# BASIC NUCLEAR INSTALLATIONS PERIODIC SAFETY REVIEWS IN FRANCE

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**Abstract –** The French regulation, especially the law Nr 2006-686 of 13 June 2006 concerning transparency and nuclear safety, called the "TSN law", requires that basic nuclear installations (BNI) are submitted to a periodic safety review (PSR) every ten years. The same regulation requires an integrated approach for PSR, having the licensees to present all the risks, radiological and/or chemical, inherent in their installations, and to take into account human and organisational factors. PSR are also the opportunity to reassess bounding accidents. These accidents are assessed by a deterministic approach that can be completed by two methods: the "operating conditions" method, that is still a deterministic method introducing some probabilistic matters like failure frequencies; and the probabilistic safety analysis (PSA) which is only used in France for nuclear power plants. The main result of a PSR is to determine whether a facility can go on operating for 10 years more or not, and to determine the works and compensatory measures that must be done by the operator to reach the goals required by an update regulation.

[11] ASN: Autorité de sûreté nucléaire = French nuclear safety authority
DRC: ASN directorate in charge of nuclear waste, research facilities (including research reactors) and fuel cycle facilities

[12] DCN: ASN directorate in charge of nuclear power plants

## Introduction

The first French regulation dealing with basic nuclear installations (BNI)[13] was a decree from 1963, that settled down procedures to create or modify a BNI, and that was called by the 1961 law concerning olfactory nuisances and air pollution. This decree did not settle down anything concerning reassessment or periodic safety review (PSR). However the French nuclear safety authority tried to put in place in the 1980s periodic reassessment for BNI, because some of them were already ageing – the most ancient BNI in France were created in the late 1940s or the early 1950s. This practice has been introduced into the French regulation by being settled down in the law concerning transparency and nuclear safety (the TSN law) promulgated on the 13 June 2006 [1] and which one of its main application texts is the decree of 2 November 2007 known as the "Procedures decree" [2].

### 1. French regulation concerning PSR – the article 29-III of the TSN law and the article 24 of the Procedures decree

The TSN law [1] requires an integrated approach for creation authorisation, dismantling authorisation and periodic safety review (PSR). This leads the licensees to present all the risks, radiological and/or chemical, inherent in their installations, and to take into account human and organisational factors (HOF).

The TSN law [1] also requires that a facility performs a PSR every 10 years. This PSR should allow to assess the actual condition of the BNI, and its capability to carry on operating, according to the regulation in force at the time of the PSR, and on the basis of an update of the risks that the BNI presents for the interests listed in the article 28 of the TSN law (public security, public health, public healthiness, nature protection and environment protection) [1].

In application of the Procedures decree [2], the draft of the ASN regulatory decision concerning PSR indicates more precisely the documents that the licensee has to provide for a PSR:

The facility description when doing the PSR;

- Its compliance with the requirements and standards accepted for its creation, the analysis of the discrepancies ;
- Compliance of the BNI actual condition with requirements and standards accepted for its creation, and the analysis of the discrepancies ;
- Suggestions of compensatory measures or modifications, or justifications of if it is not possible to achieve these regulation and standards, regarding the life-time the facility is thought to continue operation, and regarding economic factors ;
- Possible evolutions of the plant foreseen for the 10 years to come (until the next PSR) ;
- Feedback from operating and of running of similar plants in the world ;
- Updating of the risks and hazards presented by the plant for the interests listed in the article 28 of the TSN law ;
- Updating of the BNI safety documents: safety report, general operating rules, internal emergency plan, waste management, dismantling provisional plan.

[13] Basic nuclear installation = facility containing amounts of radioactive substances or of fissile materials greater than the thresholds defined in the decree Nr 2007-830 of the 11 May 2007

As the TSN law [1] requires an integrated approach for each step in the life of the BNI, the description, the updating of the risks, the feedback and the suggestion of compensatory measures or modifications must include technical matters such as nuclear safety, radioprotection and/or chemical safety, but also HOF. This last requirement is completely new compared to the previous regulation, although the ASN and the IRSN (the technical support of the ASN) have systematically taken this point into account in their assessments since the late 1990s.

## 2. Assessing the risks

In France the risks assessment is first made with a deterministic approach. But to have more realistic results, French licensees can use two iterative methods that can indicate some weak points in civil engineering or in control systems:

The "operating conditions" method, that is the oldest complementary method used by operators ;

A probabilistic safety assessment (PSA), which is at the present time only used by the nuclear power plants (NPP) and by the EDF (Electricity of France) licensee.

Historically French operators of non-NPP facilities (including FCF) have used a deterministic envelop-type approach that determines only one "envelop" accident for both design and emergency management. This envelop accident was supposed to generate the greatest consequences among all the incidental and accidental situations for the concerned facility. But recent events or assessments have shown that the consequences of some incidental scenarios, defined as minor compared to the envelop accidents, were underestimated if these incidental situations could be combined, and that it could be useful to review in a systematic way the incidental and accidental scenarios to determine the most penalizing ones. In this point of view, the CEA (French Atomic Energy Council) has taken to using the "operating conditions" method since the middle of the 2000s.

### *2.1. Using the "operating conditions" method*

This methodology is issued form the NPP safety assessments.

Previously, non-NPP nuclear installations, and in particular installations operated by CEA used the "three barriers" method to make their safety analysis. It was a deterministic approach. This method consisted in identifying three barriers, which should be materialized, leak proof, independent from each other and should separate the dangerous material from the public and the environment. But this method was limited since the barriers could not be systematically proved independent from each other. In addition the physical boundaries of the barriers could be difficult to set. That is why the CEA issued a "recommendation paper", which sets the "operating conditions" method as a standard. This method consists in designing the installation and the components in regard of the incidental and accidental situations that need to be considered.

For this method, safety functions have to be defined. For the CEA, these functions concern generally:

- containment;
- neutron reaction;
- cooling;
- radiation protection;
- management of explosive gases produced by radiolysis.

The components operating directly for a safety function, the components for which a malfunction could result in the failure of a safety function or the components considered in the safety analysis are identified as safety important components (SIC).

The “operating conditions” method, which is a deterministic method, consists in defining an initial state of the installation, adding a single internal initiating event. These operating conditions are postulated.

They will be:

- listed;
- merged into series of initiating events in order to determine bounding scenarios to analyse;
- classified in four categories according to their annual rate of occurrence (ARO) and the accumulation of the accidental situations;
- examined in regard of the acceptability of their consequences.

Regarding the four categories:

- the first one corresponds to the normal operation;
- the second ($10^{-2}$ < ARO), the third ($10^{-4}$ < ARO < $10^{-2}$) and fourth ($10^{-6}$ < ARO < $10^{-4}$) categories contain less and less probable situations;
- conditions with an ARO < $10^{-6}$ constitute the beyond design basis accidents. Nevertheless, the conditions with $10^{-7}$ < ARO < $10^{-6}$ are analysed, but the ones with an ARO < $10^{-7}$ are excluded.

The operating conditions could be classified according to:

- frequency, if it is defined by a feedback (that is less often determined for nuclear installations than for nuclear power plants);
- the number and the robustness of the lines of defence. These lines are defined as “strong” or “weak”.

To examine the acceptability of the consequences of the accidental situations, general safety objectives have to be defined. If the analysis shows that these objectives are exceeded, lines of defence should be added. It is an iterative process.

At the final step of the operating conditions method, the SIC are defined. These SIC are organised according their importance. Requirements are defined to design, build and operate these SIC.

The CEA considers that the operating conditions method has to be applied to all the PSR and to new facility projects. It was applied to the PSR of the ORPHEE reactor[14], to the design of the RJH reactor[15], of the ITER installation[16] and of the MAGENTA plant[17].

[14] Research reactor located in the Saclay site, and producing neutrons for fundamental researches

[15] Research reactor located in the Cadarache site, still under construction, and planned to irradiate materials, for example to produce medical radionuclides

[16] Prototype of a fusion reactor, located next to the Cadarache site, and still under construction

[17] Fissile materials storage, located in the Cadarache site

The external hazards are studied, according to their probability, in order to maintain the safety functions if these hazards occur and in order to design the installation.

There is at the moment not enough feedback to evaluate clearly the limits of the "operating conditions" method for laboratories and research reactors. This method could be applied to all nuclear facilities in the future. It will be considered through the application of the "BNI order", as it will require several methods to evaluate the risks and hazards presented by BNI, including probabilistic ones. Such requirements will be close to the ones already in force for installations classified for environment protection (ICPE – chemical industries), that have to use a "probabilistic" method similar to the "operating conditions" method.

### *2.2. Using PSA for French PWR (EDF operator)*

For French PWRs, the deterministic approach is completed by **PSA**. This approach is described in the Fundamental Safety Rule (RFS) Nr 2002-01 [3] that was published on 26 December 2002.

**PSA objectives:**

PSA helps to assess whether the arrangements made by the plant operator are satisfactory. It can be used to prioritise the safety problems relating to the design or operation of reactors, and is a tool for dialogue between the plant operators and the authorities.

For operating reactors, PSA contributes to assessment of their overall safety level and highlights points for which design or operating changes can be examined or even judged necessary.

The main objectives of probabilistic analysis of events are the prioritisation of events according to the conditions probability of core damage and the assessment to the pertinence of the corrective actions.

In addition, identification of the main contributions to the core damage frequency highlights any weak points for which design and operation changes can be studied, or even judged necessary. They can be ordered so as to target the priority work.

The PSA helps to put in light uncertainties and limits that would be hidden by deterministic hypotheses.

Because of the systematic investigation of accidental scenarios, PSA can point out some scenarios that haven't been thought of before, because they do not necessary involve equipments or actions classified as "safety important", but that can have very serious consequences at last.

**PSA definition:**

PSA provide a risk assessment method based on systematic investigation of accident scenarios. They provide an overall view of safety, including both equipments and operator behaviour.

PSA considers a list of initiating events which is realistic and complete as possible. In practice, the events studied can include initiating events originating inside the installation (equipment or human failures, internal fire or flooding, etc.) or originating outside (earthquake, external fire or flooding, storm, etc.) associated with the different reactor states.

It highlights operating situations covering complex events and combinations of events, including situations involving the loss of redundant systems and depending on the scope (the nature of the consequences examined and by the events studied) those involving the occurrence of an internal or external hazards.

For each initiating event, PSA establishes accidental sequences resulting from the success or failure of the operation systems and actions involved to perform the safety functions, and assesses the frequency of an undesired event which depends on the type of PSA (below). By summing all the calculated frequency values, it estimates the total frequency of the undesired event, the contribution of each initiating event to the calculated frequency, and the importance for safety of equipment and operating actions.

Three types of PSA can be produced, depending on the consequences studies:

- a level 1 PSA identifies the sequences leading to core damage and determines their frequencies;
- a level 2 PSA assesses the nature, magnitude and frequencies of releases outside the containment;
- A level 3 PSA assesses the calculated frequencies of consequences expressed in dosimetry or contamination terms (or in terms of frequencies of cancers or other effects on health).

For French NPP, the level 3 PSA have not yet been developed to date.

**Method:**

During the first step of the PSR, the reference PSA is updated, incorporating the most recent operating experience (identification and frequency of initiating events, equipment reliability data, operating profile), the standard construction condition (design and operation) and new knowledge about the behaviour of the installation obtained from the most recent studies.

An acceptable method for highlighting and prioritising the principal contributions to the core damage frequency consists in grouping elementary sequences with similar functional characteristics into “functional sequences”, then assessing the hazard associated with the latter. The priority of the grouping method is to constitute “functional sequences” whose frequency and consequences could be reduced by implementing a given provision in order to optimise the identification of opportunities for improvement.

The scope of the reference PSA and the grouping into functional sequences are likely to change at each periodic safety review. In this context, assumptions, criteria, and date have to be justified. Reliability data have to be update and extended, considering in particular feedback of the similar plants in France and abroad; in this frame, particular attention has to be devoted to common mode failures as well as to instrumentation and control systems (hardware and software).

When the conditional probability of core damage associated with an event is greater than a defined reference value, the event is called a “precursor event” and is subject to a thorough analysis.

For the most important precursor events, the plant operator defines specific processing and lead times for the implementation of corrective measures. If possible the expected improvement is assessed.

The results obtained are not used on their own: they are only one of the elements contributing to the decision to implement a corrective measure.

In the safety analysis report compiled for each PSR, the plant operator includes a summary of the reference PSA consistent with the reference and operating conditions of the reactors. This summary includes the main study assumptions and the predominant contributions to the calculated core damage frequency.

The proposed changes are then evaluated and ranked according to their cost and benefit both in terms of probability of consequences. This decision support to achieve the safety objectives set.

Following the PSR, a new version of the reference PSA is produced, taking into account the changes decided on completion of the review process.

**PSA contribution to the decision-making process:**

PSA are a decision-making aid for assessing the importance for safety of systems and equipment.

Depending on the type of use, thresholds can be defined to identify:

- Systems playing an important role with regard to safety according to their contribution to the frequency of core damage.
- The critical failure modes of equipment.

Moreover, in the technical specifications, very long equipment unavailability times should be avoided if the equipment can be repaired in much shorted times.

The analysis must take into account the frequency of the sequences, the possible consequences on containment integrity and the uncertainties.

After the review of any conservative assumptions of the PSA, this analyses results either in a status quo or in an indication of the usefulness of implementing design or operation changes. In the case where changes are made, PSA can be used to assess the advantages and drawbacks of the various solutions considered. The satisfactory character of such changes must be demonstrated by an analysis of their impact on the contributions to the core damage frequency and on the overall core damage frequency.

**PSA limits:**

Despite systematic determination of accident scenarios, PSA have identified limits in terms:

- Incompleteness for the scope (some aggressions, some human interventions processed are not taken into account).
- Uncertainties related to the PSA data and assumptions (especially the estimation of human actions, the estimations of the reliability of equipment operating beyond its qualification conditions).

The uncertainties concerning reliability data, common cause failures and human reliability have to be dealt. As the PSR, the PSA are extended to new aggressions (internal explosion, earthquake, external flooding…).

### *2.3. Bounding accidents*

Bounding accidents are a conclusion of the safety analysis and one of the bases of the authorisation decree. So it is important to identify precisely these bounding accidents.

Moreover they are reassessed at every PSR of the concerned facility, in order to take into account new standards or regulations (example: new technical Fundamental Safety Rule concerning earthquakes to take into account for BNI assessment, edited in 2001).

They depend on parameters like:

- Age of the plant ;
- Ageing management ;
- Modifications the plant was subjected to ;
- Actual condition of its containment. As a global feedback, the containment of the facility confinement is related to the age of the plant and the ageing management made by the licensee ;
- Radioactive and chemical inventories. In fact the distinctive feature of FCF is that some of them use chemical products that can be very harmful for health or environment ;
- Form (liquid, solid, gas) of the radioactive and chemical materials, and constraints generated by their storage or use ;
- Number of activities operated on the site, and eventual interactions between them.

Examples of accidents that may be postulated as possibly bounding accidents:

- $UF_6$ storage or use : accidental release of HF. Example : Eurodif BNI (uranium enrichment);
- Storages or use of high-enriched uranium and/or plutonium powder or nuclear fuel (UOx, MOX): criticality accident. Example : the Melox facility;
- Storages of highly radioactive waste or plants treating highly radioactive liquid waste : loss of containment and/or of cooling. Example : storage pool of spent nuclear fuel or of fission products tanks;
- Amounts of flammable or explosive materials (associated with radioactive materials or not) : fire or explosion. Example: the Superphenix reactor (fast breeder reactor cooled by liquid sodium, under decommissioning).

Most nuclear sites host several facilities, operating different activities linked to each other, at least because utilities are often common. So it is necessary to examine the bounding accident for each facility, as well as the most serious in terms of consequences, to determine perimeters for public protection or evacuation in case of serious events.

Example: the FBFC site in Romans-sur-Isère:

- There are 2 BNI on this site: the BNI Nr 63 that manufactures uranium fuels for research reactors; and the BNI Nr 98 that manufactures uranium fuels for French NPP.
- The bounding accident for BNI Nr 63 is a criticality reaction, because of the storage and the use of highly-enriched uranium. The bounding accident for BNI Nr 98 is an accidental release of HF, because of the storage and the use of great amounts of $UF_6$.

The most penalizing bounding accident, because generating the widest perimeter for public protection or evacuation, is the one concerning BNI Nr 98.

But, because the two bounding accidents generate different consequences requiring different types of protection measures for the public, the bounding accidents are both kept in the operator safety documents and in public-protection documents made by the local authorities.

The feedback of the Fukushima accident is hoped to cause reassessment of the bounding accidents, by taking into account new scenarios like events that are not bounding accidents for the concerned activities, but which consequences on the other activities can result in the bounding accident for the site.

### 3. PSR – practical point of view

The PSR allows having a global view on the facility, and making it easier to take decisions for both the ASN and the licensee. However this approach is quite long and heavy to develop and to analyse : the feedback for non-NPP facilities is about 1 year for the operator development, and about 1 year also for the IRSN and the ASN analysis.

**For French PWR** which are standardized into 4 technological types (900 MW, 1300 MW, 1450 MW, EPR), a PSR is assessed as follow:

- A general PSR concerning one technological type;
- A specific PSR concerning a reactor, taking into account the results of the technological type PSR and the actual condition of the reactor. But this specific PSR mostly deals with technical matters ;
- A ten-yearly specific review of a reactor, where components crucial for safety are submitted to special inspections and tests. For example, the vessel and the primary coolant systems are submitted to visual examination and under-pressure tests.

PSA results are examined and criticized by the operator (EDF), IRSN and ASN during this PSR.

As PSA has to take into account the actual condition of the plant, and as facilities have to perform a PSR every 10 years, there is no specific use of PSA for justifying extension beyond design life for NPP.

The French PWR licensee (EDF) can be submitted to specific PSR on one particular subject. For example, it is submitted every 3 years to an operational feedback PSR, and it was submitted in 2010 to a PSR concerning specifically HOF. Results of these specific PSR can be taken into account while assessing PSR for a specific reactor.

PSR **for French non-PWR facilities** look more as integrated assessments, as they combine almost all subjects listed in the chapters above, except that safety is only examined through deterministic considerations. If the licensee is the CEA, the “operating conditions” method can be used.

For these facilities, there is no design life acted in their creation authorisations. Consequently every ten-yearly PSR concerning one plant has to examine whether the plant can go on operating or not.

## 4. Conclusion

PSR in France have looked like integrated approaches for more than 10 years, even though such approaches are of regulatory matters since the publication of the TSN law that explicitly requires taking into account all the risks inherent to a facility, including HOF. Assessing a PSR requires among other things, updating accidental scenarios. This updating is based on deterministic approaches completed by an iterative method: PSA for PWR, "operating conditions" method for some other facilities. The main result of a PSR is to determine whether a plant can go on operating for 10 years more or not, and to determine the works and compensatory measures that must be done by the operator to reach the goals required by an update regulation, or to improve the safety of his installation.

The French regulation body concerning PSR has to be completed by the approval of 3 texts:

- The order concerning general regulation for BNI (known as the "BNI order") ;
- The ASN decision concerning PSR ;
- The ASN guide concerning the use of PSA for French nuclear power plants.

### References

[1] Law Nr 2006-686 of 13 June 2006 (the TSN law);

[2] Decree Nr 2007-1557 of 2 November 2007 (the Procedures decree);

[3] Fundamental Safety Rule Nr 2002-01 concerning the use of PSA for French PWR;

[4] Technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors.

### Bibliography

[a] Draft of the order concerning general regulation for BNI (known as the "BNI order");

[b] Draft of the ASN regulatory decision concerning PSR.

## Nuclear installations periodic safety reviews in France

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### Brief history

**The decree Nr 63-1228 of the 11 December 1963 :**

- ❑ 1st text regulating nuclear safety
- ❑ nothing concerning reassessment or periodic safety review

However, in the 1980s, French nuclear safety authority put in place periodic reassessments, because some nuclear installations were already ageing :

- ❑ effective for NPP
- ❑ various results for other installations



### Regulation in force concerning PSR

**The law Nr 2006-686 of the 13 June 2006 – the "TSN law" :**

- ❑ requires an integrated approach for each step in the life of a facility and for PSR
- ❑ requires a PSR every 10 years for all facilities (NPP + non-NPP)

The PSR should allow to assess the state of the facility taking into account :

- ❑ international and national best practices
- ❑ norms and regulations in force at the time of the PSR
- ❑ actualization of the facility description
- ❑ actualization of risks that the facility can generate for public and environment
- ❑ feedback of operation

The PSR must also cover suggestions of compensatory measures or of modifications made by the licensee to improve its facility safety.

**Regulation in force concerning PSR**

**The TSN law requires an integrated approach that includes :**

- ❑ nuclear safety
- ❑ when it is appropriate : chemical or biological safety
- ❑ radioprotection
- ❑ human & organizational factors

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**Assessing the risks**

The risks assessment is first made with a **deterministic approach.**
For non-NPP facilities in France, this method was historically used to determine one "envelop" accident for both design and emergency management. This envelop accident was supposed to generate the greatest consequences among all the incidental and accidental situations.

But events have shown that this method was not enough, because they have revealed that other incidental or accidental scenarios, not taken into account for design or even not considered at all, could lead to equal or greater consequences than the envelop accident.

=> The CEA has used a complementary method since the beginning of the 2000s at least for some of its reactors, and the ASN would like to extend this use.

The deterministic approach can be supplemented by :
❑ the "operating conditions" method => the oldest complementary method ; the one the CEA is using
❑ a probabilistic safety assessment (PSA) => only used by NPP

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**Assessing the risks**

**The "operating conditions" method :**

The CEA issued a "recommendation paper", which settles down the "operating conditions" method.

1. Definition of safety functions among the list :
   - ❖ Containment
   - ❖ Neutron reaction
   - ❖ Cooling
   - ❖ Radiation protection
   - ❖ Management of explosive gases produced by radiolysis

2. Identification of the safety important components (SIC)

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**Assessing the risks**

## The "operating conditions" method :

3. Definition of an initial state of the facility, and then adding of a single internal initiating event = creation of an "operating condition"

4. Evaluation of the consequences of each "operating condition"

5. Assessment of the operating conditions regarding the acceptability of their consequences and regarding their probability of occurrence



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**Assessing the risks**

## The "operating conditions" method :

4 classes of operating conditions linked to their probability of occurrence per year (ARO) :

- ❑ 1st class : normal operations
- ❑ 2nd class : $10^{-2}$ < ARO
- ❑ 3rd class : $10^{-4}$ < ARO < $10^{-2}$
- ❑ 4th class : $10^{-6}$ < ARO < $10^{-4}$

ARO < $10^{-6}$ : excluded situations (beyond design)



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**Assessing the risks**

## The "operating conditions" method :

The class of the operating condition is compared to the acceptability of its consequences.
Acceptability is defined by general safety objectives, that depend on the ARO : the more frequently the operating condition can occur, the less its consequences can be.

If the general safety objectives are exceeded, lines of defence have to be added.

The process is iterative.

## Assessing the risks

### The "operating conditions" method =

A decision-making aid for assessing the importance for safety of systems and equipments, because helping to identify :

- available lines of defence
- if these lines have weaknesses
- if there are lines of defence missing



## Assessing the risks

### The "operating conditions" method :

CEA facilities submitted to :

- ❑ PSR of the Orphée reactor (research reactor)
- ❑ design assessment of the Jules-Horowitz Reactor
- ❑ design of the ITER facility
- ❑ design of the Magenta facility (fissile materials storage)



## Assessing the risks

### PSA :

Only used in France by NPPs

Systematic investigation of accidental scenarios

PSA highlights operating situations covering complex events and combinations of events, including situations involving the loss of redundant systems or external events.

## Assessing the risks

**PSA :**

1. Choice of an initiating event
2. Determination of accidental sequences resulting from the success or failure of the operation systems and actions involved to perform the safety functions
3. Assessment of the frequency of an undesired resulting event



## Assessing the risks

**PSA :**

3 types of PSA :

- PSA level 1 : identification of the sequences leading to core damage, and determination of their frequencies
- PSA level 2 : assessment of the nature, magnitude and frequencies of releases outside the containment
- PSA level 3 : assessment of the calculated frequencies of consequences expressed in dosimetry or contamination terms (or in terms of frequencies of cancers or other effects on health)



## Assessing the risks

**PSA :**

PSA are a decision-making aid for assessing the importance for safety of systems and equipments :

- Identification of systems contributing to frequency of core damage
- Determination of the critical failure modes of equipments

### Assessing the risks

**Bounding accidents :**

Bounding accidents are a conclusion of the safety analysis.

They are reassessed at every PSR of a facility, in order to take into account new standards or regulations.



### Assessing the risks

**Bounding accidents :**

They depend on parameters like :

- ❑Age of the plant
- ❑Ageing management
- ❑Modifications the plant was subjected to
- ❑Actual condition of its containment
- ❑Radioactive and chemical inventories
- ❑Form (liquid, solid, gas) of the radioactive and chemical materials, and constraints generated by their storage or use
- ❑Number of activities operated on the site, and eventual interactions between them. The interactions can modify accidental scenarios and bounding accidents.

**=> Bounding accidents are determined case by case.**



### Assessing the risks

**Bounding accidents : example of the FBFC site**

2 facilities on the site :

❑ FBFC (facility Nr 98) manufacturing nuclear fuel for NPP => uranium enriched up to 5% ; great amounts of $UF_6$

❑ CERCA (facility Nr 63) manufacturing nuclear fuel for research reactors => uranium under metallic form, enriched up to 93.5%

The buildings of the 2 facilities are nested. But the main buildings where the fuel are manufactured are clearly separated.

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**Assessing the risks**

## Bounding accidents : example of the FBFC site



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**Assessing the risks**

## Bounding accidents : example of the FBFC site

Each facility has its bounding accident :

- ❑ FBFC : air crash on its $UF_6$ storage => major leak of HF
- ❑ CERCA : criticality accident

Theoretical bounding accident of the site : the one of FBFC, because it leads to the largest evacuation zone

But both bounding accidents are taken into account in the operator emergency documents and in emergency and public protection plan, because they do not have similar consequences.



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**Assessing the risks**

## Bounding accidents : perspectives :

The feedback of the Fukushima accident is hoped to cause reassessment of the bounding accidents, by taking into account new scenarios like :

- ❑ interactions between buildings
- ❑ chemical events occurring inside buildings that are not classified as parts of nuclear installations…

The consequences of such events could lead to a bounding accident for the site.

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## PSR : practical point of view

- global view on a facility
- helps to take decisions for both the ASN and the licensee
- quite long to perform : about 2 to 3 years between the beginning of the writing by the operator and the ASN demand letter concerning the assessment
- heavy to develop and to analyse

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## PSR : practical point of view

For NPP :

PSR are assessed in 3 steps :

- A general PSR concerning one technological type (900 MW, 1300 MW, 1450 MW, EPR)
- A specific PSR concerning the reactor itself => technical matters
- A ten-yearly specific review of a reactor, where components crucial for safety are submitted to special inspections and tests

PSA results are examined and criticized during PSR.

NPP can also be submitted to specific PSR on one particular subject (HOF, feedback from events…)

The creation authorization of one NPP settles down a precise lifetime. PSR helps to examine whether the NPP can go on operating or not beyond this initial lifetime.

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## PSR : practical point of view

**For non-NPP facilities :**

PSR look more as integrated assessments

no design life acted in the creation authorizations
=> every ten-yearly PSR concerning one plant has to examine whether the plant can go on operating or not

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