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THE OSCAR CODE PACKAGE : A UNIQUE TOOL FOR SIMULATING PWR CONTAMINATION

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ABSTRACT

Understanding the PWR primary circuit contamination by corrosion products, fission products and actinides is a crucial issue for reactor operation and design. The main challenges are decreasing the impact on personnel exposure to radiation, optimizing the plant operation, limiting the activity of the wastes produced during the reactor lifetime and preparing decommissioning.

In cooperation with EDF and AREVA NP, CEA has developed the OSCAR code package, a unique tool for simulating PWR contamination. The OSCAR package results from the merging of two codes, which simulate PWR contamination by fission products and actinides (PROFIP code) and by activated corrosion products (PACTOLE code).

These two codes have been validated separately against an extensive set of data obtained over 40 years from in-situ gamma spectrometry measurements, sampling and analysing campaigns of primary coolant, as well as experiments in test loops or experimental reactors, which are representative of PWR conditions.

In this paper, a new step is presented with the OSCAR code package, combining the features of the two codes and motivated by the fact that, wherever they originate from, the contamination products are subject to the same severe conditions (300 °C, 150 bar, neutron flux, water velocity up to 15 m.s⁻¹) and follow the same transport mechanisms in the primary circuit. The main

processes involved are erosion/deposition, dissolution/precipitation, adsorption/desorption, convection, purification, neutron activation, radioactive decrease.

The V1.1 version of the OSCAR package is qualified for fission products (Xe, Kr, I, Sr), actinides (U, Np, Pu, Am, Cm) and corrosion products (Ni, Fe, Co, Cr).

This paper presents the different release modes (defective fuel rod release, fissile material dissemination, material corrosion and release), then the processes which govern contamination transfer, and finally, we give examples of the comparison of the OSCAR package results with measurements in French PWR primary circuit obtained for representative radioisotopes : ^{133}Xe , ^{90}Sr , ^{58}Co , ^{60}Co .

In particular, we focus on the main upgrades in the OSCAR simulations compared to the PROFIP and PACTOLE codes : adaptation of the MARGARET module to assess fission product release out of fuel pellets in a defective rod, adsorption/desorption model development for strontium behaviour, multi-criteria calibration of input data which are not well known for corrosion product simulation.

1. INTRODUCTION

The assessment of the radio contamination of PWR primary and auxiliary circuits is a crucial issue for power plant operators and suppliers. It is necessary to fulfill some important challenges while operating or designing a nuclear power plant : decreasing the impact on personnel exposure to radiation, optimizing the plant operation, limiting the activity of the wastes produced during the reactor lifetime and preparing decommissioning.

The radio contamination of PWR primary circuits comes from corrosion products, fission products and actinides. Their origin is different but they are subject to the same severe conditions (300 °C, 150 bar, neutron flux, water velocity up to 15 m.s⁻¹) and follow the same transport mechanisms in the primary circuit. The main processes involved are erosion/deposition, dissolution/precipitation, adsorption/desorption, convection, purification, neutron activation, radioactive decrease.

Moreover during a reactor cycle, several kilograms of corrosion products (mainly nickel and iron) are released in the primary coolant from the base metal and represent most of the contamination mass deposited on the primary surfaces. So their behavior definitely determines the evolution of deposited radioisotopes : radioactive corrosion products of course, but also actinides.

Therefore, the two existing French contamination codes (PACTOLE [1] for corrosion products, and PROFIP [2] for fission products and actinides) have been merged in a unique tool : the OSCAR code package, dedicated to the PWR primary and auxiliary circuit contamination. This paper describes the features of this new code package, and first results of validation against experimental data.

After a general overview of the new features of the code in section 2, the source terms in the primary circuit are described in section 3 for the different radioisotopes considered. Section 4 lists the main mechanisms involved in the transport of the radioisotopes and the contamination of the primary circuit. Finally in section 5, the first validation of the code against experimental data is shown for the main isotopes.

2. GENERAL OVERVIEW

The OSCAR code has been developed within the framework of R&D cooperation between the French Atomic and Alternative Energy Commission (CEA), Electricité de France (EdF) and AREVA NP.

The OSCAR code keeps the fully object-oriented design of PACTOLE V3 and its modular structure, as well as its discretisation into control volumes. A user friendly interface for input data and visualisation of results has been developed. In the version 1.1 of the OSCAR code, the calculation for fission products and actinides on the one hand, and corrosion products on the other hand, are still independent though they can follow the same mechanisms, as for neutron activation and radioactive decay.

Concerning the transfer of corrosion products, the PACTOLE V3 modeling [1] is retained, with an improved thermodynamic database. Unknown input parameters have been assessed using a multi-criteria calibration based on a comparison with experimental data (see section 5).

Compared to the PROFIP code, the main change in the modeling of the transfer of fission products and actinides is the use of the fuel code ALCYONE and the fission gas behavior module MARGARET [3], to assess the fission product source term in the primary circuit, released from a defective rod. Mechanisms involved in the primary circuit behavior are the same as for the PROFIP code. However, a model for adsorption/desorption has been added to describe the strontium behavior.

3. THE SOURCE TERMS

In this section, the source terms taken into account in the OSCAR V1.1 code are presented for the different types of contamination products : fission products and actinides released from defective fuel rods, fission products emitted by recoil from actinides deposited on surfaces under neutron flux (see fig. 1), and corrosion products released from the base metal.

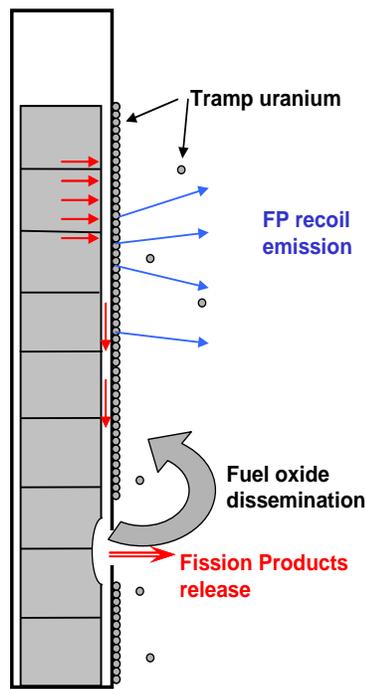


Figure 1. Fission products and actinides release

3.1 Fission Products released from defective fuel rods

In case of rod cladding defect, fission products are released in the primary circuit from the fuel rod. Three stages can be distinguished in this process : release out of the rod pellets, transfer through the fuel-cladding gap and release through the defect.

3.1.1 Release out of rod pellets

Because of the presence of water or vapor inside the defective rod, oxidation of the fuel matrix occurs, which strongly enhances the release of fission products out of the rod pellet compared to a tight fuel rod. Modeling of this phenomenon should rely on defective fuel characterization but given the lack of such experiments, the strategy is to adapt the MARGARET module developed and validated for tight fuel.

MARGARET [3] is an advanced mechanistic model of fission gas behavior in PWR fuel. It is developed by CEA within the

framework of the PLEIADES [4] platform and its fuel behaviour application ALCYONE, which supplies the thermo-mechanical data used by MARGARET.

The MARGARET module, which initially dealt only with stable gas, has first been adapted to the radioactive fission products, with validation against the experimental irradiation campaigns HATAC and CONTACT (fission product release measurements from irradiated tight fuel rods). It has been shown that the bubble trapping of fission products inside the fuel grain could be neglected for fission products with radioactive half-life shorter than 8 days. These short life isotopes are really the isotopes of interest because they represent most of the primary fluid activity. Moreover, they are used in the criteria defined in the French PWR operating specifications, allowing or not the operation to be continued in the presence of cladding defects. These isotopes of interest are : ^{133}Xe , $^{133\text{m}}\text{Xe}$, ^{135}Xe , ^{138}Xe , $^{85\text{m}}\text{Kr}$, ^{87}Kr , ^{88}Kr , ^{131}I , ^{132}I , ^{133}I , ^{134}I , ^{135}I .

Thus, as intragranular bubbles do not interact with short life fission products, the intrinsic diffusion coefficient of stable gas in the fuel oxide crystal is used for these isotopes. Like the stable gas, radioactive fission products are released from the grain by diffusion, either towards grain boundaries or towards free surfaces, that means grain surfaces connected with free volumes (fracture or porosity or fuel-cladding gap). In the former case, they may migrate towards the free volumes by percolation, when the bubbles at the grain boundaries interconnect. The free surfaces fraction is simulated by the MARGARET module, and is around 1 or 2 % in a tight fuel configuration.

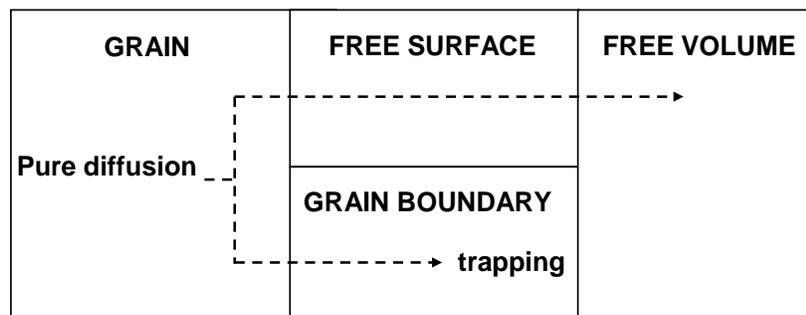


Figure 2. Radioactive fission product release out of fuel oxide grain in defective fuel rod configuration.

Considering the defective fuel rod situation, experimental data show that the fuel burnup has no impact on the radioactive fission products release from the fuel pellet. Hence the mechanisms which induce an enhancement of the release of stable gas with the burnup are not considered for defective fuel. In particular, the percolation mechanism is not

taken into account. This implies that in this situation, radioactive fission products at grain boundaries are never released in free volumes and that the release from the fuel pellet into the free volume can be expressed as the result of pure diffusion in the fuel grain, towards free surfaces (cf. figure 2). Fission product concentration at the free surface is considered to be zero.

The released fraction in free volumes can then be expressed as :

$$\frac{R}{B} = F_{free} \cdot \frac{3}{a} \cdot \sqrt{\frac{D}{\lambda}}$$

R: release rate from the grain into free volumes

B : birth rate in the grain

F_{free} : fraction of grain surfaces connected with free volumes

a : radius of the grain

D : diffusion coefficient of the fission products in the grain

λ : radioactive decay constant of the fission product.

For defective rods, oxidation of the fuel matrix leads to released fractions much higher (by a factor of roughly 30) than for non defective fuel rods. This impact from oxidation is taken into account in the simulation by the following adjustments :

- Fraction of free surface F_{free} : it is fixed to 100%, which prevents percolation from playing any role.
- Diffusion coefficient inside the grain D : a correction factor is applied to the intrinsic stable gas diffusion coefficient used for tight fuel.
- Fuel temperature : besides the fact that for a defective configuration, water vapor replaces helium in free volumes of the rod and that the internal pressure of the rod is balanced with external pressure (no further clad creeping), a degradation of the thermal conductivity of the fuel is taken into account.

The calibration of the diffusion coefficient and of the thermal conductivity degradation is performed against a unique experimental database including defective PWR rods well identified and characterized, as well as analytical irradiations in the SILOE reactor. The release rates are deduced from primary activity measurements in fission products which are linked with the release from the fuel matrix and not affected by the transfer in the fuel-cladding gap and through the defect. These fission products are ^{133}Xe (if the burnup is low enough to allow an open gap) and ^{131}I (the release is correctly assessed from transient measurements). The fission products with a shorter life time are considered to be too strongly impacted by their transfer in the gap.

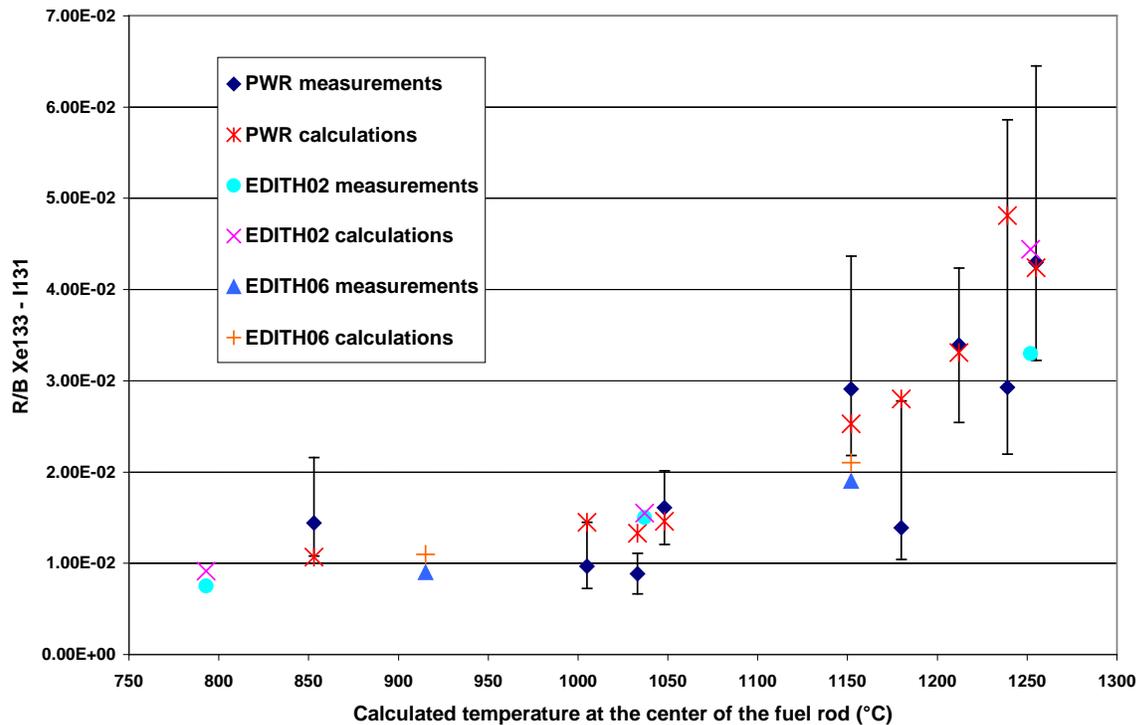


Figure 3. Comparison between experimental and calculated released fractions for UO_2 defective fuel.

Figure 3 gives a comparison between measurements and OSCAR calculations after calibration, concerning the released fractions from defective UO_2 fuel rods of the database (including EDITH 02 and EDITH 06 analytical irradiations, respectively for low and high burnup UO_2 rods). The validation domain is up to 50 GWd/t. We note a relatively good agreement between experimental and calculated released fractions. Such a fair agreement (not shown here) is also found for analytical irradiations EDITHMOX 01 (low burnup MOX rod) and EDITHMOX 02 (50 GWd/t MOX rod), which allows to widen the validation domain to MOX fuel, for the same burnup range.

3.1.2 Knockout mechanism

To fit the experimental release of short life products at low temperature, it is necessary to take into account the knockout mechanism at the grain free surface [5]. That means ejection of an atom struck by a fission fragment or another atom of the lattice, due to elastic collisions. The important parameter for this mechanism is the thickness of the layer where knockout can lead to the ejection of fission products out of the grain. This thickness is assessed about 5 nm.

The knockout flux can be expressed as an additional sink term in the knockout layer to the diffusion term. This flux is proportional to the fission rate and to the fission product concentration in the layer.

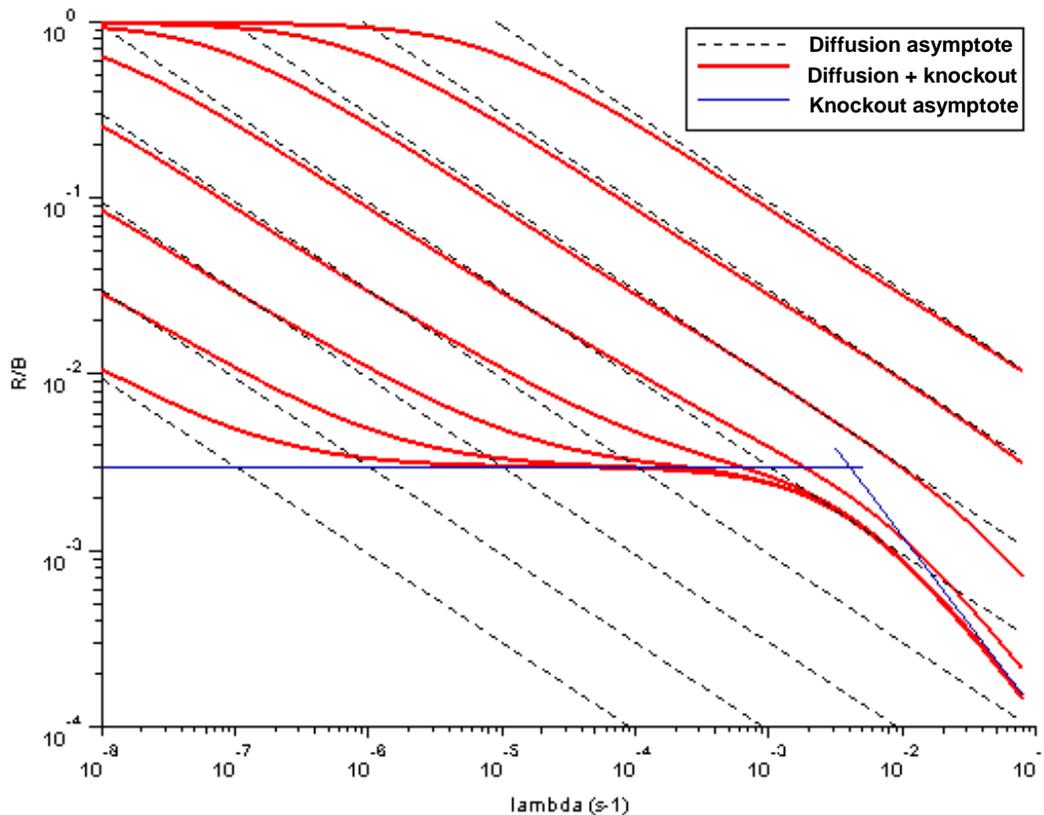


Figure 4. Effect of the knockout model.

Figure 4 illustrates the impact of this mechanism, showing the variation of the release rate for different values of the diffusion coefficient. The dotted lines represent the asymptotes for the pure diffusion mechanism and the red lines the diffusion + knockout mechanisms. As seen in this figure, knock out does not affect significantly released fraction for ^{133}Xe ($\lambda = 1.53 \cdot 10^{-6} \text{ s}^{-1}$) and ^{131}I ($\lambda = 1.00 \cdot 10^{-6} \text{ s}^{-1}$), which are used for defective fuel calibration but strongly affects shorter life time fission products, particularly at low fuel temperature.

3.1.3 Release out of the rod

Release out of the rod represents the main source term for short life fission products in the primary circuit.

For the noble gases of interest, as long as the gap is not closed, the transit in the fuel-cladding gap is fast and the release out of the rod corresponds to the release out of the fuel matrix (unless the defect size is very low).

Considering iodine isotopes, the release mechanisms are not well understood. During nominal reactor operation, a trapping occurs in free volumes, which makes the release rate out of the rod very difficult to assess. This trapping phenomenon is taken into account in the OSCAR code through a trapping coefficient. However, during power transients at low level, water enters the

free volumes and leads to the release of almost the total amount of iodine present in free volumes. The OSCAR code can be used to assess the iodine amount which is trapped in free volumes and released during a power transient. However, it can not reproduce the kinetic of the iodine release which is observed during the transient.

3.2 Fuel oxide dissemination

In the event of a significant cladding damage, fuel oxide may disseminate into a PWR primary circuit. The disseminated fuel oxide is also called “tramp uranium”. No phenomenological modeling is linked with this dissemination. However experimentally, the dissemination rate can be assessed from the ^{134}I primary volume activity measurement, which is a tracer of fissile material deposited under neutron flux (cf. § 3.3).

Thus, the rate and duration of the fuel oxide dissemination are fixed in the data set of the OSCAR code, in order to reproduce the experimental ^{134}I activity level during the dissemination reactor cycle. The isotopic composition of fuel oxide is the same as the one at the fuel pellet surface, and is calculated by the ALCYONE code.

The fuel oxide dissemination is the source term for actinides in the primary circuit. It is also a significant contribution to the source term of long life fission products (such as ^{90}Sr), which are included in the fuel oxide.

3.3 Fission Products emitted by recoil from the tramp uranium

When deposited under neutron flux, tramp uranium generates fission products, which are released in the primary circuit by the recoil mechanism.

Unlike the diffusion mechanism which is involved in the fission product release out of a defective rod, the recoil mechanism is immediate, which facilitates the release of short life fission products such as ^{134}I (radioactive half-life : 54 min).

Recoil from the tramp uranium is another source term for the noble gas and iodine isotopes listed in § 3.1. This is the main source term for fission products which are not released from the fuel oxide inside the rod (such as ^{90}Sr) or which are trapped in free volumes (such as iodine).

A reliable assessment of this source term requires a correct calculation of the tramp uranium evolution in the primary circuit, which is mainly due to the neutron flux and to the transfer through erosion/deposition mechanisms between the different surfaces. Then the release by recoil is easily obtained by means of fission cross sections, which are specially assessed for out of pellet fissile material.

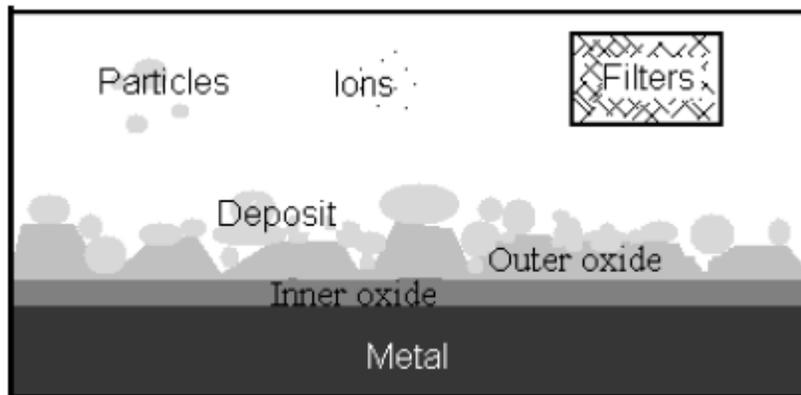


Figure 5. Media taken into account in PACTOLE code modeling [1].

3.4 Corrosion Products

An empirical approach for corrosion and ion release processes has been chosen, as for the PACTOLE code modeling [1, 6] (see figure 5). Uniform and generalized corrosion of the base metal (mainly stainless steels and alloys containing nickel or iron) leads to the formation of an oxide layer (with a given corrosion rate) and to the direct release of ions in the primary coolant (with a given

release rate). In fact, the oxide layer is composed of an inner layer, which limits the release of ions in the primary coolant but without preventing it completely, and of an outer layer, which may dissolve in the primary fluid, and so contribute to the source term.

The corrosion rate and the ion release rate are determined on the basis of test loop experiments [7]. They depend on the physico-chemical conditions in the primary circuit (pH, temperature, boron and hydrogen concentrations) and on the components (surface treatment, manufacturing process, exposure time in the circuit). Moreover an attenuation factor is applied on the release rate when the equilibrium concentration is approached in the primary fluid. This is done in order to take into account the fact that the release of corrosion products is mainly a dissolution mechanism, which occurs only when the solubility limit is not reached.

In the model, the release of corrosion products in the primary circuit corresponds to several kilograms of iron and nickel, which determine the behavior of the deposited contamination, including actinides (a few grams) and insoluble fission products (a few milligrams).

The released corrosion products are transported by the coolant to the core where they can deposit on fuel assemblies. Such corrosion product deposits can be activated and subsequently set back into the coolant where they can redeposit on out-of-core surfaces and raise radiation fields. Moreover, in the case of surfaces under neutron flux, the activation mechanism (see § 4.1) generates radioactive corrosion products in the base metal, which are directly released in the primary circuit through the release process.

4. MECHANISMS WHICH GOVERN THE CONTAMINATION BEHAVIOR

This section presents the main physico-chemical mechanisms involved in the contamination behavior in the primary circuit for the OSCAR V1.1 version. They generate transfers between the following media : ions, particles and deposit. For each of these media (as well as for metal and oxides), for each control volume and for each isotope considered, a mass balance is calculated.

Together with the source terms, these mechanisms are summarized in figure 6 for actinides and fission products and in figure 7 for corrosion products.

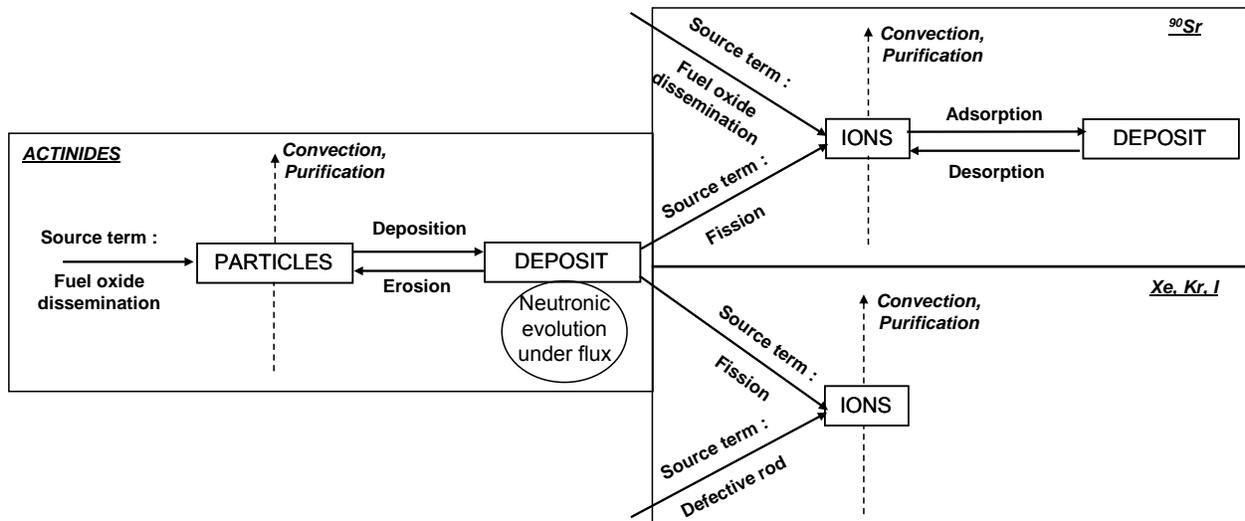


Figure 6. Main mechanisms involved in the actinides and fission products contamination.

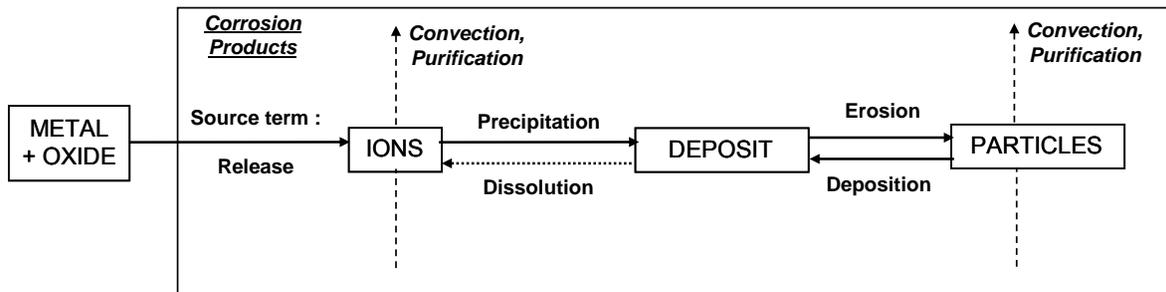


Figure 7. Main mechanisms involved in the corrosion products contamination.

Besides the mechanisms described below, the OSCAR code considers the convection, which transports ions and particles through the primary circuit, and the purification, which removes ions and particles from the circuit with a given efficiency when they go through the filter or demineralizer of the purification circuit.

4.1 Neutron activation, fission, radioactive decay

The considered isotopes are described within filiation chains, with all required data to define the reaction between two isotopes : radioactive decay constant, branching ratio, reaction nature, cross section and fission yield if necessary.

Radioactive decay and neutron capture reactions (including fission reactions) occur either inside the materials under flux, generating radioactive isotopes inside the base metal (see § 3.4), or in the deposited contamination or in the fluid. For each capture reaction (between two isotopes) and for each location (surface or fluid), reaction rate libraries (for corrosion products) or one energy group cross section libraries (for fission products and actinides) are used, obtained from APOLLO calculations and JEF 2.2 nuclear database. In these libraries the values depend on the reactor type, the fuel nature (UO_2 or MOX), the initial enrichment and the burnup.

For example, these mechanisms allow us to simulate the tramp uranium evolution when deposited under neutron flux, as well as the fission products creation from the fissile nuclides deposited under flux or the ^{58}Co creation inside the nickel deposited on the core.

4.2 Dissolution/precipitation

Precipitation or dissolution takes place, depending on whether the volume concentration of one element is higher or lower than the equilibrium concentration respectively in the considered control volume. The equilibrium concentration is calculated using the PHREEQCEA code (a modified version of PHREEQC [8]), in combination with a thermodynamic database developed by CEA. It depends on the coolant temperature in the control volume as well as boric acid, lithium, hydrogen and oxygen concentrations, but also on the composition of the outer oxide and the deposit. Indeed these two media, as well as the ions and particles, are considered for the thermodynamic equilibrium calculation.

Dissolution and precipitation induce transfers between deposit and ions in the control volume. Dissolution kinetic is calibrated from the experimental data (see § 5.4). As for the kinetic of transfer to the wall due to the precipitation process, it is supposed to be limited only by the mass transfer coefficient between the fluid and the wall, and depends on the thermo-hydraulic conditions. It is much higher than the dissolution kinetic.

In the OSCAR 1.1 version, these two mechanisms concern only corrosion products. For example, they allow us to simulate that the nickel element, which is mainly released from the InconelTM steam generators, precipitates in the hot areas of the core. Indeed the nickel solubility calculated using the CEA thermodynamic database is lower at the core exit temperature than at the entrance temperature.

Precipitation of ions on particles is also taken into account, but does not seem to have a significant influence on the contamination behaviour with OSCAR V1.1 and is not represented in figure 7.

4.3 Erosion/deposition

Erosion and deposition mechanisms concern both corrosion products and actinides.

The modeling assumes that the deposit erosion occurs only when the deposit thickness has reached a limit thickness, which depends on the thermohydraulic conditions and on the surface roughness. This mechanism generates particles in the primary fluid, with the same composition as for the deposit.

The erosion rate is calibrated from experimental data separately for actinides on the one hand and for corrosion products on the other hand. Moreover, the limit thickness considered for actinides is supposed to be zero, so that erosion occurs as soon as actinides are deposited. On the contrary for corrosion products, only surfaces for which the deposit has reached the limit thickness are concerned by the erosion mechanism.

The deposition model induces a particle transfer to the deposit. The corrosion product deposition rate is supposed to be mainly due to the turbulent diffusion and the effects of inertia [9] and depends on the thermohydraulic conditions in the control volume, and on the particle size. A log normal distribution centred on 0.5 μm is considered for the particle size distribution. Concerning the actinides deposition rate, it is calibrated from the experimental data.

For example, the nickel deposit (due to precipitation) on the top part of the core reaches the limit thickness and the erosion process generates particles rich in ^{58}Co (nickel activation product). Then these particles contaminate (through the deposition process) the out-of-core surfaces.

The erosion/deposition process is also responsible for actinides transfer between steam generators and the core, generating recoil emission of fission products, even after the contaminated core has been discharged.

4.4 Adsorption/desorption

In the primary circuit conditions, strontium is found as Sr^{2+} ion in the fluid. However important amounts are present on the primary surfaces, which can be explained by the adsorption process. Therefore, adsorption and desorption mechanisms have been introduced in the OSCAR code to simulate the ^{90}Sr behavior in the primary circuit (^{90}Sr radioactive half-life : 29 years).

The mechanism is determined by the equilibrium ratio between the activity in the fluid and the activity on the surfaces, which depends on control volume temperature and pH. During nominal reactor operation, calculated surface and volume activity of ^{90}Sr reach an equilibrium : the volume activity depends only on the source term and the purification term, as surface activity is given by the equilibrium ratio.

During an end-of-cycle shutdown, the decrease of pH and temperature leads to a significant decrease of the modeled equilibrium ratio, and then to a strong release of ^{90}Sr in the primary fluid.

5. EXPERIMENTAL VALIDATION FOR SPECIES HANDLED BY OSCAR

Considering the source terms and mechanisms described in sections 3 and 4, OSCAR is able to simulate the behaviour of four groups of contaminant species : actinides, gaseous and volatiles fission products, ^{90}Sr and lastly corrosion products. As it appears in figure 6, in case of the presence of tramp uranium consecutive to fuel oxide dissemination, actinide contamination is a source term for fission products and must be calculated prior to fission product contamination.

In the OSCAR 1.1 version, actinides and fission products simulation is still independent from corrosion products simulation.

We give here some examples of comparison with measurements, obtained from primary fluid analyses or measurements campaigns with EMECC device, a special gamma spectrometry device allowing surface activity measurements [10].

5.1 Xe, Kr, I

Gaseous and volatile fission products are totally soluble and do not interact with the primary walls in the nominal operating conditions. The Henry's law coefficient is used to evaluate the partition between liquid and gaseous phases, and to calculate the activity in the gaseous phases of the purification tank and of the pressurizer.

In figure 8, we show a comparison between measurements and OSCAR calculations for ^{133}Xe primary volume activity during a PWR cycle with occurrence of a cladding defect on a first cycle rod. A fair agreement between measurements and calculations is found.

The main uncertainty for fission products calculation is the source term coming from the defective rod. This uncertainty is important for iodine, because the trapping in the fuel-cladding

gap is difficult to model. Concerning noble gases, the uncertainty is low as far as the fuel-cladding gap is open. As soon as the gap gets closed, a mechanical retention may occur in the gap, which is not modeled.

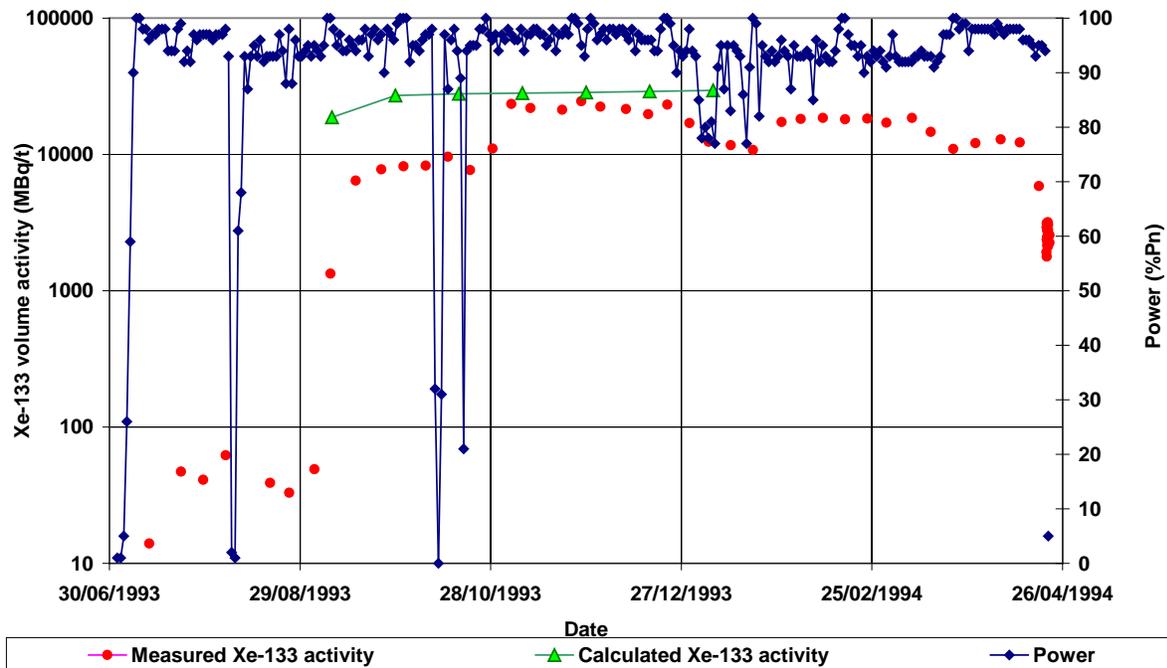


Figure 8. Calculation/Measurement comparison for ^{133}Xe primary volume activity after cladding defect occurrence.

5.2 Actinides

The actinides are considered to be totally insoluble. Their evolution in the primary circuit is due to the neutron capture reactions and to the transfers determined by the erosion/deposition process. The erosion and deposition rates are difficult to evaluate and this results in the main uncertainty for simulation of actinides distribution in the primary circuit.

Concerning actinides, OSCAR calculations are very close to PROFIP calculations, which have been presented in [2] and are not shown here.

5.3 Strontium

Strontium is considered to be found as Sr^{2+} ion in the primary circuit, and follows the adsorption/desorption process. The source term comes from recoil emission from deposited tramp uranium under neutron flux and in case of fuel oxide dissemination, from strontium included in the fuel oxide.

Figure 9 gives a comparison between measurements and calculation during a refueling shutdown. It shows that the experimental peak of ^{90}Sr primary activity at the end of a cycle is correctly reproduced by the desorption model (effect of the decrease of temperature and pH) and by the purification model, which removes ^{90}Sr ions from the primary fluid.

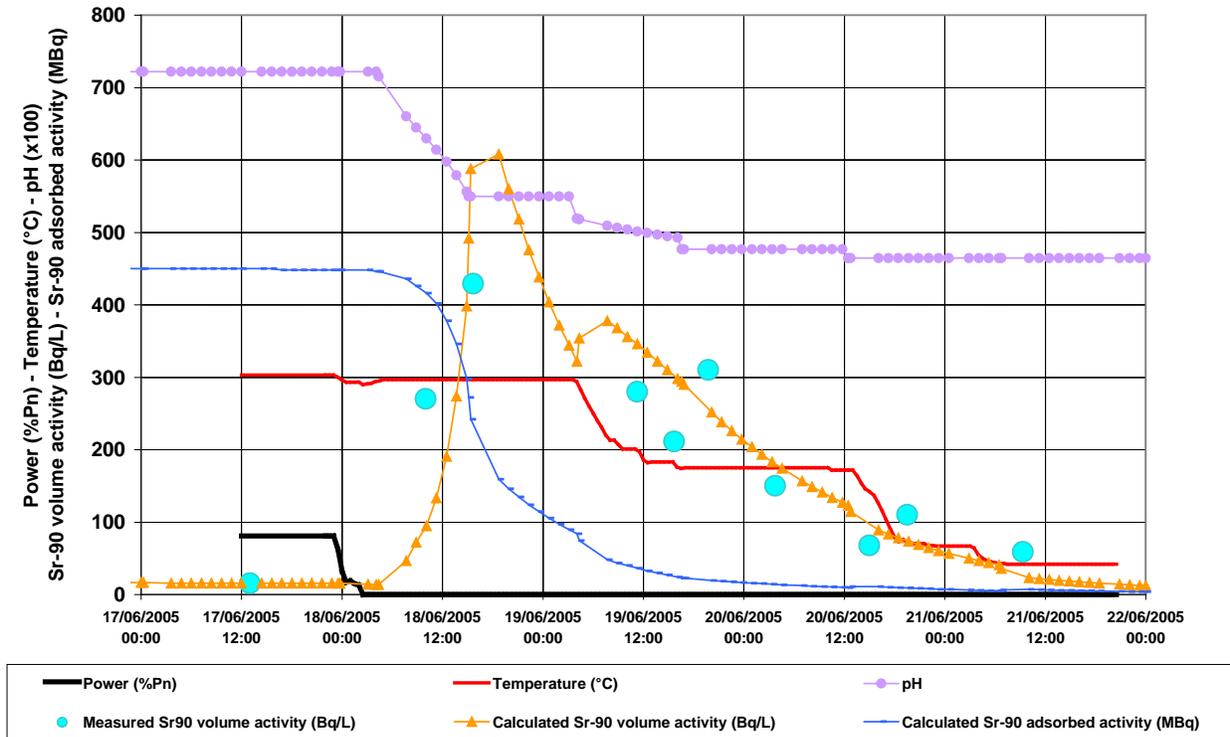


Figure 9. Calculation/Measurement comparison for ^{90}Sr primary volume activity during end of cycle shutdown.

5.4 Ni, Fe, Co, Cr

The main mechanisms involved in corrosion product transfer are dissolution/precipitation and erosion/deposition. As already mentioned, the modeling is the same as for PACTOLE V3 [1], except that the thermodynamical database has been improved. Moreover, a multicriteria calibration has been performed to calibrate the parameters which are not well known.

The parameters to be calibrated have been chosen amongst the most influent. They are the following: erosion rate, dissolution kinetic, metal release (within the range fixed by the experimental results), initial deposit mass (thought to be due to the hot functional tests prior to the first cycle). Besides, an experimental program is under progress to determine the oxide dissolution kinetic [11].

Once calibrated, the OSCAR calculations are compared for the main radioisotopes (^{58}Co , ^{60}Co) to the deposited activities on hot legs, cold legs, steam generator tubes, as well as to the volume activities. As illustrated in figure 10 for ^{58}Co and ^{60}Co deposited on steam generator tubes, the OSCAR code provides a reasonable estimation of PWR corrosion products contamination.

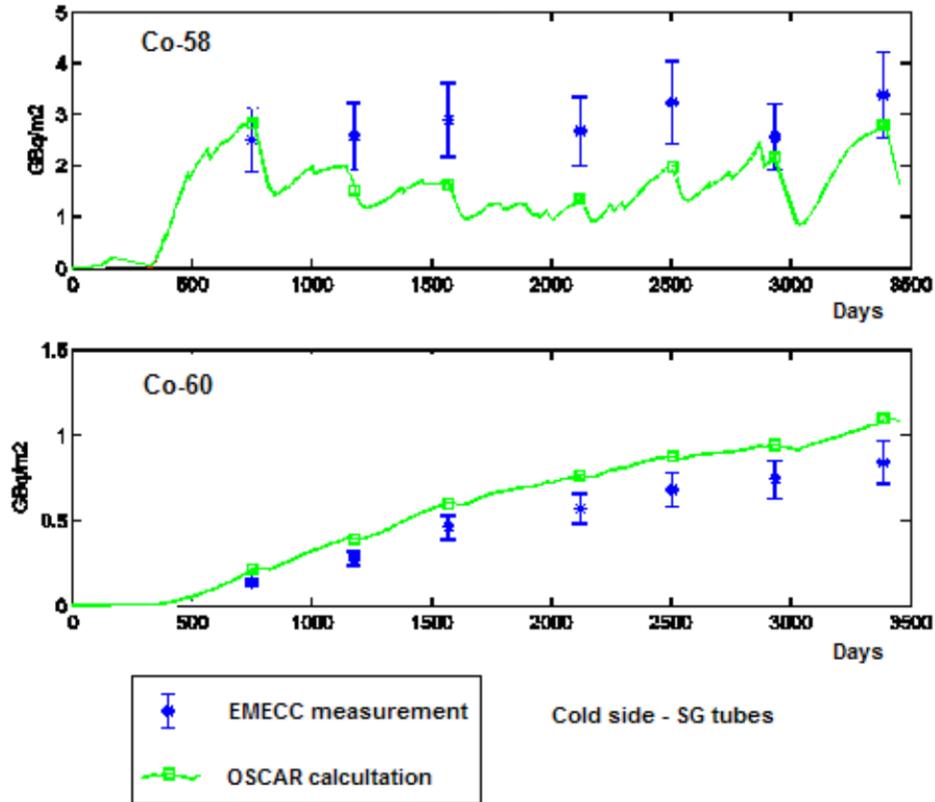


Figure 10. Activity deposited inside the SG tubes.

6. CONCLUSION AND PERSPECTIVES

The OSCAR code package is a new step in the simulation of PWR contamination. With the same code, it is now possible to simulate the contamination of corrosion products, as well as fission products and actinides. Nevertheless, the qualification of this new version needs to be extended on more experimental results (reactor cycles).

The OSCAR code package results from the merging of two codes : PACTOLE and PROFIP. Compared to these previous codes, significant improvements are obtained with the use of the ALCYONE/MARGARET code for defective fuel rod calculations, as well as the use of an improved thermodynamical database and a multicriteria calibration for corrosion products. Moreover, a model for ⁹⁰Sr behavior has been added.

However, in this first version of OSCAR, the fission products and actinides calculations are still independent from corrosion products calculations. In the next version, we will consider the transport of actinides and other non soluble fission products by the corrosion products, so that the erosion and deposition mechanisms will be determined by the corrosion products, even for actinides and fission products. This will allow to describe simultaneously in a consistent framework the coupled behavior of corrosion products and fission products/actinides.

Besides, we will consider a soluble form of actinides, specially after the primary fluid oxygenation at the end of the cycle, and the setback of actinides in water due the corrosion product dissolution. These new models will allow the simulation of alpha activity in the primary fluid during a refuelling shutdown.

Future versions of OSCAR will also be able to implement models for other isotopes as soon as they become available.

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