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ASTRID SFR PROTOTYPE STEAM GENERATOR DESIGN EVOLUTION RELATED TO SAFETY AND COST ISSUES

Antony Woaye Hune, Alain Gerber, Jean-Philippe Pirus, Laurent Soucille
AREVA NP

10 rue Juliette Récamier – F-69456 Lyon Cedex 06, France
04.72.74.73.92 – antony.woayehune@areva.com

Gilles Rodriguez, François Beauchamp
French Commission for Atomic Energy and Alternative Energy
F-13108 Saint Paul Lez Durance, France
04.42.25.74.37 – gilles.rodriguez@cea.com

Abstract

One option of the ASTRID prototype power conversion systems (PCS) is the steam cycle based on the Rankine cycle, which has been implemented in the past in France in the Phénix and Superphénix Sodium Fast Reactors, and is also being implemented in other SFRs in construction in the world (Russia, India, China).

This paper is placed in the context of the Rankine cycle ameliorations and in particular on the Steam Generator (SG), developed by AREVA, from the safety point of view without significant penalty on cost. Therefore significant progresses are awaited to reduce sodium/water reaction accident scenario impact as much by prevention as by mitigation.

Following the ICAPP 2010 paper (1), ASTRID project performed a thorough comparison of SG designs with regard to these safety requirements but also including increased sodium/water segregation (inlet and outlet of sodium implemented at the bottom of the SG), material issues and cost considerations. The design is based today on a monolithic SG with alloy 800 helical tubes but equipped with geometric specificities necessary for allowing the use of a single component per secondary loop.

I. INTRODUCTION

Sodium-cooled Fast Reactors (SFR) is one of the Generation IV reactor concepts selected to secure the nuclear fuel resources and to manage radioactive waste. In the frame of the June 2006 French act on sustainable management of radioactive materials and wastes, French Government entrusted CEA to conduct design studies of ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) prototype in collaboration with industrial partners.

One option of the ASTRID prototype power conversion systems (PCS) is the steam cycle based on the Rankine cycle, which has been implemented in the past in France in the Phénix and Superphénix Sodium Fast Reactors, and is also being implemented in other SFRs in construction in the world (Russia, India, China).

This paper is placed in the context of the Rankine cycle improvements and in particular on the Steam Generator (SG), developed by AREVA, from the safety point of view without significant penalty on cost. Therefore significant progresses are awaited to reduce sodium/water reaction accident scenario impact as much by prevention as by mitigation.

The paper (1) presented at ICAPP10 Conference was investigating several ways to achieve such a goal: research of designs favoring prevention and other mitigation. An envelope scenario was considered to determine the design conditions, i.e. the instantaneous rupture of all tubes, at least for safety extended situations (on accidents beyond category 4, occurrence probability less than 10^{-7} /reactor/year). SG modularity and main secondary circuit adaptations allowed satisfying such a new requirement.

The objective of the present paper is to provide what has been performed by the ASTRID project from the ICAPP10 paper (1) leading to the current SG design.

II. DESIGN OF ASTRID SG

The main orientations for ASTRID steam generators are:

- Master the secondary loop complexity by choosing once-through integral modules: it limits costs, maintenance and sodium fire risks on secondary loops.
- Provide possibility to demonstrate SG, secondary loop and Intermediate Heat eXchanger (IHX) integrity under postulated simultaneous DEGF (Double Ended Guillotine Failure) of all the tubes of one SG module. In that hypothesis the SWR

(Sodium Water Reaction) events are bounded by a scenario which does not depend on leak detection performance. It leads to an intrinsically safe SG design in any envelope scenario regarding SWR.

- Provide enhanced ISI&R (In-Service Inspection & Repair) capabilities and SWR detection system.
- Consider investment cost efficiency by evaluating the entire system (SG + secondary loop + cost of the SG building).

II.A. ASTRID SG Design Selection

In the first phase of ASTRID option definition, all kind of SG designs have been investigated. Then, after a first phase of evaluation, the study was concentrated on three Steam Generator types: EFR (European Fast Reactor) type Straight tubes with bellows on external shell, PFBR (Prototype Fast Breeder Reactor) type Straight tubes with bends on heat exchange tubes, modified design of the SPX (Superphénix) type Helical tubes SG (from left to right and bottom in the following Figure 1) [2].

Set of criteria was put on:

- Safety: SWR and SWAR (Sodium Water Air Reaction) management,
- ISI&R,
- R&D efforts: cost & risks, time required,
- Manufacturing: cost & risks, qualification program, industrial capabilities, control performances,
- Reliability: TRL (Technology Readiness Level) maturity level, experience feedback, design margins, corrosion and material risks, ageing effects,
- Impact on the reactor Plant Layout: secondary loop complexity, SG building surface and volume, volume of auxiliary systems,
- Investment cost.

To allow a proper Steam Generators comparison, several engineering and R&D studies were carried on from 2010 to 2013 in fields where significant knowledge gaps were highlighted. In particular an important R&D program was launched on martensitic-ferritic steel (T91) and alloy 800 austenitic steel (A800) materials. The goal was to be able to compare their behaviour in steam and sodium environment and to complete the wastage database for T91 steel thanks to experimental tests carried on with JAEA (Japan Atomic Energy Agency) and IGCAR facilities. [3]

These results contributed to adapt the related safety computing codes on those specific materials [4]: PROPANA code on wastage rates prediction; and MECTUB code for tube rupture times.

On material, R&D studies showed that the ferritic-martensitic steels are more sensitive to the generalized corrosion and present an important wastage and leaks auto-evolution rates.

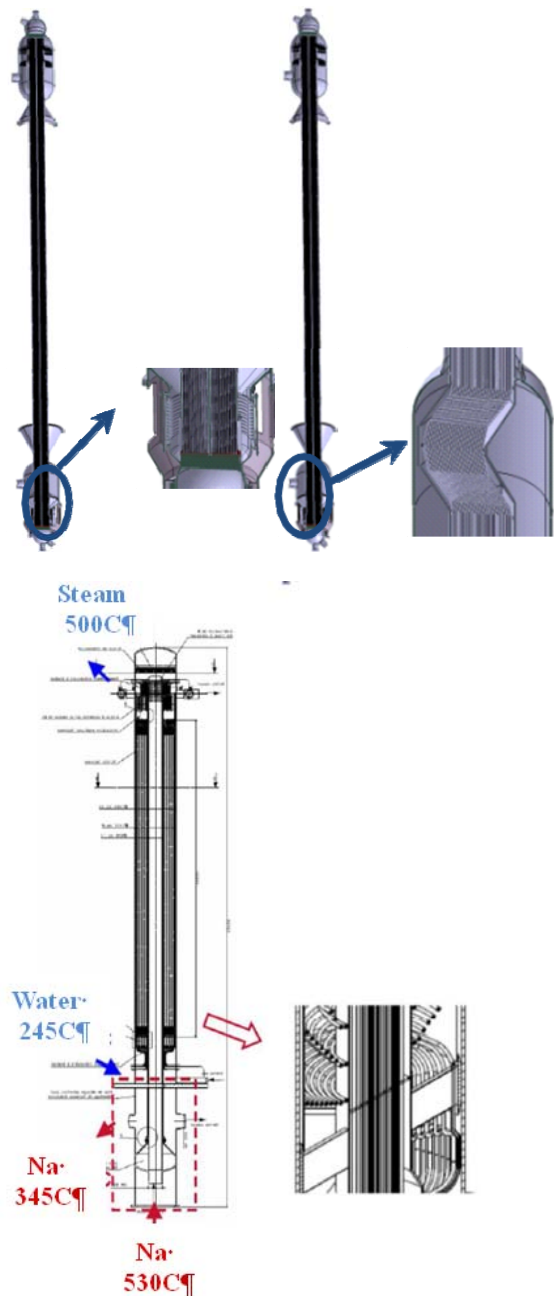


Fig. 1. The three type of SG design for ASTRID final selection (from left to right and bottom: straight tubes, with bends on straight tubes, helical tubes).

Recent experimental tests (see examples on figure 2) confirmed that wastage rates are varying in a range from 2 to 10 times quicker than A800 (depending on operating parameters and mainly temperature). The austenitic alloys have a better behaviour towards the above phenomena, but are sensitive to the stress corrosion in sodium hydroxides presence.

It can be outlined that T91 presents good mechanical properties at high temperatures. But the R&D reveals that the cyclic behaviour and fatigue of the Alloy 800 and the modified T91 are different: for the ferritic-martensitic grade, the plastic deformation during cycling at high temperature ($T \geq 500^\circ\text{C}$) induces a change in the dislocation densities and in the sub-grain sizes. A mechanical softening is then observed.

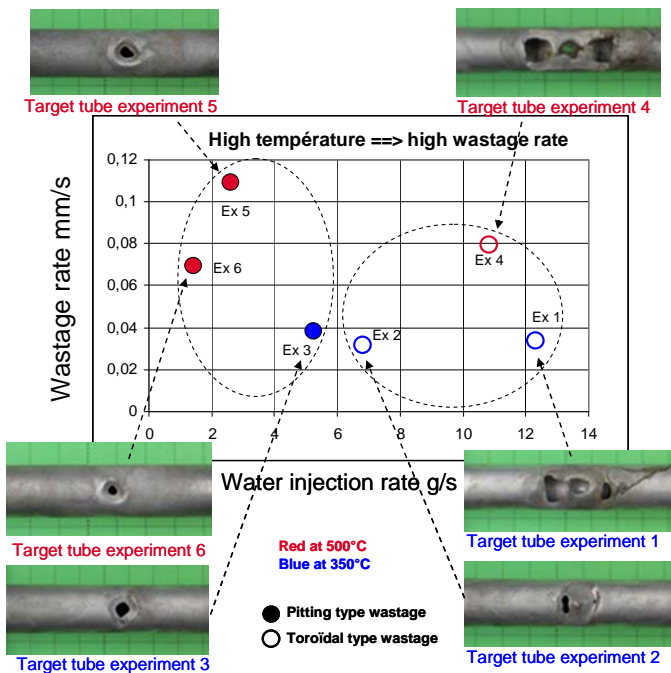
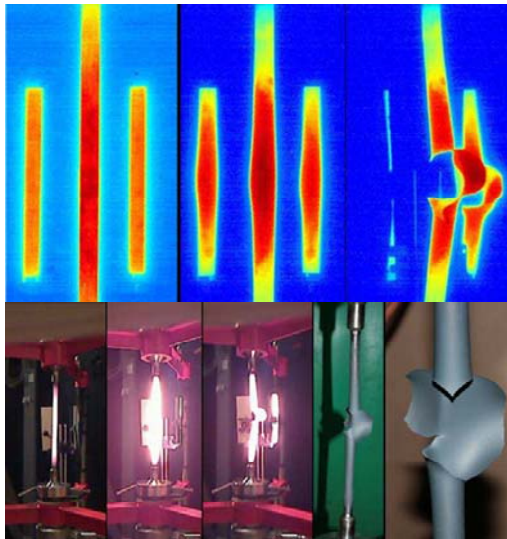


Fig. 2.

Some experimental results carried on SG tubes to adjust codes.

Up: bursting of heated tubes on the SQUAT facility (Cadarache, France). [4]

Down: Impingement test realised on SWAT-1R facility (O Arai, Japan). [3]

Such a phenomenon is completely different from what is observed in austenitic stainless steels as Alloy 800, for which the plastic deformation during cycling leads to a cyclic hardening. The consideration of component lifetime could also be a crucial point on material selection. For ASTRID, SG lifetime for design has been set to thirty years.

Concerning design, it appears that the straight tubes design with bellows on the shell is more difficult to justify towards thermo-mechanical behaviour than tubes with expansion bends designs.

The BN-600 design (Russia) [5] is a successful example of long term industrial operation of such SG, but it was built with three stage unit (evaporator / super heater / re-heater) and with a modularity aspect (8 SG per secondary loop). The straight tubes design with expansion bends on the tubes allows a higher flexibility to accommodate differential thermal expansions, especially between the different tubes. On the other hand, due to the bend on the tubes, the manufacturing sequence and the tube supporting system need to be adapted accordingly. PFBR SG design is a representative example with design and manufacturing accomplishment, but this component has been down-sized to 157 MWth. The feedback of this component in operation will be a precious gain of knowledge.

The SPX SG has demonstrated the ability to realize and operate large helical type SG unit. Diverse similar components tested in the world with limited power (a few tenths of MWth) operated also with no major problem encountered. Finally even if SPX 750 MWth with A800 alloy tubes provides a sound basis, this option lacks of long duration experience feedback.

For ASTRID, AREVA engineering company has targeted an innovative interest linked with this design. Indeed it is possible to set-up compact secondary loops designs by implementing both sodium outlet and entry at the bottom of the SG (Fig. 3). It provides a more rational separation of the sodium and steam piping systems. It allows demonstrating a better physical separation of the sodium/water risks.

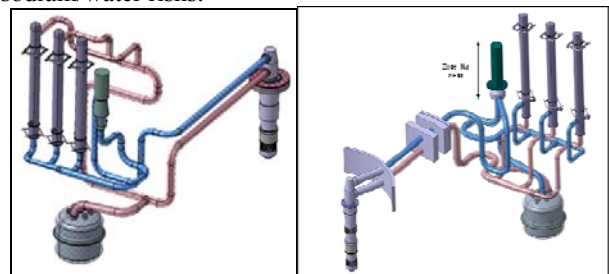


Fig. 3.

Evolution in SG Helical tube design and its secondary loop.
 Left: Cold outlet (blue) at the bottom of the SHG, hot inlet (red) at the top
 Right: Inlet and outlet at the bottom of the SG

For a same unit power, the SG design with sodium inlet and outlet at the bottom side is offering a better risk management by segregating sodium hot and cold legs at the plant layout bottom part, and hot steam inlet at the top of the steam generator. It then simplifies the drawing of physical separation with different levels of the steam generator building (separation by floors).

Regarding protection systems, they are designed in all cases to quickly detect water leak in sodium before any secondary leak ignition. In addition and complementary to the conventional hydrogen detection system, active acoustic detection are envisioned to enhance detection velocity. This is specially recommended when using ferritic-martensitic steels since it has a weaker behaviour towards leak auto-evolution and wastage than the alloy 800

In all cases the SG unit power was adjusted in order to cope with the envelope scenario of all tubes simultaneous DEGF within one SG module.

II.B. ASTRID SG Type Choice

The ASTRID SG final selection was carried out during the first phase of ASTRID conceptual design studies called AVP1 (includes design, structural analysis, thermal-hydraulics, flow induced vibrations, SWR calculations, experience feedback, ISI&R evaluation, etc.). In addition analyses on a multi-criteria grid based on the previous studies are performed. Experts' evaluations are done. Thus periodic experts meetings were conducted in the different fields over 2 years. The conclusions were that there was no reason to eliminate de facto one specific design amongst the three pre-selected. Indeed, all designs get advantages & drawbacks and the final synthesis - without taking account of the investment costs - provides results in a very tight space.

The experimental feedback of all industrial SG operations did not provide a clear advantage to one specific concept. Nevertheless it was confirmed that A800 has a real better behaviour regarding the SWR reaction (wastage). In addition the evaluation of the entire system including SG plus secondary loop plus SG building is providing an advantage to the Helical tube SG because:

- Inlet & outlet at the bottom is well fitted to reduce the length of the secondary loop,
- The drawings provide a better segregation of the sodium and steam parts,
- The SG building is reduced (20 m compared to 28 m for a Straight Tubes SG option).

II.C. ASTRID SG Unit Power Adjustment

The size of each module is such that the largest possible sodium-water reaction, corresponding to a failure of the whole tube bundle of one SG, can be accommodated without excessive loadings on the SG casings, but also on the secondary sodium piping system, and the intermediate heat exchangers within the primary system.

This avoids the need of a difficult demonstration on the limit of propagation of failures from one tube to the neighbouring tubes in a large bundle.

In terms of safety, consideration of such accident should contribute to the robustness of the demonstration as it would ensure avoidance of any cliff edge effect which might be linked to the assumptions regarding the number of tubes failures or to the kinetics of the phenomena leading to the failure of several tubes (see Ref. 1).

To meet and check this condition, parametric calculations have been performed. Progressive unit powers are assessed to study the consequences of the bounding case of a simultaneous and instantaneous rupture of all tubes within one SG module. This has been done to evaluate whether integrity of the secondary system can be maintained at intermediate heat exchanger level, at sodium loop level, and also at Steam Generator level.

By a combination of several configurations, the most influent parameters on the dynamic pressure were progressively learned: additional discharge lines on hot and cold pipes, discharge pipe diameter, positioning of the discharge line close to the SG and IHX components.

Calculations were performed with the fast dynamics code EUROPLEXUS [6], using the finite element method and an explicit resolution in time.

The validation of this code for the sodium-water reaction simulation has been performed by comparison with other codes or with analytical calculations and large scale sodium water reaction tests performed during the development of the Superphenix SG. The secondary loop is modelled with 1D tube finite elements. Tubes are filled with sodium, argon gas at the top of each SG, and hydrogen produced by the sodium water reaction are accounted of. The structures are not modelled (no fluid/structure interaction). Singular pressure drops and roughness of the pipes are taken into account and some specific elements like the rupture disks are modelled. The gravity is applied and the initial nominal absolute pressure is set to 3 bar in the whole model. The initial velocity of the sodium in the secondary loop, which is low, is not taken into account.

Initially the model of the secondary loop was adjusted with six SG modules of 100 MWth for comparison with EFR design as reference concept. Then, by adjusting the loop design, the allowable ASTRID SG unit power was progressively raised to 125 MWth (three SG per loop for

ASTRID) to finally reach a configuration with 375 MWth Helical type SG (one SG per loop for ASTRID, four in totals), see Fig. 4 and 5.

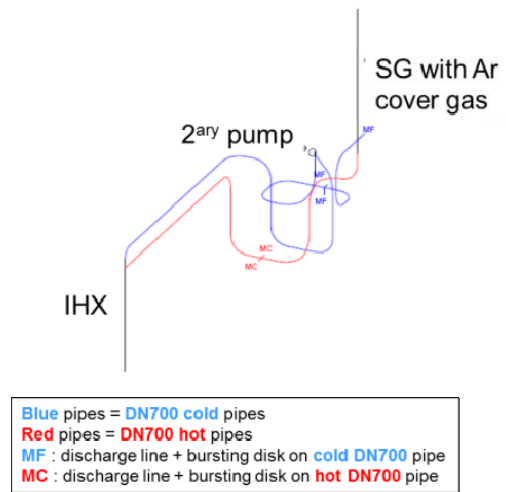
Thus, it was progressively demonstrated that the respect of the envelope SWR scenario could be achieved with any kind of helical type SG power from 100 to 375 MWth. It must be noted that the value of 375 MWth was an upper limit value. An extension to 500 MWth power (for instance in a three loop configuration, one SG per loop) did not provide satisfactory results.

Therefore the power unit adjustment could be done by considering a grid of evaluations where all different aspects were quoted and where the cost evaluation was compared with the first modular system studied for ASTRID during the AVP1 phase (three SG per loop, 125 MWth each).

This choice has proven a significant gain regarding ISIR, global arrangement and investment cost in choosing monolithic SG. That led the project to select for the AVP2 phase an ASTRID SG configuration with four 375 MWth Power helical SG with sodium inlet and outlet at the bottom part of the component (one SG per loop).

By selecting this option, ASTRID project has made the choice to strongly limit the modularity approach in steam generator design.

SG Power. 375 MWth, Helical type, Compact secondary loop



| Instantaneous Double Ended Guillotine Failure SWR pressure (bar) calculated on SG areas and connecting circuits and components | | | |
|---|---------------------------------|---------------------------------|------------------------------------|
| | 240 DEGF with 3 discharge lines | 240 DEGF with 5 discharge lines | Allowable pressure (bar) Level CID |
| SG Shell | 110 | 110 | 59/128 |
| DN700 outlet cold pipe | 89 | 89 | 37/67 |
| DN700 inlet hot pipe | 48 | 43 | 33/59 |
| SG cold discharge line | 69 | 69 | 37/67 |
| DN700 cold discharge line | 46 | 39 | 37/67 |
| DN700 hot discharge line | 30 | 16 | 33/59 |
| IHX | 46 | 22 | 27/55 |

Fig. 4. Description of the secondary loop configuration for the EUROPLEXUS Code [6] and corresponding calculation results.



Fig. 5. Evolution of the ASTRID configuration from three SG (125 MWth each) to one single SG (375 MWth).

II.C. Safety considerations and SWR management / approach for ASTRID SG

ASTRID safety orientations are based on the line-of-defence method for analysing prevention of severe accident, completed by line of mitigation method for analysing the mitigation of severe accident.

The SWR reaction was classified during the European Fast Reactor (EFR) project as follows:

- A small SG tube leak in category 2 conditions, corresponding to an occurrence frequency higher than 10^{-2} per reactor-year,
- A guillotine-tube rupture in category 3 conditions, corresponding to an occurrence frequency between 10^{-2} to 10^{-4} per reactor-year,
- A larger number of about ten guillotine-tube ruptures in category 4 conditions, corresponding to an occurrence frequency below 10^{-4} per reactor-year,
- It can moreover be underlined that SG tubes were designed to withstand a safe shutdown earthquake in category 4 conditions.

Nevertheless, due to the large number of SG tubes involved, the guillotine rupture of a tube was conservatively assumed and the loadings on the intermediate heat exchanger due to both the earthquake and the sodium-water reactions were combined.

For all the above situations, leak tightness of the secondary system, in particular at intermediate heat exchanger level, was demonstrated. More severe scenarios, considering sequential tubes failures up to the rupture of all tubes of a single SG unit were besides studied, for which integrity of the intermediate sodium loop was still maintained.

For ASTRID project, the envelop scenario of a simultaneous guillotine rupture of all tubes has been positioned in the domain of severe accident prevention (SP) corresponding to an occurrence frequency between 10^{-4} and 10^{-7} per reactor-year.

In practice, to eliminate or reduce the risk of sodium-water reaction in a steam generator, we have to prevent from effects of steam water leaks, or minimise its consequences on the intermediate system. For this, successive independent measures operating at several levels are implemented.

All these measures have associated methods; there are 3 categories:

- Prevention methods,
- Detection methods,
- Mitigation methods.

The prevention of failures in the tube bundle is based on a high quality manufacture and inspection (RCCM-R quality level 1 and compliance with nuclear pressure equipment requirements). This high manufacture and inspection qualities is combined with the choice of the suitable material of the SG tubes to prevent the worsening of a water leak in a steam generator. Indeed, it has been shown that the wastage resistance of the tube varies according to the nature of the material.

In this regard, the latest experimental results on wastage (see examples on Figs. 6 and 7) provided decisive new data on the wastage resistance of SG tubes made of 9%Cr. Those data confirm that the maximum rate of damage by wastage of a 9%Cr tube for a leak located at the bottom of the tube bundle (sodium temperature of 350°C) is around double than tubes made of alloy 800 under the same operating conditions.

For leaks located at the top of the bundle (500°C), the wastage rate of the 9%Cr tubes is significantly higher than for alloy 800 tubes, by a factor which varies (by around 6 to more than 10) according to the leak flow rate being considered.

The addition of "anti-wastage" plate type devices over the welds is a solution that can be considered but only for straight-tube SG concepts. For Helical tube SG the main action is to determine the welding zone away from thermo-mechanical stressed zone like the Departure of Nucleate Boiling (DNB) zone.

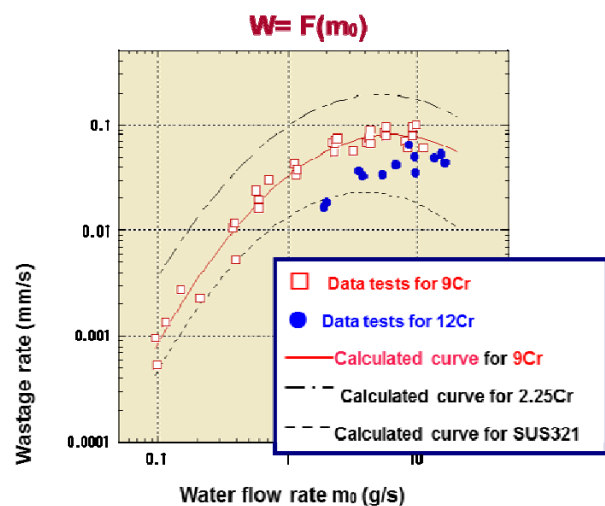


Fig. 6. Wastage rate evolution for several materials 9Cr, 12Cr and SUS321. Experimental conditions: $T_{Na} = 480^\circ\text{C}$, $T_{water} = 331^\circ\text{C}$, $P_{water} = 128$ bar, distance between leaky and target tubes = 16,2 mm

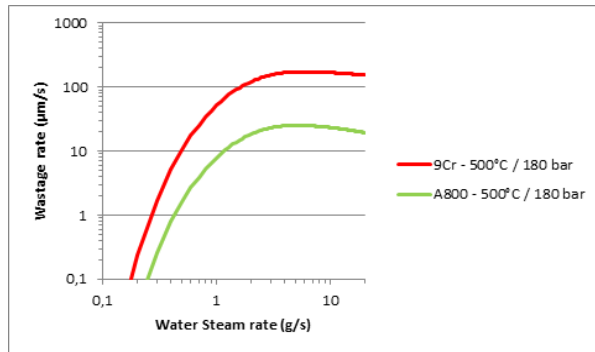


Fig. 7. Comparison of the wastage rates between 9Cr and A800 alloys (curves obtained from PROPANA code calculations)

Therefore, an appropriate choice of design – such as Helical SG tube with A800 - provides an additional margin in terms of the response times of the detection systems and of the protection operations.

The detection of failures of a steam generator tube combines redundant active systems, associated with means of quickly protecting the SGs (detailed below):

- Hydrogen detection in sodium and in Argon (Fig. 8): "Boundary" values for the increase in the measured hydrogen concentration, considered to indicate a leak, are determined by prior calibration of this measurement. If these values are exceeded, first of all (alarm threshold) a warning for the operators is triggered, then automatic actions are initiated to set the SG to a safe fall-back configuration (fast isolation and depressurization sequence described below, and sodium drainage). The main purpose of the detection system is to check that the initial leak can be detected before impingement or rupture of the neighbouring tubes leading to secondary leaks by respectively wastage or swelling-bursting effect.
- Acoustic detection can be used in addition to hydrogen detection. Its main advantage is its fast response time, as there is no transit time as with those for hydrogen detection.
- The other detection systems used to protect the SG include pressure measurements in the expansion tank (as with the UK PFR SG) to detect the increase in pressure following a large leak. There is also a disc rupture detection device (Fig. 8), based on the use of sodium detectors on the relief line downstream the rupture discs. This detection device triggers the other protection actions: fast shutdown of the reactor and fast isolation/depressurization of the steam-water in the SG, and sodium draining.

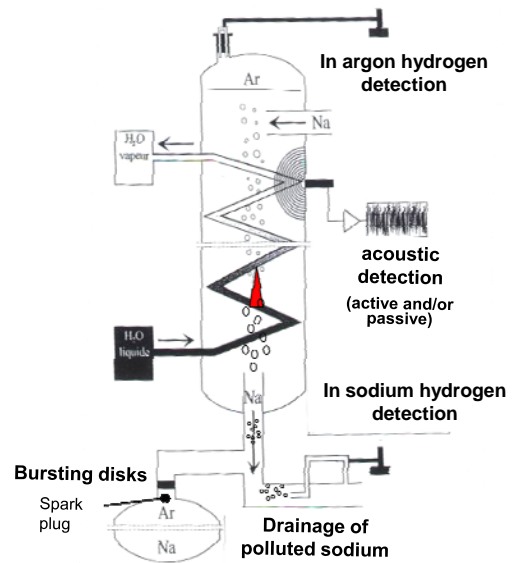


Fig. 8. Integration in a SG of the main safety devices

The Mitigation of a SWR in ASTRID SGs consists of:

- The fast isolation and depressurization system. It is actuated (and shuts down the reactor) as soon as a leak is detected (either by hydrogen detection or by rupture of the safety discs). These systems are said to be "active" because they require implementation and a diagnosis. The purpose of these operations is to prevent the sodium from entering the secondary loop in the SG tubes and the steam-water circuit. Thus, during the whole SG depressurization phase, which happens at the same time as the discharge of the sodium in the case of leaks leading to the rupture of the safety discs, the steam-water circuit is kept in excess of pressure in relation to the sodium vessel in order to prevent any pollution of the steam-water side by the sodium.
- Rupture discs and a discharge circuit. They enable decompression and fast draining of the SG on the sodium side if there is a significant sodium-water reaction. The effect of the increase in pressure following an SWR (acoustic pressure wave then followed by mass transfer) leads to these discs rupture and the sodium drains out due to gravity. This is therefore a "passive protection" system. This device is designed to protect the IHXs, which form part of the primary confinement, against the mechanical effects associated with a sodium-water reaction, and to make acceptable the pollution of the loop by the reaction products.

II.D. Summary of the ASTRID SG design for the AVP2 phase

The final selection of the ASTRID SG for the AVP2 phase is summarized hereafter (Table 1, Figs. 9 and 10).

| | | SPX GV750 MWth | ASTRID SG 375 MW (initial design) |
|-----------------------------------|------|-------------------|--------------------------------------|
| Power | MWth | 750 | 375 |
| Tube number | - | 357 | 240 |
| External tubes diameter | mm | 25 | 22 |
| Tube thickness | mm | 2,6 | 2,7 |
| Tube length | m | 91,5 | 87,3 |
| Helical bundle height | m | 11,9 | 11,06 |
| Water mass flow | kg/s | 340 | 169,5 |
| Total pressure drop in water side | Bars | 31,4 | 34,2 |
| Steam speed | m/s | 52,00 | 54,80 |

TABLE I: Synthesis of the main characteristics of the ASTRID Steam Generator

It remains to determine during the AVP2 phase a more precise design of the SG associated to the identification of sensitive points. Furthermore, the points that require additional R&D works and qualification process in specific fields such as material, fabrication rules, thermal-hydraulic, etc, will be defined.

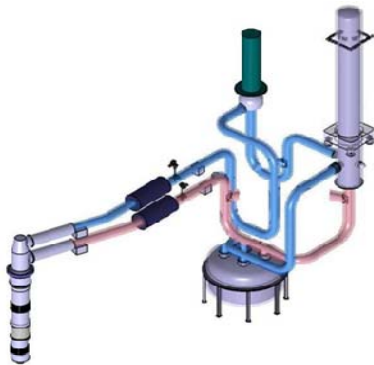


Fig. 9. ASTRID Main Secondary Loop with monolithic SG

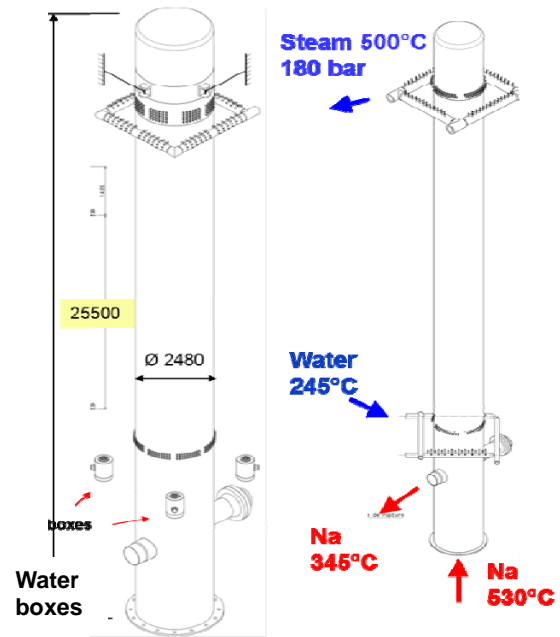


Fig. 10. ASTRID A800 helical tubes monolithic SG

III. CONCLUSION

This paper aimed to explain and justify the choice of ASTRID SG design (a 375 MWth helical tube SG with sodium inlet and outlet located at the component bottom part).

It has been shown that this design selection compared to straight tube steam generator was tricky and was finally justified by a better plant layout, and a reduced height of the SG building.

Moreover, the high tube material resistance option towards wastage effects is an additional argument for the safety approach.

ASTRID basic safety approach was to demonstrate an intrinsic safety demonstration regarding the SWR in a worst case envelope scenario.

This safety demonstration could be obtained by loop design adjustment for a 375 MWth SG leading to a choice of four SGs for ASTRID reactor (one SG per loop) with an account for techno-economic approach.

Furthermore, the points that require additional R&D and qualification process in specific fields such as material, fabrication rules, thermal-hydraulic, etc, will be refined during the coming project phase.

NOMENCLATURE

| | |
|---------|---|
| ASTRID: | Advanced Sodium Technological Reactor for Industrial Demonstration |
| AVP1/2: | Conceptual design studies, phase 1 /2 of ASTRID project |
| A800: | Alloy 800 austenitic steel |
| BN600: | 600 MWe SFR in Russia |
| DEGF: | Double Ended Guillotine Failure |
| DNB: | Departure of Nucleate Boiling |
| IHX: | Intermediate Heat eXchanger |
| ISI&R: | In-Service Inspection & Repair |
| JAEA: | Japan Atomic Energy Agency |
| PFR: | Prototype Fast reactor (<i>UK</i>) |
| PFBR: | Prototype Fast Breeder Reactor (<i>India</i>) |
| RCC-MR: | French code for Design and Manufacturing of SFR components and structures |
| R&D: | Research and Development |
| SFR: | Sodium Fast Reactor |
| SG: | Steam Generator |
| SPX: | Superphenix (<i>French SFR</i>) |
| SWR: | Sodium Water Reaction |
| SWAR: | Sodium Water Air Reaction |
| TNa: | Sodium Temperature |
| TRL: | Technology Readiness Level |
| Twater: | Water Temperature |
| T91: | Grade 91 martensitic-ferritic steel |
| 1D: | One-Dimensional |

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