FRENCH PROGRAM TOWARDS AN INNOVATIVE SODIUM COOLED FAST REACTOR

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Sodium-cooled fast reactor is considered in France as a potential candidate for a prototype of 4th generation system to be built by 2020. A detailed working program has been launched recently to identify by 2012 the potential improvement tracks for later industrial development of these reactors.

The goals for innovation are first identified: Progress of the safety with a special attention to severe accidents risk minimization and mitigation (defense in depth approach); Economic competitiveness of the system mainly by reducing the capital cost, the investment risks by enhancing in service inspection and repair capacities, and raising the availability; Sustainability with fissile material management while reducing the proliferation risk; capacity for long-lived waste transmutation.

KEYWORDS : Sodium Fast Reactors, Enhanced Safety Core, Severe Accidents, Energy Conversion, Inservice Inspection

1. INTRODUCTION

During the years 2005 and 2006, France elected its policy concerning the future of electricity-generating nuclear systems. Fast reactors, either sodium-cooled (SFR) or gas-cooled (GFR), have been chosen to save uranium and to manage radioactive waste. A coordinated program has been launched by the CEA, Areva and EDF to develop innovative Sodium-cooled fast reactors. This program is presented hereafter.

2. A STRATEGY IN FRANCE

In March 2005, an inter-departmental committee stated that France should study SFR and GFR for the long term deployment of its nuclear fleet, together with Hydrogen production using high temperature reactors. In January 2006, the French president requested for the design of a generation IV system to be operated by 2020. In June 2006, the parliament voted a law to specify the future management of radioactive waste, and included the necessity to study by 2012 the options of future nuclear systems and to start operate a prototype in 2020. Finally, in December 2006, a second inter-departmental committee agreed on a technical roadmap for SFR, GFR and fuel cycle studies leading in 2012 to gather data to choose future options. This strategy is based on the necessity to save uranium and to reduce ultimate waste in the future.

3. SPECIFIC GOALS FOR INNOVATION

One has to think about sodium-cooled fast reactors of a new generation able to be deployed by industry in 2040-2050. The research goals of this new generation of reactors are the following. They are derived from the generic objectives of the Generation IV International Forum.

At first, the safety of this type of system will be enhanced compared to what was used in the past or is used presently. Safety specifications are set up at a level at least equivalent to the one used for generation III light water reactors and applied specifically to the EPR. As sodium behaves differently from water, and fast neutron spectrum induces specific neutronic behavior of the core and fuel, those specificities are taken into account, namely for behavior under severe accident conditions and for sodium risks. The occurrence of failing of critical components will be pushed back by extended and performing In Service Inspection (ISI).

The economy of the system should be optimized to propose a plant the cost of which is acceptable for the industry taking into account the advantages coming from its ability to manage uranium and long lived radioactive waste. The European Fast Reactor (EFR), developed in the 1980's by a European association, was considered at that time period as the optimized version of a SFR. A techno economic comparison to the EPR showed nevertheless an over-cost that needed a more important interest in the management of the uranium stock, to be acceptable by the industry.

The financial risk engaged by the utility that buy the plant must be comparable to the one engaged in traditional power plants. If the investment cost is traditionally high for a nuclear plant, it is still higher for a SFR. So the availability of the plant must be assessed while the construction cost must be reduced.

The interest of fast neutrons is to allow an optimized management of nuclear matters. First, the reactor should be able to breed plutonium with a breeding ratio at least equal to zero and must have the possibility to raise this ratio in the positive values in order to allow the deployment of a fast reactor fleet by producing a sufficient mass of Plutonium. This implies in parallel the deployment of an industry that closes the fuel cycle and allows recovering Plutonium in a non-proliferating way.

The same reactor must have the capacity to transmute long lived nuclear waste. The core must accommodate the quantity of minor actinides (Americium, Neptunium and Curium) coming from the closed fuel cycle and transmute most of it. It should be able, where needed, to transmute also waste accumulated in the spent fuel of light water reactors. This implies a corresponding management of the minor actinides in the fuel cycle process.

Finally, the weak points of the sodium technology must be improved. In that domain, in service inspection is concerned together with an easy operation, repair and dismantling. One should think to the availability of the plant on a 60 years period while proving periodically that safety margins are sufficient.

A coordinated research program between the CEA, Areva and EDF was launched to answer those previous objectives. It includes the main following items.

To study and develop a reactor core that has significant advances in safety while limiting to the minimum the quantity of natural uranium used. The fuel will have the capacity to reach high burn-up and to transmute minor actinides. It will be easy to recycle it.

To enhance the resistance to severe accidents through the design of the fuel sub-assembly and the core catcher and to produce a safety demonstration more robust, that is to say covering a wide range of anticipated accidents. In the mean time, the resistance to external hazard, for instance a seismic impulse, will be re-enforced through the core and the containment designs.

To develop an optimized power conversion system that reduces the sodium risk. Alternative fluids are looked at to suppress the sodium water interaction risk, either liquid or gas. Optimization is studied through the possibility to raise the core outlet temperature. This induces using new materials for internal hot structures, for circuits and for fuel elements. Those materials are to be developed. Where water is replaced by gas as a working fluid, standard gas or supercritical CO2 are looked at. Heat exchangers optimization is part of the program, the design of a compact component coupling intermediate heat exchanger and steam generator being the ultimate way to innovate.

To discuss design options for the reactor and its components to facilitate inspection, maintenance, availability and dismantling, to reduce environmental impact and increase resistance to proliferation, to enhance performance and global economy. This includes a comparison of pool and loop type reactors, the impact of unit power and modularity. Specific components like sodium cold trap are studied again to reduce their economic impact. Fuel handling is considered as an important feature for the availability of the plant. Innovative techniques for in sodium measurements are developed.

4. AN EFFICIENT CORE WITH ENHANCED SAFETY

4.1 Reduction of the Sodium Void Effect

The core performance in terms of fissile fuel management is to produce at least the same quantity of fissile plutonium than the one burnt. A basic objective is to get a zero breeding gain (IBG=production/ consumption -1) for the fissile core, to ensure that later optimization including blankets could easily reach positive breeding gain to allow the development of a fast reactors fleet. At that time, the blankets should be designed to avoid easy plutonium separation from other actinides in a proliferation resistance way. Still for fissile material management, it is necessary to limit the mass of plutonium that is needed per electrical MW.

In parallel, future SFRs may have enhanced safety compared to past or existing projects. The sodium void effect, that induces a positive reactivity for an industrial size of the core, must be significantly reduced. The objective is to compensate it by negative effects from the Doppler and from other feedbacks like dilatation of the materials. At least, a sodium void in the core would not induce a severe damage.

Several items will be looked at. The impact of the fuel element geometry is studied while comparing pin to plate. The geometry (height/diameter, cylindrical, annular, modular) of the core is an open parameter as are size, total power and power density, the relative volume of the components (fuel, structure and coolant) and the presence of moderator to sweeten the neutron spectrum. The sub-assembly geometry is also open to modifications : diameter of the wire spacer, of the hexcan, impact of the sodium hydraulics, presence of a sodium plenum at the outlet of

Table 1. Optimized Zero Breeding Gain Cores With Respect to Past Reference

Core option	Past Reference	MOX	Carbide	Carbide + moderator
Pin diameter (mm)	6.9	9,5	9,5	9,5
Sodium void ($) EOL*	8,7	4,5	4,2	3,3
Mass of Pu (t)	8,8	10,5	8,5	9,5
IBG	-0,13	+0,04	+0,11	+0,02
Volume power (W/cm^3)	303	230	290	290
Burn-up (MWd/t)	128	106	91	105
Linear power (W/cm)	440	550	640	660

*Nota : Void Effects (Calculated Here at End-of-Life of the Whole Core, i.e. When a 200dpa Damage is Reached in the Clads) are Upper Bound: They will Depend on Core Management.

the fissile length (that voids as soon as sodium boils in the core, inducing important neutron leaks).

The nature of the fuel material is studied to compare advantage and drawbacks of oxide, carbide, metal and nitride. One needs to design a specific fuel element adapted to each fuel material in order to optimize each core. The development of such cores is conducted with a three steps program. First, neutronic calculations give reasonable materials repartition in the core. Then the fuel element and the sub-assembly design and technology are assessed. Finally, a detailed design of the core is used for several transient calculations that verify the performance and the safety of the given core.

An example of studies is summarized in the Table I below, that shows possible way of innovation for the core design according to the previous objectives. More details are given in [1].

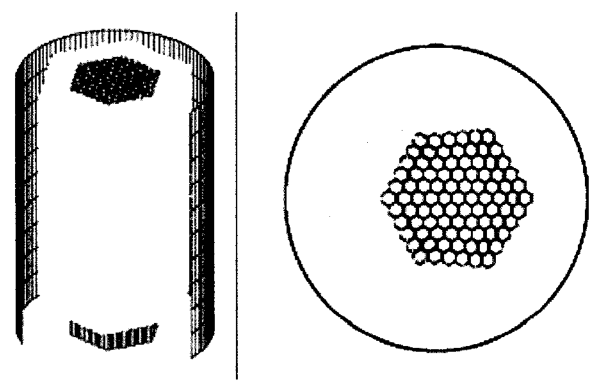

Fig. 1. 3D Core Calculations With Fluid Structure Interaction

4.2 Compaction Risk Management

Fast neutron spectrum cores are very sensitive to compaction. A special effort must be made to enhance their resistance to compaction, due to a seismic stress for example. The cores of the French SFRs (Phenix, Superphenix) are free to expand and compaction is limited by the contact at the pad level between sub-assemblies and especially dedicated stiff ones located at the core periphery. This effect could be enhanced.

The performance of a ringed core will be evaluated comparatively. Finally, the dynamic behavior of the core when a mechanical stress is applied will be studied using for instance the Symphony past experiments realized on a shaking table at CEA/Saclay. Modeling improvements either in static and dynamic situation will support these studies.

4.3 Core Instrumentation

Taking advantage of the most recent technology evolution, the core instrumentation can be revised to develop new systems with a better efficiency and a higher dynamics.

The possibility of monitoring of the power distribution through in-core high dynamic fission chambers will be assessed.

For different applications such as temperature (measurement of the temperature inside the sub-assembly head, avoiding the discrepancy coming from the mixing jets at the core outlet), detection of boiling, presence of gas, ultra sonic detectors are being studied to be used under hot sodium conditions. Difficulties are to solve wetting of the transducer by sodium in order to improve sensitiveness, and the question of a piezzo-electric material able to sustain high temperatures during long times (see also § VII E).

The clad rupture detection system can be optimized in order to reduce its response time delay. The clad rupture localization system is an expensive system with tubes that cross the closure slab. Reconsidering the complete system could allow a better performance, simplification and enhanced security.

4.4 Core Performance

One point of the core performance is the fuel burn-up. The first limitation to the lifetime of the fuel elements comes from the dose rate on the structure materials (clad, canister). A specific program, on the long term, is envisaged to improve the dose rate acceptable on the clad.

First, the present optimized austenitic steel AIM1 will be confirmed able to reach 120 to 130 dpa using recent irradiations in Phenix. To go beyond, a new cladding material ferritic-martensitic strengthened by oxide dispersion will be developed, while advanced austenitics track will be kept as a backup. This new material should allow for new geometrical features of the core design, especially by achieving a satisfactory cooling of the pins bundle because of low swelling even for very long time of irradiation, up to 200 dpa or more.

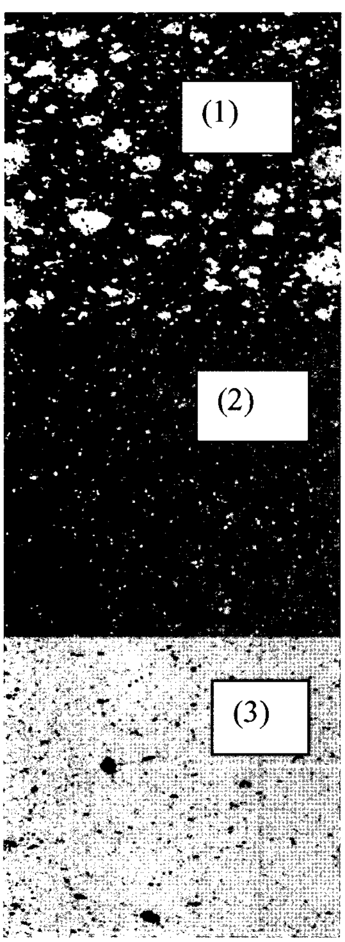

Fig. 2. Mixed (U ,Pu) Oxide Obtained With Different Fabrication Processes : (1) 11%Pu, MIMAS Process (2) 6%Pu, COCA Process
(3) 27,5% Pu COEX
Co-Precipitation Allows for a Very Homogenous Structure That is Expected to Allow for Improved Behavior When Irradiated

The hexagonal canister should withstand the same dose than the clad and possibly a higher temperature than presently. A ferritic steel like the T92 grade is foreseen.

In order to reduce the total diameter of the core, a compact lateral neutron shield must be developed. Coming from the EFR studies, a specific sub-assembly with neutrons absorber will be developed. Two problems are to be solved: the extraction of the heat produced in the material while the filling up density must be maximized, and the stability of the compounds on a sufficient long time period.

The lifetime of the control rods will be extended progressively.

As to fuel, it is anticipated that the core of the prototype mentioned in the § II, will be an oxide core, the only one that allows for sufficient knowledge in the time scale of this reactor, including in accidental conditions. Innovation on the oxide fuel will be introduced as it is anticipated to be issued from so-called COEX™ process. This process allows for coprecipitation of actinides and avoids handling of separated Pu, while allowing for simplified pelletization process. Qualification will be addressed in the timescale of the Prototype, and for that purpose an irradiation in Phenix (COPIX) is currently being prepared.

A R&D program in a more long term will be pursued on dense fuels, especially carbide, as their attractiveness comes out of the neutronics calculations (table 1 in § 4.1). Clearly, the prototype will be used in the frame of this program for experimental irradiations of such fuels, but application is foreseen in a longer timescale with the view at industrial deployment.

4.5 Minor Actinides Transmutation

The future cores of the fourth generation SFR should be able to transmute minor actinides to reduce the quantity of ultimate waste. Several technical options are available. Comparative studies of the global efficiency of these options are yet underway for various options of the future French nuclear fleet.

Minor actinides can be mixed with the driver fuel, in a homogeneous way. In such a case their relative volume in the fuel is from 1% to 5%. They are quite easy to fabricate and handle, but all the fabrication process of the fuel needs additional adaptation and protection against radiation.

They can also be put in specific sub-assemblies with a high concentration, from 15% to 45% in volume. This is the heterogeneous way. In that case the number of sub-assemblies to manufacture requires a separate facility from the one for the driver fuel.

The choice of the matrix that contents the M.A. is an essential part of the research. It depends the feasibility of the fabrication process and of the behavior under irradiation. Most of the experiments launched in the past were devoted to inert matrix, either ceramic or metal. Specific irradiation experiments are yet underway or under post irradiation examination with various materials and fuels.

A new program is starting with a MA bearing UOX matrix featuring radial blankets that could be placed around the core. Such a solution opens the way to proliferation resistant blankets.

Together with those experiments, core design studies, including specific sub-assembly design, are underway, depending on the location of the sub-assembly, inside the core or as a radial blanket.

From the point of view of fuel pins behavior, power variations of MA bearing fuels have to be accommodated and justifies dedicated studies.

Other special concerns are on the one hand the fuel handling, according to the level of residual power even for a "fresh" fuel (especially in the case of the heterogeneous route), and on the other hand the detection of a clad failure of a sub-assembly containing M.A. as release of delayed neutron emitters has to be verified.

4.6 Simulation Tools

Tools for core simulation include neutron physics, thermal-hydraulics, mechanics & fuel behavior. For neutron physics, the reference tool is ERANOS that has been validated in a wide range of situations. For thermal-hydraulics several tools are available, some are commercial like STAR-CD and others are specific like TRIO_U at CEA. For their use with sodium, they need to be validated on existing experiments from the 1980's. Core mechanics will rely on the HARMONIE tool that describes the static equilibrium of a core depending on limit conditions imposed at its boundary. The fuel behavior is simulated with the GERMINAL code. This tool was widely validated for a Phenix-like geometry. It will be extended to various future geometries and transferred into the PLEIADES platform, the French reference for nuclear fuel behavior simulation. GERMINAL will take advantage of the thermal-mechanics models yet available in PLEIADES. A special effort is intended on the long term to take into account and to validate the behavior of fuel loaded with large quantities of M.A.

For all the tools, the existing experimental basis will be revisited and a set of database will be developed.

On the medium term, coupling methods between the various physics will be introduced to simulate more precisely the core transient behavior.

5. RESISTANCE TO SEVERE ACCIDENTS AND EXTERNAL HAZARDS

5.1 Safety Approach

Generation IV systems require an enhanced safety. The safety approach for the SFR has to be made clear in this sense. Globally, the SFR safety will be of the same level as the one of the third generation LWRs. The EPR is taken as a reference and its general objectives are already very ambitious and guarantee a very high level of protection to persons and the environment. The defence-in-depth method is adopted as the basic principle to cover the risks and uncertainties inherent in this concept.

Additional requirements provide both a real and demonstrable benefit and a greater degree of assurance in the safety demonstration and therefore in its robustness. Four topics are studied:

- Allowance for degraded situations and the "practical elimination" approach,
- The robustness of the demonstration adapted to the system,
- Consideration of the specific aspects of the sodium-cooled system,
- Minimization of impacts concerning radiological protection and the environment (discharges, wastes, dismantling actions).

A complete overview of this approach is given in [2].

5.2 Scenarios and Transients Studies

The innovative design that are envisaged may lead to specific scenarios, somewhat different from the one studied before in SPX or EFR. For instance, the fourth category transients are different whether you consider a loop-type or pool-type reactor. The presence of a power conversion system driven by gas will also induce new studies (impact of large quantities of gas under pressure in case of suppression of the intermediate circuit for instance).

A special emphasis will be put on the long term influence of M.A. in the sequence of events, a field of research quite new.

5.3 Sub-Assembly Design for Core Melt-down Management

Concerning severe accidents and core degradation, a threefold strategy must be implemented that includes prevention against melting initiators (hydraulics allowing a better mitigation of a local temperature raise, design of a boiling zone capable to favor natural convection and neutrons leakages...), enhancement of protection systems (passive 3rd level shutdown system), and at the end, in case of core melt, provisions to minimize the consequences especially the risk of significant energetic re-criticality. As a matter of fact, a major discrepancy between LWRs and SFRs is the core reactivity. For a fast neutron spectrum, a compaction of the core induces a reactivity step; that is not the case in a LWR. So a specific risk linked to hypothetical core degradation is the possibility to have a recriticality that can develop a mechanical energy release.

Globally, two routes will be studied, one consisting in the dispersion of the molten core by introducing some discharge channels either among the sub-assemblies or using neutron absorber channels, the other consisting in releasing absorber material in the molten fuel to reduce its reactivity; the two solutions being potentially combined. The Fig. 3 describes a qualitative scenario of core melt-

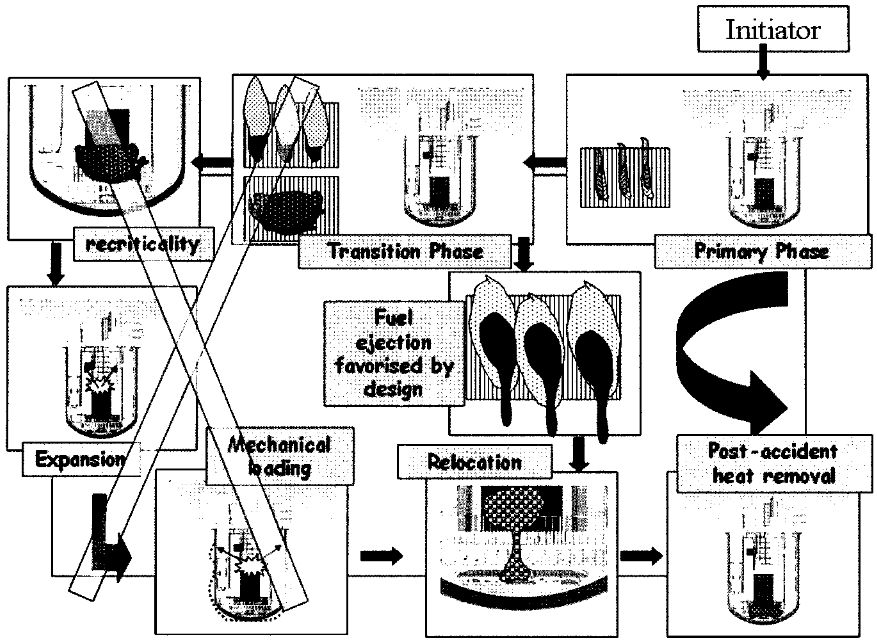

Fig. 3. Severe Accident Sequence of Events

down up to corium recovery with the objective to significantly reduce re-criticality risks or their consequences.

Finally, to analyze the various modes of degradation, several initiators, not only an ULOF, will be taken into account to study the possible degradation of the core. This item contributes to the robustness of the demonstration. One should try to cover a wide range of phenomenology to describe the mode of degradation of the core.

5.4 Core-catcher Studies

Likely they will have to cope with a reactor vessel more compact than in previous projects, while ensuring still the decay heat removal and non-criticality. The design (shape, but also location: in or ex-vessel), the materials, the modeling of the debris, are the main R&D topics to be addressed.

5.5 Strengthened Systems for Defense in Depth

In addition to provisions mentioned (in § 4.3 & 5.3) to the possibility of an upgraded monitoring of the core it will be necessary:

- to exclude by design scenarios such as the ingress of a large gas bubble in the core, a catastrophic failure of the core support structures, a compaction of the core (see also 4.2),
- to enhance the diversification of decay heat removal systems, either from the point of view of their location, the physical principles used, the architecture of the plant and of its confinement,
- to reinforce provisions against leakages and fires, reactions of sodium with fluid used for energy conversion (see also § 6),
- to protect the plant against upgraded external aggressions such as earthquakes, new plane crash hypothesis.

5.6 Accident Modeling

Accident modeling tools will be re-considered owing to their capacity to deal with the needs yielded by innovations selected for future SFRs. The basis for severe accidents involving very complex multi-physics aspects, will be treated with SAS4A and SIMMER (with a refined pin model called DPIN) codes; the so-called CATHARE CEA's code, currently being adapted for sodium applications, will be used for transient calculations and join SAS/SIMMER for the primary phase including boiling but prior to the loss of geometry.

So-called MC3D and PLEXUS CEA's codes will be available respectively for corium-coolant interaction and dynamic mechanical loads of structures. Debris beads behavior is covered by LIDEB and MC3D for some aspects.

Sodium fires will be addressed with FEUMIX and PULSAR (spray type fires).

Transfer of species (including radiotoxics) in the reactor building and releases will be assessed with CONTAIN code.

It is considered at this stage that the qualification of these tools can rely on the extended existing data bases, especially the numerous experiences using simulants of fuel and coolant, and experiences in representative situations (sodium, nuclear heat and fuel) CABRI and SCARABEE [3] , as long as oxide fuel is concerned. This does not exclude that studies to come induce some new needs, but they are not identified at now.

The situation is very different for cores that could use dense fuels such as carbide or metal; if promoted in the frame of future industrial commercial units, dedicated programs shall be requested timely.

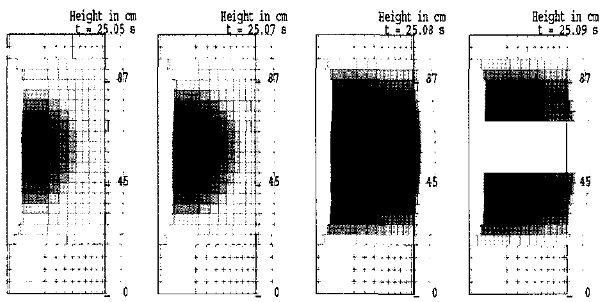

Fig. 4. Failure of a Pin as Calculated by DPIN

6. LOOKING FOR AN OPTIMIZED PCS TO REDUCE SODIUM RISK

6.1 A gas Power Conversion System

The main incentive for such an innovative option is to delete the risk of sodium water reaction and its potential consequences. Notice also that such an option opens, in the case of a loop type reactor, to delete the intermediate sodium loop, and by the way decrease the investment cost (see Fig. 5).

Nevertheless at a given core outlet temperature, classical gases (such as nitrogen, or argon, eventually mixed with some amount of helium) will require a significant effort to compete with the Rankine water/steam cycle efficiency. These efforts can be made on the pressure level, on the improvement of the (indirect) Brayton cycle through optimized and enhanced components, use of re-heating by sodium, and/or by raising the temperature level (at the core outlet). An alternative allowing for recovering an attractive efficiency (higher than 40% -Super Phenix value-) without temperature enhancement could be to use supercritical CO_2. this solution requires to develop the necessary innovative technologies concerning components an materials aspects.

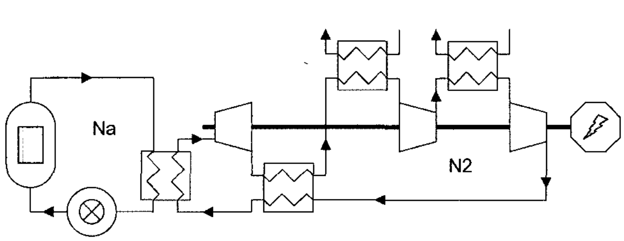

Fig. 5. Typical Layout of a Loop Type SFR, Without Intermediate Sodium Loop and Coupled (Via an IHX Na-Gas) to a Nitrogen Brayton PCS (One Turbine, Two Compressors on the Same Shaft, High Power Heat Recuperator)

The first (on going) step considers the following studies on these type of Advanced PCS, for most of which international partnership will be seek for:

-Super critical - CO_2:

- preliminary feasibility of the cycle (stability including in load follow-up hypothesis), and components (especially turbine, compressors),
- sodium - CO_2 interaction through dedicated tests (Fig. 7).

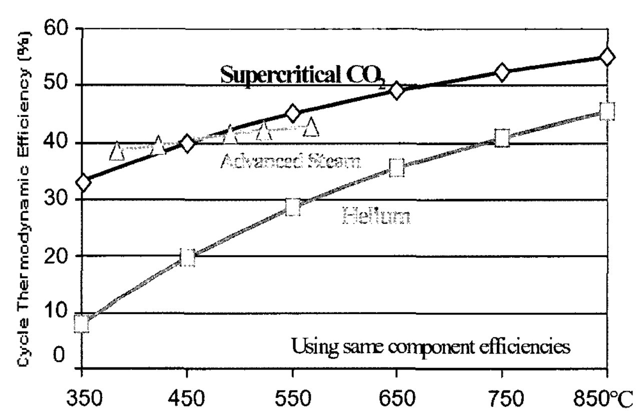

Fig. 6. Efficiency of Some PCS vs Sodium Temperature

-all gases:

- thermodynamical optimization and associated "hot" temperature level,
- protection provisions: detection of leakages, dedicated phases separator component, valves for insulation and decompression, possibility of a "short" intermediate loop between gas and primary sodium,
- safety analysis versus the risk of massive gas ingress in core,
- prospect about materials (compatibility with fluids and with required temperature level),
- preliminary studies of IHXs: heat recuperator, Na-gas IHX (including sodium plugging hazard).

A good prospect on this step would then allow to undertake heavier developments involving especially Na-gas IHX test at the scale of ~1MW exchanged, prior to larger ones if the option is definitely confirmed.

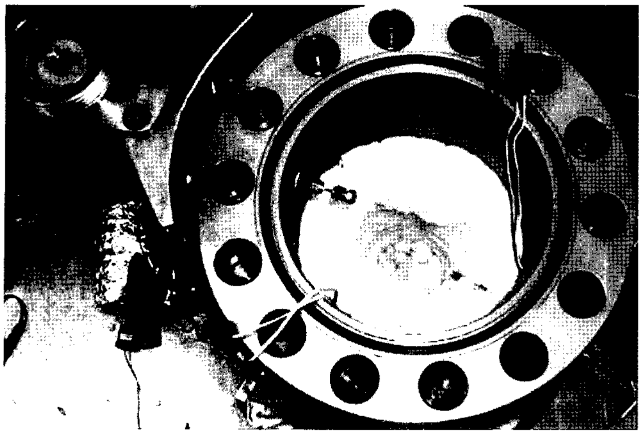

Fig. 7. Vent Sizing Package Experiment of CO_2-Na Interaction (Study of Exothermies)

6.2 Optimization of Materials Choice According to Temperatures Level

Independently of the temperature level, future SFR must be able to sustain a significant enhancement of their lifetime, up to 60 years, for those components that will not be replaceable. For coping with this requirement, feedback from Phenix reactor (after its shutdown foreseen in 2009) will be used as it includes an interesting panel of steels either austenitic and ferritic, representative of relevant families and aged for a long time in representative conditions.

For the non replaceable structures in the primary vessel, it is believed that hot and cold parts can be kept made out of the reference austenitic steel (Super-Phenix, EFR): Cr17-Ni12-Mo-Mn-(N).

Nevertheless, in case of a temperature increase (required for instance by efficiency concerns with gas PCS), by +50°C (i.e. 550 to 600°C), austenitic Cr25-20, Ni30-20 will be assessed for hot parts, with a special emphasis on creep performance and weld-ability. A more ambitious increment: (+100°C), if decided, will require a long term program on nickel based alloys. In both cases "corrosion" by sodium is a concern owing to Ni dissolution enhancement by temperature that will be checked by dedicated sodium tests.

For other components such as heat exchangers, piping, austenitic steels could be challenged by ferritic-martensitic ones. Such a choice can be justified by mechanical properties (creep resistance), but also costs concerns as thermal properties (heat conductivity, thermal expansion coefficient), could allow for a lesser level of thermal induced stresses and a reduction of masses involved. Fig 8. below expresses this potential through a so called "merit factor" plotted vs temperature level:

$$M_4 = \frac{\lambda S_r(10^5 h)(1-\nu)}{\alpha E}$$

with λ : thermal conductivity,
ν : Poisson's ratio
α : thermal expansion coefficient
Sr (10^5h): stress to rupture by creep in one hundred thousand hours
12Cr: ferritic/martensitic steel 12Cr-0,6Mo-2,5W-Cu-V-Nb
18S: ferritic/martensitic steel 9Cr-1Mo-V-Nb
1S: austenitic steel: 17Cr-12Ni-2Mo-2Mn-(N)

The program incudes the definition and optimization of a specific ferritic/martensitic grade within the range 9 to 12 chromium and to assess its attractiveness component by component.

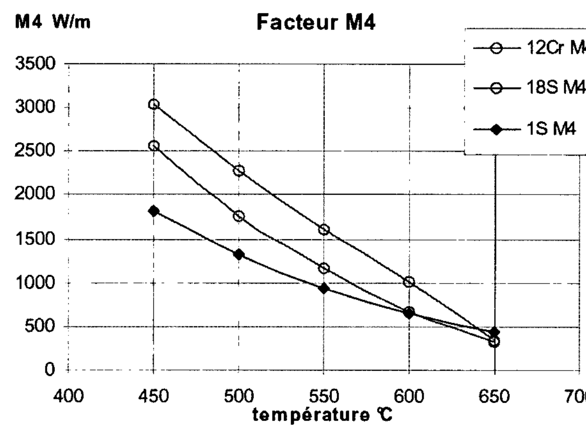

Fig. 8. Merit Factor of Candidate Materials vs Temperature Level

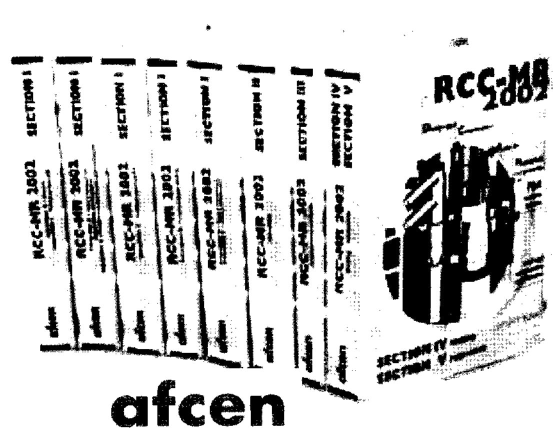

Fig. 9. RCC-MR 2002, for FBRs and High Temperature Applications, English Version

Conversely, a specific action will be devoted to evaluate the profit expectable from a limited (20°C) drop of the hot temperature of the cycle in terms of ageing of base and weld materials.

At the end it is worth to mention that the outcomes of these researches will be implemented (provisions for procuring, material data, mechanical analysis methods, construction and inspection) in the code and standards dedicated to fast reactors (so called "RCC-MR" Fig. 9). A new releases of this guide (undertaken for Super Phenix, enriched for EFR studies) is foreseen this year 2007.

It is worth at the end to mention that new tribologic materials for application in sodium will be tested with the aim at a better behavior after long duration static contact, and mitigation of activation (with regard to cobalt bearing existing products).

6.3 An Enhanced SG PCS

A first objective is to mitigate the risk of sodium-water reaction and its potential consequences; this yields a first set of actions:

- reinforce reliability by technologies such as double walled exchange tubes, modularity etc...
- assess the viability of keeping a "compact" secondary sodium loop (up to set, inside a same vessel, a SG and an IHX units, thermally coupled by a very limited amount of sodium, or by an alternative coupling fluid)
- assess the possibility of replacing secondary sodium by another fluid compatible with water and sodium. With that view different metals mixtures and some salts are envisaged. They will be tested in terms of chemical stability (for salts), reactivity with sodium, physical nature of reaction products, corrosion of materials and provisions that could allow for its control.

A second objective is to enhance the performances: with that view supercritical water cycle will be studied. It is worth to mention that such a cycle could allow for increasing performances by increasing the pressure (from 180bars for previous SFRs to 250 for instance, allow 2% efficiency earning) but a temperature increase is also to be considered (and will yield the same type of materials concerns as for gas PCS and already presented, § 6.2). Nevertheless supercritical water rises specific corrosion problems, that can be solved by use of nickel based alloys such as nickel based alloy 690 (to be checked also in sodium environment)

7. REACTOR DESIGN RE-EXAMINATION

7.1 Reactor Primary System

The previous paragraphs aimed at propose and evaluate innovations. The question is at last to see how long these innovations can participate to coherent reactor layouts, in the frame of integration studies, and to check these layouts against the high level goals in § 3, with dedicated tools (economy, safety..).

As to the primary system, a lot has been done in Europe, and especially in France about the so-called integrated primary (pool type) system. This system provides a robust design of the primary confinement, against loss of primary sodium (and by the way primary sodium fires), against loss of the primary hydraulic loop, ensures a high thermal inertia and guaranties a good natural circulation in the main vessel. The cold plenum contributes to the mitigation of thermal shocks and gas transport. It is also favorable to alleviate radioprotection concerns during operation and allows to design easily an hydraulic path for cooling down the main vessel. As to identified drawbacks, it is worth mentioning the difficulties to achieve a compact reactor block, to have an easy access to internal structures for monitoring and repair; it implies in vessel rotating components and earthquakes effects are complex because involving strong fluid-structure interaction.

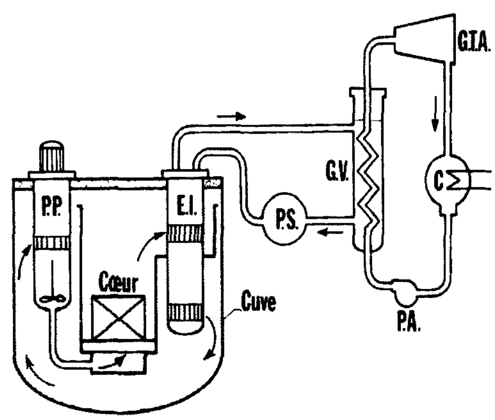

Fig. 10. Pool Type Design

Loop type system has the important potential to make easier the intermediate heat transport loop suppress. It can offer some easier maintenance and repair conditions for large components (PP, IHX) that are separated from the reactor tank, and can be integrated in a single component. There is no rotating parts in the reactor vessel and there is a potential for more compact components (main vessel).. This design is likely more easy to justify vs earthquakes because less sensitive to sloshing effects. Drawbacks concern the risks associated to the loss of a primary loop (fire, leak, flow reversal, gas entrainment), lower thermal inertia and risk of gas transportation. Keeping the main vessel below the creep regime is not easily achievable. Operation conditions can be made more difficult owing to active, double walled, primary sodium transport piping.

The program will consider both options and will address the different topics mentioned just before through integration studies which pictures in fig 12. below give an outlook

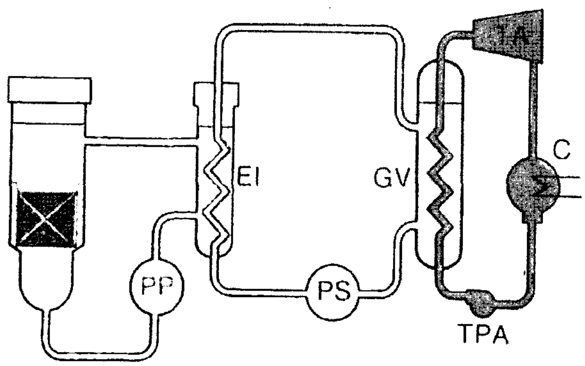

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Fig. 11. Loop Type Design

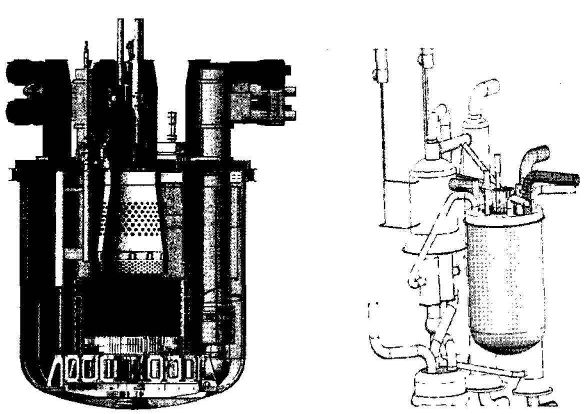

Fig. 12. Optimized Pool Type (Above), and Sketch of an Optimization of Loop Type (Below) Primary Circuits

Beyond this comparison pool vs loop, size effects will be dealt with especially in order to assess possible threshold effects than could incite to consider limited power output plants.

When precise enough, and with the input from other systems and components studies (§ 7.2, 7.3, 7.4), the designs will be compared from the points of view of economics (SEMER code), safety, inspection and repair, availability.

7.2 Intermediate System Optimization

For those reactor layouts using an intermediary loop, the target for this last will be to reduce the cost (including for maintenance and manufacturing processes), looking for compactness (using ferritic-martensitic materials) and reduction or simplification of the number of components and auxiliary circuits.

Integrated components, short loops, improvement and simplification of provisions against sodium leaks (including inert gas filled casemates) are the tracks foreseen to be followed.

7.3 Components & Systems Optimization

As for auxiliary systems, the sodium purity target and technologies for traps will be reconsidered. Sodium quality control will be adapted looking for direct measurement of impurities content (like O, H, C) in addition to conventional plugging temperature.

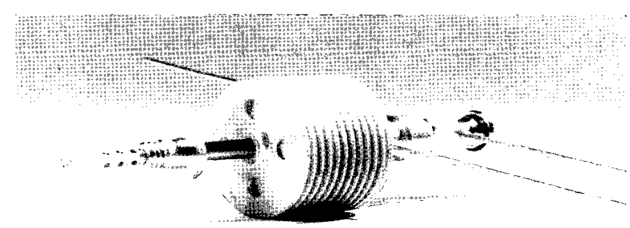

Fig. 13. Solid Electrolyte Oxygen Sensor

Tritium management will need a particular attention to be dealt with the regeneration of traps; in case of use of a gas PCS (as no hydrogen will be injected in sodium by reduction of water on the SG tube wall) a specific strategy is to be imagined.

Cover gas treatment either at the input (removal of impurities) and at the ouput (gaseous FPs), is also subjected to improvement, so is the treatment of aerosols in gas volumes above the sodium.

7.4 Fast Fuel Handling

This point is very important as it has a key contribution to the availability of the plant, and can be determining versus the design of the primary system:

(1) <u>its geometry</u>, as it must in any case allow for access to all subassemblies and as room for in-vessel storage of used fuel is necessary according to the option chosen,
(2) <u>its efficiency</u>: how fast used subassemblies can be removed out of the core and replaced by fresh ones. This is particularly important if in reactor vessel interim storage is not the option chosen, but it could be also a safety concern for instance if an inspection of the core support is needed or following an accident.
(3) The design of the handling system can even concern the layout of the complete plant, in the case when a

modular architecture appears to be attractive: as matter of fact the ex-vessel equipments, and especially any interim storage tank and washing facility, can be shared by different modules.

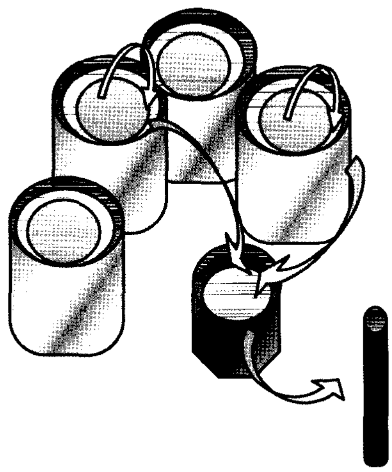

Fig. 14. Schematic Sketch of a Modular, Four Units Reactor Deserved by a Unique Washing Facility and a Unique Interim in Sodium Storage Tank

In case (1) it is necessary to think of technologies that do not compromise the compactness of the main vessel. This point comes less stringent if an in-vessel storage is chosen,

In case (2), the development of a high power/temperature technology for the transfer and the washing of used fuel are key R&D topics,

Case (3) participates to high level options of a plant. R&D actions in the program are aimed at the following:
-study three options for in-vessel fuel handling, that are not indifferently applicable to the options for the primary system (§ 7.1):
- optimization of existing system: two rotating plugs and one interim put down-take over position, with improvements concerning efficiency and high-power used SA discharge.
- assess one rotating plug plus pantograph solution
- assess direct handling using a dedicated hood.

-study transfer and washing of the used fuel subassemblies at a power ranging in between 10 and 15 KW.

7.5 Enhanced ISIR

Beyond these performances, for future SFRs, possibilities of in-service inspection and repair have to be clearly enhanced again. This will be made first by considering inspection strategies at the design stage and from the point of view of the criticality of each component or sites on this component. Strategies of access will be chosen in coherence: geometrical considerations and hatches, necessity or not to be able to empty the primary sodium, or part of it.

Developments will be pursued on under-sodium Ultrasonics technologies:

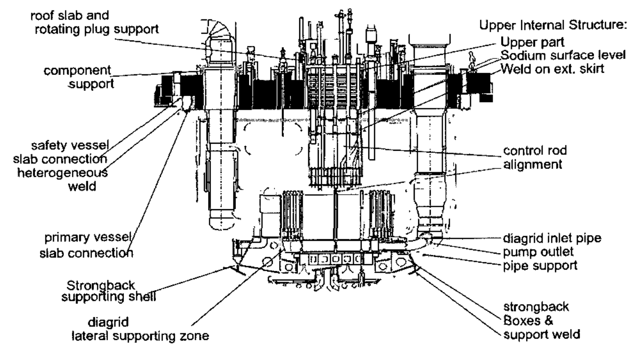

Fig. 15. Shows, Above, the Parts in the Superphenix Reactor that were Submitted to In-service Inspection in Relation with a Crossed Analysis of Expected In-service Loadings Versus Consequences that should Derive from their Failure

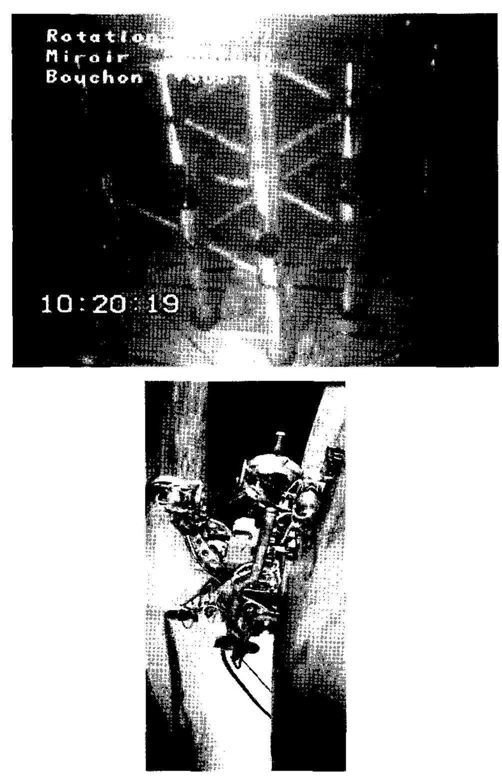

Fig. 16. Below Shows :
-Above, a View of the Internal Structure of the UIS of Phenix,
-Below, the "MIR" Device That was Developed for the Purpose of the Volumic Inspection of the Welds on the Primary Vessel of Super-Phenix

For monitoring systems used during reactor operation, a key point is to define a piezzo-electric material suitable for the high temperature of the hot pool/legs in the reactor.

For periodic examination, classically made at "cold" shutdown conditions, a key point is to enhance the quality of transmission of the US generated by the transducer to the fluid, and back from the target to the transducer.

At the same time the modelling of US propagation and reflection will be developed in order to help the optimization of dipped transducers technologies. Mono- and multi-elements will be developed as well, depending of the application:

- telemetry,
- far viewing (Fig. 17)
- close viewing
- volumic NDT against small defects (length, depth), large defects (length)

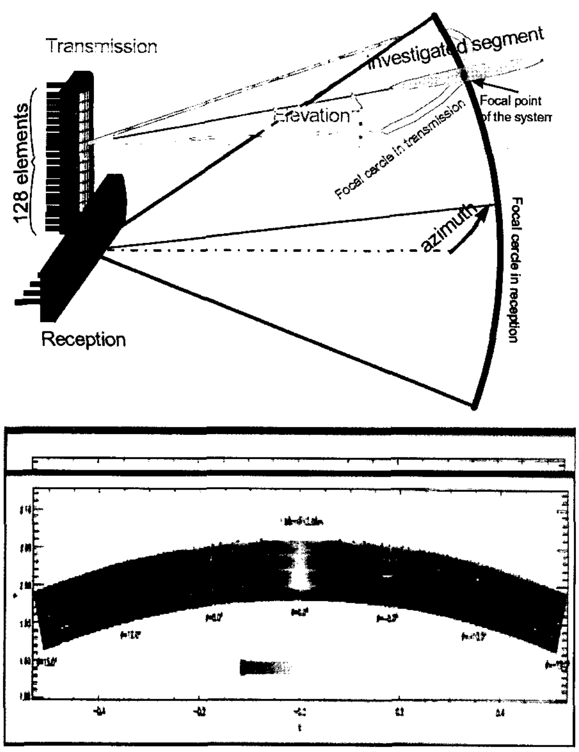

Fig. 17. Far Viewing With Multi-Elements, Two Antennas Technology and Picture of a Curved Metal Sheet

Development of distant US Technologies, allowing to check a structure from its external side, using it as a wave guide (already employed for examination of the Phenix distant welds on the conical shell supporting the core) will be pursued.

8. CONCLUSIONS

The program presented above, will be organized with regard to two short term milestones: 2009 and 2012. The first period is dedicated to propose and study innovations that will be integrated in very preliminary sketches. The 2009 milestone will be the opportunity to select promising orientations. Between 2009 and 2012, selected technologies will be studied in depth, and an integration work towards one reactor layout and one backup will be performed. Each milestone with be also the opportunity to address the question of the prototype that will have the aim to feature the technologies of the power plant of the future as far as possible. In 2012, the main specifications of this reactor are to be fixed and will take into account its mission regarding demonstrations on the fuel cycle. The facilities for its own cycle will also have to be defined, with regard to the first core, and to possibilities of cycle experiments at the scale of some subassemblies.

NOMENCLATURE

SFR	Sodium-cooled Fast Reactor
LWR	Light Water Reactor
IBG	Internal Breeding Gain
EOL	End of Life
M.A.	Minor Actinides
EPR	The AREVA 3rd generation PWR
ULOF	Unprotected Loss of Flow
PCS	Power Conversion System
NDT	Non Destructive Testing

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