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Review of experimental database to support nuclear power plant safety analyses in SGTR and LOCA domains

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ABSTRACT

In the framework of the EU R2CA project the available experimental databases were reviewed to support nuclear power plant safety analyses in SGTR and LOCA domains. The review focused on the phenomena related to fuel failure, fission products release from the fuel rods and activity transport up to the environment. Furthermore, it was shown that the phenomena were covered by different scale facilities and different experimental procedures for several reactor designs and materials. Among the tests several separate effect tests and integral tests are listed and some NPP measurements were also included. It was concluded that the reviewed database, which includes more than forty experimental programmes and measurement series can be considered as a reliable basis to support the development and validation of numerical models for SGTR and LOCA safety analyses.

1. Introduction

The safety analyses of the nuclear power plants (NPP) generally includes the numerical simulation of loss-of-coolant accident (LOCA) and steam generator tube rupture (SGTR) scenarios in design basis accident (DBA) and design extension conditions (DEC). Both scenarios could result in the release of radioactive fission products. For this reason, their environmental impact must be evaluated.

In order to carry out reliable prediction of the consequences of NPP incidents and accidents, the numerical tools must have high fidelity models, which are developed considering measured data and which are validated against experiments covering wide range of representative parameters.

The European Union (EU) Reduction of Radiological Consequences of design basis and extension Accidents (R2CA) project targets at the development of harmonized methodologies, development of innovative management approach and application of safety devices for the evaluation and for the reduction of the consequences of DBA and DEC-A events in operating and foreseen nuclear power plants in Europe. The project focuses on the analyses of LOCA and SGTR events in different NPP types.

As a starting point of the EU R2CA project, the LOCA and SGTR related experimental programs were reviewed in order to support the code validation and development activities. The OECD NEA Working Group on Fuel Safety published a state-of-the-art report on nuclear fuel behaviour in LOCA conditions (OECD NEA, 2009), which included detailed information on in-pile and out-of-pile LOCA tests. A detailed overview on experimental programs on core melt progression and fission product release was conducted by Lewis et al., 2008. The experimental programmes and experience from reactor transients on in-vessel core degradation in water cooled reactors (BWR, PWR and VVER)

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Nomenclature

ACRR	Annular Core Research Reactor
AEKI	Atomenergia Kutatóintézet
ANL	Argonne National Laboratory
APROS	Advanced Process Simulation Software
ARTIST	AeRosol Trapping In Steam generator
ASTEC	Accident Source Term Evaluation Code
ATHLET	Analysis of Thermal Hydraulics of Leaks and Transients
BIP	Behaviour of Iodine Project
BWR	Boiling water reactor
CD	Core degradation
COCOSYS	COntainment COde SYStem
CODEX	Core degradation experiment
CRL	Chalk River Laboratories
DBA	Design basis accident
DEC-A	Design extension conditions without significant fuel degradation
DEC-B	Design extension conditions with core melting
EDF	Electricité de France
FIRST-Nuclides	Fast/Instant Release of Safety Relevant Radionuclides from Spent Nuclear Fuel
FGR	Fission gas release
FP	Fission product
FRAPCON	Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup
FRAPTRAN	Computer Code for the Transient Analysis of Oxide Fuel Rods
FSCB	Full Scale Containment Blowdown experiments
FUMEX	Fuel Modelling at Extended Burnup
EK	Centre for Energy Research
EU	European Union
IAEA	International Atomic Energy Agency
IFA	Instrumented Test Assembly
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
LOCA	Loss of coolant accident
LOFT	Loss of fluid test
MFPR-F	Model for Fission Product Release - France
MTA	Magyar Tudományos Akadémia
MRBT	Multi-Rod Burst Tests
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
NRU	National Research Universal reactor
NUGENIA	The Nuclear Generation II & III Alliance
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory
PBF	Power burst facility
PWR	Pressurized water reactor
REBEKA	Reactor typical Bundle Experiment Karlsruhe
R2CA	Reduction of radiological consequences
RING	Release of iodine and noble gases
SGTR	Steam generator tube rupture
SNL	Sandia National Laboratories
STEM	Source Term Evaluation and Mitigation
THAI	Thermal-hydraulics, Hydrogen, Aerosols and Iodine
VVER	Water-cooled water-moderated energetic reactor
WGFS	Working Group on Fuel Safety

during severe accidents were summarised in a NUGENIA technical report (Haste et al., 2018). In the present review it was intended to cover both LOCA and SGTR scenarios without covering those experimental programs which focused on severe core damage falling in DEC-B category. The review also focused on the inclusion of the most recent experimental programs with new data on the related phenomena.

The authors considered experimental programs from national and international projects as well as information from the open literature. In most of the cases, some R2CA partners were involved in these test series or participated in the evaluation of the data.

This paper reviews the LOCA and SGTR experimental databases from the point of view of phenomena covered by the experiments (Section 2.), characterisation of test conditions (Section 3.) and utilisation of measured data (Section 4.).

2. Phenomena covered by the experiments

The review of phenomena that take place in SGTR and LOCA scenarios was grouped into three main categories characterising the fuel failure mechanism, activity release from the fuel rods and activity transport to the environment.

2.1. Fuel failure in LOCA events

It is expected that the loss of fuel integrity in LOCA scenarios starts with plastic deformation of the cladding, leading to ballooning and burst. The zirconium oxidation by steam at high temperatures may result in the embrittlement of cladding tubes and due to mechanical and thermal loads brittle failure may take place. The water quenching of the dry core may play an important role in the degradation process. The fragmentation and dispersal of fuel pellets may cause release of fissionable isotopes into the primary coolant. The fuel fragments with undefined and unpredictable geometry may have an effect on core coolability.

Large number of separate effect tests was performed in different laboratories to map cladding burst conditions. The EDGAR tests (Forgeron et al., 2000), the COCAGNE tests (Dominguez et al., 2022), the REBEKA tests (Erbacher et al., 1990), the AEKI/MTA EK tests (Hózer et al., 2005), the JAERI and JAEA tests (Nagase and Fuketa, 2006), the MRBT tests (Crowley, 1982), the UK burst tests (Hindle and Mann, 1982), the Russian burst tests (Yegorova, 1999), the ANL burst tests (Billone et al., 2008), the Studsvik LOCA tests (Askeljung et al., 2013) and the EDF burst tests (Thieurmel et al., 2019) provided detailed information on cladding ductile failure due to high internal pressure at high temperatures typical for LOCA scenarios. Single rods with and without pellets, and small fuel bundles were used in those tests. Different irradiated and non-irradiated cladding alloys were tested with internal or external electrical heating. In some cases, the samples were precharged with hydrogen or pre-oxidised in steam. The effects of temperature increase and pressurisation rates were also investigated in isobar and isothermal test series. 1409 data points were collected from the above listed experimental series and were selected by IRSN for the definition of burst criteria in numerical models. The measured pressure difference on the cladding wall is shown in different colours corresponding to the scale on the right hand side of Fig. 1. From the measured data the engineering stress associated with clad burst was derived, which shows a strong decrease with the increase of temperature.

In the traditional burst tests pressure and temperature histories are measured and the final deformation of the ballooned tubes are determined. In order to provide more detailed information on the ballooning and burst process, new measurement techniques have been introduced, which could support the development of finite element multi-dimensional models. High speed cameras were applied to track the very fast process of the formation of burst opening in the cladding (Nagy

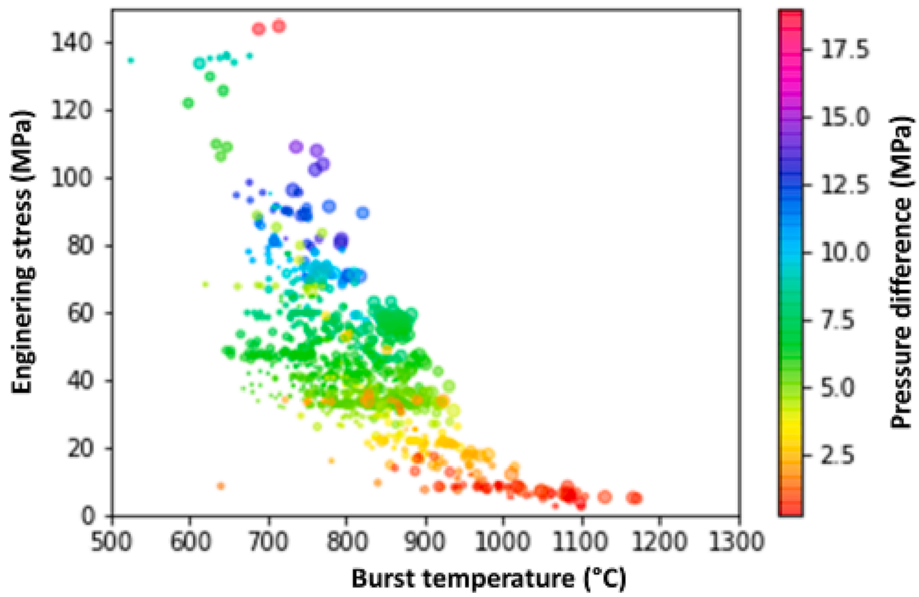

Fig. 1. Engineering stress and pressure difference at burst as function of temperature in separate effect tests.

et al., 2018). In the EIGEN experiments the full-field multi-dimensional deformation of Zircaloy-4 cladding tubes was measured in-situ and real-time (Kim et al., 2020). Laser engraved specimens with digital image correlation technique was used to record in-situ the change of cladding geometry and quantify the strain during burst experiments (Kane et al., 2022). Infrared observation in high temperature tests showed that during crack propagation, the crack tip temperature is significantly higher than the temperature of the rest of the tube (Nagy et al., 2021).

Cladding burst was also observed in out-of-pile integral tests facilities. Electrically heated fuel bundles were used in the CORA (Hofmann et al., 1997), QUENCH-LOCA (Stuckert et al., 2020), CODEX-LOCA (Hózer et al., 2021) and PARAMETER (Salatov et al., 2003) integral tests, which simulated LOCA scenarios with different conditions. Some integral tests with low internal pressure in the fuel rods indicated that during DBA conditions representatives for some reactor designs, cladding burst did not take place (Hózer et al., 2021; Salatov et al., 2003). However, the simulation of DEC conditions normally leads to cladding burst even with low internal pressure, since the cladding temperatures reach high values. The integral tests helped characterising the effect of axial power and temperature profiles on the development of ballooning and burst.

The in-pile tests demonstrated that cladding burst observed in out of-pile test may happen with real irradiated fuel rods under accident conditions. The PBF (Broughton et al., 1983), FR-2 (Karb et al., 1980), PHEBUS-LOCA (Grandjean, 2005), Halden LOCA (Kolstad et al., 2013), NRU MT-4 (Wilson et al., 1993), LOFT LP-FP (Fell and Modro, 1990) and FLASH (Bruet et al., 1993) tests were performed in research reactors. Data on cladding burst in NPPs was not found in the review. However, the event in a cleaning tank analysed in the OECD-IAEA Paks Fuel Project (Hózer et al., 2010) showed burst of fuel rods in power plant fuel assemblies at elevated temperatures. Focusing on in-pile tests results and taking into account a number of separate effect tests with irradiated fuel samples an empirical correlation for fuel rod rupture was derived which determines the burst temperature as a function of stress, heating rate and hydrogen content in the cladding (Meyer and Wiesenack, 2022).

Zirconium oxidation takes place in all tests where high temperature steam is in contact with the surface of claddings. The fuel transient behaviour codes have reliable correlations for Zr alloy oxidation, which were developed on the basis of oxidation kinetics experiments. However, the prediction of brittle failure of oxidised cladding tubes is still a challenge for the codes. The indication of cladding failure is typically correlated with the corrosion state (oxidation, hydriding) instead of mechanistic simulation of loads associated, for example, with water quench. Brittle failure of highly oxidised cladding tubes was observed in several small scale separate effect tests.

In the CORA, QUENCH-LOCA and CODEX-LOCA tests water quench was applied and the brittle failure was also observed. The fragmentation of NPP fuel assemblies was seen in the OECD-IAEA Paks Fuel Project. These experimental results together with the regulatory limits on peak cladding temperature (PCT) and equivalent cladding reacted (ECR) values could serve as the basis for the prediction of brittle fuel failure. The bundle tests data can be used for the estimation of blockage formation due to ballooning in the bundle, too.

The fragmentation and dispersal observed in the Halden LOCA tests raised important questions on the mechanism and potential consequences fissionable material release into the primary circuit. The fragmentation phenomena were observed also in separate effect tests with irradiated fuel (ANL burst tests, Studsvik LOCA tests) and in a number of earlier in-pile tests (PBF, FR-2, LOFT LP-FP, ACRR (Gasser et al., 1997), FLASH). These tests indicated that after cladding failure, beyond the release of fission products, the dispersal of small fuel fragments may happen. This effect should be taken into account in modern LOCA safety analyses with specific models. If the scope of the safety analysis covers high burnup fuel, the pellet fragmentation, relocation and dispersal should be also taken into account. The results of ANL burst tests, Studsvik LOCA tests and Halden LOCA tests could support the estimation of the amount of released fuel material (Table 1).

2.2. *Fuel failure in SGTR events*

In SGTR scenarios the maximum cladding temperatures are much lower compared to LOCA cases. For this reason, only modest oxidation of cladding is expected and the ductile failure due to ballooning can be excluded. However, if defective fuel rods were in the core, the loads associated with pressure and temperature changes can lead to further degradation in those rods.

The degradation phenomena taking place in defective fuel rods were investigated in the following experimental series. The DEFECT experiments were carried out in the water loops of the Siloe reactor with the aim of measuring and interpreting the release rate of fission gases and iodine under a range of experimental conditions of linear power, defect type and gap dimensions (Seveon et al., 1984). In the experiments fuel

Table 1
Phenomena resulting in fuel failure in LOCA and SGTR events.

PHENOMENA	fuel failure during LOCA					fuel failure during SGTR				
EXPERIMENTS	cladding oxidation	ballooning and burst	brittle failure after heavy oxidation	water quench	fuel fragmentation and dispersal	secondary defect	brittle failure	hydrogen uptake by Zr	local hydriding of Zr clad	water logged fuel rod
Edgar tests	X	X								
COCAGNE tests		X								
REBEKA tests	X	X								
AEKI/MTA EK burst tests	X	X	X							
JAERI and JAEA burst tests	X	X		X						
UK burst tests	X	X								
MRBT (ORNL) burst tests	X	X								
Russian burst tests	X	X								
ANL burst tests	X	X		X	X					
EDF burst tests	X	X		X						
PBF tests	X	X		X	X					
FR-2 tests	X	X			X					
PHEBUS-LOCA test	X	X		X						
Halden LOCA tests	X	X			X					
ACRR (SNL) tests	X				X					
NRU MT-4 test	X	X		X						
LOFT LP-FP tests	X	X		X	X					
FLASH tests (Grenoble, Siloe)	X	X		X	X					
Studsvik LOCA test	X	X		X	X					
CORA tests	X	X	X	X						
QUENCH-LOCA integral tests	X	X	X	X						
CODEX-LOCA integral tests	X	X	X	X						
PARAMETER tests	X	X		X						
MTA EK H uptake test								X		
DEFECT tests with defective fuel							X			X
DEFEX secondary defect test						X	X	X	X	X
Halden IFA-631 secondary degradation test						X				
OECD-IAEA Paks Fuel Project	X	X	X	X						

rods with artificial defects were used. The DEFEX tests were performed in the R2 reactor with the aim to study secondary hydriding of cladding caused by water from a simulated primary defect in test rods (Grounes et al., 1997). Severe hydriding and secondary failure occurred in the high power rods. In the Halden IFA-631 secondary degradation test the primary failure of the test rods were simulated with a special water-ingress device connected to the upper end plug (Wright et al., 2017). Some days after the initiation of primary failure simulation, the hydrogen content in the gas phase significantly increased.

The hydrogen uptake by the cladding during normal operation before the accidents is considered as significant effect on the local embrittlement of cladding tubes in the defective rod. To support the numerical simulation of hydrogen pick-up at reactor operational temperatures the MTA EK H uptake tests (Novotny et al., 2015) were performed (Table 1).

2.3. *Activity release from fuel during LOCA*

After loss of cladding integrity radioactive fission products are released from the damaged fuel rod. The basic part of gap activity in the gas volume of the fuel rod is produced during normal operation as release/migration from fuel pellets. The data from Halden FGR tests (Volkov et al., 2014) are widely used for the model development on fission gas release under operational conditions. During LOCA event the released activity is higher than gap activity. It was observed in the FLASH and LOFT LP-FP integral tests and also in the OECD-IAEA Paks Fuel Project. This additional activity release must be taken into account in the safety analyses.

In the HEVA (Iglesias et al., 1999) GASPARD (Pontillon et al., 2004), VERDON (Gallais-During et al., 2017), VERCORS (Ducros et al., 2013), ITU FP test (Hiernaut et al., 2008), ORNL FP tests (Iglesias et al., 1999) and CRL FP tests (Hunt et al., 1994) small irradiated fuel segments were heated up to high temperatures and the release of different isotopes was recorded. The wide range of experimental conditions covered the most important mechanisms of fission product release, including diffusion in the fuel pellet, vaporization of fission products from the pellet surface and influence of cladding. The observations in the ITU FP tests indicated that the oxidation of the fuel matrix results in the increase of fission product release (Hiernaut et al., 2008). The CRL experiments have shown that hydrogen production from the Zr-steam reaction can significantly reduce the oxygen potential and volatile release (Iglesias et al., 1999). The chemical reactions with fission products may also affect the release kinetics.

Very high temperatures were reached in most of the FP release tests, which are not representative for LOCA conditions. However, the first phases of the tests with moderate pellet temperature provided very valuable data for modelling of the release of gaseous, volatile and semi-volatile fission products.

The FLASH experiments simulated in-reactor fuel behaviour during a LOCA accidental sequence and indicated the importance of fuel burnup on fission product release. In the ACRR tests time-resolved data were obtained on fission product release from irradiated fuel rods under severe accident conditions. The LOFT LP-FP experiments contributed significantly to the knowledge base of fission product release during loss-of-coolant accidents and indicated high fission product release during core reflood in some scenarios.

The high burnup fuel is characterized by increased activity release from the rim region (Table 2).

2.4. Activity release from fuel during SGTR

In case of SGTR event, activity release is expected from defective fuel rods. The gap activity of defective fuel rods is obviously smaller than that of intact rods. The accumulation of gap activity in the intact fuel rods can be estimated on the basis of data from the Halden FGR test series simulating fission product release from the fuel pellets. The estimation of the gap activity in the defective rod needs additional information on the release fraction of fission products from fuel rod.

At CRL an in-reactor research program was carried out with irradiated fuel elements containing various sizes and types of cladding defects, with the variation of linear power and using fuel samples with different burnups (Lewis et al., 1993). The CRL tests provided a better understanding of defective fuel behaviour in nuclear reactor and supported the development of numerical models for power plant applications with unique data. During the SGTR events the power and pressure changes generate the iodine spiking effect. This transient activity release was observed in the DEFECT, DEFEX and CRL tests.

There are some data on the iodine spiking phenomena from VVER (Smiesko et al., 2005; Hózer, 2014) and PWR (Adams and Atwood, 1991) nuclear power plants, which could be used to simulate activity release from defective fuel rods in SGTR events (Fig. 2.). The data points for 1970 s and 1980 s were published on US PWR reactors, while the data after 1998 are originated from VVER units. The 191 points in the figure indicate a decreasing trend in coolant activity concentrations, which was probably a common result of the introduction of lower activity concentration limits and of the decreasing number of defective

Table 2
Activity release from fuel rods during LOCA and SGTR events.

PHENOMENA	**activity release from fuel during LOCA**					**activity release from fuel during SGTR**					
EXPERIMENTS	**noble gas release from the fuel rod - steady state**	**noble gas release from the fuel rod - transient**	**volatile fission product release from the fuel rod - transient**	**semi-volatile fission product release from the fuel rod - transient**	**fission product release from high burnup structure**	**noble gas release from the fuel rod - steady state**	**noble gas release from the fuel rod - transient**	**volatile fission product release from the fuel rod - transient**	**semi-volatile fission product release from the fuel rod - transient**	**leaching of fuel pellets by water**	**fission product release from high burnup structure**
Halden LOCA tests			X								
ACRR (SNL) tests			X	X	X						
LOFT LP-FP tests		X	X	X							
FLASH tests (Grenoble, Siloe)		X	X								
GASPARD tests		X	X		X						
HEVA tests		X	X	X							
VERCORS tests		X	X	X	X						
VERDON tests		X	X	X	X						
ITU FP tests		X	X	X							
ORNL FP tests		X	X	X							
CRL FP tests		X	X	X							
Halden FGR tests	X				X	X					X
FIRST-Nuclides leaching tests										X	
CRL defective fuel tests						X	X	X		X	
DEFECT tests with defective fuel						X	X			X	
DEFEX secondary defect test						X	X			X	
VVER NPP iodine spiking								X		X	
PWR NPP iodine spiking								X		X	
OECD-IAEA Paks Fuel Project		X	X	X							

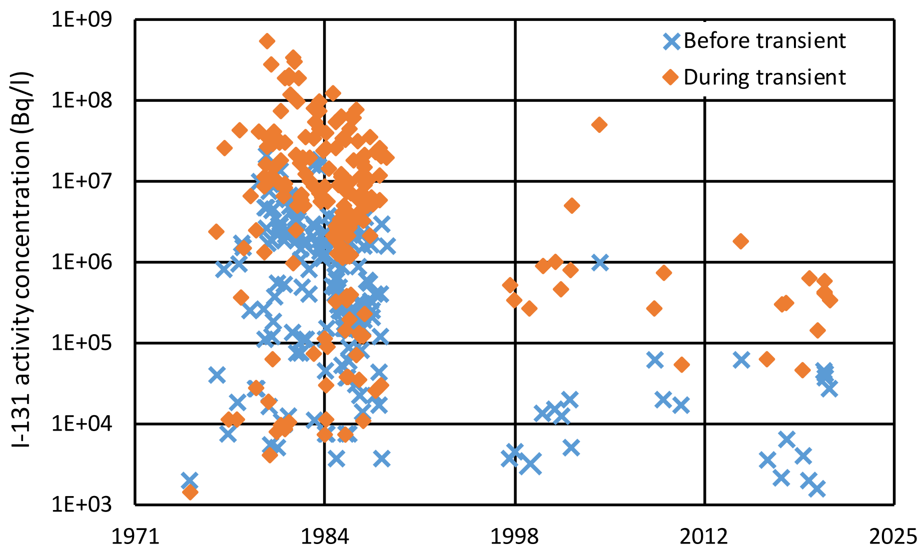

Fig. 2. Collected iodine spiking data from PWR and VVER NPPs.

fuel rods in NPPs. The increase of activity concentration in the coolant during the transient could reach up to two orders of magnitude in some cases compared to the steady state values before the transient.

The activity release can be enhanced by water penetration into the defective rod and the wash-out of radionuclides from pellet and cladding surfaces can increase the released activity. The leaching tests in the FIRST-Nuclides project (Lemmens et al., 2017) provided valuable information on the dissolution of different fuel types and different burnup levels (Table 2).

2.5. Activity transport during LOCA

Once the fuel is damaged, noble gases and volatile fission products (mainly iodine and cesium) from the fuel rod gap get released into the primary coolant. Then most of them, except for noble gases, undergo different processes along their transport that make vapors nucleate into primary particles, the agglomeration of which results in bigger multi-component particles. Along their transportation both vapors and particles are partially removed from the gas carrier by several mechanisms (chemi-sorption, thermo-phoresis, bend/inertial impaction, etc.). Once the airborne fraction reaches containment, other relevant phenomena become dominant: steam condensation usually drives further growth of particles, which became more sensitive to a process like gravitational settling. Still a fraction of the fission products released (particularly iodine), other than noble gases, can enter containment in vapor form. Hence, the eventual radiological inventory and composition available to be released to the environment if containment fails will depend on physical and chemical phenomena (the latter being a transport boundary condition along the entire pathway). It is worth mentioning that phenomena resulting from drastic changes in the prevailing boundary conditions might cause the remobilization of previous deposited fission products, causing a delayed source term into containment.

The data from VERCORS and VERDON tests can be used for the development of transport and deposition and re-evaporation models. The LOFT LP-FP tests offers possibilities for the simulation of fission product behavior in the primary circuit. The results of BIP (Glowa and Moore, 2011), MARVIKEN FSCB (Sokolowski and Kozlowski, 2010), THAI (Gupta et al., 2016) and STEM (Mun et al., 2015) projects provided information on the significant effects of activity transport in different components of the NPP, including the containment. The OECD-IAEA Paks Fuel Project results also included data on the release of gaseous and volatile fission products and the Rivne VVER non-closure of the pressurizer safety valve event (Klyuchnikov et al., 2011) showed a possible activity release path for a full scale nuclear power plant (Table 3).

2.6. Activity transport during SGTR

In case of SGTR event the activity transport after the break continues in the steam generator. In some reactor designs there are possibilities for containment bypass through secondary circuit (e.g. opening of blow-down valves). The activity transport starts from the defective fuel rods, as it was shown in the DEFECT and DEFEX tests. The deposition in the steam generator is a special effect in the SGTR scenario. The fission product transport in steam generators was addressed in the ARTIST project (Lind et al., 2019). SGTR events took place in the Rivne VVER (Groudev et al., 2004) and in the Doel PWR (Aksan, 2008) units. There were limited activity releases, but the simulation of activity transport using the conditions for the two accident scenarios could be considered as representative for real nuclear power plants (Table 3).

3. Characterisation of test conditions

The review in the framework of R2CA project also included the characterisation of test conditions (Tables 4 and 5). Among the test types, separate effect tests (with irradiated or non-irradiated materials), integral tests (in-pile or out-of-pile) and NPP measurements could be specified. Most of the tests included scaled down approach, while the MARVIKEN test and the NPP measurements were full size representative for reactor conditions. The test atmosphere could contain steam, inert gas, hydrogen, or was specified as oxidising or reducing. There was a large variety of tested samples including small cladding samples, small pellet samples, single rods, bundles and non-core materials in irradiated and non-irradiated state. Different cladding types (Zircaloy-4, Zirlo, M5, E110, pre-charged with H, pre-oxidised) and fuel types (UO_2, MOX, high burnup) were used in the reviewed experiments. The heating methods could be nuclear and electric, the latter with internal heaters, furnace or induction method. The fission product transport could be characterised by pH, dose rates, temperatures and by the transport mechanisms of natural circulation, gravity and forced flow. The transport phenomena could be assigned to liquid phase, gas atmosphere or to gas–liquid interface.

Some phenomena were investigated in several facilities and experimental programmes with different test conditions. Even testing the same materials, different final results can be obtained in different facilities. For example, the cladding burst can be influenced by the variation of

Table 3
Activity transport from the reactor core to the environment during LOCA and SGTR events.

PHENOMENA	**activity transport during LOCA**									**activity transport during SGTR**						
EXPERIMENTS	**transport in the primary circuit (from core to break)**	**deposition in the primary circuit, retention by primary circuit components**	**transport in the containment**	**deposition on the containment wall**	**deposition in the containment sump water**	**transport to the environment outside of containment**	**noble gas transport**	**volatile fission product (I, Cs) transport**	**semi-volatile fission product transport**	**transport in the primary circuit (from core to break)**	**deposition in the primary circuit, retention by primary circuit components**	**deposition in the steam generator, retention by steam generator**	**transport to the environment outside of containment**	**noble gas transport**	**volatile fission product (I, Cs) transport**	**semi-volatile fission product transport**
LOFT LP-FP tests	X	X					X	X	X							
VERCORS tests	X	X					X	X	X							
VERDON tests	X	X					X	X	X							
CRL defective fuel tests														X	X	
DEFECT tests with defective fuel														X		
DEFEX secondary defect test														X	X	
BIP			X		X			X				X			X	
MARVIKEN FSCB	X	X	X		X	X		X		X	X	X	X		X	
THAI			X		X			X							X	
ARTIST												X	X			
STEM	X	X	X		X			X	X							
Rivne NPP SG collector cover lift-up										X	X	X	X	X	X	X
Rivne NPP non-closure of the pressurizer safety valve	X	X	X	X	X	X	X	X	X							
SGTR at Doel NPP										X	X	X	X	X	X	X
OECD-IAEA Paks Fuel Project							X	X								

Table 4
Characterisation of experimental programmes (part I.).

TEST CHARACTERISATION	type			scale		atmosphere					tested sample						
EXPERIMENTS	separate effect test	integral test	NPP measurement	scaled down	full scale	steam	inert gas	hydrogen	oxidising	reducing	small cladding sample	small pellet sample	single rod	bundle	non-core material	irradiated	non-irradiated
Edgar tests	X			X		X			X		X		X				X
COCAGNE tests	X			X			X				X		X	X			X
REBEKA tests	X			X		X	X		X		X		X	X			X
AEKI/MTA EK burst tests	X			X		X	X		X		X		X	X			X
JAERI and JAEA burst tests	X			X		X			X		X		X			X	X
UK burst tests	X			X		X			X		X		X	X		X	X
MRBT (ORNL) burst tests	X			X		X			X		X		X	X			X
Russian burst tests	X			X		X			X		X		X			X	X
ANL burst tests	X			X		X	X		X		X	X	X			X	X
EDF burst tests	X			X		X			X		X		X				X
PBF tests		X		X		X			X					X		X	X
FR-2 tests		X		X		X			X		X	X	X			X	X
PHEBUS-LOCA test		X		X		X			X		X	X		X			X
Halden LOCA tests		X		X		X			X				X			X	X
ACRR (SNL) tests		X		X		X	X	X		X					X	X	X
NRU MT-4 test		X		X		X			X					X			X
LOFT LP-FP tests		X		X		X			X					X		X	
FLASH tests (Grenoble, Siloe)		X		X		X	X		X				X			X	
GASPARD tests	X			X		X	X						X			X	
HEVA tests	X			X		X		X					X			X	
VERCORS tests	X			X		X	X	X	X	X			X			X	
VERDON tests	X			X		X	X	X	X	X			X			X	
ITU FP tests	X			X			X		X			X				X	
ORNL FP tests	X			X			X	X	X				X			X	
CRL FP tests	X			X			X	X	X			X	X			X	
Studsvik LOCA test		X		X		X			X				X			X	
CORA tests	X	X		X		X	X		X				X	X			X
QUENCH-LOCA integral tests		X		X		X	X		X					X			X
CODEX-LOCA integral tests		X		X		X			X					X			X
PARAMETER tests		X		X		X	X		X					X			X
Halden FGR tests	X			X									X			X	
FIRST-Nuclides leaching tests	X			X								X					
MTA EK H uptake test	X			X				X		X	X						
CRL defective fuel tests		X		X		X							X			X	
DEFECT tests with defective fuel		X		X		X			X				X			X	
DEFEX secondary defect test		X		X		X							X			X	
Halden IFA-631 secondary degradation test	X			X		X	X	X					X			X	
BIP	X	X		X					X						X		
MARVIKEN FSCB		X			X												
THAI		X		X											X		
ARTIST	X			X			X								X		
STEM	X														X		
VVER NPP iodine spiking			X		X									X		X	
PWR NPP iodine spiking			X		X									X		X	
Rivne NPP SG collector cover lift-up			X		X									X		X	
SGTR at Doel NPP			X		X									X		X	
Rivne NPP non-closure of the pressurizer safety valve			X		X									X		X	
OECD-IAEA Paks Fuel Project			X		X	X			X					X		X	

Table 5
Characterisation of experimental programmes (part II.).

TEST CHARAC-TERISATION	cladding							fuel			heating method					FP release and transport							
EXPERIMENTS	Zircaloy-4	Zircaloy-2	Zirlo	M5	E110	precharged with H	pre-oxidation	UO_2	MOX	high burnup	nuclear	electric	internal	furnace	induction	pH	Dose rate	Temperature	transport by natural circulation/gravity	transport by forced flow	transport in gas atmosphere	transport in liquids	interfacial mass transfer
Edgar tests	X			X		X	X					X											
COCAGNE tests	X					X	X					X											
REBEKA tests	X				X							X											
AEKI/MTA EK burst tests	X				X		X					X		X									
JAERI and JAEA burst tests	X		X	X		X	X							X									
UK burst tests	X											X											
MRBT (ORNL) burst tests	X												X										
Russian burst tests					X							X		X									
ANL burst tests			X			X		X						X									
EDF burst tests	X					X								X									
PBF tests	X							X			X												
FR-2 tests	X							X			X												
PHEBUS-LOCA test	X							X			X												
Halden LOCA tests	X	X		X	X			X		X	X	X											
ACRR (SNL) tests	X							X		X	X												
NRU MT-4 test	X							X			X												
LOFT LP-FP tests	X							X			X						X	X		X	X	X	X
FLASH tests (Grenoble, Siloe)	X							X			X							X					
GASPARD tests	X							X	X	X				X	X			X					
HEVA tests	X							X						X				X					
VERCORS tests	X			X				X	X	X				X	X			X					
VERDON tests	X			X				X	X	X				X	X			X					
ITU FP tests								X		X				X	X			X					
ORNL FP tests	X							X						X				X					
CRL FP tests	X							X						X				X					
Studsvik LOCA test			X					X		X				X									
CORA tests	X				X		X	X															
QUENCH-LOCA integral tests	X		X	X		X						X											
CODEX-LOCA integral tests					X							X	X										
PARAMETER tests					X							X	X										
Halden FGR tests								X	X	X	X		X										
FIRST-Nuclides leaching tests								X	X	X													
MTA EK H uptake test	X				X							X		X									
CRL defective fuel tests	X							X			X						X					X	
DEFECT tests with defective fuel	X							X			X						X					X	
DEFEX secondary defect test		X						X			X												
Halden IFA-631 secondary degradation test		X					X	X			X												
BIP												X				X	X	X		X	X	X	
MARVIKEN FSCB												X						X					
THAI												X						X	X		X	X	X
ARTIST																		X		X	X	X	X
STEM												X						X		X	X		
VVER NPP iodine spiking					X			X			X												
PWR NPP iodine spiking	X							X			X												
Rivne NPP SG collector cover lift-up					X			X			X												
SGTR at Doel NPP	X							X			X												
Rivne NPP non-closure of the pressurizer safety valve					X			X			X												
OECD-IAEA Paks Fuel Project					X			X			X						X						

temperature field around the sample or the oxidation process of zirconium in steam may depend on the steam content in the atmosphere or on the heat-up and cool-down rates.

All these test conditions are important factors, when the scope of validation of computer codes has to be evaluated. The materials of a given reactor design are obviously of interest. The representativeness of test conditions for the analysed accident scenario is also crucial. Details of test characterisation are summarised in Tables 4 and 5.

4. Utilisation of measured data

In order to carry out numerical analyses of an experiment, measured data providing the initial and boundary conditions for the simulation are needed, and some of the measured data could be compared with the calculated results. There are basically two groups of data:

- On-line data (temperature history, pressure history, fission product release monitoring from fuel, FP release monitoring from primary circuit, FP release monitoring from steam generator, FPs concentration in containment atmosphere, FPs concentration in containment sump, FPs deposition on surface, pH, dose rate).
- Post-test examination data (clad deformation, clad corrosion state (oxidation, hydrogen content), FP inventory in the gap, cumulative FP release from fuel rod, cumulative FP release from primary circuit, cumulative FP release from steam generator, cumulative FP release from containment, FPs deposition on surface).

Not all measured data are available in the open literature or in open access databases. The members of the EU R2CA project, however, had access to significant part of the experimental data needed for code validation purposes in the safety analyses in the SGTR and LOCA domains.

The suitability of experimental data for code validation purposes depends on the applied instrumentation and on the amount and resolution of measured data. For example, the number and position of thermocouples in an integral bundle test determine the temperature profile, which is an important boundary condition for the numerical model. Detailed information on the failed fuel geometry may be also required for some models. For this reason, some additional measurements were taken using sophisticated tomography techniques to characterise the 3D geometry of samples from the AEKI/MTA EK burst tests in the framework of the EU R2CA project.

The experimental data from 11 test series were included in the R2CA database: REBEKA, AEKI/MTA EK burst, UK burst, Russian burst, NRU MT-4, CORA, QUENCH-LOCA, CODEX-LOCA, MTA EK H uptake, DEFEX and OECD-IAEA Paks Fuel Project.

Data for other 26 test series were accessible for some of the project partners, but could not be shared within the project with all partners for different reasons: Edgar, COCAGNE, JAERI and JAEA burst, MRBT (ORNL) burst, ANL burst, FR-2, PHEBUS-LOCA, Halden-LOCA, LOFT LP-FP, FLASH, GASPARD, VERCORS, VERDON, Studsvik LOCA, Halden FGR, FIRST-Nuclides leaching, DEFECT, BIP, MARVIKEN, THAI, ARTIST, STEM, VVER NPP iodine spiking, Rivne NPP collector cover lift-off, DOEL SGTR, Rivne NPP non-closure of safety valve.

Some of the experimental data are stored in international databases (e.g. IFPE of OECD NEA (Menut et al., 2000), OECD NEA joint projects, IAEA FUMEX (Killeen et al., 2006)), while some others are stored by the owners of data in private databases. The data of some old test series or experiments carried out outsides of Europe (USA, Russia) were not accessible for the project, but were listed in the database as significant contributions to our knowledge on the SGTR and LOCA related phenomena.

The experimental data were used for the support of R2CA tasks in several areas including model development and validation activities:

- burst tests' data were used for the improvement and validation of DRACCAR, TRANSURANUS and FRAPTRAN transient fuel behaviour codes,
- integral LOCA tests were used for the further validation of DRACCAR, TRANSURANUS and FRAPTRAN transient fuel behaviour codes,
- FP test data were crucial for the testing of fission gas release model in fuel behaviour codes TRANSURANUS, SCIANTIX and MFPR-F (Zullo et al., 2023),
- FP transport experiments provided unique possibilities for the validation of numerical models used in ATHLET-CD, ASTEC, COCOSYS, SOPHAEROS and APROS codes,
- iodine spiking data were used to develop and improve activity release models applied in SGTR analyses in computer codes MELCOR, RING and TRANSURANUS,
- hydrogen uptake data were useful for the simulation of secondary degradation in defective fuel rods in the SHOWBIZ and FRAPCON codes (Feria and Herranz, 2023)

5. Conclusions

The experimental data on reactor incidents and accidents provide the basis for code validation and development activities and serve as the background of current knowledge on related phenomena. The EU R2CA review of experimental databases covered a large number of tests, which characterizes the phenomena taking place during LOCA and SGTR events in PWRs and VVERs. Among the tests several separate effect tests and integral tests are listed, and some NPP measurements were also included.

- Fuel failure during LOCA is well covered by experiments, since many test series have been carried out under different conditions with all important cladding types. In addition to burst type failure – which took place in more than twenty reviewed test series –, the brittle failure of Zr alloy claddings due to thermal and mechanical loads was observed in some tests. The fuel pellet fragmentation and dispersal was indicated by several tests (PBF, FR-2, ACRR, ANL, FLASH, Halden LOCA, Studsvik LOCA).
- Fuel failure during SGTR normally is not expected for intact fuel rods. The related experiments simulate the behaviour of defective fuel rods, which may suffer from secondary defects during the accident. The available experimental data characterise the hydrogen uptake by Zr alloys in the defective fuel rods and its embrittlement effect.
- Activity release from fuel during LOCA conditions was simulated in several separate effect tests (HEVA, VERDON, VERCORS, GASPARD, ITU FP, ORNL FP and CRL FP) and also by integral tests (ACRR, FLASH, Halden-LOCA, LOFT LP-FP). The available experimental data cover a wide range of parameters for different fission products. The Halden fission gas release (FGR) tests are also important for this topic, as they may provide part of the gap source term in case of fuel failure.
- Activity release from fuel during SGTR conditions is supported by iodine spiking experience at PWR and VVER NPPs and by separate effect tests on leaching of fuel pellet samples. The DEFECT, DEFEX and CRL defective fuel rod test series simulated the behaviour of defective fuel rods in research reactor conditions and provided valuable information on secondary defects and water logged fuel rod phenomena.
- Activity transport during LOCA includes several phenomena in the primary circuit and containment, which were investigated in the VERCORS, VERDON, BIP, THAI and STEM projects. The ARTIST project focused on aerosol trapping in steam generators. Some important data can be drawn from the OECD-IAEA Paks fuel project and from the Rivne NPP event with non-closure of pressurizer safety valve.

- Activity transport during SGTR is characterised by complex path configurations, which were studied in the VERCORS, VERDON, BIP, THAI, ARTIST and STEM projects. The BIP, MARVIKEN FSCB and STEM test series simulated fission product transport in the steam generator, too. The primary-to-secondary transport phenomena were also observed in the Doel and Rivne NPP events with steam generator tube rupture and collector cover lift-up, respectively.

The common use of data from small scale separate effect test and integral tests provides possibilities for the improved simulation of several phenomena (e.g. cladding burst, oxidation, hydrogen uptake, activity release and transport) under very different conditions. Most of these conditions can be considered as typical for LOCA and SGTR events, and some of them cover even wider parameter ranges than those that could be expected in the analysed accidents.

The reviewed experimental series were executed mainly with such materials, which are used today or were used earlier in the NPPs. In order to support the introduction of new fuel types and accident tolerant fuel designs, some of the test series could be repeated with new materials, or new experimental programs could be launched to investigate their behaviour under LOCA and SGTR conditions.

CRediT authorship contribution statement

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Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

The authors do not have permission to share data.

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