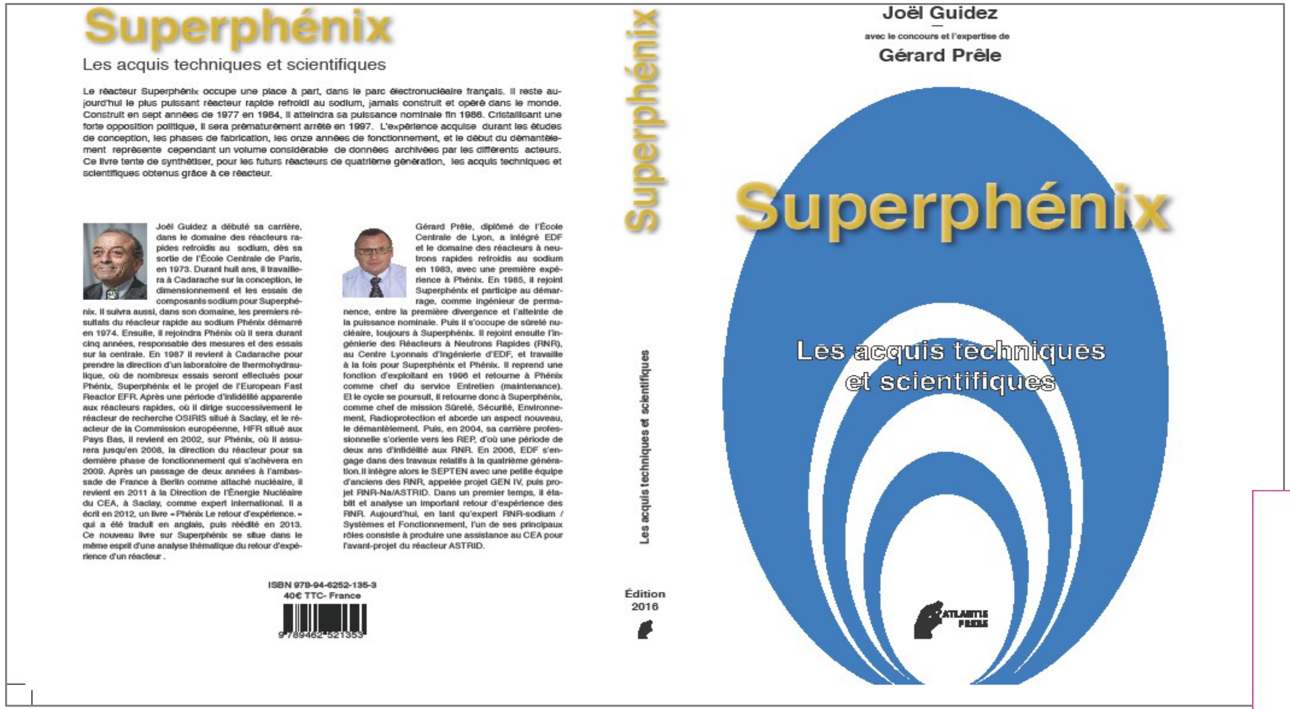

Superphénix

Les acquis techniques et scientifiques

Le réacteur Superphénix occupe une place à part, dans le parc électronucléaire français. Il reste aujourd'hui le plus puissant réacteur rapide refroidi au sodium, jamais construit et opéré dans le monde. Construit en sept années de 1977 en 1984, il atteindra sa puissance nominale fin 1986. Cristallisant une forte opposition politique, il sera prématurément arrêté en 1997. L'expérience acquise durant les études de conception, les phases de fabrication, les onze années de fonctionnement, et le début du démantèlement représente cependant un volume considérable de données archivées par les différents acteurs. Ce livre tente de synthétiser, pour les futurs réacteurs de quatrième génération, les acquis techniques et scientifiques obtenus grâce à ce réacteur.

Joël Guidez a débuté sa carrière, dans le domaine des réacteurs rapides refroidis au sodium, dès sa sortie de l'École Centrale de Paris, en 1973. Durant huit ans, il travaillera à Cadarache sur la conception, le dimensionnement et les essais de composants sodium pour Superphénix. Il suivra aussi, dans son domaine, les premiers résultats du réacteur rapide au sodium Phénix démarré en 1974. Ensuite, il rejoindra Phénix où il sera durant cinq années, responsable des mesures et des essais sur la centrale. En 1987 il revient à Cadarache pour prendre la direction d'un laboratoire de thermohydraulique, où de nombreux essais seront effectués pour Phénix, Superphénix et le projet de l'European Fast Reactor EFR. Après une période d'infidélité apparente aux réacteurs rapides, où il dirige successivement le réacteur de recherche OSIRIS situé à Saclay, et le réacteur de la Commission européenne, HFR situé aux Pays Bas, il revient en 2002, sur Phénix, où il assurera jusqu'en 2008, la direction du réacteur pour sa dernière phase de fonctionnement qui s'achèvera en 2009. Après un passage de deux années à l'ambassade de France à Berlin comme attaché nucléaire, il revient en 2011 à la Direction de l'Énergie Nucléaire du CEA, à Saclay, comme expert international. Il a écrit en 2012, un livre « Phénix Le retour d'expérience. » qui a été traduit en anglais, puis réédité en 2013. Ce nouveau livre sur Superphénix se situe dans le même esprit d'une analyse thématique du retour d'expérience d'un réacteur .

Gérard Prêle, diplômé de l'École Centrale de Lyon, a intégré EDF et le domaine des réacteurs à neutrons rapides refroidis au sodium en 1983, avec une première expérience à Phénix. En 1985, il rejoint Superphénix et participe au démarrage, comme ingénieur de permanence, entre la première divergence et l'atteinte de la puissance nominale. Puis il s'occupe de sûreté nucléaire, toujours à Superphénix. Il rejoint ensuite l'ingénierie des Réacteurs à Neutrons Rapides (RNR), au Centre Lyonnais d'Ingénierie d'EDF, et travaille à la fois pour Superphénix et Phénix. Il reprend une fonction d'exploitant en 1996 et retourne à Phénix comme chef du service Entretien (maintenance). Et le cycle se poursuit, il retourne donc à Superphénix, comme chef de mission Sûreté, Sécurité, Environnement, Radioprotection et aborde un aspect nouveau, le démantèlement. Puis, en 2004, sa carrière professionnelle s'oriente vers les REP, d'où une période de deux ans d'infidélité aux RNR. En 2006, EDF s'engage dans des travaux relatifs à la quatrième génération. Il intègre alors le SEPTEN avec une petite équipe d'anciens des RNR, appelée projet GEN IV, puis projet RNR-Na/ASTRID. Dans un premier temps, il établit et analyse un important retour d'expérience des RNR. Aujourd'hui, en tant qu'expert RNR-sodium / Systèmes et Fonctionnement, l'un de ses principaux rôles consiste à produire une assistance au CEA pour l'avant-projet du réacteur ASTRID.

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Superphénix

Les acquis techniques et scientifiques

Édition 2016

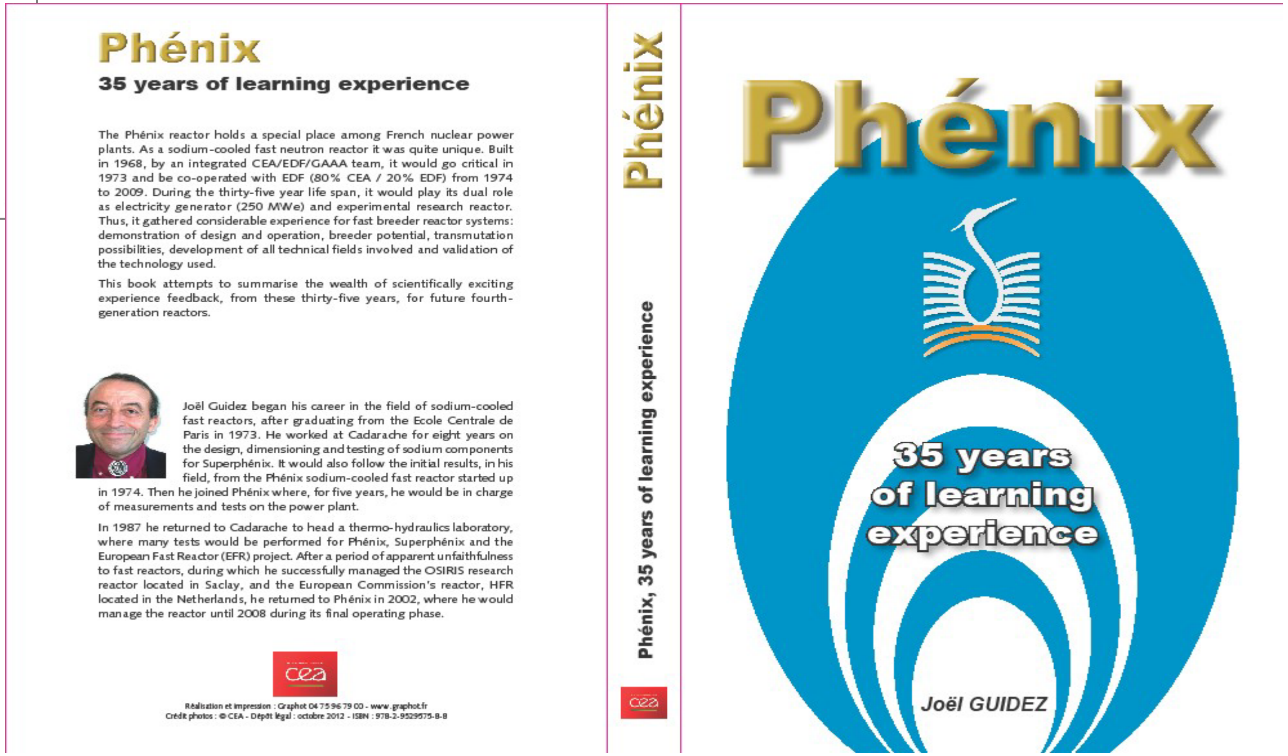
Joël Guidez

avec le concours et l'expertise de

Gérard Prêle

Superphénix

Les acquis techniques et scientifiques



Phénix

35 years of learning experience

The Phénix reactor holds a special place among French nuclear power plants. As a sodium-cooled fast neutron reactor it was quite unique. Built in 1968, by an integrated CEA/EDF/GAAA team, it would go critical in 1973 and be co-operated with EDF (80% CEA / 20% EDF) from 1974 to 2009. During the thirty-five year life span, it would play its dual role as electricity generator (250 MWe) and experimental research reactor. Thus, it gathered considerable experience for fast breeder reactor systems: demonstration of design and operation, breeder potential, transmutation possibilities, development of all technical fields involved and validation of the technology used.

This book attempts to summarise the wealth of scientifically exciting experience feedback, from these thirty-five years, for future fourth-generation reactors.

Joël Guidez began his career in the field of sodium-cooled fast reactors, after graduating from the Ecole Centrale de Paris in 1973. He worked at Cadarache for eight years on the design, dimensioning and testing of sodium components for Superphénix. It would also follow the initial results, in his field, from the Phénix sodium-cooled fast reactor started up in 1974. Then he joined Phénix where, for five years, he would be in charge of measurements and tests on the power plant.

In 1987 he returned to Cadarache to head a thermo-hydraulics laboratory, where many tests would be performed for Phénix, Superphénix and the European Fast Reactor (EFR) project. After a period of apparent unfaithfulness to fast reactors, during which he successfully managed the OSIRIS research reactor located in Saclay, and the European Commission's reactor, HFR located in the Netherlands, he returned to Phénix in 2002, where he would manage the reactor until 2008 during its final operating phase.

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Phénix, 35 years of learning experience

Phénix

35 years of learning experience

Joël GUIDEZ

Summary

1) ESFR SMART history
2) Methodology to increase safety
3) Core design
4) Containment and primary circuit
5) Decay heat removal
6) Secondary loops
7) Final Layout
8) Status of simplifications and improvements
9) Status of passive systems
10) Necessary R&D
11) Conclusion

ESFR SMART history

- In 1988, while the European Superphénix reactor was in operation, a new European (Sodium) Fast Reactor (EFR) project, with a slightly higher power of 1500 MWel, was launched in collaboration between France, Italy, Germany, and the UK. This project was stopped by the shutdown of Superphénix reactor and was closed by a final file summarizing all the options selected.
- On this basis, a project called CP ESFR (Collaborative Project on ESFR), was initiated a few years later to "groom" EFR options and integrate the new technical developments.
- It is on this new basis that a project called ESFR-SMART started at the end of 2017 mainly with the objective of integrating the new safety rules resulting primarily from the Fukushima accident.
- This project tries to include all the long European experience on SFR, not only from Phenix to Superphenix but also from project studies such as EFR or ASTRID.

Working horse or concept car?

- The ESFR-SMART project is what in the Anglo-Saxon world is called a "working horse" or a ”concept car.” Its role is to introduce, outside of any constructive planning, new ideas for the future, which can be valuable guides for R&D. Unlike in an “industrial” project, which initially had a construction schedule, one can introduce innovative ideas, even if their lower technological-readiness level would require development and time.
- For these new ideas, research and first calculations have been performed to check their general feasibility and the absence of major impossibilities.
- A status of necessary R&D to provide is given in the conclusion.

Methodology to reach new safety requirements post-Fukushima

- Try to avoid dedicated systems for incidents: simple is safe
- Use the advantages of sodium especially for natural convection
- Use the feedback experience of reactors (PX, SPX) and of projects as EFR, ASTRID
- Use the practical elimination methodology, to try as far as possible, to suppress by design a lot of known possible incidents.
- A final verification of accordance of the final design to new safety rules post Fukushima has been provided in a dedicated task.

Feed-back experience of reactors

- The EFR project was initiated during the end of SPX1 operation, the biggest SFR built and operated in the world.
- On the basis of this SPX experience, modifications were proposed in EFR design on several SPX design options. These propositions are very interesting because they were made on the basis of this huge and unique experience.
- They are mainly in the ESFR SMART design as:
  - Massive metallic roof
  - Non oscillating primary pumps
  - Modular steam generators

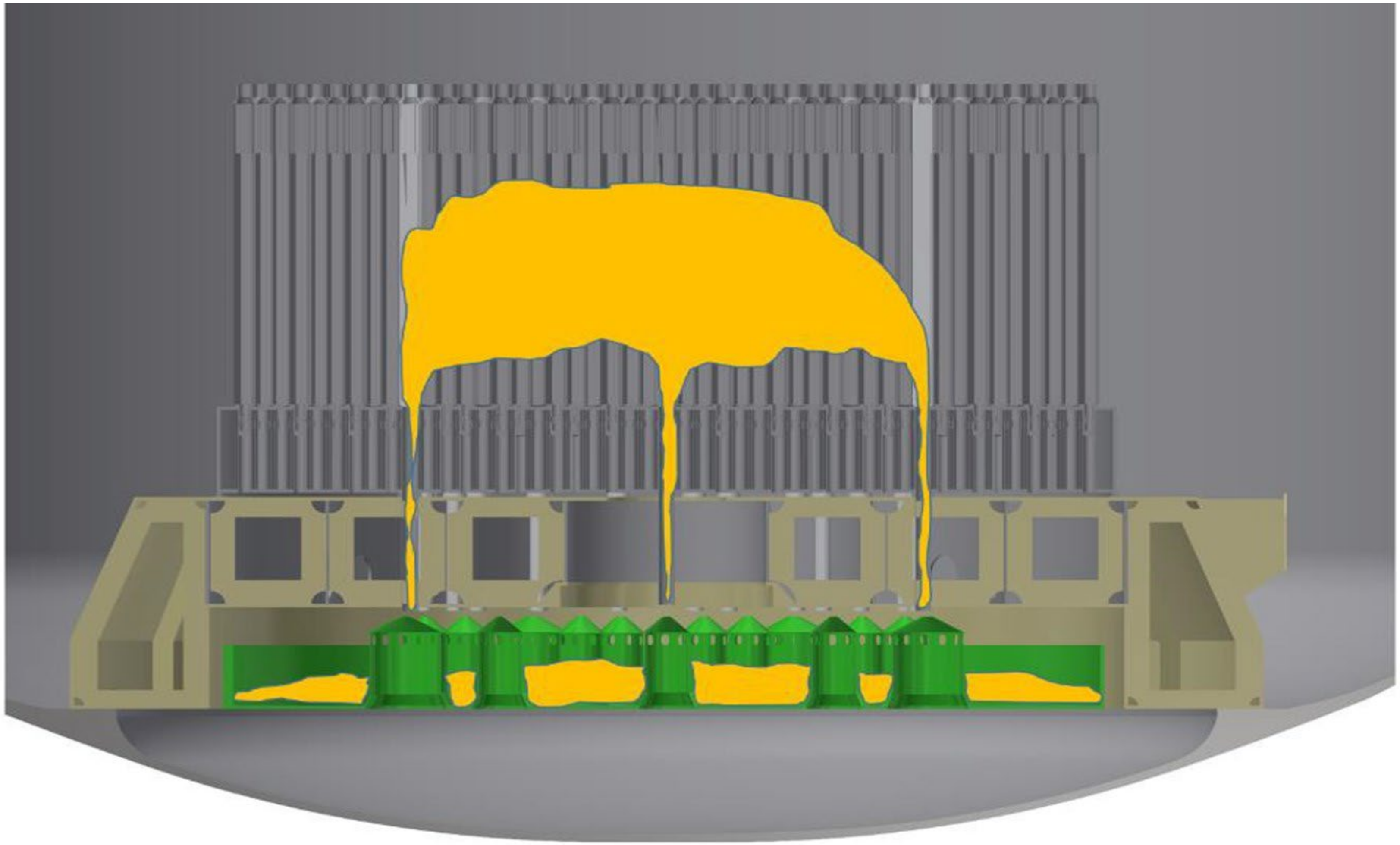
Simple is safe

- We tried to propose a design of reactor very simple because simple is safe.
- So rather than adding devices to improve safety, we simplified the reactor design to promote passive or intrinsic safety, with a great grace period for the operator.
- We will resume in conclusions, how we simplified the design (suppression of the dome, of the safety vessel, of the DHRS systems inside the primary vessel, ..) and how we increased the passive systems (control rods, passive decay heat removal, thermal pumps,...)

Core design (1)

- Void effect
  - The core design is issued of ASTRID works to minimize the void effect. A lot of various dispositions as diameter of pins, plenum above the core, mix of fertile and fissile inside the core, etc. allow to obtain a global almost zero void effect.
- Passive control rods
  - These 12 control rods must stop the reactor if a physical parameter is reached (temperature or flowrate). They operate without any control command order.
- Tubes for corium discharge
  - Several tubes are in the core arriving above the core catcher
  - In case of severe accident with core melting, they should conduct the corium to this core catcher

Artistic view of the preferential ways for the melted core



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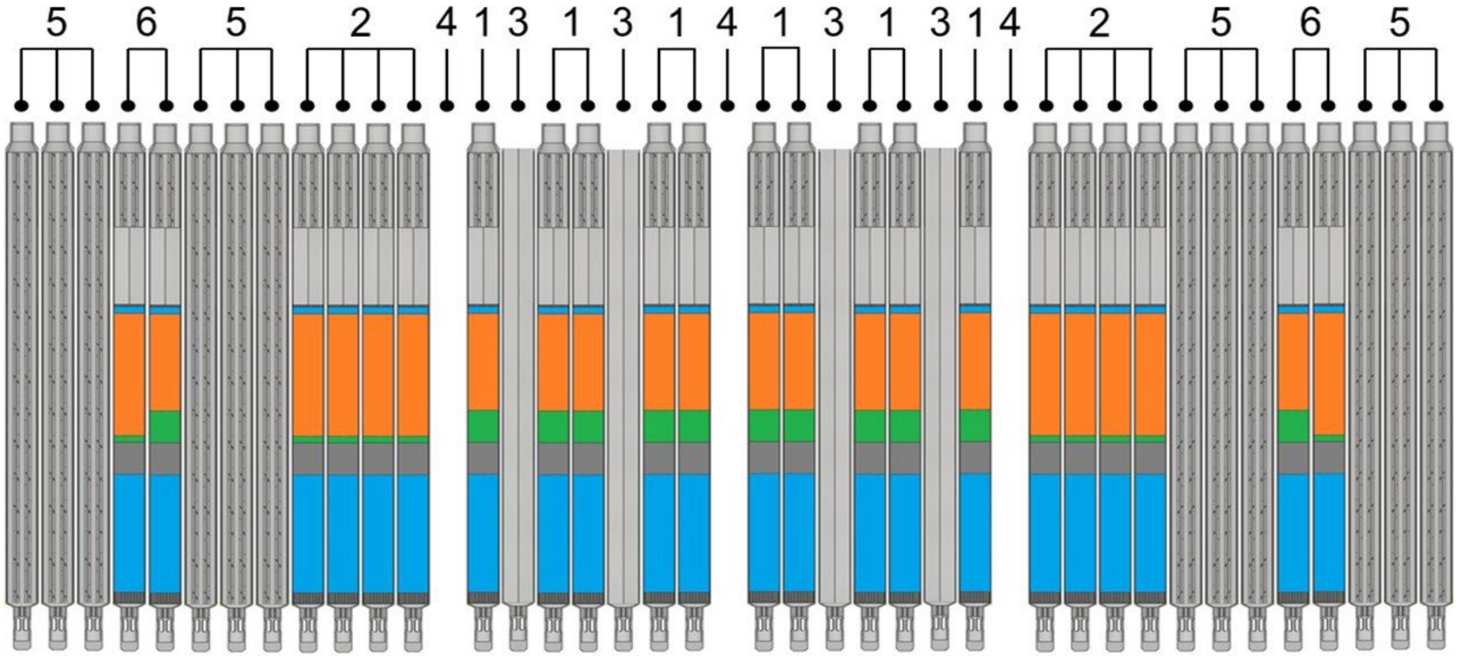
Core design (2)

- We have in this core, 24 active control rods responding to usual safety criteria, especially for diversification, with two different types of control rod.
- As for other SFR as PX or SPX, pads on the fuel assembly are in contact when the power of the reactor begins to increase. So, a compaction of the core is not possible when the reactor is in operation. To verify that there is no modification of the global core geometry during the operating cycle, some ultra sonic measurements can be provided.

View of ESFR SMART core



1 – Inner zone SA
2 – Outer zone SA
3 – Control assembly
4 – Corium discharge path
5 – Shielding SA
6 – Internal spent fuel storage

Fissile fuel (~18% Pu content)
Fertile blanket
Steel blanket
Fission gas plenum
Sodium plenum
Shielding (absorber)

Containment and primary circuit

The design of the primary circuit is mainly characterized by following improvements

- Suppression of safety vessel
- Suppression of possibilities of primary sodium leaks that allows suppression of dome or polar table
- Suppression of dedicated circuits for decay heat removal inside the primary circuits
- New core catcher for better mitigation

Safety vessel functions

- All existing sodium fast reactors, built or operated, have a safety vessel around the main reactor vessel.
- The safety vessel function is to contain the primary sodium in case of the main vessel leakage, while avoiding lowering the primary, sodium-free level below the exchanger inlet windows which would have the effect of interrupting the cooling of the core by natural convection.
- In this accidental situation, the reactor will never start again and an unloading of the core is necessary. But this unloading will take at least one year because it is necessary to wait until the decrease of assemblies' residual power. The other function of the safety vessel during this accidental situation is to maintain this sodium without leakage in the pit during more than one year.

View of SPX safety vessel

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Why to suppress the safety vessel ?

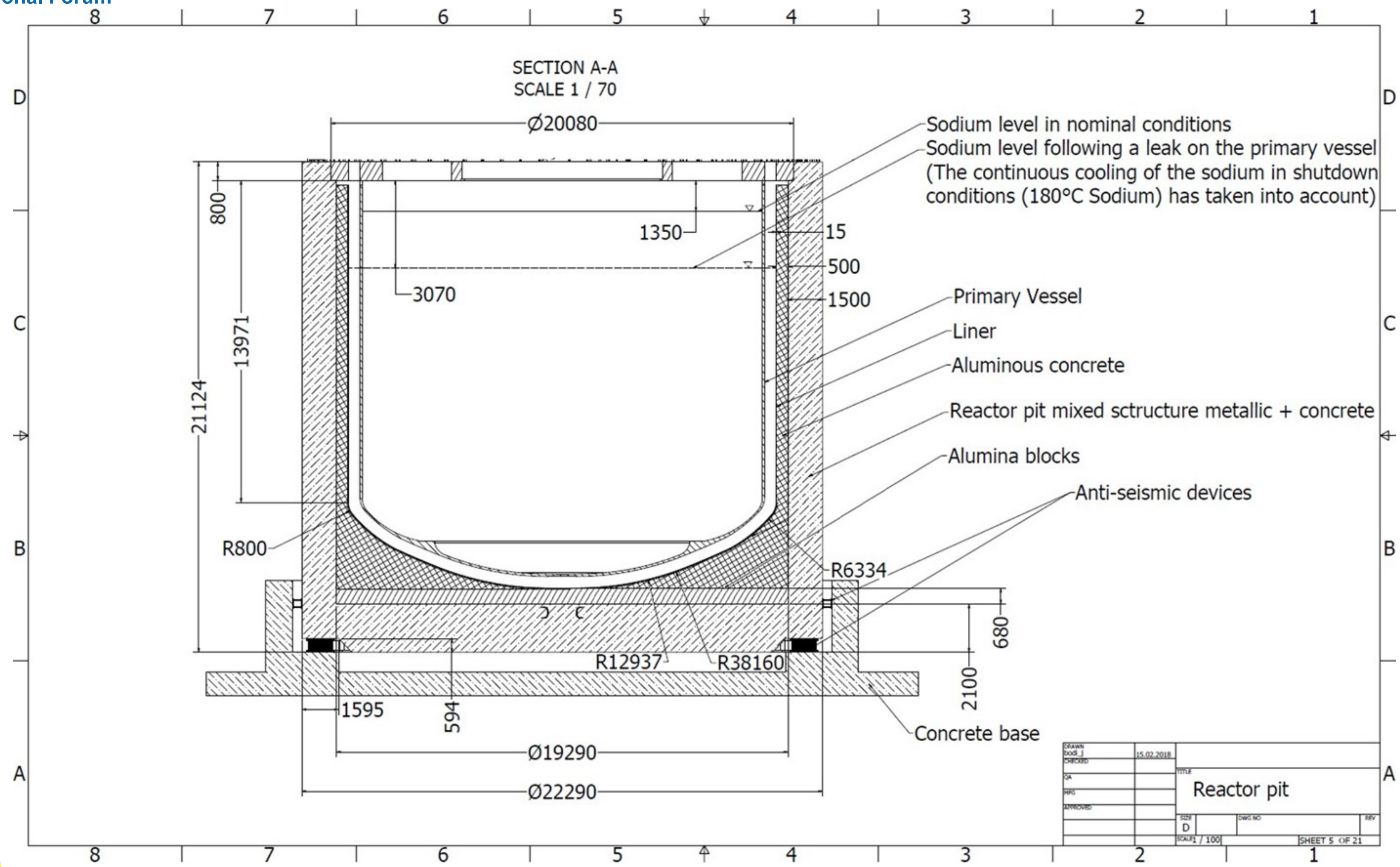
- A number of measures have been taken to prevent leakage of the safety vessel, as slight overpressure between the two vessels to detect a possible initial leak and choice of different materials to avoid a common failure mode on corrosion.
- But the scenarios of main vessel leakage are diverse, from corrosion leakage to leakage on a severe accident with mechanical energy release. This leads to uncertainties in the temperatures and leakage rates, which make it difficult to demonstrate the safety vessel mechanical strength against the corresponding thermal shocks
- The evolution of safety standards leads us to look at other options where its functions could be directly taken over by a reactor pit capable of withstanding a sodium leak, and thus a long-term mitigation situation. It was an option that had already been looked at in the EFR project with a vessel anchored in the pit, which option was later abandoned for reasons of feasibility and design difficulties.

Advantages of suppression of safety vessel

- To regain with the pit design, the safety vessel functions, it is necessary to have a reactor pit allowing to withstand the reception of this sodium leak and to bring the pit closer to the main vessel so that the volume between vessel and pit remains identical to the volume between safety and main vessels.
- This design presents several advantages:
  - The heat screen from the safety vessel is suppressed and the power removal through the reactor pit is facilitated, which will play an interesting role to increase the decay heat removal possibilities.
  - It is answered to a question from the safety authority relating to a double leak of the two vessels.
  - The main vessel In-service inspection remains still possible
  - The final structure is better adapted to the mitigation functions because this pit design is now able to support any sodium leak.

ESFR SMART pit design proposition

1) A mixed concrete/metal structure with a water-cooling system inside the concrete supports the thick metal slab to which the reactor vessel is attached. Together with the reactor roof, it provides a sealed containment which must keep its integrity in all the cases of normal or accidental operations.
2) Above the bottom of the concrete/metal structure, blocks of insulating materials (non-reactive with sodium) are installed in the lower part. Alumina is selected as reference material for these insulation blocks.
3) A metallic liner is placed on the surface of these insulation blocks. The gap between the reactor vessel and the liner is small enough to avoid decrease of the primary sodium free level below the intermediate heat exchanger (IHX) windows in case of sodium leakage from the reactor vessel and after sodium temperature drop.
4) An oil cooling system is installed on this the liner in front of the main vessel. This oil is compatible with sodium (no hydrogen in case of reaction) and can operate at high temperatures.
5) Finally, a special concrete with alumina (aluminous concrete) which could withstand, without significant chemical reaction with sodium, a leakage of the liner could be used between the liner and the external structure.



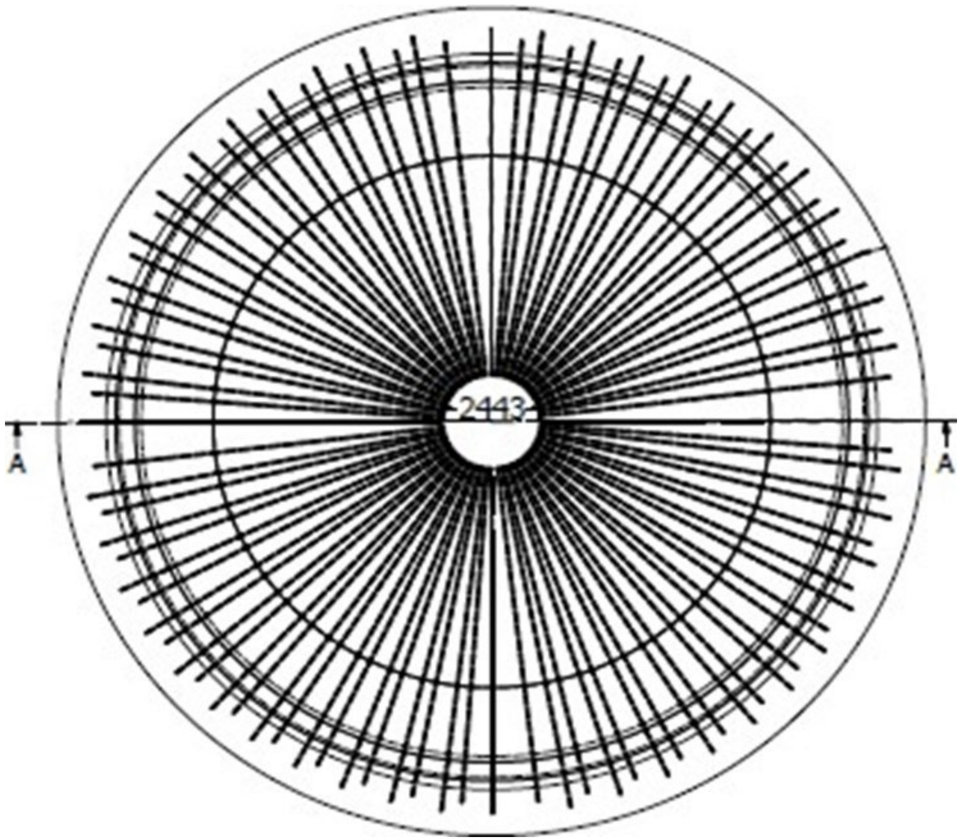
SECTION A-A
SCALE 1 / 70
Ø20080
Sodium level in nominal conditions
Sodium level following a leak on the primary vessel
(The continuous cooling of the sodium in shutdown
conditions (180°C Sodium) has taken into account)
800
1350
15
500
1500
3070
13971
21124
Primary Vessel
Liner
Aluminous concrete
Reactor pit mixed sctructure metallic + concrete
Alumina blocks
Anti-seismic devices
R800
R6334
680
R12937
R38160
2100
1595
594
Concrete base
Ø19290
Ø22290
Reactor pit
15.02.2018
SHEET 5 OF 21

Two cooling systems

- Two independent active cooling systems are proposed in the reactor pit:
    1) The oil cooling system close to the liner The oil under forced convection can remove the heat transferred by radiation from the reactor vessel at high temperature. Conversely to water, the adopted synthetic oil is resistant to high temperatures above 300°C and reacts with sodium without producing hydrogen. As an example, the commercial oil called “Therminol SP” can be used in normal operation at temperatures up to 315°C.
    2) The water-cooling system for the concrete cooling is installed in the concrete and aims at maintaining the concrete temperature under 70°C in all possible situations, especially if the oil system is lost.
- Both oil and water circuits work during normal operation and must maintain the concrete temperature below 70°C. This margin is intended to ensure the concrete integrity and to protect it from any thermal degradation
- In case of the reactor vessel leak and loss of the oil system, the water system is able to remove the decay heat generated by the core and to maintain alone the concrete below 70°C.

View of the oil system

The oil system is made of several independent oil circuits on the liner



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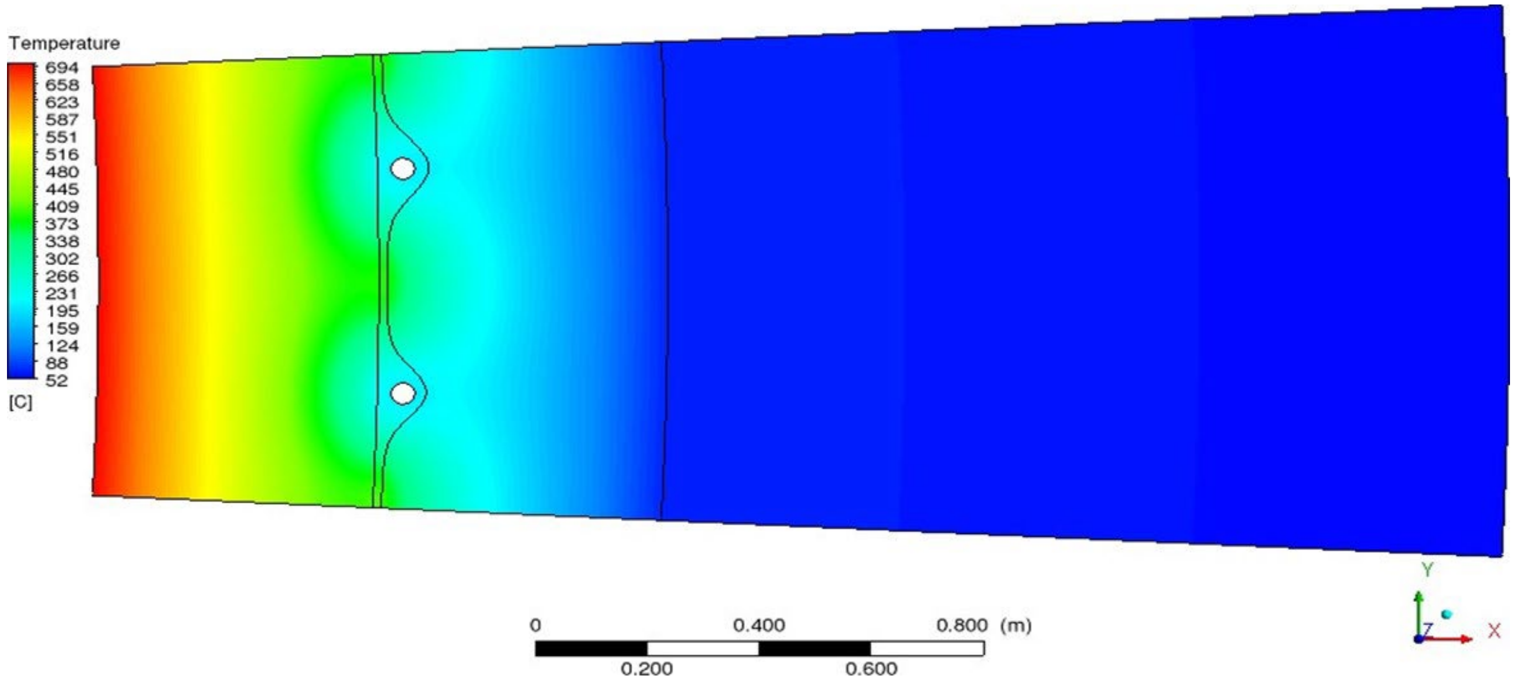
Preliminary thermal calculations

Preliminary thermal calculation of the pit were provided by JRC for the following three main scenarios:

- Scenario 1: Normal operation: The main vessel is at about 400°C. The operation of the oil cooling system is sufficient to maintain the correct thermal conditions in the pit (i.e., less than 70° C for the concrete of the mixed structures).
- Scenario 2: Operation in exceptional decay heat removal regime: The safety studies should take into account exceptional situations of successive losses of decay heat removal systems. In this case, in exceptional situations of Categories 3 and 4, the reactor vessel is allowed to reach temperature of 650°C. The two cooling systems (oil and water) must make it possible to maintain the concrete temperature below 70°C while playing an important role in the decay heat removal.
- Scenario 3: Operation in accident situation of sodium leakage: In this situation, vigorous sodium cooling is possible with the redundant and available DHRS, to bring the sodium to a temperature corresponding to the handling temperature (180°C).Therefore, the temperature of the sodium in the gap should not exceed 200°C. The demonstration of the oil cooling system availability in case of reactor vessel leakage is difficult and we assume as hypothesis that the oil cooling system is no longer available. The operation of the water-cooling system alone must be sufficient to maintain the concrete temperature below 70°C

Pit thermal calculation

A modelisation of the pit has been provided for calculation and presented in ICAPP 2019

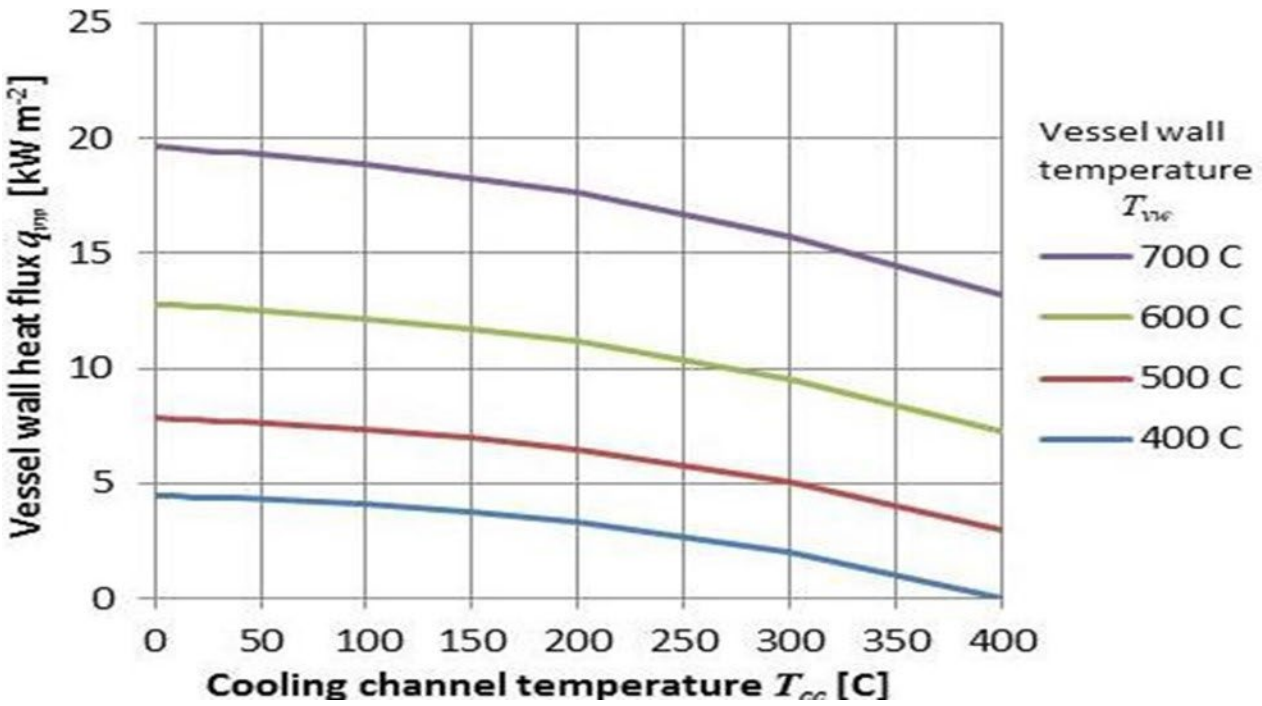


Calculation results

- Scenario 1: At nominal power, about 3 Mw are extracted by the oil circuit if we maintain this oil at 200°C. In this situation, the concrete of the concrete/metal structures is under 70°C even if the water circuit is not operating.
- Scenario 2: In exceptional regime of decay heat removal (situations in category 3 and 4) the system can then remove (the main vessel being at 650°C), a power of about 15 MW. The liner remains at about 200°C and the concrete remains easily under 70°C.
- Scenario 3: in case of sodium leakage, we don't even try to demonstrate that the oil circuit are always in operation. But with the water circuit alone, we maintain the concrete temperature under 70°C.

Improved decay heat removal capabilities

- The figure below gives values of power removed by the oil cooling system at different temperatures of the reactor vessel and of the oil cooling channel.
- The heat screen of safety vessel is suppressed, that allows to increase the decay heat removal possibilities of the pit. With a vessel at 650°C, we can extract about 15 MW that means, that after about three days this oil circuit can assure alone the decay heat removal of the core.



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Conclusion on safety vessel suppression

- The studies allowed to comfort the innovative design of the reactor pit for the European Sodium Fast Reactor. The aim of the innovations is to eliminate the safety vessel and to keep its safety functions by modifying the reactor pit design.
- Two active (forced-convection) cooling systems are proposed: an oil cooling system close to the metallic liner and a water-cooling system in the reactor pit concrete structure. Preliminary thermal calculations showed that the oil cooling system could alone and be able to maintain the concrete temperature below 70°C in all scenarios. The water circuits are redundant and useful only in case of loss of the oil circuits.
- The proposed reactor pit design has several advantages: elimination of the safety vessel, better efficiency of the decay heat removal by the reactor pit cooling systems, and safer configuration in case of accidental or mitigation situations.

To know more on pit organization and calculation

- These results have been presented in ICAPP 2019 symposium, jointly with JRC/Petten, EDF and PSI.

  “European Sodium Fast Reactor: innovative design of reactor pit aiming at suppression of safety vessel.”

  –Joel Guidez, Antoine Gerschenfeld (CEA),Aleksander Grah , Haile Tsige Tamirat (JRC Petten), Konstanton Mikityuk, Janos Bodi (PSI), Enrico Girardi (EDF)

- They are also presented in the NERS papers as “New Reactor Safety Measures for the European Sodium Fast Reactor Part I : Conceptual Design” and “New Reactor Safety Measures for the European Sodium Fast Reactor Part II: Preliminary Assessment”

Primary sodium confinement/ Massive metallic roof

- Superphenix experience feedback leads to recommend hot slabs at their bottom part (so as to minimize the aerosol deposits) and with no water cooling. This last point is also part of the demonstration of practical elimination of sodium/water reactions on the primary side.
- The EFR thick slab is therefore taken over, which presents many advantages:
    - Simplicity of operations
    - Neutron protection
    - Mechanical oversizing in the event of a severe accident
    - Limitation of thermal flows controlled by the conduction
- Its thickness will be defined by the industrial manufacturing contingencies, but should be about 80 cm. In the upper part, a heat insulator will eventually be installed so as to limit the heat flows to be evacuated during nominal conditions by airflow in forced convection or even natural convection.

Primary sodium confinement

- A lot of other dispositions are taken to avoid any primary sodium above the massive metallic roof (ref 1)
  - Components are firmly bolted and welded
  - Rotating plugs have eutectic seals solidified during operation
  - No circulation of sodium outside the vessel (integrated cold traps)
  - Low argon pressure to avoid any sodium fountain effect.
- All these measures do enable to prevent any primary sodium leakage outside the roof, and so any overpressure possibility, due to primary sodium fires.
- It allows to suppress expensive and complex systems as dome (SPX) or polar table (ASTRID), generating cost and later difficulties for the operation of the plant

View of SPX dome (22m diameter)

Suppression of decay heat removal systems inside the primary vessel

Decay heat removal is assured by systems in the secondary loops. This disposition has several advantages compared to the independent systems (DHX) located in the primary circuit (as in SPX or CP ESFR)

- No slab penetrations required. Gain on the main vessel diameter.
- A cold column is maintained in the intermediate heat exchanger, which is the guarantee of a good natural convection in the primary circuit. It was not the case with DHX with complicated transient situations.
- This circuit uses the already existing purification circuit of the corresponding secondary loop. This minimizes the number of sodium circuits to be managed by the operator. Each DHX was a sodium circuit to manage with its purification and draining system
- Less risk of sodium leak out of the primary vessel
- More resilient in case of energy release in the core

In vessel core catcher

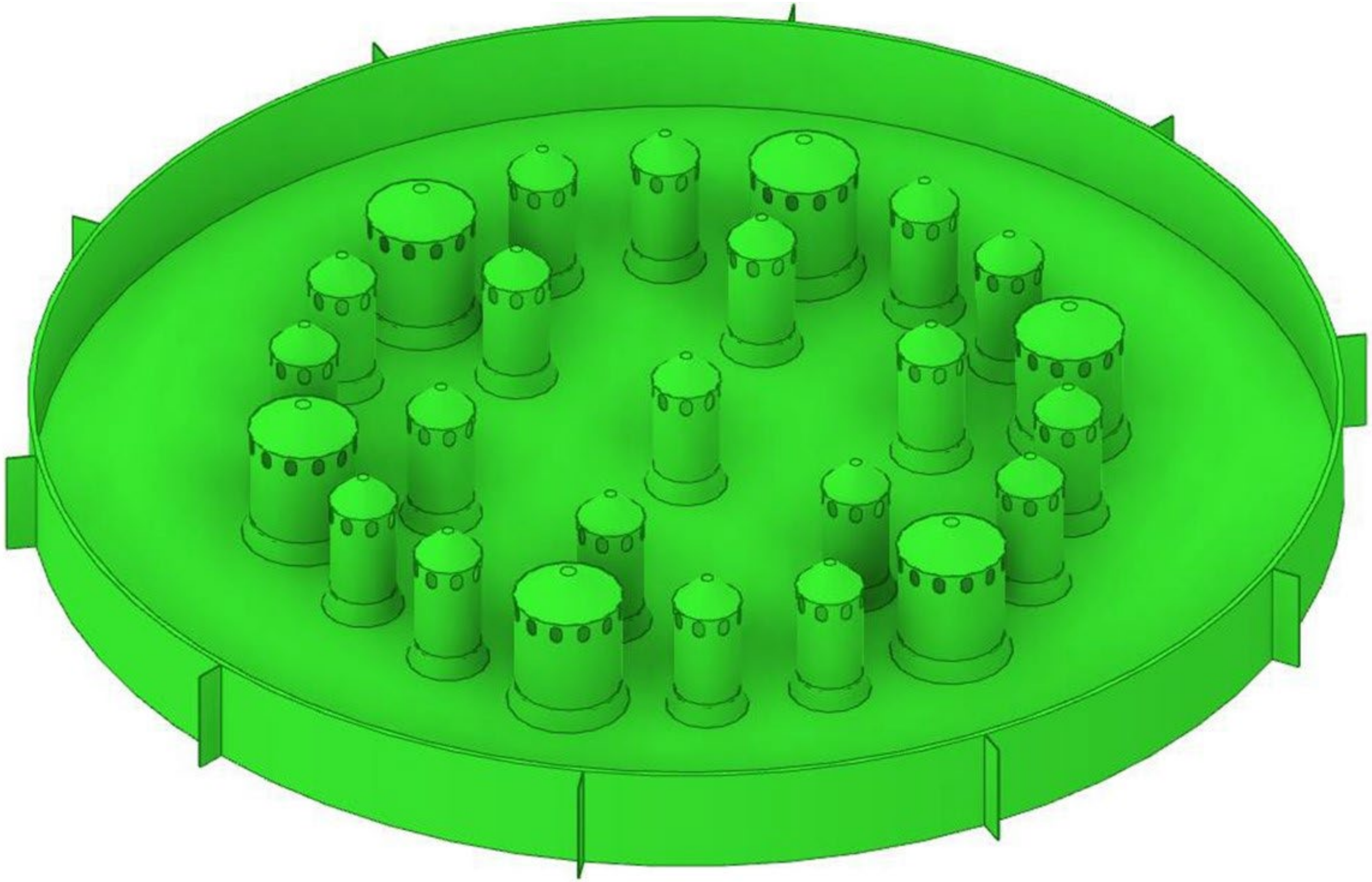
- The mitigation of a severe accident with core meltdown is achieved by means of a core catcher, located at the bottom of the vessel, under the core support plate (also called a strongback).
- Guide tubes, coming from the core, emerge above this core catcher so as to channel the molten corium. The tubes arrive above conical hats which disperse this corium.
- The use of molybdenum could prevent melting of the core catcher structures and facilitate the heat removal by conduction. The use of hafnium-type poisons in the core catcher structures enables to avoid any re-criticality.
- The core catcher volume will be dimensioned for the whole fissile core meltdown and for his cooling by natural convection of sodium.

Choice of core catcher material

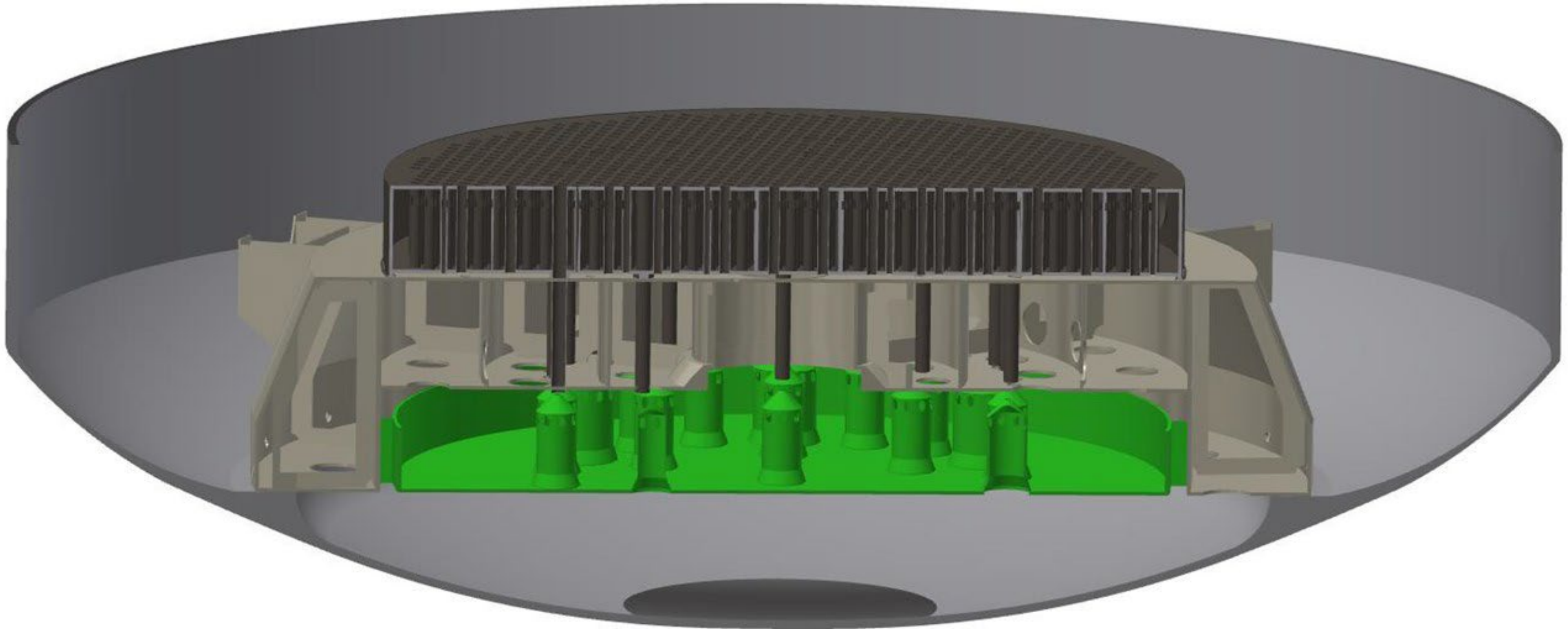
As for the core catcher of the Russian fast breeder reactor BN 800, we choose molybdenum as core catcher material

- good resistance to corium ablation
- good chemical compatibility with sodium
- very high melting point (above 2000° C)
- material available, weldable and affordable
- very good conduction coefficient
- excellent mechanical resistance against thermal shock

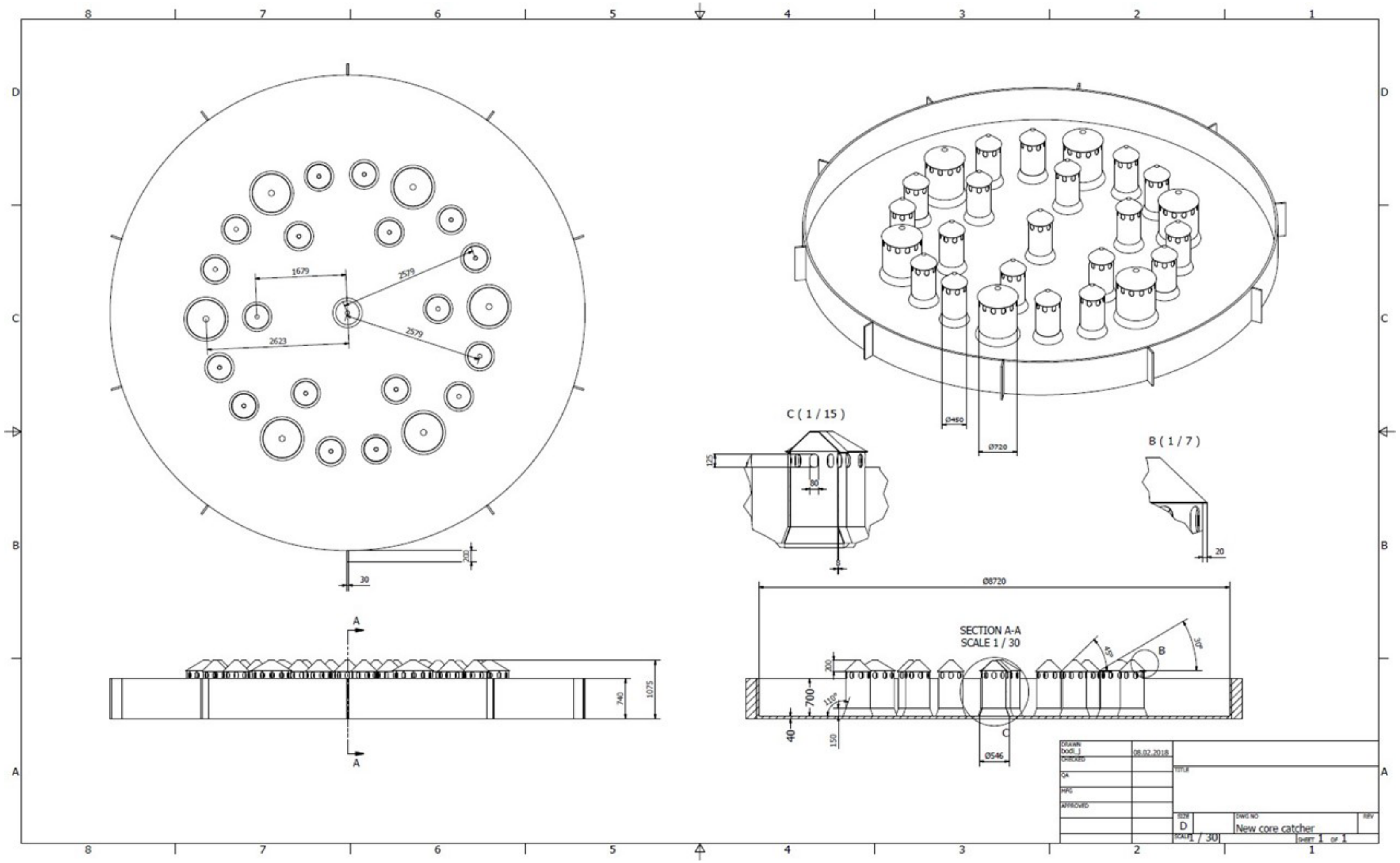
View of the core catcher



Core catcher disposition



Core catcher drawings



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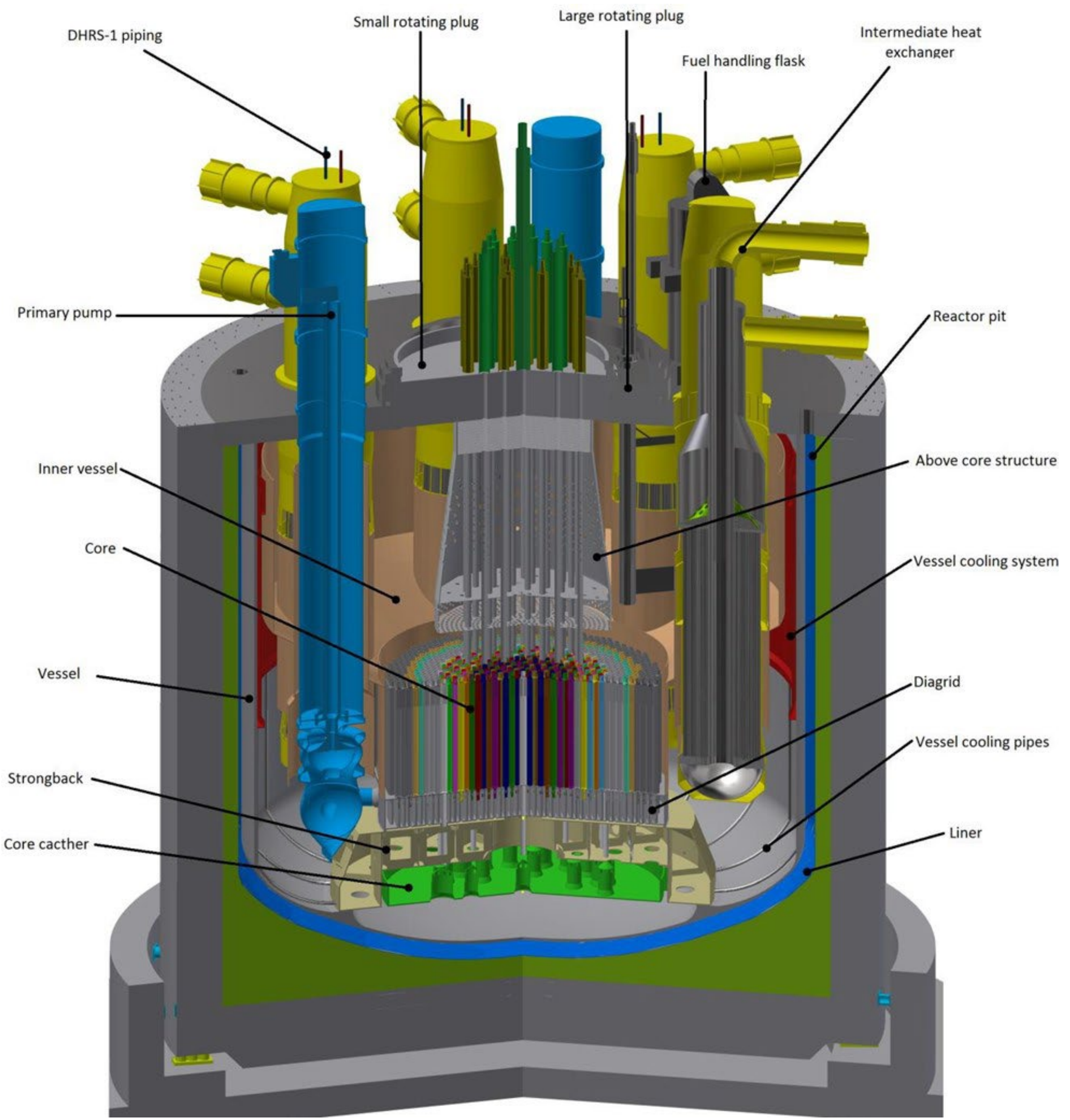
Final view of the primary vessel



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Decay heat removal systems (DHRS)

The reactor has three DHRS systems, without any DHX systems inside the primary circuit

- Six secondary loops able to evacuate in active and passive way (air cooling). (DHRS 2)
- Two active circuits in oil and water in the pit. Each circuit has several independent fluid circulation systems. The water circuit is able alone to maintain the concrete temperature under 70° C (DHRS 3)
- Six passive systems (DHRS 1) able to remove power even if the secondary loop is drained in natural convection (air cooling).

DHRS 2

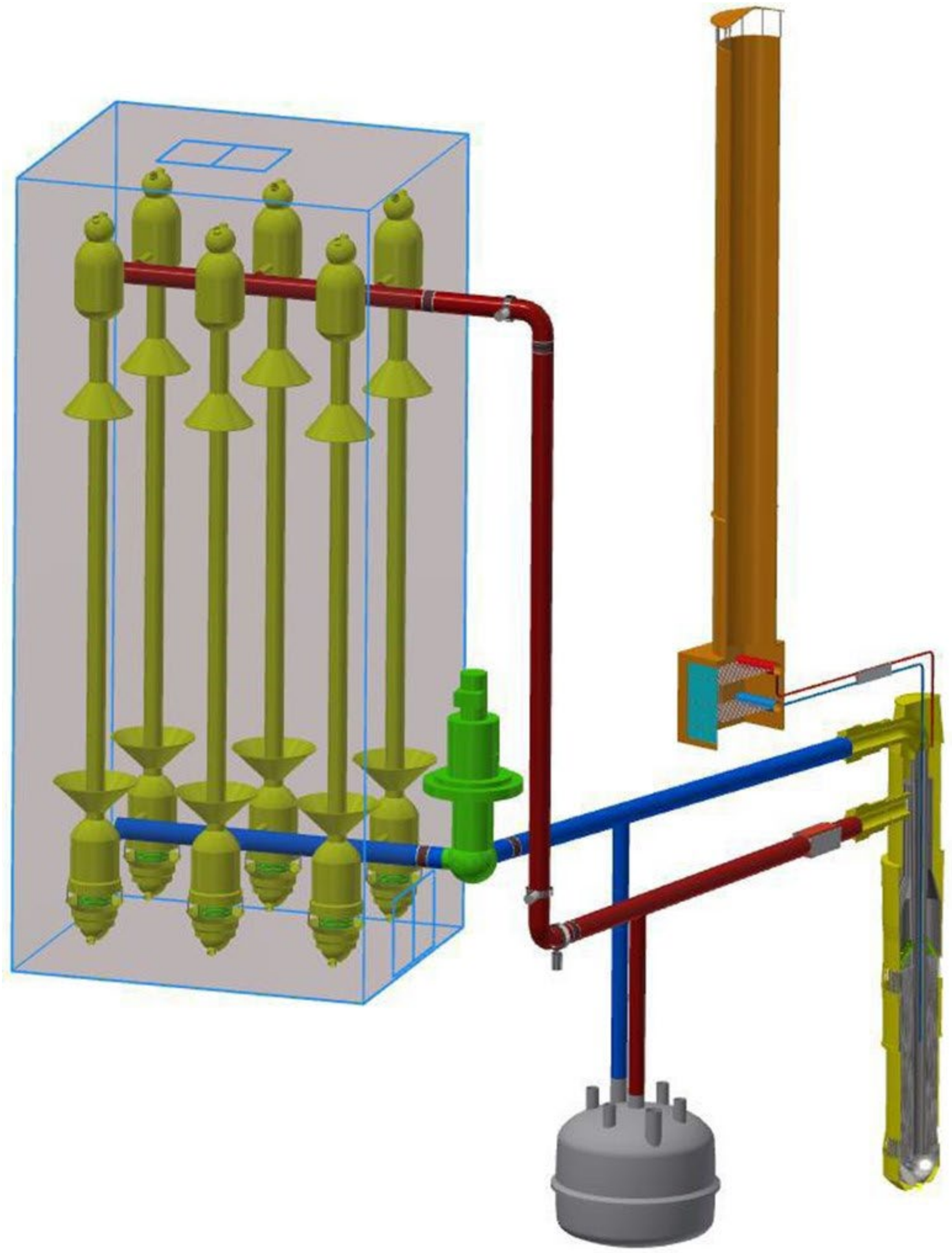
- The secondary circuits are the normal power removal circuits. Their use for this decay heat removal is very useful because that allows creating, in the intermediate heat exchanger, a cold column essential for the establishment of a good natural convection in the primary circuit.
- Their design will be optimized so as to enable a good power removal by air in natural convection, that is to say, in the extreme situation when both the cooling water and the electrical power supply have been lost.
  - The secondary loop design enables an easy establishment of natural convection.
  - One thermal pump is installed to increase the natural convection circulation flow rate (thermal pumps are passive pumps with no need of electricity supply).
- The CP ESFR option for steam generators was modular, with six modules per loop. It takes advantage of the related large exchange surface, to have opportunities for cooling these modules by air in natural convection (through hatch openings likewise at Phenix reactor). This will be the loop heat sink.

DHRS 2

There are six loops, but all calculations are made with the principle that one loop is out of order.

- In normal operation one loop is sufficient to assure the decay heat removal
- If we loose water in the steam generators, circulation of secondary sodium with open casings is OK to assure decay heat removal
- If we loose also electricity supply (Fukushima situation), low sodium circulation (low speed 100 R/Min of secondary pumps, with batteries) in the loops is able to assure alone decay heat removal
- If we stop totally the secondary pumps, natural convection in the loops is able with DHRS 3 to assure all decay heat removal
- After several hours DHRS 2 is able, without DHRS 3, to assure alone DHR

Secondary loop as DHRS 2



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DHRS 3

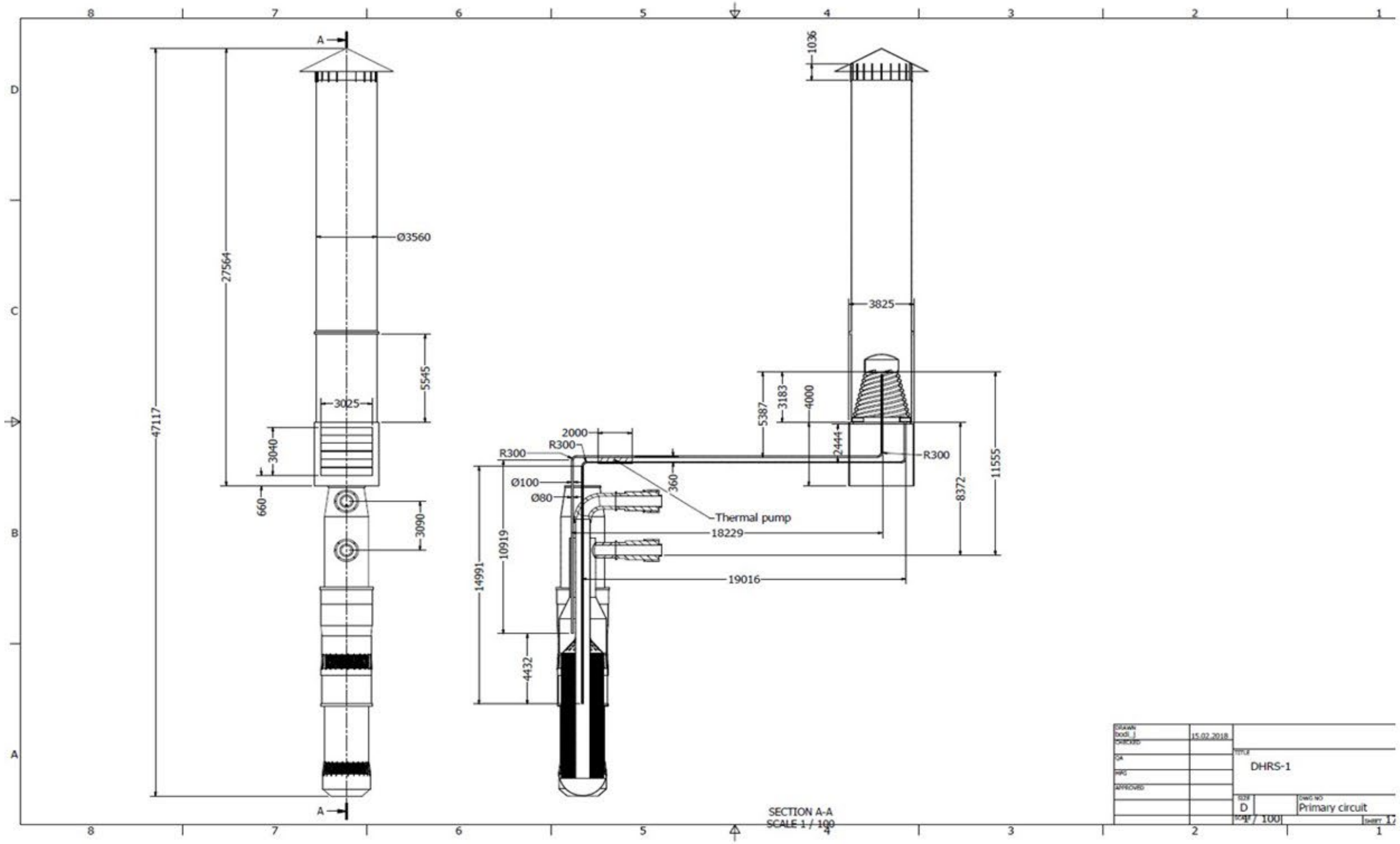
- In addition to these secondary loops, redundant cooling active circuits (located in the reactor pit) (DHRS 3) are capable to maintain the entire pit at temperatures below 70 °C.
- Suppressing the safety vessel makes these devices much more efficient, and able to assure a more efficient decay heat removal.
- These circuits are redundant (3 or more) and can assure when the primary vessel is at 650°C, a power extraction of about15 MW
- There are active circuits.
- After three days, these circuits can assure alone the decay heat removal of the plant.

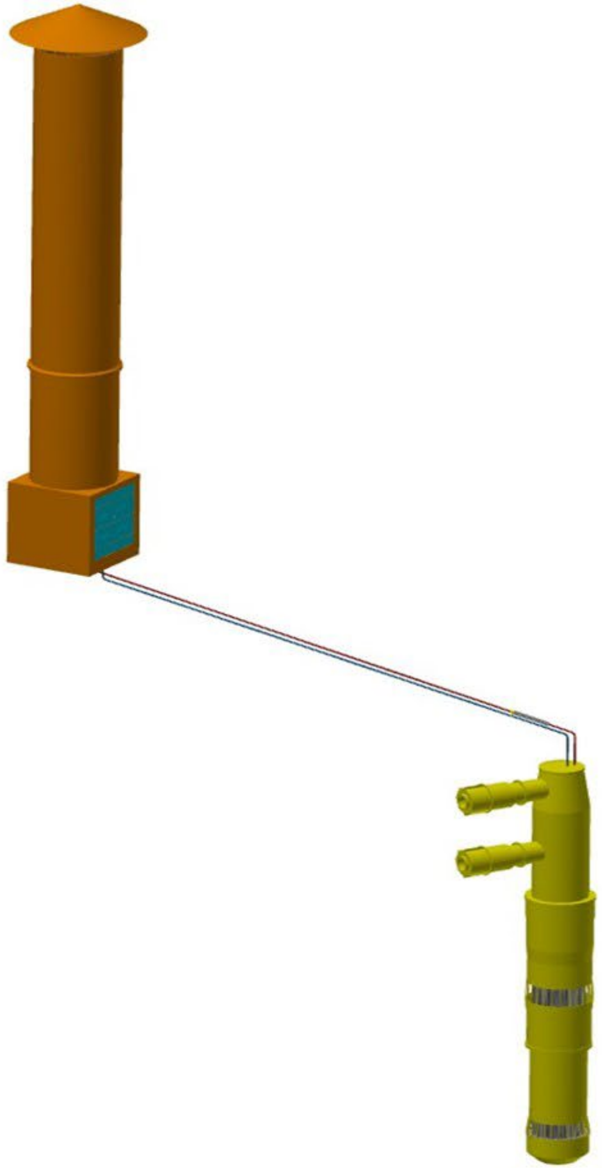
DHRS 1

- DHRS1 are cooling circuits, with sodium/air heat exchangers connected to the intermediate heat exchangers and out of the primary vessel.
- This DHRS circuit operate in natural convection, but the addition of a thermal pump can further increase its capabilities and help for the starting of the operation without freezing risk.
- This circuit remains active even if the loop is drained.
- In case of ADC, these circuits are out of primary vessel and protected.

Advantages of DHRS1

- These circuits, DHRS 1 have several advantages compared to the independent systems
- (DHX) located in the primary circuit (as in SPX or CP ESFR)
- No slab penetrations required. Gain on the main vessel diameter.
- A cold column is maintained in the intermediate heat exchanger, which is the guarantee of a good natural convection in the primary circuit. It was not the case with DHX with complicated transient situations.
- This circuit uses the already existing purification circuit of the corresponding secondary loop. This minimizes the number of sodium circuits to be managed by the operator. Each DHX was a sodium circuit to manage with its purification and draining system.
- Less risk of sodium leak out of the primary vessel
- More resilient in case of energy release in the core





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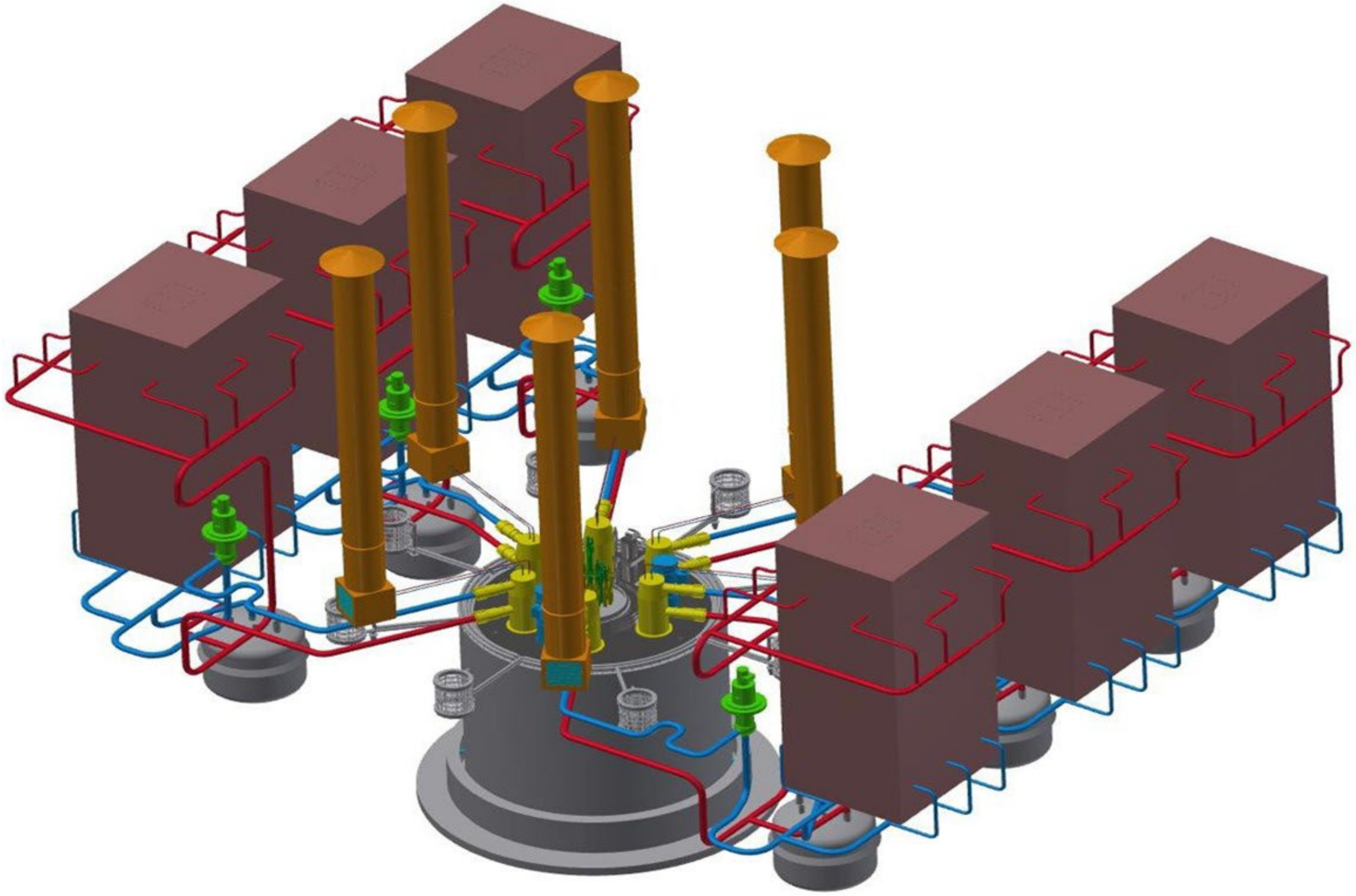
DHRS Capabilities

- The secondary loops (DHRS2) and the pit oil circuits (DHRS3) are independent. DHRS3 can assure about 15 MW and after 3 days can assure alone the decay heat removal.
- DHRS2 can assure in natural convection more than 20 MW. It is able together with DHRS 3 to assure the decay heat removal of the plant in all situations.
- With a low speed of secondary pumps (100 mp) it is able alone to assure DHR
- Even if these circuits were lost, the DHRS 1 is able to assure alone the function without need of water or of electricity.
- DHRS1 are six independent systems, totally passive and using air always available. These six DHRS 1 are able together to assure 100% of the decay heat removal (36 MW with sodium at 650°C)

Global safety acceptance

- The probability of prolonged loss of residual power systems must be less than 10-7.
- For this it is admitted in a simplified way, that the system of evacuation of residual power must comprise two lines of strong defenses (i.e., reliable and capable to assure alone the evacuation of residual power) and a line of weak defense (rule 2a + b)
- In our case, the first calculations made it possible to demonstrate, in broad outline, that DHRS 1 and 2 could constitute the strong lines of defense and that the DHRS 3 the weak line of defense.

View of the three DHRS systems (in the disposition with flexible pipes)



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To know more

- ICAPP 2018 symposium: "New safety measures considered for European Sodium Fast Reactor in Horizon-2020 ESFR-SMART project" Joel Guidez (CEA), Andrei Rineiski (KIT), Sophie Esther (AREVA), Enrico Girardi (EDF), Konstantin Mikityuk (PSI)
- ICAPP 2019: "Status of new safety measures considered for European Sodium Fast Reactor in the ESFR-SMART project" Joel Guidez (CEA), Janos Bodi, Konstantin Mikityuk (PSI), Enrico Girardi (EDF), Andrei Rinieski (KIT)
- ICAPP 2021: "Innovative decay heat removal systems in the ESFR SMART project." Joel Guidez, Antoine Gerschenfeld (CEA)Konstantin Mikityuk, Bodi Janos (PSI, Enrico Girardi, Jeremy Bittan (EDF), Aleksander Grah (JRC/Petten)
- ASME NERS special ISSUE 2022 : " New safety measures for the ESFR Part II Preliminary assessment"

Secondary loops

- As already explained, the secondary loops were designed to assure decay heat removal without help of DHX inside the primary vessel.
- Another big proposition was to suppress flexible pipes and to use straight pipes with bellows
- Another proposition was to use thermal pumps
- The secondary loop design increase the safety against water/sodium reactions easier

  to detect and to manage with modular steam generator
- The sodium fires in case of leakage are also easier to detect and to manage with straight pipes option.

Difficulties with flexible pipes

Feedback from operating experience on PX and SPX shows a certain number of difficulties with flexible pipes

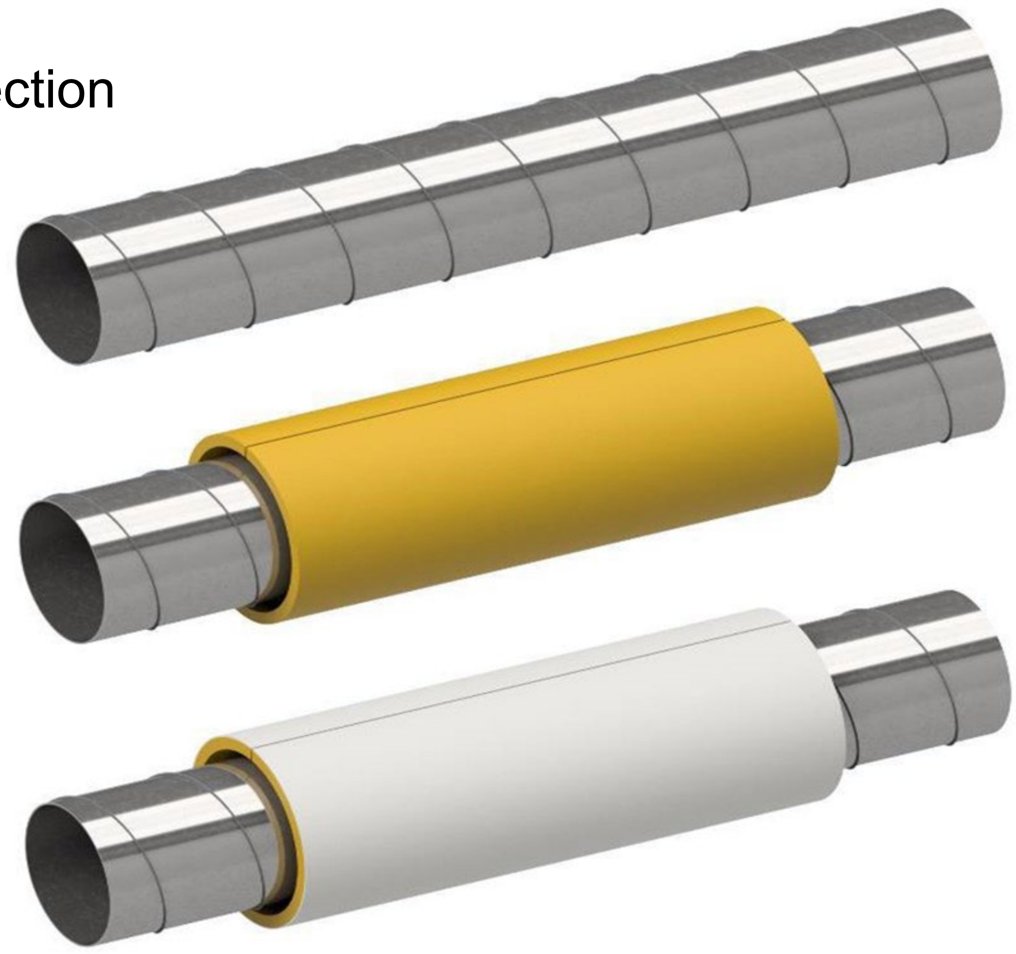
- The expansion of the pipes are important and difficult to manage to allow this free expansion
- With new safety rules on seismic conditions, the pipes fixations are not compatible with these expansion
- The length of the pipes is increased, and also the number of welds, and so the risk of sodium leak
- The insulation is difficult to separate from the pipes because of the differences in expansion. This complicates the leak detection with a lot of false alarms
- In case of sodium leak corrosion effects can occur between sodium/insulation and pipes

Advantages of straight pipes

- Straight pipes have a lot of potential advantages:
  - Reduced length of piping (220m to 120m), reduction of the sodium volume (116 m3 to 57 m3)
  - Reduced number of welds and reduced risk of sodium leaks at this level.
  - Easier operation and possibility of credible anti-seismic systems
  - Possibility of reducing the distances between fixed points / Significant savings on secondary buildings
  - Possibility of having separate insulation from the piping / simpler and safer sodium leak detection
- Reduced risk of pipes corrosion with the insulation in case of sodium leak
- Better possibilities of leak detection with an offset thermal insulation

View of external and removable insulation

- Possibility of improvement of sodium leak detection
- No false alarms



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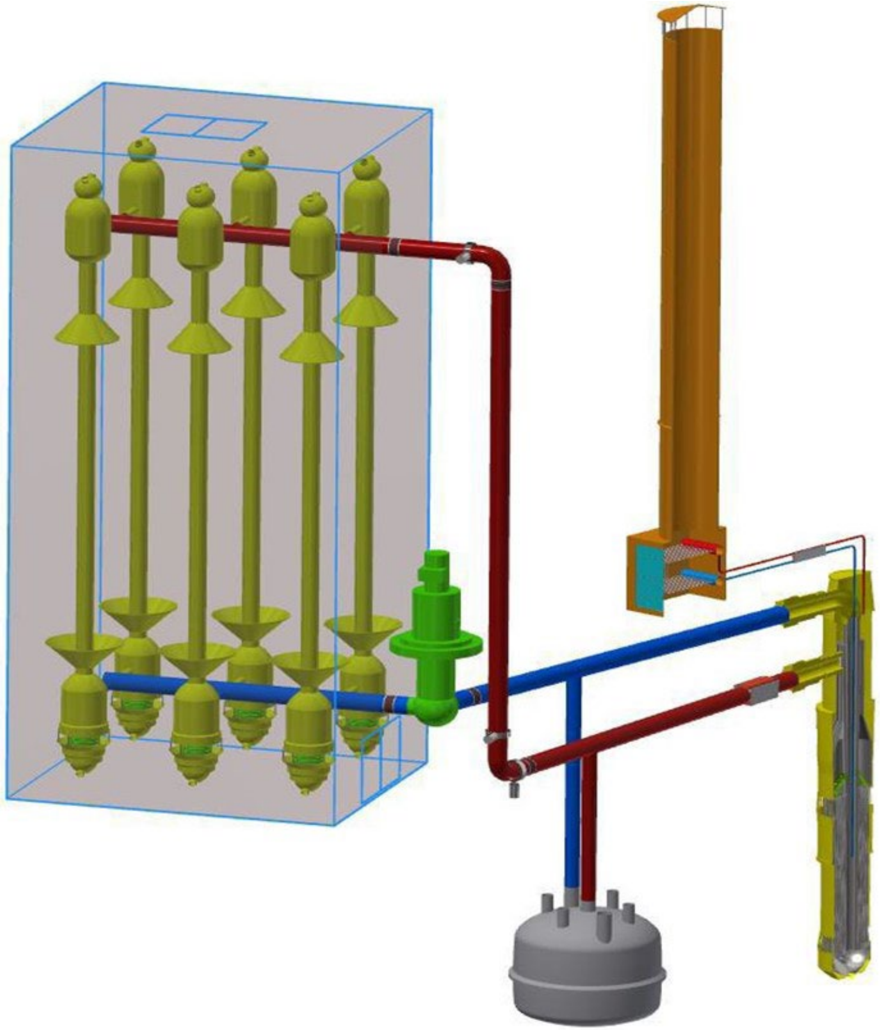
View of a secondary loop with straight pipes and bellows

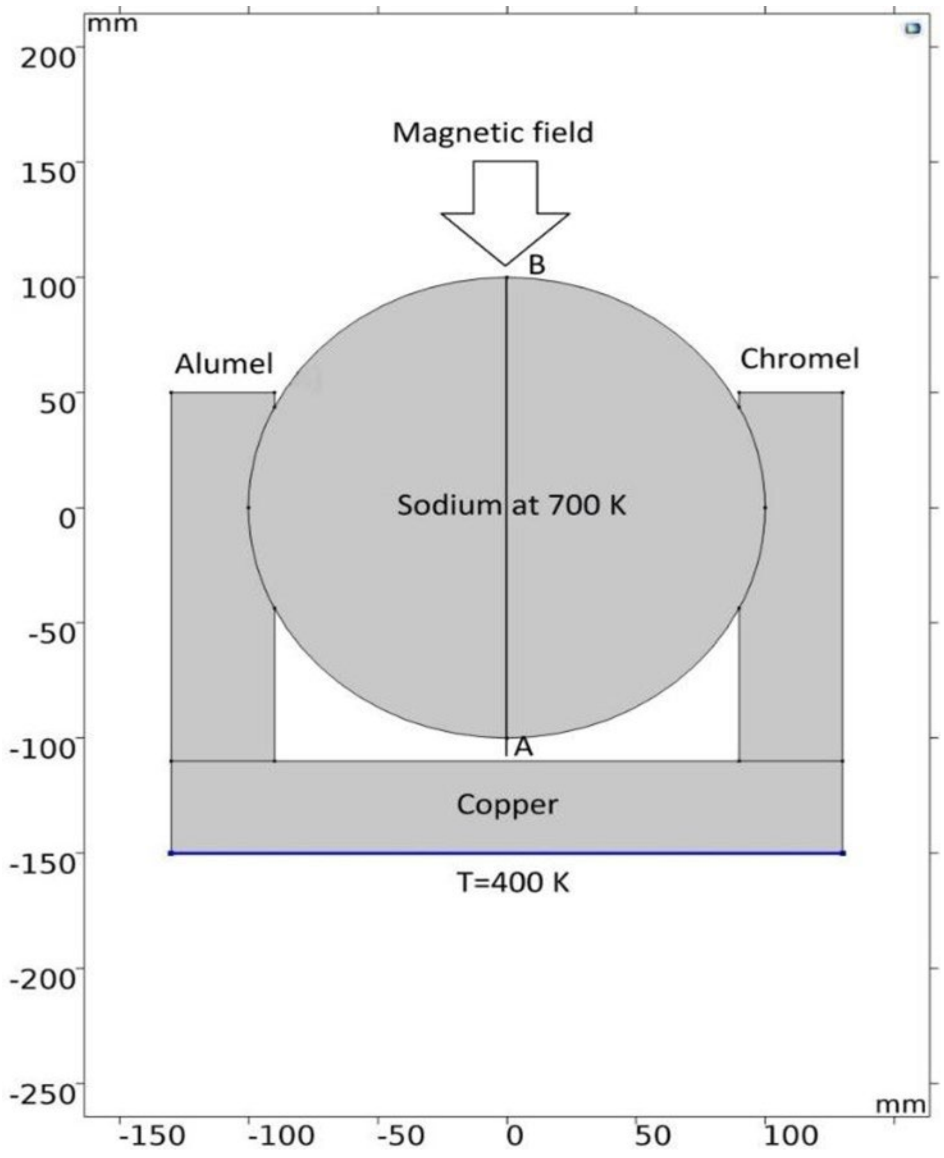


R&D on bellows

- With straight pipes it is possible to bring near components, to reduce the pipes length and so to reduce dilatations
- Some materials can be used with lower dilatation coefficient
- Remaining dilatation is taken by bellows
- Some R&D is necessary for these bellows of diameter 850 mm in terms of dilatation capacity and lifetime. However, the use of bellows in sodium is not very new. These bellows exist on many sodium valves especially in Phenix and Superphenix, and inside the Phenix heat exchangers.
- A bellow of large diameter (approximately 800 mm) was installed in Superphenix on the internal part of the hot collector of the intermediate exchangers to take up the differential expansions with the external part. On the ESFR SMART steam generator module itself, a bellow of large diameter (750 mm) exists to allow relative dilatation between the external wall of the SG, and its internal bundle

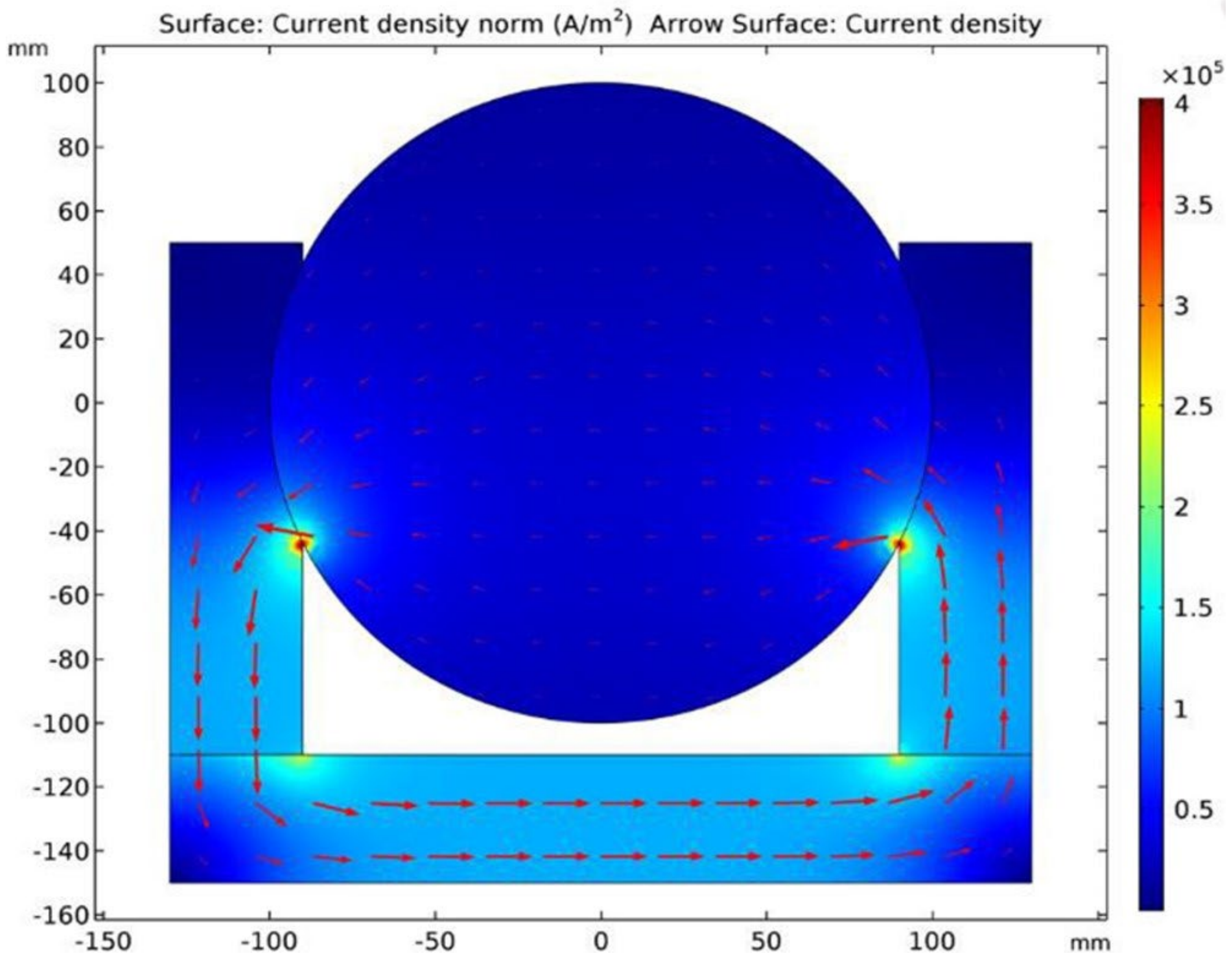
Thermal pumps principle

- The thermal pump is a passive electromagnetic pump, which uses thermoelectricity generated by the difference in temperatures (hot sodium and atmospheric air). This pump does not need an external electricity supply and provides a supplementary flow rate.
- A magnetic field is created by permanent magnets. An electric current is produced by the attached thermo-elements, being exposed to a temperature gradient. The resultant magnetic field and electric current initiate a pressure increase and a flowrate in the liquid metal coolant.

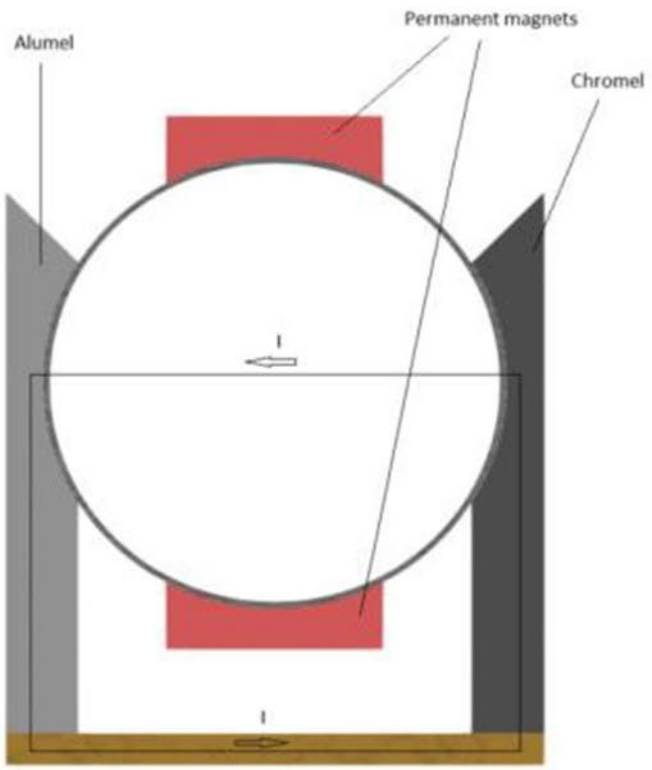


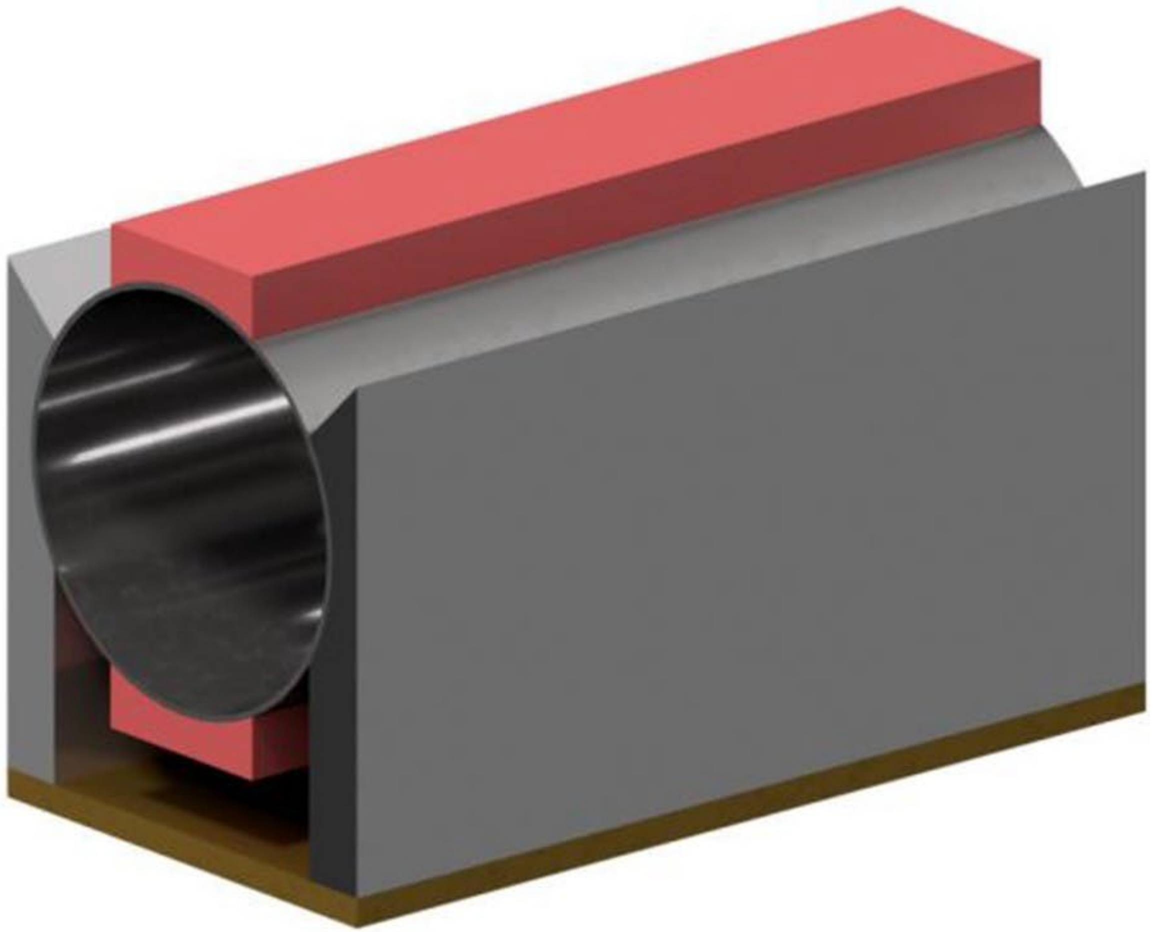
Thermal pumps utility

- These pumps provide a passive flowrate to increase the flowrate issued from natural convection in the DHRS 1 system and also on a secondary pump pipe, if useful.
- They could be replaced by electromagnetic pumps with emergency electric current. It would assure the same function but more expensive, more complicated and without passivity.



DHRS1 thermal pump





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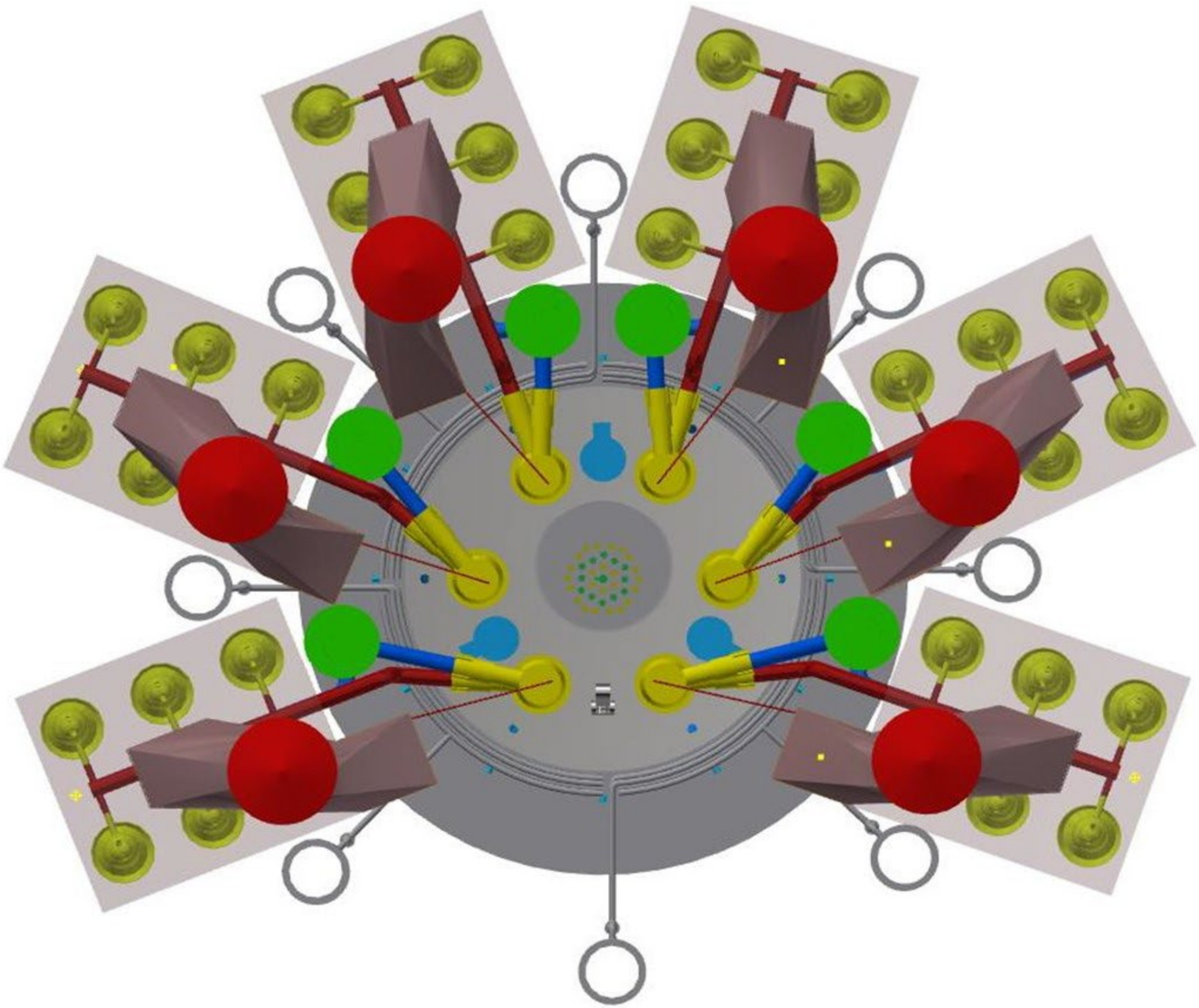
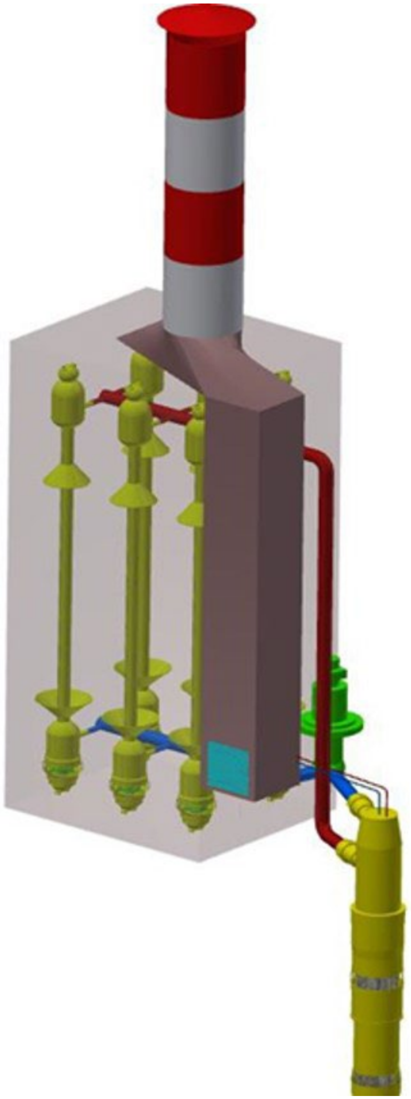
Final ESFR SMART layout

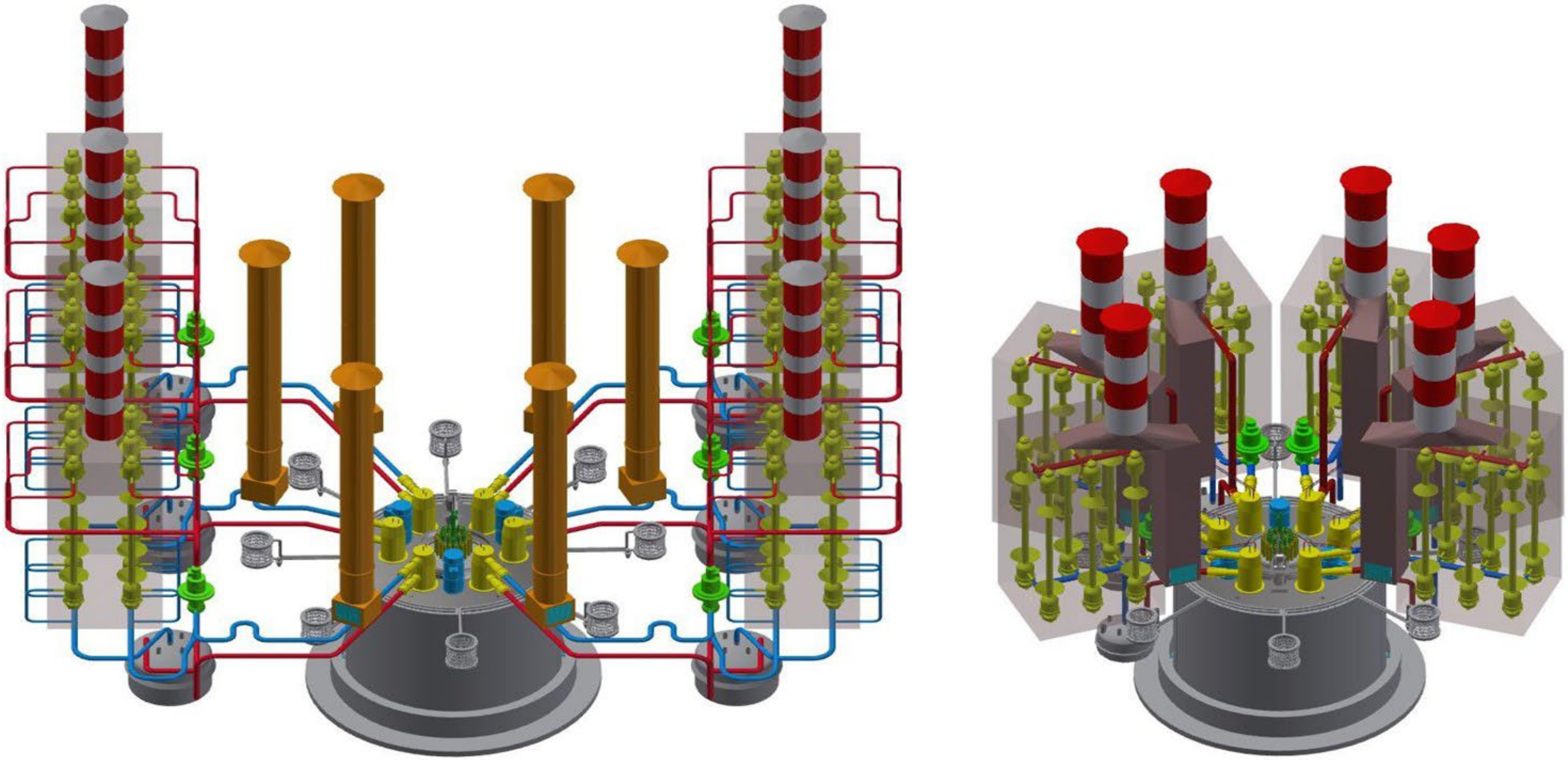
- Secondary loops with straight pipes allow a circular disposition of the loops around the primary vessel This disposition allows:
  - Reduction of the length of the pipes by a factor two
  - Reduction of the secondary sodium volume by a factor two
  - Reduction of all related auxiliary systems (sodium storage,..)
  - Significant savings on the size of the secondary buildings (their area being proportional to the square of the distance between reactor vessel and steam generator casing)
  - With this disposition, DHRS 1 and casing are very near and it is possible to have the save chimney for the two components
- Note: in order to reduce costs, the layout of the buildings is made so that the handling building can serve two reactors.

Circular disposition of the six loops with only one chimney per loop





Original plant layout with flexible pipes versus new circular layout with straight pipes



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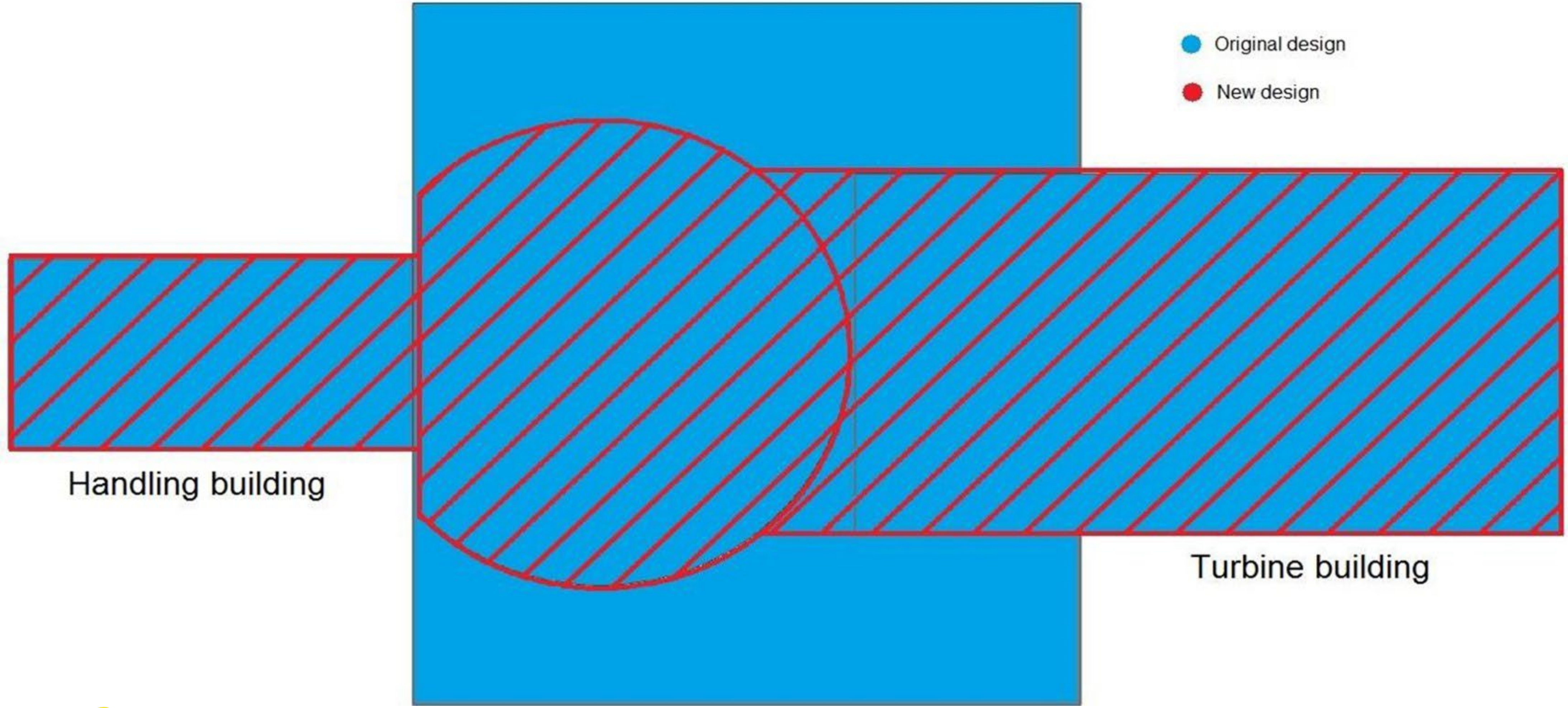
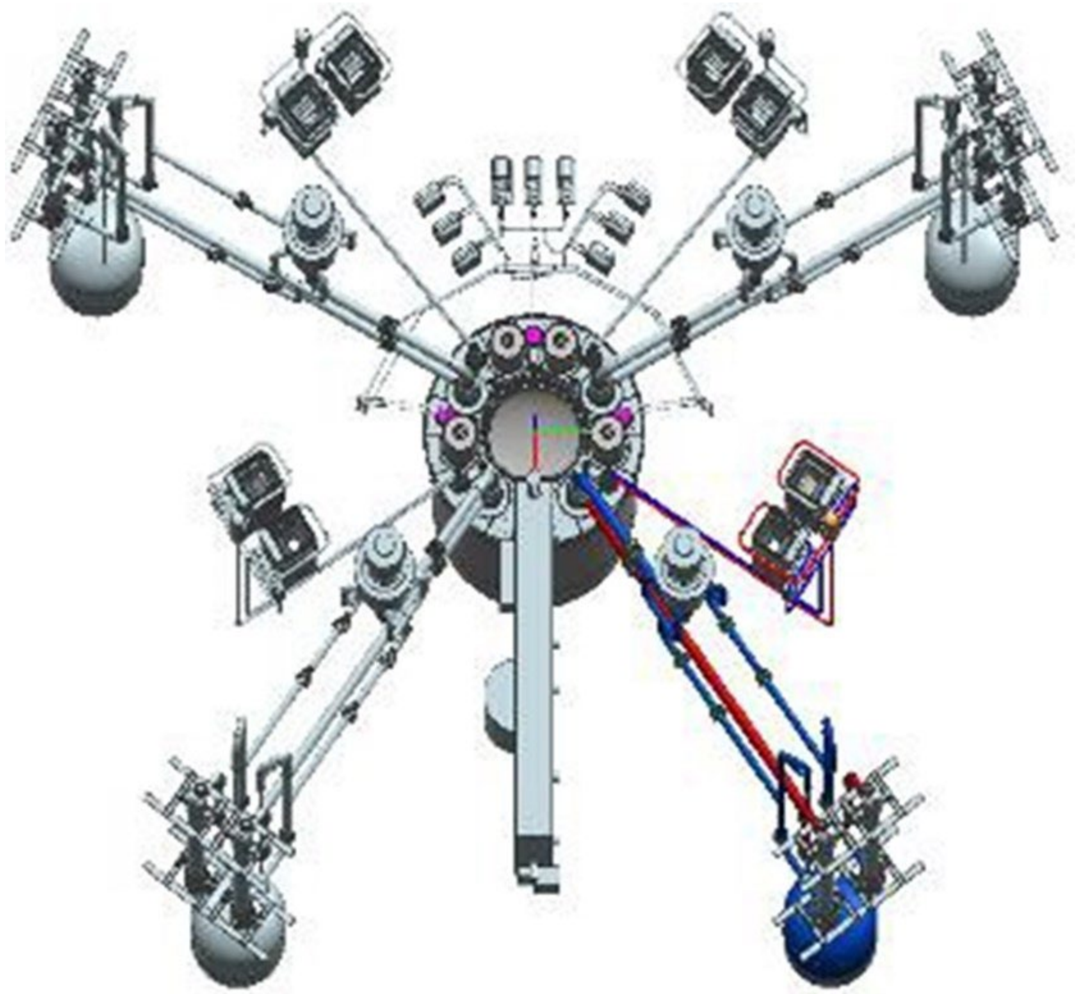
Comparison of the secondary building between the two options



This disposition (straight pipes and circular disposition) is also proposed in the BN 1200 project



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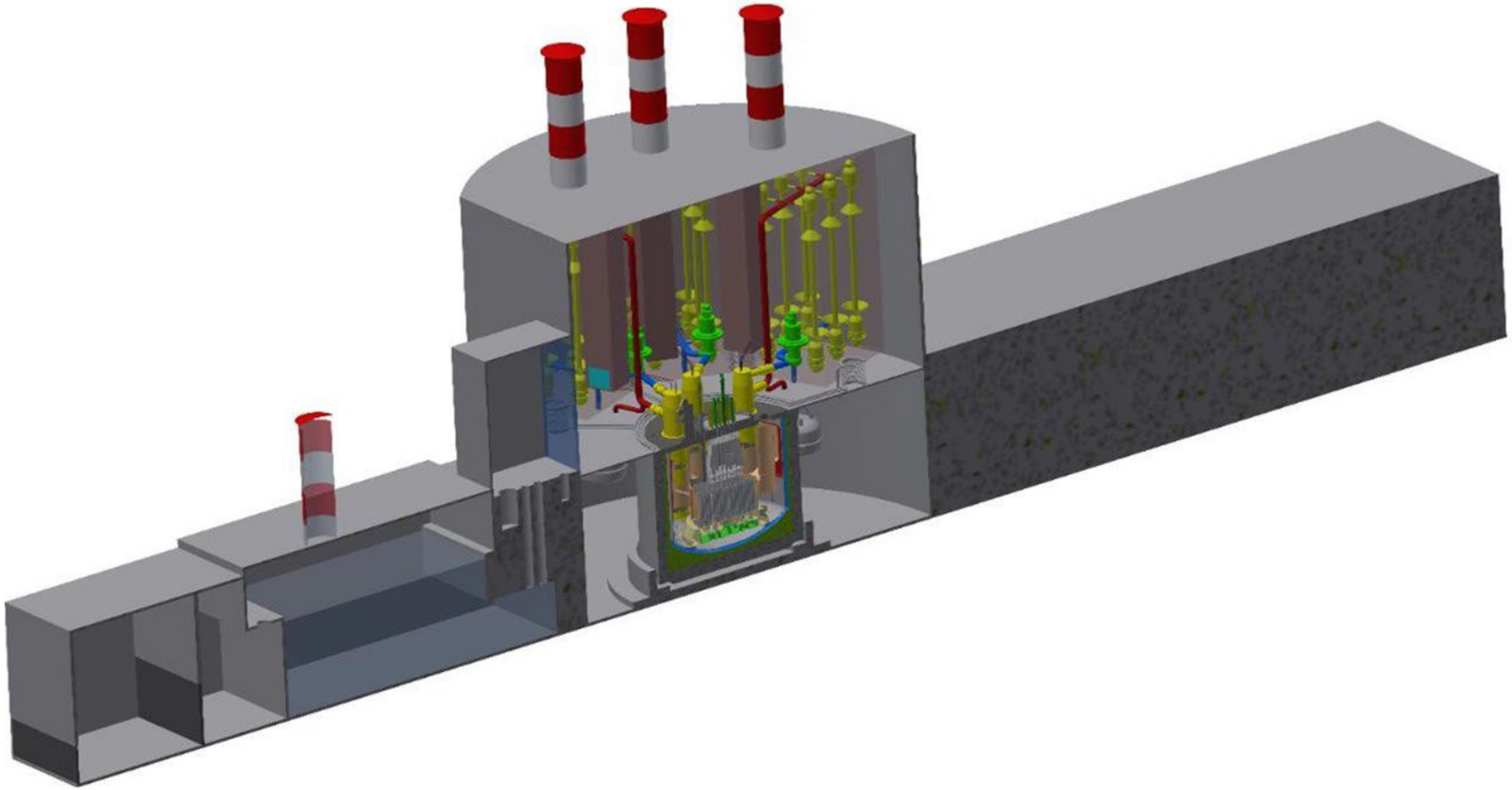
Final layout of the plant , with the secondary building around the primary vessel, the turbine on the right and the handling building on the left



To know more

- ASME NERS spécial ISSUE 2022 : “Secondary loop optimization of the ESFR as part of the ESFR SMART project”

- FR 22 “ESFR SMART Secondary loops optimization”

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Status of simplifications and improvements

- Suppression of safety vessel: functions taken over by the reactor pit
- Suppression of dome (or polar table) due to primary sodium containment improvement with a massive metallic roof and other dispositions.
- Natural convection cooling enhancement in the secondary side
- Optimized and simplified DHRS circuits (no DHX system in the primary vessel, and no supplementary sodium circuits to manage).
- Gain of about 50% on the secondary loops with straight pipes
- General lay out gain with circular disposition of these secondary loops

Note: no cost exercise was provided, but these simplifications should also provide, together with safety improvements, gain on cost

Status of passive systems and intrinsic safety

- Passive control rods stop the plant on physical parameters and without any I&C order
- Low void effect in the core help to cope with severe transients (ULOF, etc.)
- Passive decay heat removal by DHRS 2 and 1 (12 independent loops in natural convection) using only air, always available.
- Thermal pumps totally passive to increase flow rate in natural convection and the decay heat removal systems capabilities
- Long delay before necessity of operator action, even in case of simultaneous loss of water and of electricity supply.

Status of mitigation situation

Due to the core design with a void effect very low, the probability of severe accident is reduced, and, in case of severe accident, the energy released is lower. Nevertheless, a more robust design is proposed for this severe accident mitigation:

- A core catcher is provided at the bottom of the vessel, designed for the whole core meltdown.
- Mitigation devices inside the core (corium discharge tubes) are intended to channel the molten fuel to the core catcher.
- The re-criticality of this core should be made impossible by using dedicated material such as hafnium inside this core catcher or discharge tubes.
- The coolability of this core catcher should be provided by natural convection in sodium
- The reactor pit should accept sodium leakage and, with its massive metallic roof, should form a solid, tight and that-can-be-cooled containment system. There is no primary sodium ejection.
- The corium long-term cooling should be managed by the diversified cooling measures provided in the SG and in the pit (DHRS-2 and DHRS-3).
- The use of DHRS-1 circuits may be done as a supplement so as to continue the reactor block cooling even with the six secondary circuits being drained and pit cooling lost
- These DHRS1 systems are protected in case of ADC because out of primary vessel
- The temperature of pit concrete remains easily under 70°C even if the oil circuit is lost

Necessary R&D

- The ESFR SMART design requires few R&D that would relate mainly to the following points.
  - **Industrial confirmation of the proposed organization for the reactor pit.**
- The organization proposed is based on developments already made for EFR, as test of a dedicated concrete without reaction with sodium. The global organization of the pit should be validated.
  - **Industrial validation of the manufacturing method of the EFR-type thick slab.**
- Thick slab needs thick welds. This type of operation has already been manufactured. But the global organization of the slab fabrication, with a part in factory, and final welding on site, must be managed by the industrial
  - **Qualification of low-expansion materials and large-diameter bellows for the secondary circuit**
- **R&D on thermal pumps**

Thermal pumps are not new and were already used on the Siloe reactor to ensure the flow in the test loops. However, and although some calculations were performed on the thermal pump of the DHRS-1, a full-scale test on a sodium loop would be necessary for final validation and industrial demonstration of the results.

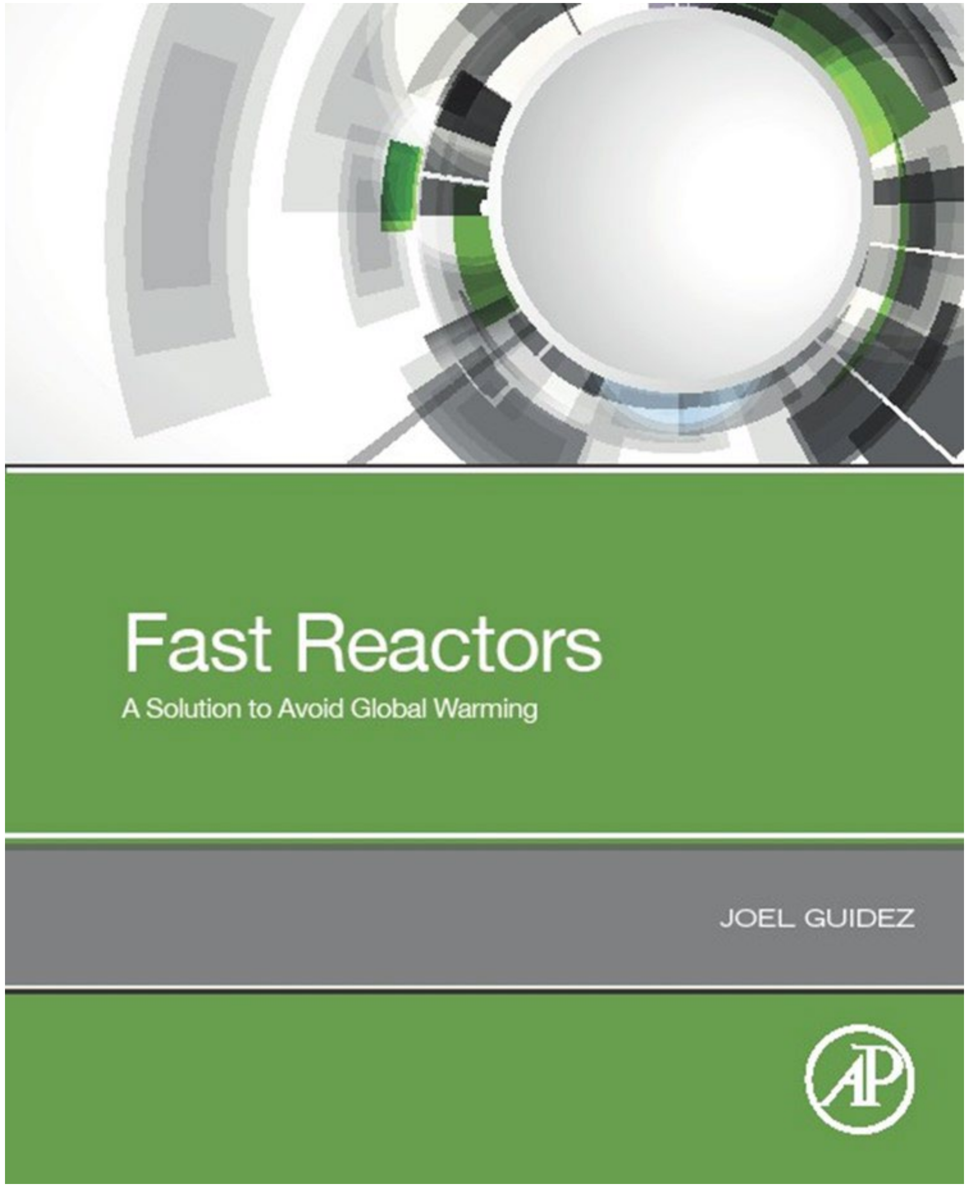
Conclusion : application to SMR

- Nobody will today directly build a SFR with a power of 1500 MWe
- SMR will be necessary in a first step
- All the ESFR SMART improvements can be used for a SMR.
- But some simplifications are possible when the power decreases. For example, if this power decreases, DHRS 2 becomes able to assure alone DHR by natural convection. If power continues to decrease, DHRS 3 become able to assure alone the function. So, simplifications on global DHRS systems are possible with, for example, suppression of DHRS 1.

Conclusion: design and calculations available

- All CAD drawings of the plant are available .
- First calculations (NERS Papers) were provided for first assessment and are also available
- On this basis, a start up was created for further designs using these improvements
- A detailed presentation of project results will be also available in the forthcoming Elsevier Book

“Fast reactors: A solution to avoid global warming”



Upcoming Webinars

Date	Title	Presenter
24 February 2022	AI in support of NE Sector	Prof. Nawal Prinja, Jacobs, UK
23 March 2022	Scale Effects and Thermal-Hydraulics: Application to French SFR	Mr. Benjamin Jourdy, CEA, France
19 April 2022	GIF/IAEA joint Webinar: Role of Nuclear Energy in Reducing CO_2 Emissions	Dr. Shannon Bragg-Sitton, INL Mr. Huang Wei, IAEA Ms. Diane Cameron, NEA