MATERIALS FOR SODIUM FAST REACTORS AND PROSPECT FOR RCC-MRX CODE

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ABSTRACT

Sodium Fast Reactor (SFR) is considered in France as the most mature technology of the different Generation IV systems. Regarding the potential tracks of improvement, the materials have a key role to play.
This paper is focused for one part on the challenges asked to the materials to fulfill the requirements of new SFRs and for another part to show how the construction and design code will be improved in the future.
After recalling the choices of materials done in the past projects, the ways of improvements will be first developed : ferritic steels as candidates for steam generators and possibly sodium circuits, optimization of materials and fabrication processes to improve safety and risk management, extension of material databases to take into account the 60 years life duration including irradiation and ageing effect, use of Co free materials for hard-facing….Then, it will be explained how the RCC-MRx code, replacing the RCC-MR code, will take into account all those improvements.

INTRODUCTION

In France an important program was launched by the three partners CEA, AREVA and EDF in order to develop an innovative Sodium Fast Reactor (SFR) concept. The SFR concept allows sustainable carbon-free energy production, through maximal use of natural uranium ore, and allows stabilization or even reduction of the nuclear waste stockpile. This concept benefits already of a very significant R&D, licensing and operational feedback. It is however estimated that some innovations are necessary, in order to improve robustness of the safety demonstration and of the operating capability (particularly as regards in service inspection and maintainability), and to improve economics.

In this frame, the prototype ASTRID is planned to enter operation after 2020, in line with the plan for a commercial Generation IV reactor of 2500-3600 MWth unit power to be put in operation after 2040. The first step of the program associated to this 1500 MWth SFR prototype is the conceptual design that will end in 2014, and the second step, the basic design, planned for 2017. Most of the design options studied for the SFR pool type reactors is of interest for ASTRID.

This paper presents first the candidate materials for the new SFRs, the associated challenges and describes the possible improvements to bring to the RCC-MRx design and construction code.

SFR POOL TYPE CONCEPT AND ASSOCIATED CANDIDATE MATERIALS

The plant design is based on an industrial sodium cooled pool reactor, with a target efficiency of 42% and a lifetime of 60 years. The design is guided by the following objectives:
• Simplification of structures.
• Improved In Service Inspection and Repair.
• Improved manufacturing conditions for cost reduction and increased quality.
• Reduction of risks related to sodium fires and to the water/sodium reaction.
• Improved robustness

The key design options of the SFR are the followings:

A single sodium cooled vessel is used with a safety vessel and all the primary components inside. It is hung to the roof and bears the core, the core support structures and the primary sodium. The core, cylindrical, is supported by a strongback. The diagrid, maintaining the core sub-assemblies feet, has a bottom sodium inlet. To allow the removal of the sub-assemblies, the locations in the diagrid are coated by Stellite. An ACS (Above Core Structure) is there to guide control rods, support instrumentations, and calm sodium above the core. The primary circuit is closed at the top by a forged roof, pierced by penetrations in which are inserted the removable components such as pumps and exchangers. Two eccentric rotating plugs are able to take any of the fuel assemblies by means of charge machines. The Primary Circuit contains IHX (Intermediate Heat Exchangers) connected to secondary loops, Primary Pumps, and DHX (Decay Heat Exchangers. Each secondary loop is equipped with sodium/water Steam Generators Unit (SGU).

Regarding operating conditions, the temperatures at normal operation are the same as for previous projects: the sodium temperature varies from 400°C at the input of the primary circuit to 550°C at the outlet of the core. The Steam Generator Units are submitted to a lower temperature that is expected to be around 530°C at the exchange zone entrance.

For the materials in contact with sodium, such as main vessel and internals, the criteria for the choice of materials are the following:

- Good properties at high temperatures (for internals operating at 550°C)
- Low interaction with sodium
- Intercristalline corrosion resistance
- Good weldability
- Good structural stability
- Reduced activation for ISI&R and dismantling

These criteria have led to choose austenitic stainless steels 316L with controlled nitrogen (X2CrNiMo17-12-2 with controlled nitrogen, denominated 316L(N) in the following). A good confidence is acquired for this material thanks to the experience and large R&D database available.

Concerning components, innovative Steam Generator Unit concepts are being developed and assessed with the important objective to improve the robustness regarding prevention and mitigation of sodium/water reactions. Different types of components are investigated: the modular Straight-tube SGU with reinforced external shell and bellows, the modular SGU with bended tubes, the modular SGU with simplified helical bundle... Having small SG combines the advantages to need a reduced time for maintenance and re-qualification, and to intrinsically limit the consequences of a sodium/water reaction.

The main motivation guiding the choice of a material for components devoted to heat exchanges as the SGU, is to have good thermal properties in order to increase the compactness. The ferritic/martensitic steels are therefore of interest, and the Cr-Mo grades are the main candidates thanks to their good mechanical properties at moderately high temperatures. The use of modified 9Cr1Mo can bring some improvements. It combines good resistance in creep, good resistance to water and water vapor corrosion and correct weldability. Such a material is well fitted to straight or bended tubes SGU, but has never been used in past projects.

Regarding the incentive to diminish the activation products to make easier the ISI&R and the dismantling, the Stellite used in the friction zones of the primary circuit is not satisfying. Alternative materials are nevertheless not defined at present time.

Table 1 gives an overview of the candidate materials for the SFRs

Table 1: Candidate materials

Component	Specificities	Candidate Materials
Structures in contact with sodium	Internals and hot circuits T ≤ 550°C	X2CrNiMo17-12-2 with controlled nitrogen (316L(N))
	Auxiliary pipings little loaded T ≤ 400°C Strongback, diagrid	X2CrNiMo17-12-2 with/without controlled nitrogen (316L(N)) or X2CrNi18-9 with/without controlled nitrogen (304L(N))
	Main Vessel T≤400°C	X2CrNiMo17-12-2 with controlled nitrogen (316L(N))
	Intermediate Heat Exchangers T≤ 550°C	X2CrNiMo17-12-2 with controlled nitrogen (316L(N))
Upper closure of vessel	Outside sodium (T< 200°C)	A42, 16MND5 according to the design features
SGU and vapor pipes	Vapor and Sodium environment T ≤ 490°C	modified 9Cr1Mo if straight tubes design, or Z5NCTA33-21 for helical bundle design (A800)

MATERIAL CHALLENGES

60 Years Lifetime

The past projects have qualified the materials for a lifetime of 30 or 40 years. An important issue for the new generation of plants is the demonstration that these materials are still adequate up to 60 years. This passes through the identification of the damaging modes able to be significantly modified by a life duration extension.

Representativeness of the experimental tests is one point to manage, as it is used to say that for the most stable materials such as 316L(N), an extrapolation factor, between real operation and experiment, equal to 3 can be envisaged, obliging to start the experiments very early :

- 40 years (280 000 h) of operation lead to experimental tests of 11 years (93 000 h)
- 60 years (420 000 h) of operation lead to experimental tests of 16 years (140 000 h)

The time-temperature equivalence allows a compensation of the time increase by a rise of the testing temperature, but only if it can be guarantied that the rise in temperature does not change the mechanisms of viscoplastic deformation and of damaging. As an example, for 316L(N) at 550°C, a range of 6°C corresponds to a factor 1.5 on the time and at about a variation of 5% on the stress level.

The qualification of the materials has to be done checking in particular that ageing factors are still valid for 60 years and that damage mechanisms by creep-fatigue are still the same. If a lot of long-term and cycling data exist for 316L(N) showing that the material is very stable, the situation is different for Modified 9Cr1Mo which displays a change in its microstructure after ageing or cycling at high temperatures. Base metal and welded joints of Modified 9Cr1Mo are likely to be affected: the appearance of a discrepancy between the beginning and the end of life of the instantaneous properties (especially tensile strength, toughness and short-term creep strength to rupture) has to be evaluated.

Some of the PHENIX plant structural materials issued from long-term exposure should be examined in the next years and should provide valuable information.

Irradiation Issues

The irradiation effect can be harmful for the ductility, the toughness, the creep behavior and the creep-fatigue resistance. The fluence levels of the past projects were such that irradiation was not a real issue for internals.

But this consideration shall be re-assessed for the new SFR and ASTRID, as the configuration of the core and compactness could be more challenging. Moreover, the issues will be different between a “hot” structure and a “cold” structure, considering He production.

Thermal Striping Issues

The locations where fluids at different temperature are mixed are numerous in a SFR. A good practice, applied in other reactors is to have mixing device at junctions where temperature differences are too large, or if possible to move away the mixing zone from the component walls. Nevertheless, such provision is difficult to apply in the primary circuit where thermal striping can also occur and involves structures located just above the core. These structures are submitted to streams coming from fissile and fertile assemblies at different temperatures, able to induce thermal striping if the temperature differences are significant.

A solution to increase the acceptable temperature variations is to have a more resistant material in the sensitive zones than 316L(N). But qualification of the material has to be done in the range of operating conditions of the component including the welding/joining issues.

Manufacturing Issues

Regarding manufacturing, there is an advantage to use large forged pieces in order to reduce the number of welds. To do that it is necessary to optimize the forgings quality, in particular to be sure that property gradients in the thickness are low and that grain size is adequate regarding the examinations criteria. Fabrication and complete testing of a prototypic part would be required in order to verify that all properties fit with the expectations. For Modified 9Cr1Mo, this is a real key issue as no thick plate (greater than 250 mm) has been manufactured with the specification required by nuclear codes.

Successful welding process with satisfactory properties of welded joint (HAZ and weld metal) is crucial.

The criteria for the selection of the weld metal chemical composition associated to Modified 9Cr1Mo are hot cracking resistance, adequate impact toughness and reference transition temperature of the weld metal. Indeed, the French regulation concerning Nuclear Pressure Equipments requires a level of 60 J for Charpy V impact values at 0 °C for base metal and weld metal (together with 20 % as minimum elongation at room temperature) for the maximum level required (N1). First R&D results have shown that industrial filler materials fail to satisfy such requirements and that a specific optimized filler material has to be developed.

Another key issue regarding welding is the post weld heat treatment. Care has to be taken as modified 9Cr1Mo is not as tolerant as other ferritic steels more usually employed. Failure to obtain precise microstructure will seriously degrade the alloy high-temperature properties. Experience feedback from high temperature conventional plants has displayed premature failures of components after long term operation: the failures were attributed to type IV cracking in HAZ and to the presence of soft zones in base metal. It appears nevertheless that at least the latter phenomenon could be limited or avoided by taking care of the fabrication stage.

Modified 9Cr1Mo High Temperature Cyclic Behavior Issues

Another question turns around the cyclic behavior of modified 9Cr1Mo at high temperatures. Plastic deformation during cycling at high temperature ($T \geq 500°C$) induces a change in the dislocation densities and in the grain sizes. A mechanical softening is observed together with an acceleration of the creep deformation. Such a phenomenon is completely different of what is observed in austenitic stainless steels as 316L, and necessitates reconsidering the way to take it into account in the design analysis. See on Figure 1 an illustration of its cyclic behaviour.

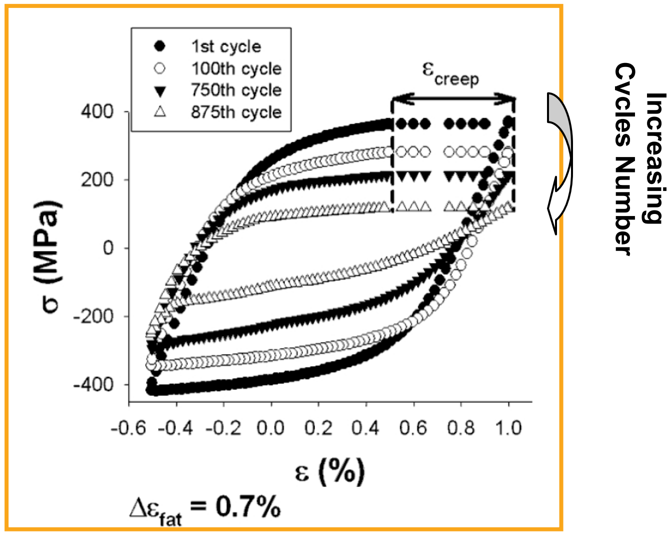

Fig. 1: Stress-strain cyclic behavior of Modified 9Cr1Mo

PROSPECTIVE IMPROVEMENTS FOR RCCMRX CODE

The AFCEN (Association Française pour les règles de Conception et de Construction des Matériels des Chaudières Electro-nucléaires) proposes a complete collection for the design and construction codes for nuclear plants and related equipments, initiated by RCC-M for Pressurized Water Reactors (PWR).

A draft of the fifth edition of the RCC-MR code, named RCC-MRx 2010 (draft), has been issued on December 2010. This RCC-MRx Code is the result of the merger of the RCC-MX 2008 developed in the context of the research reactor Jules Horowitz Reactor project, in the RCC-MR 2007 which set up rules applicable to the design of components operating at high temperature and to the Vacuum Vessel of ITER. This is a non-public document established in order to prepare the fifth edition which will be published in French and English by AFCEN in 2012. In parallel, the draft is being reviewed by Europeans in the frame of a CEN Workshop. Identification of possible changes to the code regarding non-French projects is in progress.

The RCC-MR code was written originally to collect the feedback on the design and construction of Superphenix. Thanks to successive projects, the code has been continuously improved. It has been adopted by the European countries (France, Italy, Great Britain and Germany) associated with the project EFR (European Fast breeder Reactor), with the support of the WGCS (Working Group on Codes & Standards from the European Commission). It was then extended to include the experience in design and construction gained in collaborative European countries and supported by the large cooperative R&D programme. It appears to be fully appropriate to be used in the frame of the new SFR projects. Nevertheless, as the technologies and the requirements have evolved, the code has to be completed and the results coming from the R&D programme underway have to be capitalized.

In relation with the material challenges set out above, a set of improvements of the code text can be envisaged. The most significant work has to be done relatively to modified 9Cr1Mo, but also to update the manufacturing specifications for 316L(N) and to update the databank for a 60 years lifetime.

Improvements Tracks For Modified 9Cr1Mo Steel

Due to the softening feature of the cyclic behaviour of the material, some design rules have to be re-assessed:

- The ratcheting rule. In RCC-MRx it is based on the efficiency diagram built originally with tension/torsion test results on austenitic materials (such as 316L) and extended to other cyclic hardening materials and other types of tests. Experimental results carried out on softening materials shall be confronted with the results given by the present methodology.
- The creep-fatigue rule. In RCC-MRx this rule has been fully validated for austenitic steels, and the practice is to use cyclic curves of the material with coefficients representative of cyclic behaviour. For modified 9Cr1Mo, a specific coefficient representative of cyclic softening is provided. Creep laws (primary and secondary) are taken into account considering the material as not-softened.

It appears that those rules are conservative, in some cases too conservative, and do not represent fully the real phenomenon observed for modified 9Cr1Mo.

Boundary curves, as negligible creep curves, are needed for the designer to know if the effects of creep have to be considered in the structural analysis. Such a curve does not exist for the modified 9Cr1Mo (only a maximum temperature is supplied whatever is the duration). The principle of building the curve has to be established (it could be different from what is done for austenitic steels), and the curve by itself defined.

Associated to the design rules, an update of the databank could be necessary specially if the in-service conditions (after cycling at high temperature or long-term-exposition at high temperature), do affect the instantaneous properties (tensile strength, toughness and creep strength to rupture), and the creep laws. The possibility to define an "aging coefficient" to be applied on admissible stress, and a specific criterion for the RTNDT could be discussed.

Characterization of welded joints allow to define weld coefficients to be integrated in the structural analysis of structures supposed exempt of weld. In particular, the creep properties of welded joints have been defined in the RCC-MRx through Jr coefficients to be multiplied to creep rupture stress of base metal. Confirmation of the data would be necessary in particular considering representative products (base and filler materials taking into account the thermal treatments during welding). Same types of coefficients relatively to fatigue strength have to be defined (Jf coefficients).

Improvements Tracks Relative to Procurement, Specifications, Welding and NDE

As said before, there is a need to update the procurement specification (referred to as reference technical specifications in the code) to make them fully compatible with the industrial practices. As an example, the procurement of flat austenitic products of low thickness may be done by continuous tapping whereas the specifications of the code are presently written for a production by ingots. Such a possibility could be examined and introduced in the code with its specific parameters.

For forging, the present practice is to go towards larger parts. It is likely that reference specifications have to be created for specific part/component of SFR/Astrid such as tubular plates in modified 9Cr1Mo steel.

The RCC-MRx does not propose reference sheet for modified 9Cr1Mo filler material. According to the discussion above, a specific material has to be developed, knowing that "on the shelf" filler materials do not provide joints with enough toughness and ductility. Reference sheets have to be created at least for TIG and SMAW processes. Dispositions for PWHT have to be re-assessed and possibly refined to introduce more stringent parameter ranges in order to be sure to prevent any subsequent long-term damage. Of course the realization of welded joints using those specifications have to be used for characterizing the resistance of the joint against fatigue and creep and then to determine the weld coefficients (fatigue Jf and creep Jr).

Also, as regards 316L(N), a refinement of the filler material specification or an update of the reference sheets should be done if another process of welding than TIG and SMAW is envisaged (automatic processes for instance).

The possibility to use different types of methods for Ultrasonic Testing examinations is proposed in the specific appendix devoted to ITER vacuum vessel. A progress could be to generalize those methods (creeping waves, tandem, phase-array) in alternative to the single probe reflexion method.

To be noted also that if new hard facing materials are defined, it will be necessary to introduce in the code the specificities relative to the coating process.

Negligible Irradiation Boundary

The limits expressed in RCCMRx for negligible irradiation criterion are based on experimental data mainly carried out on base metal. Confirmation of these limits would be beneficial for the code even if there are difficulties to obtain new data particularly corresponding to welded materials (ductility and impact testing).

Another track of improvement would be to refine the criterion when creep is significant. Today, the creep-fatigue damage ratio required is equal to 0.1 if irradiation is significant, meaning that very little creep is allowed when irradiation is present. Even if this topic should not be of concern for SFR, some new data will allow to a better understanding of the phenomena when little creep is present with few irradiation.

Modifications Linked to 60 Years Lifetime

The designers will need databases with characteristics up to 4.2 10^5 h (60 years). The completion of the Appendix A3 (base materials) and A9 (weld coefficients) has to be done, for 316L(N), modified 9Cr1Mo at least.

This concerns the following properties:

- Creep rupture strength
- Ageing coefficient if any
- Creep-fatigue interaction diagram (if changed)
- Creep strain laws
- Fatigue curves
- Weld coefficients

Other Prospectives

Other reflexions are in course of progress: bellows are usually set up in a high temperature plant in order to facilitate the thermal expansion between components at different temperatures. Two methods of designing are today proposed in the code, one is a usual method employed by non nuclear industries. The other is based on the classical "shell" analysis of the RCC-MRx. In the 2 cases, some questions arise: adequation of the first method as regards nuclear specifications, adequation of the second method when the material (316L(N)) is not solution annealed after forming and presents some inhomogeneities of microstructure.

Another topic underway concerns the possibility to use the European standards for the design and construction of a quality level 3 component. Up to now RCC-MR code has specific rules for these types of components. The question has to be posed to let the possibility to the designer to use EN13445 for instance, possibly restricting it to insignificant creep domain.

CONCLUSION

New projects such as SFR and ASTRID bring technical topics of industrial interest, which merit to be considered in the Design and Construction Code RCC-MRx, code devoted in particular to high temperature reactors. This is an opportunity to develop the code around several topics: updating of the database for a 60 years lifetime, improvement of the design and construction rules for Modified 9Cr1Mo, introduction of new industrial technologies….

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