REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

# REVIEW OF EXPERIMENTAL DATABASES FOR SGTR AND LOCA ANALYSES

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# Introduction

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

The experimental data on reactor incidents and accidents

- background of current knowledge on related phenomena
- basis for code validation and development activities.

EU R2CA project focused on the analyses of two main categories:

- Loss Of Coolant Accidents (LOCA) and
- Steam Generator Tube Rupture (SGTR) accidents.

# Considered test series

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- tests and measurements in national or institutional projects,
- OECD NEA and EU projects,
- publications in scientific journals, conference materials and other open libraries (IAEA and NUREG reports).


*OECD NEA (2009),* ***Nuclear Fuel Behaviour in Loss-Of-Coolant Accident (LOCA) Conditions****. State-of-the-Art Report OECD/NEA - NEA No. 6846,*

*Haste, T., Barrachin, M., Coindreau, O., Fichot, F., Repetto G., Journeau, Ch., Pontillon, Y., Beuzet, E., Torkhani, M., Hózer, Z. P.D.W. Bottomley,P.D.W., Hollands, Th., Austregesilo, H., A Miassoedov, A., Steinbrück, M., Stuckert, J., Bechta, S., Horváth, G.L., Fernandez-Moguel, L., Lind, T., Birchley, J., Buck, M., Pohlner G.(2018)* ***In-Vessel Core Degradation in Water-Cooled Reactor Severe Accidents*** *State-of-the-Art Report Update (CoreSOAR), 1996-2018, NUGENIA Technical Repost, PSN-RES/SAG/2018-00239, ISBN 978-2-919313-13-6.*

*Lewis, B. J., Dickson, R., Iglesias, F. C., Ducros, G., & Kudo, T. (2008).* ***Overview of experimental programs on core melt progression and fission product release behaviour.*** *Journal of Nuclear Materials, 380(1-3), 126-143*

# Considered test series

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

1. Edgar tests
2. COCAGNE tests
3. REBEKA tests
4. AEKI and MTA EK burst tests
5. JAERI and JAEA burst tests
6. UK burst tests
7. MRBT (ORNL) burst tests
8. Russian burst tests
9. ANL burst tests
10. EDF burst tests
11. PBF LOCA tests
12. FR-2 tests
13. PHEBUS-LOCA tests
14. Halden LOCA tests
15. ACRR (SNL) tests
16. NRU MT-4 test
17. LOFT LP-FP tests
18. FLASH tests (Grenoble, Siloe)
19. GASPARD tests
20. VERCORS tests
21. VERDON tests
22. Studsvik LOCA tests

# Considered test series

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

23. CORA tests
24. QUENCH-LOCA integral tests
25. CODEX-LOCA integral tests
26. PARAMETR tests
27. Halden FGR tests
28. FIRST-Nuclides leaching tests
29. MTA EK H uptake tests
30. DEFECT tests with defective fuel rods
31. DEFEX secondary defect tests
32. Halden IFA-631 secondary degradation test
33. OECD BIP
34. MARVIKEN FSCB
35. OECD THAI
36. ARTIST
37. OECD STEM
38. VVER NPP iodine spiking
39. PWR NPP iodine spiking
40. VVER NPP SG collector cover lift-up
41. SGTR event at Doel NPP
42. VVER NPP non-closure of the pressurizer safety
43. OECD-IAEA Paks Fuel Project
44. PSB-VVER and other thermal-hydraulic loops
45. HEVA tests
46. ITU FP tests
47. ORNL FP tests
48. CRL FP tests
49. CRL defective fuel tests

# Fuel failure

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

LOCA:

- large number of burst test series under different conditions with all important cladding types
- brittle failure of Zr alloy claddings due to thermal and mechanical loads observed in some tests
- fuel pellet fragmentation and dispersal in several tests (PBF, FR-2, ACRR, ANL, FLASH, Halden LOCA, Studsvik LOCA).

SGTR

- no failure expected for intact fuel rods
- behavior of defective fuel rods, secondary defects
- hydrogen uptake by Zr alloys in the defective fuel rods and embrittlement

# Activity release from fuel

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

LOCA

- separate effect tests (HEVA, VERDON, VERCORS, GASPARD, ITU FP, ORNL FP and CRL FP)
- integral tests (ACRR, FLASH, Halden-LOCA, LOFT LP-FP)
- wide range of parameters for different fission products
- Halden FGR tests: part of the gap source term in case of fuel failure

SGTR

- iodine spiking experience at PWR and VVER NPPs
- separate effect tests on leaching of fuel pellet samples
- DEFECT, DEFEX and CRL defective fuel rod tests in research reactors: secondary defects and water logged fuel rod phenomena

# Activity transport

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

LOCA:

- includes several phenomena in the primary circuit and containment investigated in the VERCORS, VERDON, BIP, THAI and STEM projects
- important data from the OECD-IAEA Paks fuel project and from the Rivne NPP event with non-closure of pressurizer safety valve

SGTR:

- complex path configurations studied in the VERCORS, VERDON, BIP, THAI, ARTIST and STEM projects
- ARTIST project: focus on aerosol trapping in steam generators
- BIP, MARVIKEN FSCB and STEM test series simulated fission product transport in the steam generator
- primary-to-secondary transport phenomena observed in the Doel and Rivne NPP events with steam generator tube rupture and collector cover lift-up

# Content of the review document

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

Short description on each test series:

- Objectives
- Tested materials, test facility
- Measured para-meters, PIE data
- General conclusions
- References



## 1. Edgar tests

Edgar tests have been performed for several decades at CEA (France) in order to study and model zirconium based alloys creep and burst in LOCA conditions. Extensive description of some series of EDGAR tests can be found in reference [1] and in its update from 2018 (to be published) [2].

### 1.1 Objectives

The main objective of EDGAR tests was to be able to model creep and burst of zirconium alloys pressurized tubes in LOCA conditions for large and intermediate breaks.

### 1.2 Tested materials, test facility

Several materials were tested in the EDGAR facilities but mainly Zircaloy-4 and M5 alloys with as-received and pre-hydrided samples. Tested tubes are 300 mm long, directly heated by Joule effect in steam atmosphere. EDGAR tests were performed on different facilities over decades. The EDGAR-2 facility, is described in [1] as follows it "operates in the temperature range from 350 to 1 400 °C in steady-state (creep tests) or transient conditions. The heating rates can be controlled between 0.1 and 250 K/s. The maximum internal pressure is limited to 200 bars. Pressure rates can be controlled between -10 to +10 bar/s in transient conditions". For pressure steady-state test, the pressure is maintained constant thanks to buffer volumes.

### 1.3 Measured parameters, PIE data

From [1], the measured parameters are mainly temperature, internal pressure and diametral deformation. "The temperatures and the diametral deformation are measured at the same axial location of the specimen, around the middle of the useful part of the specimen. The temperature gradient in the useful part is less than 5 K. The external clad temperature is measured by optical pyrometers or by spot-welded Pt-PtRh type thermocouples. The internal gas temperature is measured by an internal K type thermocouple. The change in the cladding diameter during the test is continuously measured by a laser device that operates, like the pyrometers, through a quartz window fitted in the vessel."
Post-test measurements are made with thin paper ribbons at three cladding elevations, one at the largest diameter in the ballooned section and 20 mm above and below the edges of the burst opening. The total elongation is defined as the circumferential strain at the largest diameter. The largest of the other two diameters defines the uniform elongation. This approach is illustrated on Fig. 1.

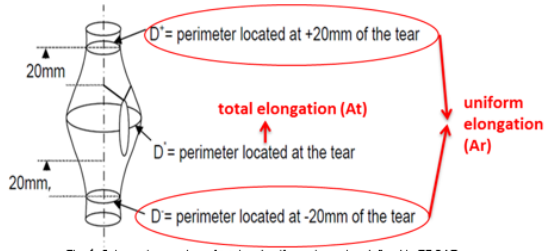



Fig. 1. Schematic overview of total and uniform elongation defined in EDGAR programs.





### 1.4 General conclusions

Only a few data are available in the open literature; the results are not complete for each test but only partial data such as the total elongation versus burst temperature (as illustrated on Fig. 2) or time to burst versus initial hoop stress. Some R2CA partners (IRSN, EDF) own detailed data for some tests but these data cannot be shared with other R2CA partners.

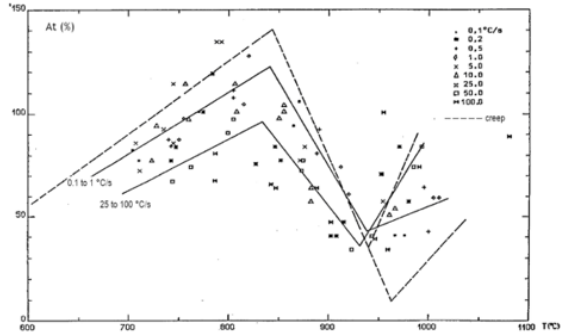



Fig. 2. Total elongation (%) versus burst temperature (°C) in EDGAR tests (Zy-4) [1].

Main conclusions from EDGAR tests and taken from [1] are listed hereafter:

- Very large maximum circumferential strain were observed (higher than 50 % for burst under 900°C),
- Transient tests lead to lower burst strain than creep tests if burst temperature is considered,
- M5 (RXA) seems to lead to lower burst strain than Zy-4 (SRA),
- Hydrogen seems to lower creep resistance and ductility.

These conclusions have to be taken carefully since phase transition from alpha to beta has a significant impact on creep properties and burst characteristics. It is therefore complicated to compare the influence of temperature transients, alloys composition, hydrogen impact and so on.

### 1.5 References

[1] *Nuclear Fuel Behaviour in Loss-of-Coolant Accident (LOCA) Conditions, NEA No. 6846, 2009, NEA.
[2] ALSO update of Nuclear Fuel Behaviour in Loss-of-Coolant Accident (LOCA) Conditions. NEA, To be published
*available in the R2CA database

# Matrices

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

Three large matrices have been compiled to support detailed discussions on

- phenomena,
- test characterization and
- data availability.

| PHENOMENA | fuel failure during LOCA | | | | | fuel failure during SGTR | | | | | activity release from fuel during LOCA | | | | | activity release from fuel during SGTR | | | | | | activity transport during LOCA | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 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 | 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 produc 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 produc release from high burnup structure | transport in the primary circuit (from core to break) | deposition in the primary circuit, retention by primary circuit components | tranposrt in the containment | deposition on the containment wall | deposition in the containment sump water | transport to the enironment outside of containment |
| Halden LOCA 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 | | | | | | | | | | | | | | | | | | | | | | | |
| CODEX-LOCA integral tests | X | X | X | X | | | | | | | | | | | | | | | | | | | | | | | |
| PARAMETER tests | X | X | | X | | | | | | | | | | | | | | | | | | | | | | | |
| Halden FGR tests | | | | | | | | | | | X | | | | X | X | | | | | X | | | | | | |
| FIRST-Nuclides leaching tests | | | | | | | | | | | | | | | | | | | | X | | | | | | | |

# Phenomena matrices – fuel failure

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

**Fuel failure during LOCA:**

- cladding oxidation
- ballooning and burst
- brittle failure after heavy oxidation
- water quench
- fuel fragmentation and dispersal

**Fuel failure during SGTR:**

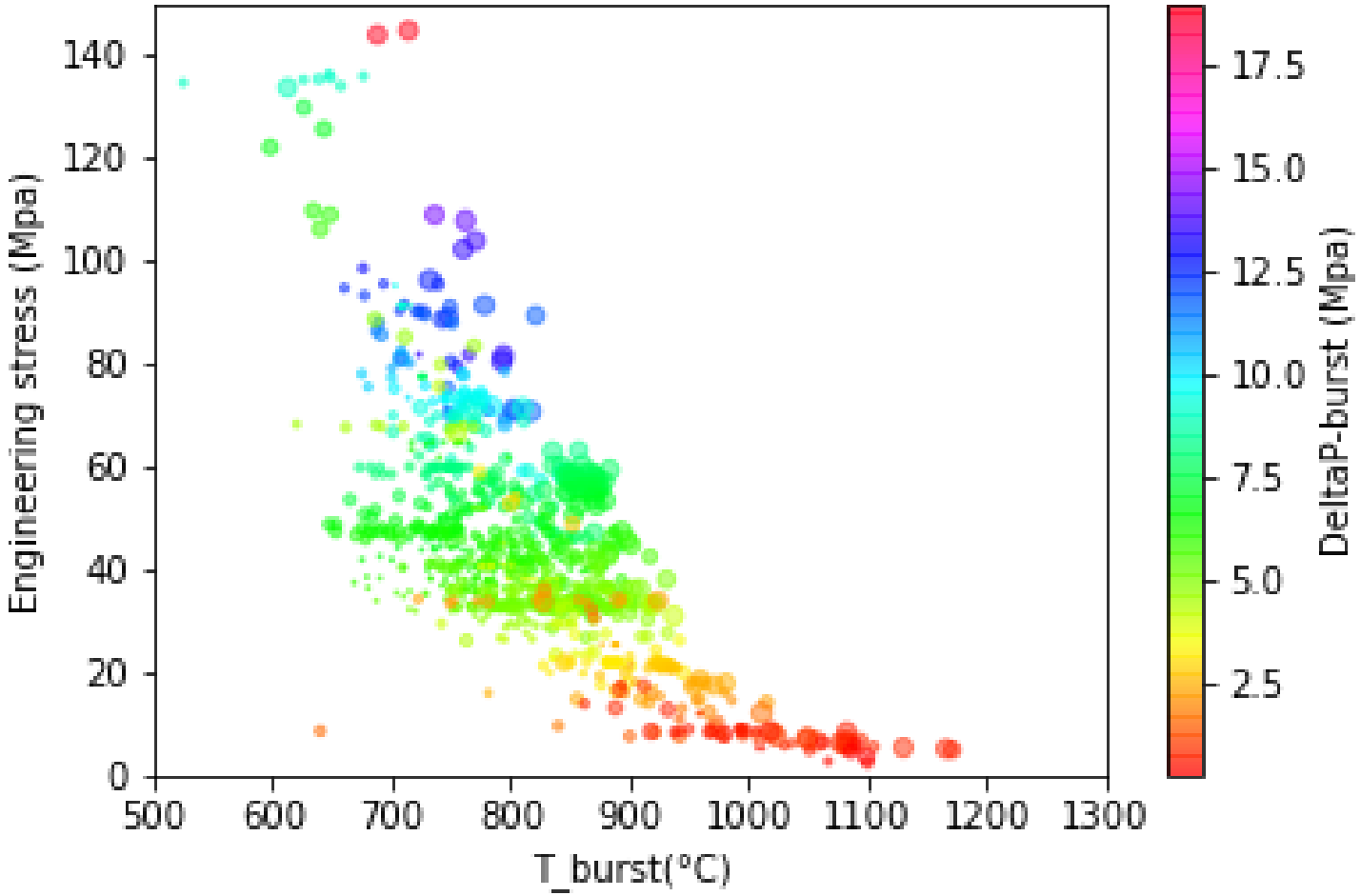
- secondary defect
- brittle failure
- hydrogen uptake by Zr
- local hydriding of Zr clad
- water logged fuel rod

# Ballooning and burst tests

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

Engineering stress and burst pressure as function of temperature in separate effect tests

(1409 data points)



# Phenomena matrices – activity release

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

**Activity release from fuel during LOCA:**

- 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

**Activity release from fuel during SGTR:**

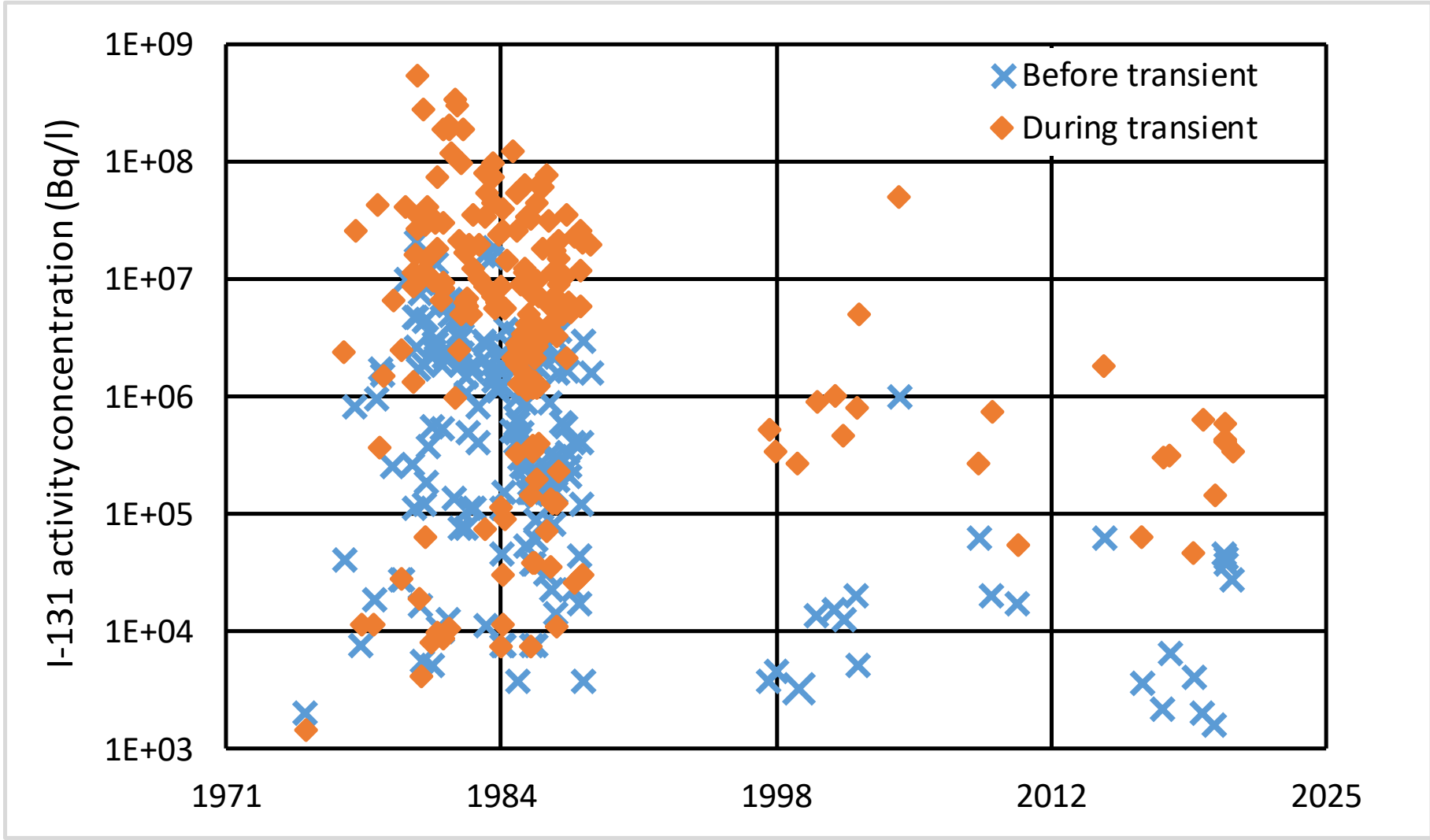
- noble gas release from the fuel rod - steady state
- noble gas release from the fuel rod - transient
- volatile fission product release from the fuel rod - transient
- semi-volatile fission product release from the fuel rod - transient
- leaching of fuel pellets by water
- fission product release from high burnup structure

# Iodine spiking data from NPPs

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

Collected iodine spiking data from PWR and VVER NPPs

(191 data points)



# Phenomena matrices – activity transport

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

**Activity transport during LOCA:**

- 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

**Activity transport during SGTR:**

- 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

# Test characterisation matrices

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- test types (separate effect tests, integral tests and NPP measurements)
- scale (scaled down or full size)
- atmosphere (steam, inert gas, hydrogen, oxidising, reducing)
- tested sample (small cladding sample, small pellet sample, single rod, bundle, non-core material, irradiated, non-irradiated)
- cladding type (Zircaloy-4, Zirlo, M5, E110, pre-charged with H, pre-oxidised)
- fuel type ($UO_2$, MOX, high burnup)
- heating method (nuclear, electric, internal, furnace, induction) and
- fission product transport (pH, dose rate, temperature,transport by natural circulation/gravity, transport by forced flow, transport in gas atmosphere, transport in liquids, interfacial mass transfer).

# Data matrices

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- **Availability** of data (available for the R2CA project, included into the Appendix of the database report, or accessible for some R2CA partners, but could be shared within the project)
- **On-line data** (temperature history, pressure history, fission product release monitoring from fuel, fission product release monitoring from primary circuit, fission product release monitoring from steam generator, etc.)
- **PIE data** (clad deformation, clad corrosion state (oxidation, H content), fission product inventory in the gap, cumulative fission product release from fuel rod, cumulative fission product release from primary circuit, etc.)
- **Validation** (the measured data were used by R2CA partners before the project for code validation purposes or the measured data were intended be used by R2CA partners in the project for code validation purposes)

# Utilisation of measured data

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- **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
- **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

# Conclusions

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- most of the LOCA and SGTR related phenomena were investigated in experimental test series
- behavior of water logged fuel under transient loads was not covered by the database
- common use of data from small scale separate effect test and integral tests provided possibilities for the improved simulation of several phenomena
- reviewed experimental series: materials used today or were used earlier in the NPPs.
- 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 behavior under LOCA and SGTR conditions

# Public document and publication

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

European Commission | Horizon 2020 European Union funding for Research & Innovation

REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

| | |
|---|---|
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| Related Task | T2.1.3. Review of experimental database (MTA-EK) |
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## Review of experimental database to support nuclear power plant safety analyses in SGTR and LOCA domains

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Activity release
Activity transport

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.

https://ec.europa.eu/research/participants/documents/downloadPublic?documentIds=080166e5d44315b1&appId=PPGMS

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