IRSN
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Towards Optimized Criticality Safety Limits

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Towards Optimized Criticality Safety Limits

- **OBJECTIVE:**

  - **To detail the various stages leading to the choice of the criticality control modes and to the dimensioning of the apparatuses.**
  - **To identify the possible ways allowing an optimization of the criticality constraints, with respect to the criticality-safety requirements and the economic objectives.**
  - **To identify the conditions to guarantee the parameters required for the nuclear criticality safety, in terms of knowledge of the processes and in terms of validation of the criticality calculations.**
  - **Finally, to identify the lacks of knowledge and to try to define the researches to be carried out to gather the required missing data.**
  - **As an illustration, several examples resulting from the main operations carried out in the current Fuel Cycle Installations operated in France will be exposed and some tracks for the future Fuel Cycle will be proposed.**

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1 - PRELIMINARY DESIGN STAGE:

This stage must be carried out for each apparatus in which a risk of criticality is identified.

- **To describe the physicochemical operations to be carried out and to identify the available processes.**
- **To describe the fissile materials which will be handled.**
  - **qualitative description: which isotopes? and which chemical forms?**
  - **quantitative description: which quantities and which flows of production?**

    (these materials have to be representative of normal operations AND possible disturbed operations identified at this stage)
- **To make the (strategic) choice of a criticality control mode.**

  This choice is facilitated when "standard" critical parameters (masses, geometrical dimensions etc) are available and if some "experience feedback" of the implementation of similar processes with similar fissile materials is available.
- **To calculate the values (critical and sure) of the control parameters (masses, geometrical dimensions…) for the various identified fissile mediums.**

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1 - PRELIMINARY STAGE OF DESIGN (2):

- **At the end of this stage, sufficient information required for the preliminary design of the apparatuses are available, which makes it possible to confirm the feasibility**
- **If a satisfactory solution cannot be not obtained, it may be necessary to carry out some iterations by modifying the initial data (range of the fissile materials to be handled), by modifying the criticality control mode (geometry + mass limitation …) or by using some neutron poisons etc.**
- **At the end of this stage, it should be possible to analyze more finely the disturbances resulting not only from one apparatus but from the operation of a whole of apparatuses and, IF NECESSARY, to better specify the range of the fissile materials to be taken into account.**
- **At the end of this stage, a "consolidated" safety analysis must be carried out in order to make sure that the reference fissile mediums used for the calculations are well envelopes of the situations representative of disturbed operations considered.**

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❑ 2 - DETAILED DESIGN STAGE:

- In order to limit the number of later calculations, one can also identify an upper bound "reference fissile medium": which leads to the most constraining values of the criticality control parameters.

- At the detailed design stage, it is necessary to know the sensitivity of calculated Keff, with the calculations models parameters and with the nuclear data.

- It is also useful to be able to evaluate the available safety margins. Thanks to these studies, it will be possible to evaluate if there are some "reserves" making it possible to consider future evolutions (ability to be adapted to new fissile mediums…)

- At this stage, the questions relating to the validation of the computational tools and the confidence in the calculations take a great importance.

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3 - COMPLEMENTARY OPTIMIZATION STAGE:

We now will examine how a thorough knowledge of the processes and fissile materials really handled can make it possible to optimize the design of the installations of the fuel cycle with respect to the constraints of criticality safety.

- **This optimization is made possible by the fact that, in a real application, the range of variation of some parameters can be restricted compared to their theoretical range of variation.**
- **So that this optimization is not made against the safety, the restricted values taken by some parameter should be guaranteed and controllable.**
- **Two complementary ways are detailed and concrete examples coming from some existing French nuclear installations are now given.**

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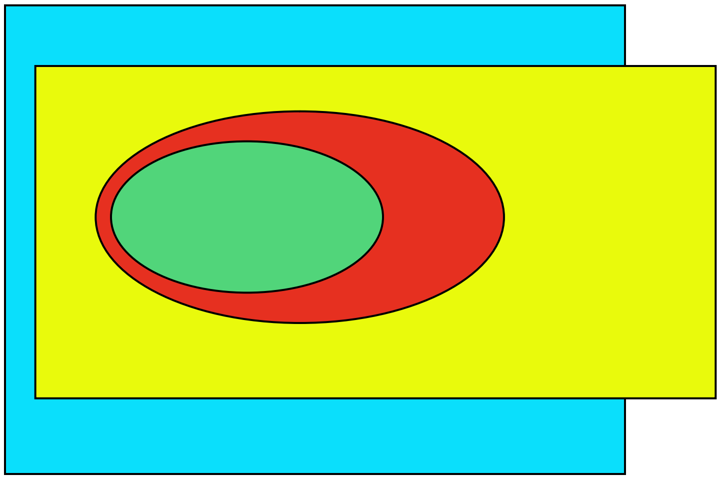
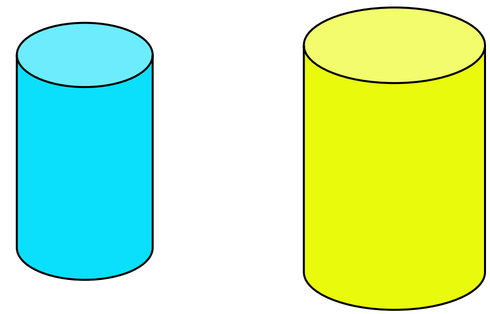
- **INITIAL DIMENSIONING STAGE:**

Leading to dimensions





Normal Conditions

Abnormal Conditions

Reference Fissile Medium 1

Reference Fissile Medium 2

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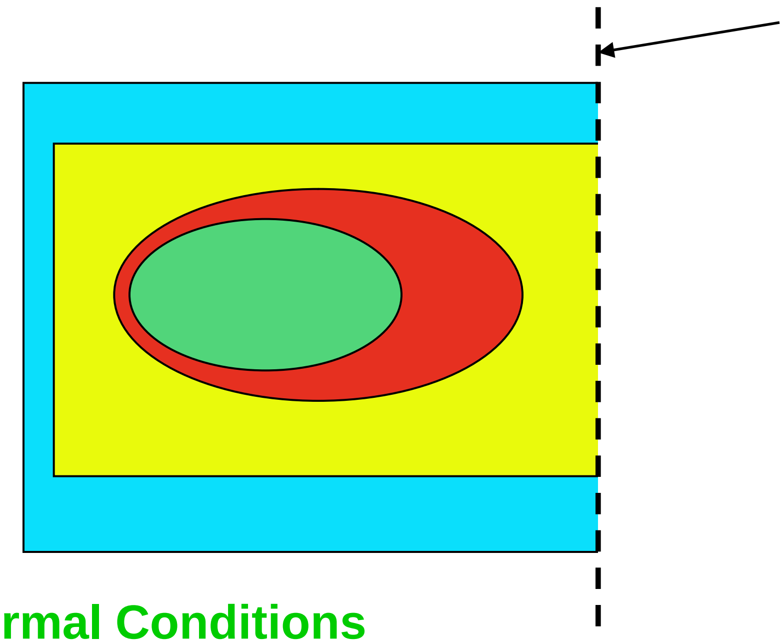
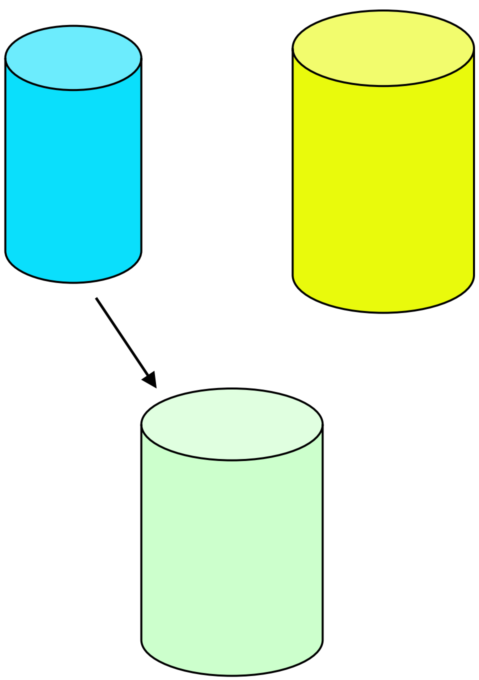
- **OPTIMIZATION AT DIMENSIONING STAGE:**



Limitation guaranteed by Process
Lead to optimized dimensions



Normal Conditions

Abnormal Conditions

Reference Fissile Medium 1modified

Reference Fissile Medium 2 modified

Warning: the optimal moderation is possibly changed and as a consequence, the validation need

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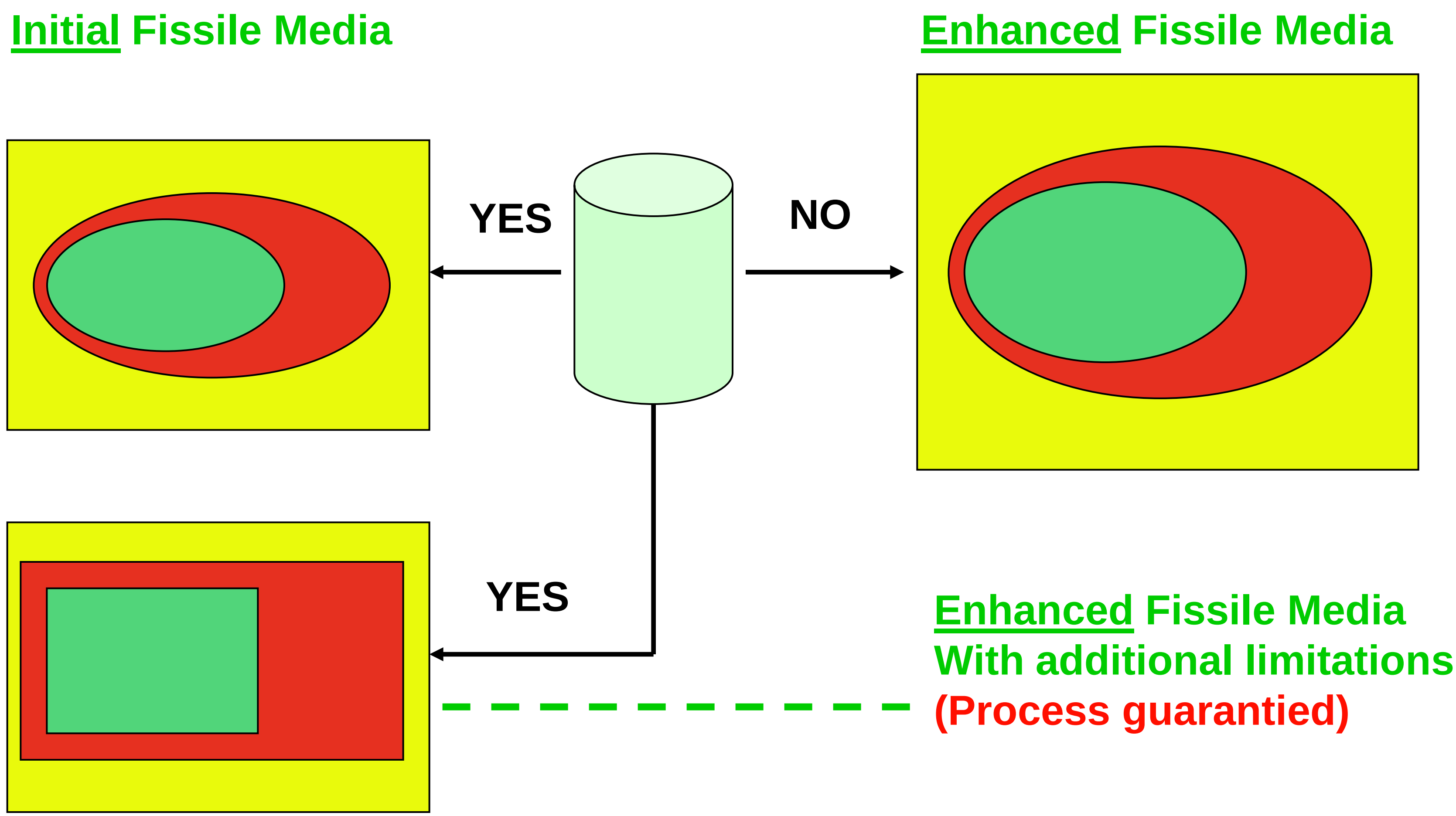
- **FURTHER OPTIMIZATION POST DIMENSIONING STAGE:**



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- **Ex1 - POOLS & TRANSPORT CASKS – UO2 PWR FUELS - initial**

In the Eighties, the bases of dimensioning of the pools and theses of the transport casks were very simple.

- **The criticality control mode of the transport casks and the pool baskets was based on their geometry.**
- **It was taken account of the presence of a fixed neutron poison (Boron) in the walls of the cells of transport casks and in the pool baskets.**

- **The enrichment of Uranium was to the maximum of 3,5% in 235U.**
- **No minimal value of irradiation was required (fresh fuel assumption).**
- **The assemblies could contain any number of rods, which means that any number of rods could be removed in a fuel assembly (any moderation).**
- **The dimensions of the assemblies, the rods diameter and the pitch of the lattice were those of a "standard" assembly.**

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- **Ex1 - POOLS & TRANSPORT CASKS – UO2 PWR FUELS – evolution (1)**

  The maximum initial enrichment of Uranium becomes higher than 3,5% in 235U.

  - **The criticality control mode of the transport casks and the pool baskets is unchanged.**

  - **To compensate the increasing of uranium enrichment, it is taken account of the irradiation of the fuels (Burn-up credit).**

  - **An additional optimization is obtained by limiting the number of rods removed in an assembly (to a maximum of 10 missing rods)**

  - **From a technical point of view, criticality calculations are carried out by taking account of the composition of irradiated fuels with some conservative assumptions.**
    (Burn-up credit - Actinides only)

    The range of the fuel rods parameters are more precisely defined (diameter, thickness of the cladding material …)

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- **Ex1 - POOLS & TRANSPORT CASKS – UO2 PWR FUELS – evolution (2)**

PRACTICAL APPLICATION

The requirement (minimum Burn-up) is applied as a mean value to the less irradiated 50 cm of the assemblies.

- **The control of the irradiation of all assemblies is carried out on the power plant site before loading the transport casks (physical control - not only administrative).**
- **For a low value of the minimal irradiation required: (guaranteed by only one cycle in the reactor)**
  => the control of this minimal irradiation is made by a simple gamma measurement (loading error detection).
- **For a minimal irradiation required higher than one cycle in the reactor:**
  => the burn-up of each assembly is measured by "gamma scanning"

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- **Ex1 - POOLS & TRANSPORT CASKS – UO2 PWR FUELS – evolution (3)**

  FURTHER OPTIMIZATION (on going)

  The current use of "actinide only" Burn-up credit should be improved, taking account some FISSION PRODUCT (6 or more) according to the following conditions:

  - **Validation of the Fissions Products nuclear data.**
  - **Validation of the FP content in the Spent Fuels.**
  - **Ability to Define an "upper bound" Burn-up profile according to the range of irradiation history in reactors.**
  - **Ability to measure the profiles in order to guarantee the parameters used for criticality calculations.**
  - **Improvement of the criticality calculation schemes. (End effect, source convergence …)**

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- **Ex1 - POOLS & TRANSPORT CASKS – ENHANCED UO2 PWR FUELS**

CONCLUSIONS:

The Nuclear Criticality Safety of POOL BASKETS and TRANSPORT CASKS designed initially as "Geometrically Safe" for UO2 PWR fuels assemblies containing U (3.5%), have been demonstrated for ENHANCED PWR FUELS (U > 3.5%) thanks to an optimization process (on going) including various results of researches:

- **Validation of some Actinides and Fissions Products nuclear data.**
- **Validation of the Spent Fuels content.**
- **Study of Spent Fuel "Burn-up profiles" according to various irradiation history in reactors.**
- **Development of some Techniques for measuring the profiles in order to guarantee the parameters used for criticality calculations.**
- **Development of new criticality calculation schemes. (End effect, source convergence …)**

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- **Ex2 - UO2 PWR SPENT FUEL DISSOLUTION**

A PRELIMINARY DESIGN of the dissolver of UO2 PWR spent fuel as been carried out according to the following assumptions:

- **The criticality control mode of the dissolver was based on its geometry.**
- **It was taken account of the stainless steel in the basket.**

- **The enrichment of Uranium was to the maximum of 3,5% in 235U.**
- **No minimal value of irradiation was required (fresh fuel assumption).**
- **The presence of cladding material was ignored.**

THIS SET OF ASSUMPTIONS LEAD TO A NON ACCEPTABLE DESIGN:

➔ TO SMALL DIMENSIONS

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- **Ex2 - UO2 PWR SPENT FUEL DISSOLUTION**

IN A SECOND STAGE, some assumptions and basic data have been changed:

- **The criticality control mode of the dissolver was based on its GEOMETRY + a MASS LIMITATION.**
- **A Burn-up credit (actinide only) was taken in account.**
- **The MASS limitation depends on the Initial Enrichment (Ei) and the Burn-up.**
- **A minimum of 3N Nitric Acid was assumed.**
- **A maximum concentration of Uranium was assumed (Cu < 400 g/l)**

THIS NEW SET OF ASSUMPTIONS LEAD TO AN ACCEPTABLE DESIGN.

In addition the ABILITY of FUTURE EVOLUTION have been checked

=> INITIAL ENRICHMENT up to 4.5% 235U

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❑ **Ex2 - UO2 PWR SPENT FUEL DISSOLUTION**

PRACTICAL APPLICATION and POST OPTIMIZATION:

- **The MASS LIMITATION was in the main cases (standard SPENT FUEL) due to the PROCESS and NOT to CRITICALITY requirements.**
- **A set of various conditions have been studied:**
  Maximum MASS versus INITIAL ENRICHMENT and BURN-UP.
- **The licensing was extended to MOX Fuels.**

CONCLUSION: THANKS TO RESEARCHES (PROCESS KNOWLEDGE, BURN-UP measurement, CRITICAL EXPERIMENTS (HTC and PF programs), an OPTMIZED DESIGN of the Spent Fuel Dissolver was carried out.

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- **Ex3 - CYCLES OF PURIFICATION BY SOLVENT - CURRENT PRACTICE**

The criticality control mode of the principal apparatuses of extraction by solvent is the geometry.

- **For this case, the phase of dimensioning is particularly important and the choice of the “upper bound” parameters defining the fissile medium as well as the choice of the scenarios of “abnormal” situations is determining.**
- **For the current facilities of La Hague (UP3 and UP2-800), the major data retained for the “criticality design”of the extraction columns was that 240Pu content of the plutonium was at least 17%.**
- **A secondary data was that the residual enrichment of uranium in 235U was lower than 2%.**
- **Including for the calculation of the pulse columns of the first cycle of extraction, the Pu/U+Pu ratio was supposed to be unspecified.**

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- **Ex3 - CYCLES OF PURIFICATION BY SOLVENT TRACKS FOR FUTURE EVOLUTION**

  The current rules of dimensioning have leads to the realization of annular pulsed columns containing, in their internal part, a neutron poisoned core made with borated concrete. This « design » was necessary to abble the flow rate of the solution to be purified.

  TRACKS OF EVOLUTION FOR FUTURE FACILITIES (open question)

  - **Which isotopic composition of plutonium?**
  - **Which range of variation for the Pu/U+Pu ratio?**

- **TRACKS FOR FUTURE RESEARCHES**

  - **Taking in account the "experience feedback" relating to the spent fuel composition,**
  - **Based on the knowledge of the solvent extraction process,**

  an IMPORTANT OPTIMIZATION should be possible

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- **Ex3 - CYCLES OF PURIFICATION BY SOLVENT: TRACKS FOR FUTURE RESEARCHES**

  The current "upper bound" parameters used for the criticality design should be improved in order to optimize the dimensions of the solvent extraction apparatuses according to the following conditions:

  - **Validation of high order Pu isotopes nuclear data.**
  - **Validation of Pu+U solutions in a large range of variation for the Pu/U+Pu ratio.**
  - **Definition and Validation of a DILUTION LAW for Pu+U organic solutions**
  - **Good knowledge of the range of the PROCESS PARAMETERS with the ability to guarantee the "upper limits" used for the criticality calculations.**

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- **GENERAL CONCLUSION:**
  TRACKS FOR FUTURE RESEARCHES

  The current criticality practices and methods leads to a good level of safety. However, some improvements of theses methods should be possible in using "upper bound" parameters better connected with the real situations without reducing the level of safety. The conditions of success involve

  - **Validated and well known processes.**
  - **Validated nuclear data and criticality codes.**
  - **Good knowledge of the range of the PROCESS PARAMETERS with the ability to guarantee the "upper limits" used for the criticality calculations.**
  - **Ability to monitor and to measure the control parameters in order to detect the "abnormal situations in a short time.**