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DISPARITIES IN THE SAFETY DEMONSTRATIONS FOR RESEARCH REACTORS AND THE NEED FOR HARMONIZATION

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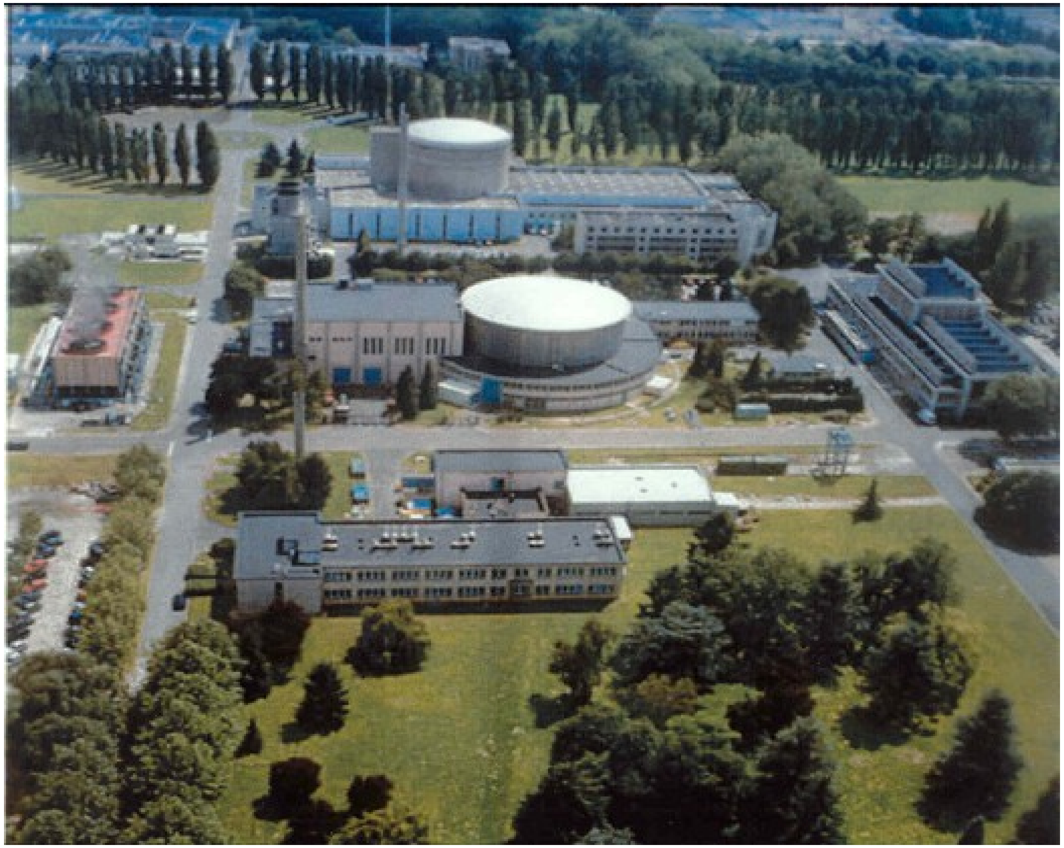

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- Disparities observed in the safety analyses, especially in the:
    - Approaches and methods
    - Envelope accidents
    - Data used for source term evaluations
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- Conclusion

INTRODUCTION (1/2)

- **272 operational research reactors in 58 countries:**
  - **Great diversity in design, power level and operating modes.**
  - **Consistency between the safety principles adopted for the design and safety reassessments.**

INTRODUCTION
(2/2)

- **Important disparities in the safety analysis methods, data and computational tools used for similar research reactors.**

- **This may result in different conclusions on the safety level of such reactors.**

USE OF DIFFERENT METHODS FOR THE SAFETY ANALYSIS (1/3)

- **DTERMINISTIC METHOD:**
  - Most frequently used method
  - Selection of limited number of events for analysis.
  - Accidents are postulated without the need to precisely identify their causes.

USE OF DIFFERENT METHODS FOR THE SAFETY ANALYSIS (2/3)

- **PRABABILISTIC METHOD:**
  - Often used as a complement to deterministic method.
  - Able to take into account common failure modes.
  - Some uncertainties in the available generic reliability data which could be not valid for a given research reactor.

USE OF DIFFERENT METHODS FOR THE SAFETY ANALYSIS (3/3)

- Safety analyses for French research reactors are based mainly on deterministic approach with, however,

- probabilistic evaluations concerning external events (air crashes, explosions,..).

DISPARITIES IN THE TYPES OF ENVELOPE ACCIDENTS (1/2)

Envelope accidents taken into account in the safety analyses for similar research reactors cover a range including :

- **The meltdown of a fuel element, or**

- **the partial or total meltdown of the reactor core.**

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DISPARITIES IN THE TYPES OF ENVELOPE ACCIDENTS (2/2)

- Reactivity insertion accidents resulting in core fuel meltdown are not always analyzed in a consistent manner.

- The same type of reactivity accident is considered as DBA for some reactors and as BDBA for others._

ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT IN THE SAFETY ANALYSIS (1/4)

Reactor	Fuel (Power)	Envelope accident
BR2	U-Al 93% (100 MW)	200 MJ reactor excursion leading to the core meltdown followed by a water-aluminum interaction (DBA)
FRM II	U_3Si_2 90% (20 MW)	Complete meltdown of the core due to the loss of primary cooling system or to a reactivity accident with failure of shutdown system (BDBA)

ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT IN THE SAFETY ANALYSIS (2/4)

Reactor	Fuel (Power)	Envelope accident
HIFAR	U-Al 60% (10 MW)	- Complete meltdown of the core due to a LOCA (MCA) - Complete meltdown of the core with unsealed containment (BDBA)
OPAL	U_3Si_2 19.75% (20 MW)	- Melting of 36 U-Mo targets due to a loss of coolant flow (BDBA) - Melting of 3 fuel plates due to flow blockage (BDBA)

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ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT IN THE SAFETY ANALYSIS (3/4)

Reactor	Fuel (Power)	Envelope accident
OSIRIS	U_3Si_2 19.75% (70 MW)	135 MJ reactor excursion leading to complete meltdown of the core followed by a water-aluminum interaction (DBA)
RHF	U-Al 93% (57 MW)	- 135 MJ reactor excursion leading to complete meltdown of the core followed by a water-aluminum interaction (DBA) - Uncovering and complete meltdown of the core (DBA)

ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT IN THE SAFETY ANALYSIS (4/4)

Reactor	Fuel (Power)	Envelope accident
RSG-GAS	U_3Si_2 19.75% (30 MW)	- Meltdown of a fuel element caused by a coolant channel blockage (DBA) - ATWS leading to the meltdown of 5 fuel elements (BDBA)
SAFARI-1	U-Al 87%-93% (20 MW)	Complete meltdown of the core accompanied by the loss of all ventilation systems (MHA)

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ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT FOR FRENCH RESEARCH REACTORS (1/2)

- **The DBA taken into account in France for pool type research reactors is a BORAX type explosive reactivity accidents.**
- **Main assumptions:**
  - **complete core meltdown under water**
  - **Energy release of 135 MJ, including 9% in the form of mechanical energy.**

ENVELOPE ACCIDENTS TAKEN INTO ACCOUNT FOR FRENCH RESEARCH REACTORS (2/2)

- **These assumptions were used for the different research reactors in prescriptive manner for the design of the reactor pool and containment building.**

- **Analyses are currently in progress at the IRSN to obtain more precise data on the consequences of explosive reactivity accidents**

THE NEED FOR HARMONIZATION

Harmonization is needed for the:

- types of envelope accidents to be taken into account for research reactors having similar technical characteristics.
- Terminologies used to designate these accidents. _

DISPARITIES IN THE DATA USED FOR SOURCE TERM EVALUATIONS (1/3)

- Disparities concerning:
  - Fission products release fractions from molten fuel into pool water and then into the reactor building atmosphere.
  - Hypotheses used to determine the release to the environment.

RELEASE FRACTIONS FROM MOLTEN FUEL INTO POOL WATER (1/2)

	BR2	FRM II	OPAL	OSIRIS	RSG-GAS
Kr,Xe	1	1	1	1	1
I	0.5	0.75	0.3	0.8	0.5
Br	0.5	0.75		0.8	0.5
Cs	0.1	0.25	0.3	0.8	0.25
Rb	0.1	0.25	0.3		0.25
Te	0.01	0.001	0.01	0.8	0.25

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RELEASE FRACTIONS FROM MOLTEN FUEL INTO POOL WATER (2/2)

	BR2	FRM II	OPAL	OSIRIS	RSG-GAS
Ru	0.01	0.001	0.01	0.1	
Ba,Rh				0.1	
Sr		0.001		0.1	
Actinides		0.001		0.01	
Lanthanides				0.01	

RELEASE FRACTIONS FROM POOL WATER INTO THE REACTOR BUILDING ATMOSPHERE (1/2)

	FRM II		OPAL
	prompt release	delayed release	
Kr, Xe	1		1
I	$5\ 10^{-4}$	$5\ 10^{-6}$ /h	0.5
Br	$5\ 10^{-4}$	$5\ 10^{-6}$ /h	
Cs	$1\ 10^{-5}$	$5\ 10^{-7}$ /h	0.01

RELEASE FRACTIONS FROM THE POOL WATER INTO THE REACTOR BUILDING ATMOSPHERE (2/2)

	OSIRIS		RSG-GAS
	prompt release	delayed release	
Rb	0		$1\ 10^{-5}$
Te	$5\ 10^{-4}$	$5\ 10^{-6}$/h	$1\ 10^{-5}$
Ru	0		
Sr	0		
Actinides	0		

RELEASE FRACTIONS IN THE CASE OF CORE MELTDOWN FOLLOWING A LOCA (1/2)

	HIFAR	RHF	SAFARI-1
Noble gases	1	1	1
I	0.3	0.8	1
Br		0.8	
Cs	0.3	0.8	0.163
Te	0.01	0.8	0.192

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RELEASE FRACTIONS IN CASE OF CORE MELTDOWN FOLLOWING A LOCA (2/2)

	HIFAR	RHF	SAFARI-1
Rb	0.3	0.01	
Ru	0.01	0.1	0.005
Ba, Rh, Sr		0.1	
Actinides		0.01	0.1
Other		0.01	

DISPARITIES IN THE DATA USED FOR SOURCE TERM EVALUATIONS (2/3)

There is a need to:

- **Examine the possibility of adopting a single conservative database on release fractions to be used for research reactors.**

DISPARITIES IN THE DATA USED FOR SOURCE TERM EVALUATIONS (3/3)

There is a need to:

- Complete experimental data on Fission product release fractions for silicide fuel.
- Get such data for the U-Mo fuel currently being qualified for Research reactors.

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DISPARITIES RELATING TO THERMAL-HYDRAULIC CALCULATIONS (1/2)

- Diverse thermal-hydraulic codes, originally drawn up for power reactors, were adapted and used, in the safety analyses for research reactors.
- Disparities in the mathematical models and correlations used in the different codes.
- Disparities in their validation levels.

DISPARITIES RELATING TO THERMAL-HYDRAULIC CALCULATIONS (2/2)

Suggestion for:

- Experimental validations with the use of a fuel element instrumented with thermocouples.
- Comparative calculations with different codes for a « reference core » case.

CONCLUSION (1/2)

It is important to:

- **Examine and solve the disparities related to fission product release fractions.**
- **Investigate the possibility of elaborating a common database on this subject.**

CONCLUSION (2/2)

- Obtain experimental data on fission product release fractions, in particular for U-Mo

- Continue the validation work concerning computational tools.