« ASTRID SAFETY DESIGN: RADIOLOGICAL CONFINEMENT IMPROVEMENTS COMPARED TO PREVIOUS SFRS »

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Abstract. ASTRID is the Advanced Sodium Technological Reactor for Industrial Demonstration which is intended to prepare the Generation IV reactor, with improvements in safety and operability.

In order to meet the objectives of the Generation IV reactors and comply with the related specifications, the ASTRID project integrates innovative options. In the earlier phase of ASTRID project, a specific safety approach was set and its main guidelines were agreed by the French Nuclear Safety Authority.

The basic safety design guide is currently applied as reference for the choices of the design options. The paper presents application of the safety approach, called "Top-down" approach, relating to the "confinement" safety function. The confinement design of ASTRID has several safety objectives from both radiological point of view and sodium chemical risk, and its design is based on "plant state" approach.

As concerns potential radiological risk, main objectives are to postpone a hypothetical off-site release of radiological material coming from core degradation and also to decrease its health and environmental possible consequences.

As concerns the sodium chemical risk, main objectives are to prevent, by design, hazards (e.g., overpressure) on the containment, introducing drawbacks in terms of confinement, and also to cope with the risk of off-site release of soda aerosols with possible health effect.

In order to meet all these objectives, design provisions are taken, considering the different release ways inside the confinement. The paper presents the lessons learned from the previous sodium fast reactor confinement and the method applied to choose consistent design options for ASTRID. The design of the confinement provisions takes into account the lessons learned from Fukushima event, in order to prevent any cliff edge effect in terms of radiological consequences.

Key Words: severe accident - confinement - safety design

1. Introduction

The objective of the Generation IV International Forum (GIF), in which France is involved, is to prepare the future nuclear sector in an international framework by jointly developing the R&D of Generation IV reactors.

In that frame, six reactor technologies have been selected, among which the sodium fast reactor (SFR), supported by the French community because of favourable technical characteristics and a significant and satisfactory industrial experience feedback.

A prototype of Generation IV SFR is under development, its name is ASTRID (Ref. 1): Advanced Sodium Technological Reactor for Industrial Demonstration. Its aim is to demonstrate, at a sufficient scale, technological progress by qualifying the innovative options during its operation, in particular in the fields of safety and operability.

In this frame, the severe accident is specially studied. It corresponds to an entire core meltdown accident. Therefore it is the accident with the most significant radiological inventory released from the core in the confinement. This hypothetical accident is retained according to the fourth level of defense in depth.

A part of the studies aims to improve the prevention of this accident.

Another part, which is the topic of this article, takes into account the occurrence of a severe accident, despite the high level of prevention, in order to demonstrate that the consequences of this accident on the public is limited in space and time. In terms of off-site consequences, goals are: limitation in space of sheltering measures, absence of emergency evacuation beyond the immediate vicinity of the plant, absence of long-term restrictions of food consumption and absence of permanent rehousing.

The design approach consists to limit and delay the potential releases towards the environment by the implementation of adequate provisions.

2. ASTRID overview

ASTRID is a 600 MWe reactor. Its size is sufficient to allow extrapolation to commercial reactors, however without being excessive, in order to limit the cost and the industrial risk.

It is a pool type reactor, sodium-cooled, with an intermediate sodium circuit (Figure 1).

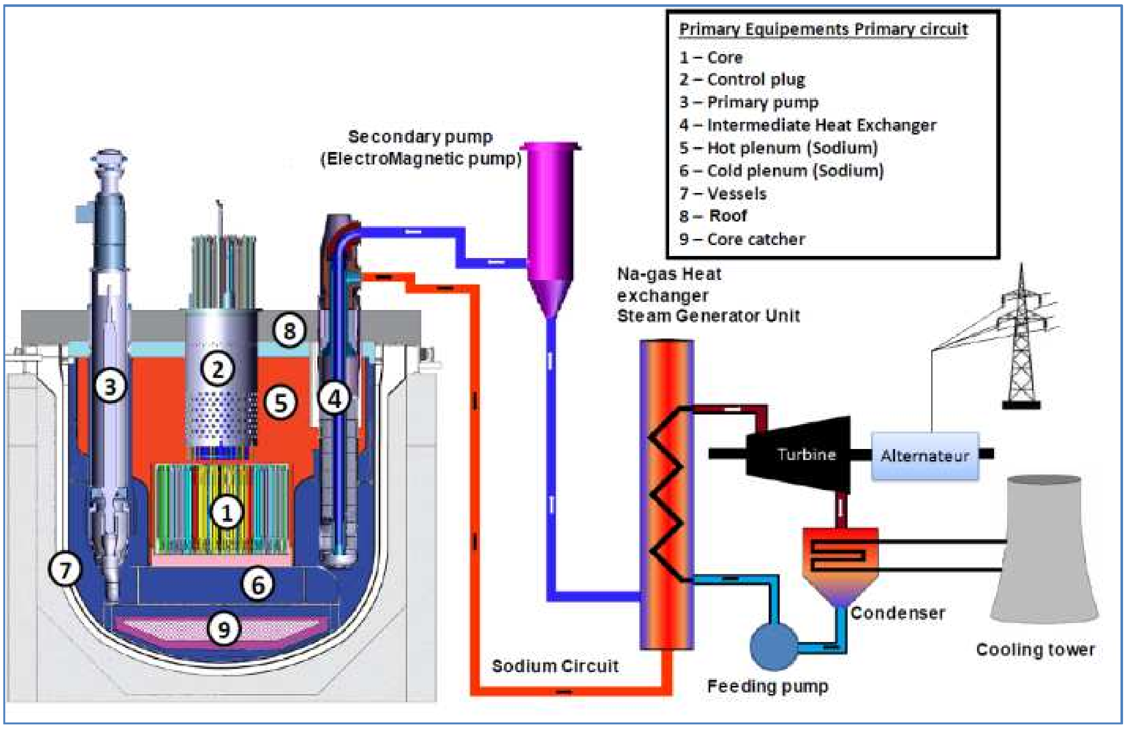

Figure 1: ASTRID general scheme

Mixed oxide fuel $(U,Pu)O_2$ is considered as the reference fuel for the core. An axially heterogeneous core with an upper sodium plenum is employed to achieve a low sodium void effect and limit the potential mechanical energy release in case of severe accident. The primary vessel contains a core catcher whose object is to avoid the damage of confinement system in case of severe accident.

An internal storage is implemented in order to reduce the residual power of the spent fuel assemblies before they leave the reactor.

ASTRID will also have capacities of radioactive waste transmutation in order to demonstrate the feasibility of such transmutation on a significant scale.

3. Severe accident presentation and study methodology

3.1. Severe accident presentation

The term "severe accident" refers to an event causing significant damage to the core and resulting from more or less complete core meltdown. Severe accidents are very unlikely events generally caused by a cooling failure within the core.

A series of complex phenomena then occur, according to various scenarios and depending on the initial conditions of the accident. The accident leads to release fissions products from the core and could lead to a mechanical energy release that may damage the primary circuit and then the confinement features.

3.2. Assessment methodology

3.2.1. The Top-down approach

The Top-down approach aims to minimize the radioactivity released in the environment during an accident, whatever the cause of this accident is. The goal is also to avoid any cliff-edge effect resulting from assumptions and uncertainties in the knowledge of the relevant phenomena being able to occur during the accident. This approach has been developed in order to maximize the robustness of the reactor taking into account the inherent difficulty to provide a well-defined scenario of the accident. This is due to the difficulty to define the sequences of events which cause the severe accident, and also because of the complex phenomena which could occur during the severe accident scenario, in particular due to the potential reactivity effects likely to occur.

In case of severe accident, a significant mass of fission products escapes from the molten core. Then, the release paths are identified and the conception of the reactor is improved in order that the releases in the environment are as low as reasonably achievable. The by-passes are also identified, regardless the reason and the scenario of the accident, in other to strengthen them.

The Top-down approach provides an optimum confinement as less dependent as possible from the possible accidental scenario.

3.2.2. ASTRID release pathway

During the severe accident, the core melts. The fissions products are released in the sodium. The major part of the corium is kept in the primary circuit, in particular thanks to the implementation of the core catcher which avoids any risk of primary circuit failure in the bottom part. A part of mobile fission products as gas or volatile compounds are trapped in sodium, in particular a more or less large fraction of iodine and cesium which have strong chemical affinities with it. The remaining fraction moves to the cover-gas plenum.

The transfer of the mobile radiological inventory in the plant and potentially towards the environment is mainly consecutive:

- to the pressurization of the cover-gas plenum due to:
  - the release of fission gases and volatile isotopes not trapped by sodium;
  - heat produces by the fission products which are in the cover-gas plenum;

- to the possible primary sodium circuit leaks in case of an hypothetical important mechanical energy release, in particular on the level of the main vessel and the roof. These sodium leaks can growth the pressure of the rooms where they take place, especially in case of sodium fire.

The analysis which guides the choice of the confinement provisions is based on the identification of the release ways, by taking into account the risk of degradation of the impacted barriers, in particular for the identification of the risks of confinement by-pass. The experience feedback from the former reactors is integrated into this analysis.

The Figure 2 introduces the main elements for the confinement analysis: the core in the primary vessel, the cover-gas plenum and the connected argon circuit, the roof which closes the cover-gas plenum, the secondary sodium circuits, the above roof area, the polar table which closes the above roof area and the crane hall. A retention room is implemented in the crane hall in order to collect all the releases from the gas circuits before they are released to the stack.

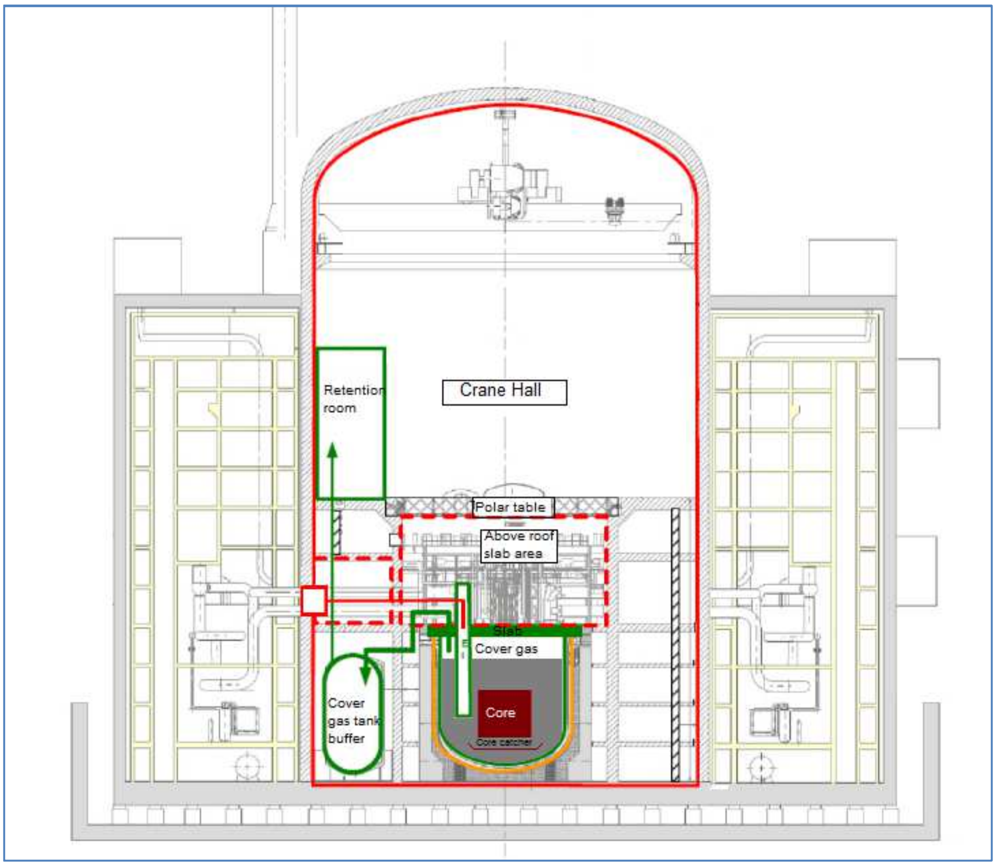

Figure 2: Simplified scheme of ASTRID reactor building

Note that, as the objective is to avoid by design high energetic accidents, the most realistic way of releases is the argon circuit, via its safety valves. Nevertheless, the other potential ways, resulting from the failure of barriers, are considered in accordance with the application of the Top-down approach.

3.2.3. Simulation approach

The evaluation of radiological consequences is done in three steps:

- in the first one, the radiological inventory is evaluated taking into account the neutron flux in the core and the evolution of fuel under irradiation,
- in the second one the transfer of the mobile inventory from the core to the environment is simulated,
- then, the evaluation of radiological consequences on population is estimated, this part depends on the location of the plant (in particular, considering climatic conditions and eating habits).

The remainder of this paper is based on the first two steps, as the results of the second step are sufficient to compare different design option.

4. Evaluation of the mobile radiological inventory

The core radiological inventory is evaluated with assumptions that lead to maximize the mass of fission products in the fuel pins:

- the irradiation ratio taken into account is the higher value,
- the reactor is permanently operated at nominal power,
- the in-vessel fuel storage is assumed to be damaged.

The presence of some subassemblies with high minor actinide content is taken into account in the assessment of radiological inventory.

The fission products are separated in four main groups depending on their chemical properties: noble gases, halogens, alkaline metal, and the other fission products.

During the severe accident, it is considered that a fraction of fission products in each group is released from the fuel to the cover-gas plenum through sodium. These fractions depend on their solubility in sodium and chemical interaction with other products (sodium and other fission products, Ref. 2).

There are two kinetics of fission products release from the core during severe accident:

- instantaneous release when the accident occurs which correspond to ejection of a fraction of fission products at the surface of primary sodium,
- differed release from sodium by vaporization of some fission products from the surface of primary sodium or though sodium aerosol production.

The chemically inert gases, in particular noble gases, are instantaneously transferred from the core to the cover-gas plenum.

In the fuel, in normal operation, iodine combines with cesium to form a stable composite (CsI). If the fuel melts, CsI dissolves quickly in sodium to form CsI and NaI. They are very chemically stable: only a limited fraction could be transferred towards the cover-gas plenum.

It is similar for other alkaline metals, in particular for rubidium that is very soluble in sodium.

Note that uranium and plutonium oxides are not significantly soluble in sodium. For the most part, they are kept inside the primary vessel. However, in case of an energetic accident, fuel particles can be entrained by sodium vapor flow in the cover gas area.

Another feature of ASTRID concept is the in-vessel spent fuel storage. This storage is around the core, separated from the core by neutron reflector assemblies and neutron shielding assemblies. It can contain about half of a core. During the severe accident, it is considered that the in-vessel spent fuel storage is sufficiently separated from the molten core that it doesn't melt itself. Nevertheless, it is penalizing assumed that during the accident all the fuel clads of the spent fuel subassemblies in the in-vessel storage fail and that the fission gas and volatiles are released.

For example, more than 400 kg of xenon (all isotopes included) are released in the cover-gas plenum during the accident. When arriving in the cover-gas plenum, the gases are supposed to be at the sodium temperature (about 550 °C).

When the mobile radiological inventory is in the cover-gas plenum, it may be transferred in the plant by different ways:

- the argon circuit,
- through the roof if it is damaged,
- through the primary vessel if it is damaged, then in the inter-vessel plenum and its connected nitrogen circuit,
- the other circuits connected to the primary circuit if they are damaged.

5. Potential release way analysis

5.1. Argon circuit

The argon circuit has in particular as function to limit the variations of cover-gas plenum pressure under normal operations. It is thus a circuit made up of several high volume tanks. It is provided with safety valves for limiting the maximum pressure in the cover-gas plenum. In normal operation the cover gas circuit can be connected to the stack for discharging purified argon, in particular during the start-up procedure.

The argon circuit of ASTRID is designed taking into account the feedback from other French SFR, in particular SuperPhénix. Three weaknesses were identified on the SuperPhénix design: the argon circuit by-passed the primary containment, the argon circuit releases were collected in a retention room which couldn't be isolated and the safety valves by-passed the tanks of the argon circuit.

For ASTRID, the confinement measures implemented on this circuit are then the following ones:

- to recover in a retention room the contaminated gas from the argon circuit and possibly released by its safety valves; the retention room is not fully leak-tight, leakages from the retention room towards the crane hall can occur;
- to allow the isolation of the retention room; this is a major improvement compared to the SuperPhénix design;
- to locate the whole argon circuit up to its isolation valves and the retention room inside the confinement system.

The maximal pressure in the argon circuit is reached if the roof is not damaged. All the fission products released from the core and the primary circuit pass through the argon circuit.

The evolution of the relative pressure in the argon circuit is presented Figure 3. Assuming an instantaneous release (penalizing assumption), the pressure peak in the cover-gas plenum is fast and is about 1.4 10^5 Pa (relative pressure). In the other volumes of the argon circuit, the pressure is significantly lower (about 0.5 10^5 Pa).

The pressure in the retention room, which is a big volume, doesn't increase significantly. The leaks from the retention room to the crane hall are therefore reduced.

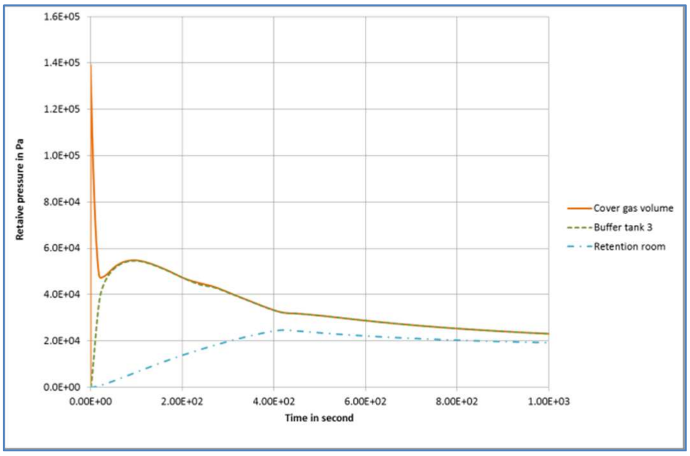

Figure 3: Relative pressure in function of time in the argon circuit during the first minutes of the accident

This design of the cover-gas circuit allows the limitation of the pressure loadings and therefore the risks of leakage. Moreover, any hypothetical leaks would be inside the confinement system without risk of by-pass. The risk of confinement by-pass is thus focused on the isolation of the retention room.

5.2. Roof

Under normal operation, the roof ensures the leak-tightness of the cover-gas circuit in particular at the level of its penetrations which are provided with specific devices.

The severe accident could lead to a mechanical energy release likely to damage the sealing of the roof. Despite the design option aiming to limit the mechanical energy release, the consequences of this hypothetical event are assessed in order to identify possible cliff-edge effects (Top-down approach). The design of the confinement system takes into account the possibility of a leak between the cover-gas plenum and the above roof area. This leak could lead to a release of gas towards the above roof area and to a primary sodium ejection which can burn in contact with air.

The selected confinement provisions have two objectives:

- to prevent that the pressure increase due to the sodium fire does not damage the confinement system;
- to create a room confining the leakages through the roof.

To this end, the above roof area is contained in a volume as small as possible and distinct from the confinement system.

In case of roof leakage, a part of the fission products is released in the above roof area, then in the confinement system. The main confinement measure at this level is the isolation of the confinement system. This is also a major difference with the SuperPhénix options where the secondary containment was not isolated in case of severe accident. Only the primary containment was isolated.

A limited leakage of the confinement system is considered.

The scheme of the model including the leakage through the roof is presented Figure 4.

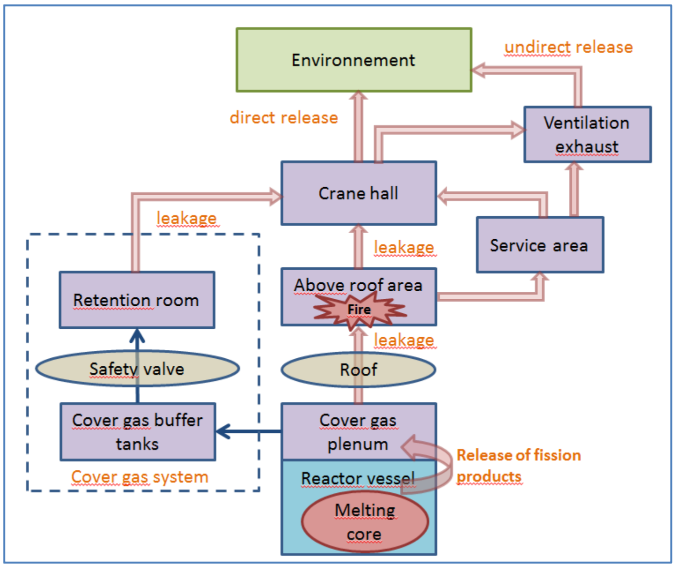

Figure 4: Scheme for severe accident simulation included roof leakage.

The evaluation of the leakages toward the roof depends on different assumptions: the size of the leak compared to the argon circuit pipes diameter and the back pressure in the above roof area due to primary sodium fire.

According to the simulation, with a large leakage through the roof and a significant fire in the above roof area, the gas pathway during the simulation can be divided in three principal phases.

In a short first phase (about 10 s), the fission products principally go from the cover-gas plenum trough the above roof area.

In a second phase (from 10 to about 25 s), the sodium fire in the above roof area leads to improve significantly the pressure in this volume: the pressure therefore becomes greater than in the cover-gas plenum. The flow rate through the roof is then stopped and the cover-gas plenum is depressurized only in the argon circuit. The duration of this phase depends on the mass of sodium which burns into the above roof area. The greater is this amount of sodium, the longer is the fire duration. So the amount of fission products in the above roof area depends on the back pressure due to the sodium fire.

The third phase begins when the fire ends. The pressure in the cover-gas plenum becomes then greater than in the above roof area and the flow rate through the roof starts again until the pressure is equilibrated.

The pressure in the above roof area and in the crane hall is presented Figure 5.

The maximum pressure is obtained in the above roof area in the first seconds and is due to the sodium fire. The conception of this volume leads to limit the pressure in the crane hall: the maximum pressure in the crane hall is quite low (about 5.10^3 Pa) which contributes to limit the releases in the environment.

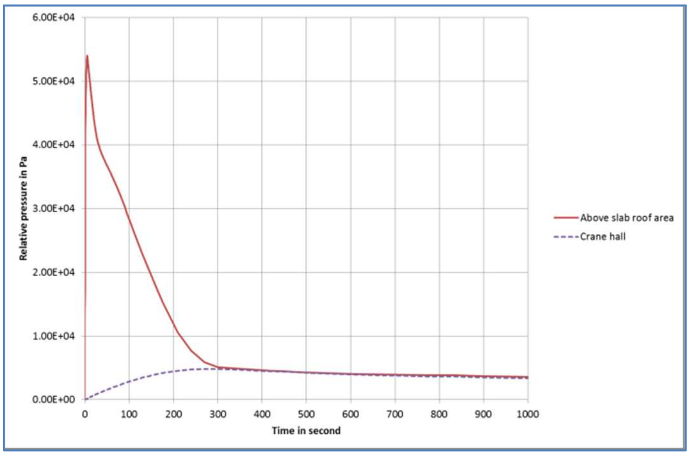

Figure 5: Relative pressure in function of time in the above roof area and in the crane hall during the first minutes of the accident

5.3. Primary vessel

Because of a hypothetical mechanical energy release during the accident, the leakage of the main vessel could occur. It is surrounded by a safety vessel which is far enough from the primary vessel so that it isn't damaged during the accident.

The volume between main and safety vessels is filled with nitrogen. The nitrogen circuit has in particular the function to maintain the pressure between the primary vessel and the safety vessel in an acceptable value, in particular thanks to a safety valve connected to the cover-gas circuit. Other safety valves are implemented and connected to the retention room.

The main confinement measure is to isolate the nitrogen circuit to limit the contamination coming from the vessel or the cover-gas plenum (gas or sodium leaks or opening of the safety valve).

As for the cover-gas circuit, the whole nitrogen circuit up to its isolation valves is located inside the confinement system what allows to limit the risk of confinement by-pass.

5.4. Circuits connected to the primary circuit

The fluid circuits (except sodium circuits) connected to the primary circuit are isolated. This concerns in particular:

- Gas circuits ensuring the leak-tightness of the roof penetrations;
- Roof cooling system: it is an air circuit with a heat sink which is located in the reactor building.

The secondary sodium circuits are designed to remain leak-tight even in case of a hypothetical mechanical energy release. Nevertheless, leaks of the heat exchangers between primary sodium circuit and secondary sodium are postulated because they are equipment impacted by the accident. In this case, the confinement is ensured by the part of the secondary circuit which is not affected. However, provisions to manage these possible leaks are under investigation (e.g., draining and isolation of the damaged circuits).

One option being considered is to purify the primary sodium with a system outside the vessel. In this case, the purification system has a double jacket and is located in the reactor building.

Concerning the other sodium circuits which are not entirely located in the reactor building (circuits for extraction of the power and for decay heat removal), the approach consists in being ensured by design that they are not damaged by the hypothetical mechanical energy release and to study provisions to manage the possible leaks through the most impacted areas.

5.5. Confinement system

In order to avoid the risks of by-pass, the confinement system includes the whole ways of releases up to the isolation devices, as well as the primary circuit.

The confinement system is carried by the reactor building. The risks of by-pass are limited by:

- the isolation of reactor building ventilation;
- the implementations of leak-tight devices at the singular points of the building in particular the secondary sodium loop penetrations and the building openings. Moreover, these singular points are located at lower part of the building, in front of adjacent buildings.

The selected design provisions ensure that the severe accident does not lead to a significant pressure and temperature increase inside the building. This one is designed to resist to loadings more important than those induced by the severe accident (earthquake, aircraft crash…).

The pressure inside the reactor building depends on its volume, its leakage rate and its exchange area with the above roof area. The current design leads to have a maximal overpressure in the reactor building of about 150 mbar.

Finally, the design of the ASTRID confinement includes:

- containment of the radiological potential allowing to limit the transfer of the radioelements in the plant and to profit from delay effects;
- the set up of a confinement system which contains all the potential release ways of radioactive materials and which recovers the possible leaks of the primary circuit.

A special attention is given to the singular points in order to avoid any risk of by-pass through the confinement system.

6. Conclusions

In the earlier phase of ASTRID project, a specific safety approach was set and its main guidelines were agreed by the French Nuclear Safety Authority. This safety design guide is currently applied for the choices of the design options.

The goal of this approach and the selected provisions is to enhance the application of the defense in depth principle.

The lessons learned from the previous SFR design, operations and safety assessment, are taken into account to correct the identified weak points in term of both methodology and design. In particular, the design take care to limit the by-pass risks and to avoid direct release.

This paper presents the simulation approach in order to evaluate the radiological consequences of a severe accident and the main hypothesis.

An analysis of the potential release ways leads to an evaluation of the pressure in the principle volumes of the installation.

The pressure peak in the cover-gas plenum is fast and is about $1.4\ 10^5$ Pa. However, as a result of the cover-gas circuit design, the pressure in the retention room doesn't increase significantly. The retention room conception reduces and delays the release.

In case of an energetic accident, the above roof area pressure could be important, as a result of a fire due to sodium ejection from the vessel. Nevertheless, the design of the above roof area leads to limit the pressure in the crane hall, and then limit the releases to the environment.

The analysis of the different leakage way leads to conclude that the current design minimizes the release in the environment. Such studies will be continued, especially in case of major modification of the design.

Appendix 1: References

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