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

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1. INTRODUCTION

There are at present 272 operational research reactors in 58 countries. The different reactors have a wide disparity of design, power levels and operating modes. The risks related to these reactors and the safety levels also differ widely.

The activities carried out within the framework of international organizations and the various technical exchanges between safety organizations from different countries have contributed to ensuring some consistency between the safety principles adopted for research reactor design or safety reassessments. However, in some cases, the approaches and analysis methods, data and computational tools used to demonstrate the safety of similar research reactors show important disparities, which may result in different conclusions as to the safety level of these reactors.

This paper concentrates only on pool-type and tank-type research reactors using aluminide and silicide fuels, the most frequently used.

2. USE OF DIFFERENT METHODS IN THE SAFETY ANALYSIS

The purpose of the safety analysis is to demonstrate that the safety requirements such as ensuring the integrity of the radioactive products containment barriers are met for the different postulated internal and external initiating events and in the different operating conditions. The demonstration of the efficiency of each barrier in normal operating conditions and in accident conditions must take account of the measures related to prevention, monitoring, protection and safety actions.

Two complementary methods are used to perform the safety analysis of research reactors. A brief reminder is given below.

2.1. The deterministic method

The deterministic safety analysis method is the most frequently used method at present. It consists mainly in studying a limited number of events that are selected and analyzed in accordance with rules that aim at ensuring the conservatism of the safety analysis results. In general, deterministic analyses are based on a list of Postulated Initiating Events and cover the related Design Basis Accidents in greater detail. The selection of the accidents to be analyzed is often based on the engineer's experience and judgment. In most cases, the accidents are postulated without the need to precisely identify their causes and the analyses performed mainly concern the consequences of these accidents.

The deterministic analyses also take account of multiple failures in the most serious DBA scenarios in order to obtain core fuel degradations causing the release of fission products in the primary coolant system.

For research reactors, the envelope accidents analyzed in different safety reports are named differently: Design Basis Accident (DBA), Reference Accident, Maximum Credible Accident (MCA) and Maximum Hypothetical Accident (MHA).

In general, accidents that are more serious than DBAs are known as Beyond Design Basis Accidents (BDBA). BDBA analyses are taken into account for the purpose of emergency planning and accident management. Therefore, they aim in particular at obtaining a very conservative determination of the source terms and the corresponding radiological impacts. The disparity between the DBAs and the BDBAs selected for certain research reactors is discussed in section 3.

2.2. The probabilistic method

The probabilistic safety analysis method, often used as a complement to the deterministic method, has the advantage of being able to take into account common failure mode analyses, and judging the relevance of the list of events selected for the deterministic safety analysis methods. However, the conclusions of the probabilistic analyses must be considered with caution, notably because of the uncertainties in the generic reliability data that is presently available for research reactors, and that are not necessarily valid for a given facility.

It should be noted that the safety analyses of French research reactors are based mainly on a deterministic approach, with, however, probabilistic evaluations concerning external events such as earthquakes, aircraft crashes, explosions, etc.

3. DISPARITIES IN THE TYPES OF ENVELOPE ACCIDENTS

The envelope accidents taken into account in the safety analyses for similar research reactors operating in different countries cover a range including either the meltdown of one fuel element or the total or partial meltdown of the reactor core. In this latter situation, the percentage of core meltdown is often difficult to justify.

For accidents due to rapid reactivity insertions resulting in core fuel meltdown, the mechanical effects of the interaction between the molten fuel and the coolant water are not always taken into account consistently, especially as regards the dimensioning of the pool and the containment. In particular, the possible consequences on the reactor containment building of internal missiles likely to be projected as a result of steam explosion during accidents of this type have not been systematically analyzed. Moreover, the same type of reactivity accident leading to steam explosion is considered as a DBA for some reactors, but for other reactors it is considered as a BDBA, with no clear justification. These disparities are illustrated in table 1 showing a synthesis of the available information on envelope accidents taken into account for certain research reactors. Some information was given directly by the operating organizations, while other information was taken from the safety analysis reports of the reactors concerned.

Table 1: Envelope accidents taken into account in the safety analysis

Reactor	Reactor type	Power (MW)	Location	Fuel	Envelope accident

BR2	Tank	100	Belgium (Mol)	- U-Al enriched to 93%	- 200 MJ reactor excursion leading to meltdown of the core followed by a water/aluminum interaction (DBA)
FRM-II	Tank	20	Germany (Garching)	- U_3Si_2 enriched to about 90%	- Complete meltdown of the core due to the loss of primary cooling system or to a reactivity accident with failure of shutdown system (BDBA)
HIFAR	Tank	10	Australia (Lucas Heights)	- U-Al enriched to about 60%	- Complete meltdown of the core due to the loss of primary cooling system (Maximum Credible Accident) - Complete meltdown of the core with unsealed containment (BDBA)
OPAL	Pool	20	Australia (Lucas Heights)	- U_3Si_2 enriched to 19.75%	- Melting of 36 U-Mo targets in the reactor due to a loss of coolant flow (BDBA) - Melting of 3 fuel plates due to partial blockage of coolant channels in a fuel assembly (BDBA)
OSIRIS	Pool	70	France (Saclay)	- U_3Si_2 enriched to 19.75%	- 135 MJ reactor excursion leading to complete meltdown of the core followed by a water/aluminum interaction (DBA)
RHF	Tank	57	France (Grenoble)	- U-Al enriched to 93%	- 135 MJ reactor excursion leading to complete meltdown of the core followed by a water/aluminum interaction (DBA for pool and containment design) - Uncovering and complete meltdown of the core (DBA)
RSG-GAS	Pool	30	Indonesia (Serpong)	- U_3Si_2 enriched to 19.75%	- Meltdown of a fuel element caused by a coolant channel blockage (DBA) - ATWS leading to the meltdown of 5 fuel elements (BDBA)
SAFARI-1	Tank	20	South Africa (Pelindaba)	- U-Al enriched to 87%-93%	- Complete meltdown of the core accompanied by the loss of all ventilation systems (Maximum Hypothetical Accident)

It should be mentioned that the Design Basis Accident taken into account up to now in France, for pool type research reactors using aluminide and silicide fuels and having a potential core reactivity exceeding 2% Δk/k, is a BORAX type explosive reactivity accident. The main assumptions associated with this accident were:

- Complete core meltdown under water,
- An energy release of 135 MJ, including 9% in the form of mechanical energy participating in deformation and destruction of the internal structures and the ejection of a water column outside the pool. The mechanical energy released by the destruction of experimental devices (hot and cold neutron sources, pressurized irradiation loops, etc.) is also added to the abovementioned value.

These assumptions were used prescriptively for the design of the reactor pools and containment buildings without considering in a detailed manner the scenarios that may lead to a fast and significant reactivity injections.

Until now, the value of 135 MJ has been adopted for the different French research reactors, without taking into consideration their other characteristics (core size, core composition, safety engineered features, etc.). Analyses are currently in progress at the IRSN with the aim of obtaining more precise data on the consequences of explosive reactivity accidents for the purposes of the safety assessments to be performed, in particular on the Jules Horowitz reactor (RJH ; 100 MW) to be constructed at Cadarache.

The above discussion highlights the need for a certain harmonization of the types of envelope accidents to be taken into account for research reactors with similar technical characteristics. For the sake of clarity, the terminologies used in the safety analysis reports to designate these envelope accidents should also be harmonized.

4. DISPARITIES IN THE DATA USED FOR THE SOURCE TERM EVALUATION

The evaluation of the source term used in the assessment of the radiological consequences of accidents leading to fuel damage (cladding failure or meltdown) requires knowledge of the kind and extent of fuel damage and the release pathways and amounts of fission products released from damaged fuel into the reactor building atmosphere. The second phase in assessing the radiological consequences consists in determining the release of fission products from the reactor building into the environment and the doses at different distances from the facility. In the case of certain reactors, the health detriment in the exposed population (risk of radiation-induced cancer) was assessed. These elements, which depend in particular on the specific characteristics of the reactor building (leaktightness) and the associated ventilation (flowrate, filtration system efficiency) and the specific characteristics of the site, are evaluated on a case-by-case basis.

In the case of fuel meltdown under water, the fission products are released into the pool water, from which a fraction is released into the reactor building atmosphere instantaneously then in a delayed manner. In the case of fuel meltdown in the air, the fission products are released directly to the reactor building atmosphere.

Disparities exist in the hypotheses used to determine the release to the environment. These mainly concern whether the deposition and resuspension of fission products on surfaces are taken into account and the effectiveness of the filtration system.

4.1 Fuel meltdown under water

Some differences were noticed in the release fractions of fission products taken into account for the source term evaluations concerning similar research reactors. Tables 2 and 3 below show these differences for the case of fuel meltdown under water. In these tables, the absence of data for certain nuclides does not necessarily mean that these nuclides were not taken into account in evaluating the source term.

Table 2: Release fractions from molten fuel into pool water

	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
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	

Table 3: Release fractions of fission products from water pool into the reactor building atmosphere

	FRM-II		OPAL	OSIRIS		RSG-GAS
	(prompt release)	(delayed release)		(prompt release)	(delayed release)*	
Kr, Xe	1		1	$5\ 10^{-2}$	$2\ 10^{-2}$/h	1
I	$5\ 10^{-4}$	$5\ 10^{-6}$/h	0.5	$5\ 10^{-4}$	$5\ 10^{-6}$/h	$5\ 10^{-4}$(**)
Br	$5\ 10^{-4}$	$5\ 10^{-6}$/h		$5\ 10^{-4}$	$5\ 10^{-6}$/h	$5\ 10^{-4}$
Cs	$1\ 10^{-5}$	$5\ 10^{-7}$/h	0.01	$5\ 10^{-4}$	$5\ 10^{-6}$/h	$1\ 10^{-5}$
Rb	$1\ 10^{-5}$	$5\ 10^{-7}$/h	0.01	0		$1\ 10^{-5}$
Te	$1\ 10^{-5}$		0.01	$5\ 10^{-4}$	$5\ 10^{-6}$/h	$1\ 10^{-5}$
Ru	$1\ 10^{-5}$		0.01	0		
Sr	$1\ 10^{-5}$			0		
Actinides	$1\ 10^{-5}$			0		

* An additional transfer by evaporation was also taken into account.

(**) $5\ 10^{-2}$ for organic iodine whose proportion is taken to be equal to 50% of the released iodine.

It should be noted that the disparities mainly concern the release fraction values used for noble gases, iodine, cesium, tellurium and actinides.

As regards the OSIRIS reactor, the values adopted for the iodine and cesium release fractions are based on evaluations made following the meltdown of six U-Al fuel plates in the SILOE reactor in 1967, which was attributed to the loss of the coolant flowrate at the inlet of a fuel element.

4.2. Core meltdown following a LOCA

Table 4 below shows the disparities between the release fractions used for the HIFAR, RHF and SAFARI-1 reactors.

Table 4: Release fractions in the case of core meltdown following a LOCA

	HIFAR	HIGH FLUX REACTOR	SAFARI
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
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	

These disparities concern the different nuclides, except the noble gases.

4.3. The need for consistency

It would be useful to look into the cause of the abovementioned disparities and to examine the possibility and the usefulness of adopting a single conservative database on release fractions, that could be used in source term evaluations for fuel meltdown accidents in the different research reactors. This type of analysis must consider the different experimental results available, which show that the release fractions depend in particular on fuel burn-up, the maximum temperature reached by the fuel and the ambient medium (water, air, steam-air, etc.). In this respect, we must point out that there is very little experimental data available for U_3Si_2 type fuel and that experiments on this subject will have to be performed for the U-Mo fuel currently being qualified for research reactors.

5. DISPARITIES RELATING TO THERMAL-HYDRAULIC CALCULATIONS

Different thermal-hydraulic calculation codes originally drawn up for power reactors have been adapted for use in the safety analyses of research reactors in normal operating conditions and in transient and accident conditions. The mathematical model and correlations used in these codes and their degree of validation in the specific conditions of research reactors show a certain level of disparity.

In this context, we should emphasize the interest of performing an experimental validation of the calculation results for the fuel cladding temperatures with the use of a fuel element instrumented with thermocouples, and the usefulness of performing comparative calculations with different thermal-hydraulic codes for a reference core case.

CONCLUSION

In the present situation, there is a need to harmonize the accident analyses relating to research reactors and to improve their technical consistency. Efforts must be made to analyze the disparities existing in the fission product release fractions used for different reactors and to examine the usefulness of adopting a common database that could be used for evaluating the radiological consequences of core fuel meltdown accidents. In this respect, we must emphasize the importance of acquiring experimental data on the release fractions, in particular for the U-Mo type fuel that is currently being qualified.

As concerns thermal-hydraulic safety analyses, it is necessary to continue to validate the computational tools used. In fact, the availability of recognized and validated tools and agreed rules and methods would enhance the consistency of the safety analyses for research reactors.