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Experimental Data Requirements for Fuel Performance Modelling

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OECD Nuclear Energy Agency
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Experimental Data Requirements for Fuel Performance Modelling

Expert Group on Reactor Fuel Performance (EGRFP)

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Foreword

Under the guidance of the Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC), the Working Party on Scientific Issues and Uncertainty Analysis of Reactor Systems (WPRS) studies the reactor physics, fuel performance, and radiation transport and shielding in present and future nuclear power systems. The NEA WPRS Expert Group on Reactor Fuel Performance (EGRFP) performs specific tasks associated with fuel performance aspects of present and future nuclear power systems, with a focus on normal operating conditions.

A key objective of the group is to help identify, evaluate and preserve experimental data necessary for developing separate-effect models and integral fuel rod codes and updating validation databases. In this context, the Expert Group monitors, steers and supports the International Fuel Performance Experiments (IFPE) data collection. This working paper on the experimental data requirements for fuel performance modelling was initiated and developed in service of the EGRFP's stewardship of the IFPE.

Acknowledgements

This working paper evolved over almost a decade, engaging a wide number of experts of the EGRFP at various periods. Glyn Rossiter (EGRFP former Chair) initiated and co-ordinated the activity. Experts who contributed to the writing and review of this paper include (* indicates affiliation at the time of contribution):

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List of abbreviations and acronyms

AECL	Atomic Energy of Canada Limited
AGR	Advanced gas-cooled reactor
AOO	Anticipated operational occurrence
BDBA	Beyond-design-basis accident
BWR	Boiling water reactor
CAGR	Commercial advanced gas-cooled reactors
CANDU	CANada Deuteruim Uranium
CEA	French Alternative Energies and Atomic Energy Commission (France)
CNEA	National Atomic Energy Commission (Argentina)
CRP	Co-ordinated Research Project
DBA	Design-basis accident
DBTT	Ductile-to-brittle transition temperature
EDF	Électricité de France
EGRFP	Expert Group on Reactor Fuel Performance (NEA WPRS)
EPMA	Electron probe micro-analysis
FGR	Fission gas release
FUMAC	FUel Modelling in Accident Conditions
FUMEX	Fuel Modelling at Extended Burn-up
HBS	High burn-up structure
HBU	High burn-up
HBWR	Halden Boiling Water Reactor
HCP	Hexagonal close-packed
HRP	Halden Reactor Project
IAEA	International Atomic Energy Agency
IBRAE	Nuclear Safety Institute of the Russian Academy of Sciences
IFA	Instrumented fuel assembly
IFPE	International Fuel Performance Experiments
INR	Institute for Nuclear Research (Romania)
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute (Japan)
JRC-Karlsruhe	Joint Research Centre-Karlsruhe
KIT	Karlsruhe Institute of Technology (Germany)

LOCA	Loss-of-coolant accident
MOX	Mixed oxide
MTA EK	Hungarian Academy of Sciences, Centre for Energy Research (Hungary)
MTR	Material testing reactor
NEA	Nuclear Energy Agency
NINE	Nuclear and Industrial Engineering (Italy)
NRI	Nuclear Research Institute (Czechia)
NRU	National Research Universal
NSC	Nuclear Science Committee (NEA)
NSRR	Nuclear safety research reactor
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory (United States)
PCMI	Pellet-cladding mechanical interaction
PCI	Pellet-cladding interaction
PHWR	Pressurised heavy water reactor
PIE	Post-irradiation examination
PWR	Pressurised water reactor
R&D	Research and development
SCC	Stress-corrosion cracking
SCK CEN	Belgian Nuclear Research Centre
SIMS	Secondary ion mass spectrometry
US DOE	United States Department of Energy
V&V	Verification and validation
VVER	Water-water energetic reactor
WPRS	Working Party on Scientific Issues and Uncertainty Analysis of Reactor Systems (NEA)
XRF	X-ray fluorescence

Executive summary

Reliable prediction of fuel rod behaviour is a basic requirement for safe operation of nuclear power plants, for fuel design, and to enable fuel performance assessments. Therefore, it is essential to develop an understanding of fuel rod performance and to embody this knowledge in computer codes to provide best-estimate or conservative predictions of fuel behaviour. Experimental data are essential for developing separate-effect models and integral fuel rod codes and for updating validation databases. The utilisation of such experimental data is therefore an integral part of the code development and validation process.

This working paper aims to describe the data requirements for fuel performance modelling, and to determine the types of experimental data that can be used to meet these requirements. The paper captures the experience and specialised knowledge of the experts working under the NEA Expert Group on Reactor Fuel Performance (EGRFP) operating under the guidance of the Working Party on Scientific Issues and Uncertainty Analysis of Reactor Systems (WPRS) over the last decade. The paper outlines data requirements for fuel performance codes considering their evolution, the development of new measurement techniques, the availability of new data, and the emergence of new phenomena such as fuel micro-fragmentation during temperature transients at high burn-up. Through this paper, the EGRFP experts aim to advise those who generate and capture experimental information to ensure that experimental outputs can provide a meaningful and complete picture necessary for validation of fuel rod simulation tools. The results will also guide the continuous development of the NEA International Fuel Performance Experiments (IFPE) data collection. The authors consider that the paper establishes data requirement expectations that often exceed the precedent of legacy experimental programmes and has the ambition to advance the quality and completeness of data captured in future experimental programmes.

The working paper focuses on the needs of fuel rod simulation tools for uranium dioxide and (U, Pu) mixed oxide fuel in zirconium-based cladding under light and heavy-water reactor conditions, but aspects discussed may be applicable to other fuel types. It starts from the simulation needs for normal operation and operational transient conditions. Based on the related data sets already available in the IFPE, the paper also draws preliminary conclusions for design-basis accident (DBA) conditions. Although the paper intends to provide a common basis for all fuel rod simulation tools, there is no intention to meet the specific needs of multi-dimensional fuel rod codes such as dedicated boundary conditions that include azimuthal variations.

1. Introduction

Reliable prediction of fuel behaviour is a basic requirement for the safe operation of nuclear power plants, for fuel design, and to enable fuel performance assessments. Therefore, it is essential to develop an understanding of fuel performance and to embody this knowledge in computer codes to provide best-estimate and bounding predictions of fuel behaviour. This, in turn, leads to an increase in operating margins, flexibility in fuel management, and improved economics. The goal is a fully comprehensive understanding of fuel behaviour in both normal and off-normal conditions. From this knowledge, operating rules can be derived to prevent fuel failures and the release of fission products to the environment, and, in extreme cases, to prevent escalation of fuel and core damage and the consequential hazards.

Fuel performance codes have been developed independently in many countries by fuel vendors, utilities, research and development (R&D) organisations and/or regulators (Van Uffelen and Suzuki, 2012). Existing commonly used fuel performance codes, such as TRANSURANUS (Lassmann, et al., 1995) and FRAPCON (Geelhood, et al., 2015), have also been customised by interested parties for their own use. These codes have been developed with differing objectives and for various reactor types; some have been designed as research and development tools, while others are used for fuel design and licensing. Differences exist in both the level of development and the extent of verification and validation (V&V).

Experimental data are essential for developing fuel models and updating validation databases. The utilisation of such data is therefore an integral part of the code development and validation process, which must always be ongoing to ensure codes become, or remain, fit for purpose. This working paper aims to describe the data requirements for fuel performance modelling, and to determine the types of experimental data that can be used to meet these requirements. The results will also guide the continuous development of the NEA International Fuel Performance Experiments (IFPE) data collection. The paper is essentially an update of the report produced in 1995 (Turnbull, 1995), which takes into account the various changes that have occurred in the almost two decades since the original report was written. These changes include the evolution of fuel performance codes, the development of new measurement techniques, the availability of new data, and the emergence of new phenomena consideration in modelling, such as fuel micro-fragmentation during temperature transients at high burn-up. Although the paper intends to provide a common basis for all fuel rod simulation tools, there is no intention to meet the specific needs of multi-dimensional fuel rod codes such as dedicated boundary conditions that include azimuthal variations.

2. Background

2.1. History of IFPE data collection

The International Atomic Energy Agency (IAEA) first organised a Co-ordinated Research Project (CRP) to support fuel performance code development in 1982 (IAEA, 1987). It was then recognised in the early 1990s that further development and improvements were needed in the modelling of nuclear fuel.

In collaboration with the OECD Halden Reactor Project (HRP) in Norway, the IAEA initiated a second CRP on fuel performance code development – FUMEX (Fuel Modelling at Extended Burn-up) – in 1993 (IAEA, 1998). In this CRP, modellers performed blind analysis of experiments carried out in the Halden Boiling Water Reactor (HBWR); the combined predictions were then compared to the in-reactor and post-irradiation examination (PIE) measurements (which included fuel temperature, rod internal pressure and fission gas release) and the modellers had the opportunity to modify and improve their codes.

At around the same time, the NEA Nuclear Science Committee (NSC) set up a Task Force on Scientific Issues of Fuel Behaviour. During discussions at the first meeting in December 1993, it was agreed that there was a need for a better understanding of the underlying basic phenomena. Members were charged with the task of identifying areas of high priority that would benefit from international co-operation, and to prepare a report providing advice on the developments needed – in the form of experiments, data and models – to meet the requirements for a better understanding of fuel behaviour.

One of the high priority actions identified, and later approved by the NSC, was the setting up of an international database of well characterised experiments from different sources as a support for fuel model development and validation. As a first step, it was agreed to prepare a report reviewing existing datasets that could potentially be included in such a database. This report – which was completed in early 1995 (Turnbull, 1995) – reviewed in detail the data available from the OECD Halden Reactor Project, while also briefly mentioning data available from other sources. Considering the experience during the FUMEX exercise, the NEA actively supported this initiative and made available both data and funds for what is now known as the International Fuel Performance Experiments (IFPE) data collection. The initial data were stored in the IFPE data collection by the end of 1995.

The collaboration between the NEA and the IAEA developed further at this time: following the FUMEX CRP, three NEA hosted symposia were held in France on topics that had been identified during the FUMEX CRP as of key importance, namely:

- thermal performance of high burn-up light water reactor fuel (NEA, 1998);
- fission gas behaviour in water reactor fuels (NEA, 2002);
- pellet-clad interaction in water reactor fuels (NEA, 2005).

The IFPE data collection was well established by the year 2000. Experimental data from many countries had been collated and prepared for use, and covered all the major reactor systems, including pressurised water reactors (PWRs), boiling water reactors (BWRs), the Russian-design PWRs (known as water-water energetic reactors [VVERs]), pressurised heavy water reactors (PHWRs, including the CANada Deuteruim Uranium [CANDU] units) and commercial advanced gas-cooled reactors (CAGRs).

In 2002, the IAEA initiated a third CRP on fuel modelling, FUMEX-II (IAEA, 2012), to try to further improve fuel modelling, particularly in the area of fission gas release (FGR) and pellet-cladding mechanical interaction (PCMI). All the experimental data either came from, or was included in, the IFPE data collection. Many of the participants in FUMEX-II were able to provide experimental data from their local organisations or countries, and during the period of the project to 2007 many new datasets were added. Furthermore, the exposure of the IFPE data to many fuel modellers allowed many questions to be asked on the completeness and accuracy of some of the data, and several datasets were updated in answering these queries. At this time, the data used in the first FUMEX CRP were also added to the IFPE data collection to allow fuel modellers who had not participated in this CRP to access the well-described data.

The successful collaboration between the IAEA and the NEA continued through a new IAEA project, "Improvement of Computer Codes used for Fuel Behaviour Simulation (FUMEX-III)" (IAEA, 2013), which ran from 2008 to 2012. This project was wide-ranging, and the fuel modelling teams were able to attempt to model fuel failure due to power ramping, which was made possible through the presence in the IFPE of Studsvik ramp data, originally collected in the mid-1990s and some of which was updated during the project. Further data were added to the IFPE data collection by the participants.

The IAEA CRP on FUel Modelling in Accident Conditions (FUMAC) (2014-2018) aimed to enhance understanding of fuel behaviour during accidents (IAEA, 2019). The CRP involved collecting and analysing well-verified results of accident simulation experiments using modern fuel performance codes, including separate-effect clad ballooning tests (PUZRY from the Hungarian Academy of Sciences, Centre for Energy Research [MTA EK]), out-of-pile single rod loss-of-coolant accident (LOCA) tests (Studsvik 192, 198), in-reactor LOCA tests with single rods (IFA 650.9–11 from the Halden Project), a Karlsruhe Institute of Technology (KIT) out-of-pile bundle LOCA test (QUENCH-L1), and a bundle under severe accident test (CORA 15).

2.2. Description of the IFPE data collection

The aim of the IFPE data collection is to provide, in the public domain, a comprehensive and well-qualified data collection on experiments to support fuel performance model development and code validation. One of the driving forces for the construction of the data collection was the recognition that there had been many important fuel experiments carried out in the previous decades, but that the data were not being fully utilised by the international community and there was the possibility that the information might be lost.

The IFPE data collection is restricted to fuel performance in thermal reactors, with the emphasis on Zircaloy-clad UO_2, $(U,Gd)O_2$ and mixed-oxide (MOX) fuel in water-cooled and water-moderated plants, although the addition of advanced products with fuel and clad variants is not ruled out. Further, efforts are underway to include select fast reactor fuel performance datasets. Experiments encompass both normal and off-normal operation, and include prototypic commercial irradiations as well as experiments performed in material testing reactors (MTRs). To date, the data collection contains about 1 450 individual cases, providing data from both in-reactor and post-irradiation examination (PIE) measurements. The former include fuel centreline temperatures (from fuel thermocouple or expansion thermometer measurements), dimensional changes (from extensometer or diameter gauge measurements) and rod internal pressures (from pressure transducer measurements), while the latter include dimensional changes (from profilometry), FGR (from rod puncturing), cladding oxide thicknesses (generally from eddy current measurements), clad hydrogen contents, fuel grain sizes (from ceramography), fuel density/porosity volume fractions (from fuel density measurements or ceramographic image analysis) and isotopic or

elemental concentration profiles in the fuel pellets (from electron probe micro-analysis [EPMA], secondary ion mass spectrometry (SIMS), or X-ray fluorescence [XRF]).

Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. Of particular interest to fuel modellers are data on fuel temperatures, FGR, fuel swelling, clad deformation (e.g. creep-down, ridging) and pellet-cladding interaction (PCI). Data on these issues are of great value if measured in-pile by dedicated instrumentation; in this respect, the IFPE data collection is fortunate in having access to several diverse experiments.

The first data to be included in the data collection came largely from experiments carried out at the research reactors at Halden, Risø and Studsvik, and covered a wide range of fuel behaviour in normal and transient conditions. In each case, and for all subsequent additions to the data collection, the datasets contain information on the pre-characterisation of the fuel pellets and cladding, the fuel rod geometry, the irradiation history (presented in as much detail as the source documents allow) and any in-pile or PIE measurements that were made. Special emphasis is placed on data relevant to current issues such as behaviour at high burn-up. Besides the compilation and evaluation of the experimental data, the data collection also contains the detailed primary documents from which the data were derived.

The work in assembling and disseminating the IFPE data collection is carried out in close co-operation and co-ordination between NEA, the IAEA and the OECD Halden Reactor Project. The original authors of the data collection spent much time and effort in identifying and collecting suitable data from providers around the world. The time taken to arrange the release of a dataset for public use and to prepare it for inclusion in the data collection often took many years. All these data were analysed, documented, subjected to stringent quality assurance procedures and formatted in a way to be readily usable by fuel modellers.

The data collection is now widely used in about 100 institutions in more than 30 countries. Feedback from users has been essential to ensure that the data collection improves with its use.

Table 1 shows the current contents of the IFPE data collection and notes when a particular dataset was added or updated; also indicated is where cases have been used in the IAEA FUMEX, FUMEX-II and FUMEX-III and FUMAC CRPs. HRP experiments are listed by their instrumented fuel assembly (IFA) number. As the most recent data package, in April 2024 the three LOCA tests examined in the IAEA's FUMAC benchmark exercise (IFA-650.9, 10, 11) were included.

Archives (unprocessed datasets) are also available for:

- the Babcock and Wilcox extended burn-up gadolinia-doped fuel lead test assembly irradiation programme;
- AREVA data on grain boundary fission gas concentrations required for interlinkage;
- the IFA-508 and IFA-515 experiments conducted by the Japan Atomic Energy Research Institute (JAERI) at HRP (where the PCMI behaviour of rods with different cladding thicknesses was investigated by means of a clad diameter rig).

Table 1. Data currently available in the IFPE data collection

Experiment title	Date released / last updated	Scenario	Experiment primary purpose	Reactor or fuel type	IAEA CRPs in which data were used
AEAT-IMC NFB-8 and NFB-34: post-irradiation annealing tests (22 samples)	03/1999 01/2023	Temperature ramp	Fission gas release (FGR)	CAGR	
Atomic Energy of Canada Limited (AECL)-CANDU elements irradiated in National Research Universal (NRU) (36 rods)	02/2000 01/2023	Steady state	Fission gas release	PHWR	FUMEX-III
AECL-CR EXP-BDL-406 (NRU experiments): performance of CANDU natural UO_2 fuel irradiated at low linear powers to burn-ups above 450 MWh/kgU (3 rods)	06/2009 01/2023	Steady state	CANDU fuel	PHWR	FUMEX-III
AEKI Experimental Database of E110 Claddings under Accident Conditions (~ 200 samples)	02/2008 01/2023		Cladding mechanical properties	VVER	FUMAC
Belgonucleaire and the Belgian Nuclear Research Centre (SCK CEN) PRIMO: ramped MOX fuel rod BD8 (1 rod)	05/2007 07/2009	Power ramp	Pellet-clad mechanical interaction (PCMI)	PWR/MOX	FUMEX-III
Belgonucleaire Beznau-1: Irradiated MOX fuel rod M109/D3 (1 rod)	05/2008	Nuclear power plant operation + PIE	MOX fuel	PWR/MOX	
Belgonucleaire Beznau-1: Irradiated MOX fuel rod M501/D10 (1 rod)	10/2011	Nuclear power plant operation + PIE	Fuel behaviour for high Pu MOX	PWR/MOX	FUMEX-III
Belgonucleaire GAIN programme: Ramp tests of $(U,Gd)O_2$ fuel (4 rods)	03/2002 01/2023	Steady state + power ramp	Fuel behaviour for Gd doped fuel	PWR	FUMEX-III
BR-3/DOE High Burn-up Fuel Rod Hot Cell Programme (5 rods)	09/2003 01/2023	Steady state	High burn-up (HBU)	PWR	
CANDU experiment FIO-130 (1 rod)	06/2010 01/2023	LOCA transient	Fuel behaviour under LOCA	PHWR	FUMEX-III
CANDU experiment FIO-131 (1 rod)	04/2007 01/2023	LOCA transient	Fuel behaviour under LOCA	PHWR	FUMEX-III
CANDU IRDMR experiments FIO-118 and FIO-119: fuel behaviour under LOCA conditions (7 rods)	04/2007 01/2023	Power ramp	Rod dimensional response on power	PHWR	FUMEX-III
The French Alternative Energies and Atomic Energy Commission (CEA) defect fuel experiments (8 rods)	02/2000 01/2023	Steady state	Fission gas release for defect fuel	PWR	
CEA defect fuel experiments EDITH-MOX 01: test of failed PWR rod irradiated in SILOE (1 rod)	06/2000	Steady state	Fission gas release for defect fuel	PWR/MOX	
CEA/EDF/FRAMATOME Contact 1 and 2 (3 rods)	03/1999 01/2003	Steady state	Fission gas release; thermal performance; clad creep	PWR	FUMEX-II
CEA/EDF/FRAMATOME PWR and OSIRIS ramped fuel rods (4 rods)	02/2000 01/2006	Power ramp	Pellet-clad mechanical interaction (PCMI)	PWR	FUMEX-III
National Atomic Energy Commission (CNEA) six power ramp irradiations with MOX fuels (5 rods)	10/2000	Power ramp	Pellet-clad mechanical interaction (PCMI)	PHWR/MOX	FUMEX-III
US Department of Energy (DOE)	01/2007	Steady state	Integral	MOX	

Experiment title	Date released / last updated	Scenario	Experiment primary purpose	Reactor or fuel type	IAEA CRPs in which data were used
Fissile Materials Disposition Programme in the Advanced Test Reactor (2 cases)			performance of MOX fuel		
Halden Cases for FUMEX-I (6 rods)	09/2003 01/2023	MANY	Benchmark package	MANY	FUMEX-I
Halden IFA-429 (7 rods)	11/1997	Power cycling + steady state	Gas absorption; fission gas release (FGR); thermal behaviour; power cycling	PWR	
Halden IFA-432 (5 rods)	06/1995 01/1996	Steady state	Fuel integral performance	BWR	
Halden IFA-507: transient fuel temperatures during a power increase (6 rods)	10/2004	Power ramp	Thermal performance	LWR	FUMEX-II
Halden IFA-519.9: PWR rods irradiated to high burn-up (3 rods)	10/2011 09/2020	Power cycling + steady state	Fission gas release; high burn-up structure (HBS)	PWR	FUMEX-III
Halden IFA-533.2 (1 rod)	11/1997	Steady state + power ramp	Thermal performance	LWR	
Halden IFA-534.14 rod 18 and 19 (2 rods)	11/2002 06/2005	Steady state	Fuel grain size; fission gas release (FGR)	PWR	FUMEX-II
Halden IFA-535.5 &.6 (4 rods)	11/1997 01/2023	Power ramp	Fission gas release (FGR)	BWR	FUMEX-III
Halden IFA-562.1 (12 rods)	11/1997	Steady state + power ramp	Thermal performance; pellet-clad mechanical interaction (PCMI)	LWR	
Halden IFA-585: in-reactor creep behaviour of Zircaloy-2 and Zircaloy-4 under variable loading conditions (2 rods)	03/2008	Steady state	Cladding creep	BWR and PWR	
Halden IFA-597: Hollow and solid MOX rods experiments (2 rods)	10/2006	Steady state with uprate	Pellet design effect on MOX fuel; fission gas release (FGR)	PWR/MOX	
Halden IFA-597.3 rods 7, 8 and 9 (3 rods)	11/2002 04/2003	Steady state	Integral experiments data	BWR	FUMEX-II
Halden IFA-629.1: Ramp testing of two pre-irradiated MIMAS MOX rods (2 rods)	10/2011 01/2023	Steady state	fission gas release (FGR); swelling	PWR/MOX	FUMEX-III
Halden IFA-650.1&.2: LOCA testing with fresh fuel (2 rods)	10/2011	LOCA transient	Qualification LOCA test rig	PWR	FUMEX-III FUMAC
Halden IFA-650.9,.10,.11: LOCA testing with irradiated fuel (3 rods)	04/2024	LOCA transient	High burn-up fuel LOCA testing	PWR and VVER	FUMAC
Halden/IMC(UK): Swelling data from ramping CAGR UO_2 fuel in the Halden BWR (13 rods)	10/2003 06/2006	Power ramp	Fuel swelling	Advanced gas-cooled reactor (AGR)	
Halden/JAEA IFA 514/565: MOX fuel irradiation test in the Halden BWR (6 rods)	02/2009 01/2023	Steady state	Fuel creep; fission gas release (FGR)	BWR/ MOX	
Halden/JAEA IFA 591: Power ramp tests of MOX fuel rods pre-irradiated in the Fugen advanced thermal	04/2007 01/2023	Power ramp	Pellet-clad mechanical interaction (PCMI);	MOX	

Experiment title	Date released / last updated	Scenario	Experiment primary purpose	Reactor or fuel type	IAEA CRPs in which data were used
reactor (11 rods)			fission gas release (FGR)		
HATAC in SILOE reactor (2 rods)	11/2005 11/2009	Steady state + Power ramp	Fission gas Fission Fission gas release (FGR); sweeping in rods	PWR	FUMEX-II
High burn-up effects programme (81 rods)	11/1997 01/2023	Steady state	High burn-up (HBU)	LWR	FUMEX-II
Idealised cases to illustrate the functional dependence of FGR predictions (7 cases)	04/2004	Idealised modelling cases	Benchmark package	MANY	FUMEX-II
Institute for Nuclear Research (INR) Pitesti RO-89, RO-51: In-reactor measurement of rod internal pressure (2 rods)	04/2002 01/2023	Steady state	Effect of fuel density on fission gas release (FGR)	PHWR	FUMEX-III
KOLA-3 (32 rods)	09/1996 03/1999	Nuclear power plant operation + PIE	Fission gas release (FGR); rod elongation at PIE	VVER	FUMEX-II
KOLA-3 MIR: Ramp tests in MIR of rods pre-irradiated in Kola-3 (9 rods)	03/2006 10/2011	Steady state + power ramp	Fission gas release (FGR); thermal performance	VVER	FUMEX-II FUMEX-III
NFIR-1 data for rods irradiated in BR3 under different operating regimes (6 rods)	03/2006	Steady state	Fuel design on fission gas release (FGR)	PWR	
Novovoronezh-5 VVER 1000 fuel behaviour data (317 rods)	03/2006 01/2023	Nuclear power plant operation + PIE	Rod elongation at PIE	VVER	
NRU MT-4 and MT6A (33 rods)	11/2003	LOCA transient	LOCA Ballooning and blockage of rod bundle	PWR	
JAERI tests FK-1, FK-2 and FK-3 in Nuclear Safety Research Reactor (NSRR): Irradiated BWR fuel under reactivity-initiated-accident conditions (3 cases)	02/2005	RIA transient	Fuel behaviour under RIA	BWR	FUMEX-III
REGATE L10-3: FGR and mechanical behaviour during a power ramp in SILOE (1 rod)	04/2004	Power ramp	Fuel swelling; fission gas release (FGR)	PWR	FUMEX-II
RISOE-1 (11 rods)	03/2005	Steady state + bump	Fission gas release (FGR)	LWR	
RISOE-2 Transient Fission Gas Release Project (15 rods)	12/1995 04/2005	Power ramp	Fission gas release (FGR)	BWR	FUMEX-III
RISOE-3 (16 rods)	09/1995 04/2005	Power ramp	Fission gas release (FGR)	BWR and PWR	FUMEX-II FUMEX-III
Siemens PWR rods irradiated in GINNA (17 rodlets)	03/2000	Nuclear power plant operation + PIE	Pellet design on pellet-clad mechanical interaction (PCMI)	PWR	FUMEX-III
SOFIT VVER fuel irradiation programme (12 rods)	11/1997 01/2023	Steady state	Thermal performance	VVER	
Studsvik DEFEX: Secondary defect formation as a consequence of primary defects (11 rods)	04/2008	Steady state with clad primary defect	Secondary clad degradation	BWR	
Studsvik DEFEX-II DEMO project: Simulation of primary defect and conditions leading to secondary	04/2008	Steady state with clad primary defect	Secondary clad degradation	BWR	

Experiment title	Date released / last updated	Scenario	Experiment primary purpose	Reactor or fuel type	IAEA CRPs in which data were used
failure of the cladding by hydriding (2 rods)					
Studsvik DEMO-RAMP I (5 rods) & II (8 rods)	03/2001	Power ramp	Pellet-clad interaction (PCI)	BWR	
Studsvik INTER-RAMP BWR Project (20 rods)	11/1997	power ramp	Failure operational Limits	BWR	FUMEX-III
Studsvik OVER-RAMP PWR Project (39 rods)	11/1997 03/2006	Power ramp	Pellet-clad mechanical interaction (PCMI) failure analysis	PWR	FUMEX-III
Studsvik ROPE-1: BWR cladding creep-out under influence of rod overpressure (3 rods)	04/2008 01/2023	Lift-off at steady state	Clad creep-out at rod overpressure	BWR	
Studsvik ROPE-II: PWR cladding creep-out under influence of rod overpressure (6 rods)	04/2008	Lift-off at steady state	Clad creep-out at rod overpressure	PWR	
Studsvik STEED-I: Stored energy/enthalpy determination (2 rods)	04/2008	Steady state + scram	Stored energy determination	PWR	
Studsvik SUPER-RAMP BWR (16 rods) and PWR (28 rods)	11/1997 01/2023	Power ramp	Pellet-clad mechanical interaction (PCMI) failure analysis	BWR and PWR	FUMEX-III
Studsvik TRANS-RAMP I (5 rods), II (6 rods), IV (7 rods)	04/2023	Power ramp	Pellet-clad mechanical interaction (PCMI) failure analysis	BWR and PWR	
TRIBULATION (19 rods)	11/1996 03/2002	Steady state + power ramp	High burn-up (HBU)	LWR	FUMEX-II
US-PWR 16x16 LTA Extended Burn-up Demonstration Programme (9 rods)	06/2005	Nuclear power plant operation + PIE	High burn-up (HBU)	PWR	FUMEX-III
Zaporoshye VVER1000 fuel behaviour data (312 rods)	10/2004 01/2023	Nuclear power plant operation + PIE	Fission gas release determination at PIE	VVER	

2.3. 1995 review of existing fuel performance data

The 1995 NEA report (Turnbull, 1995) reviewed in detail the fuel performance data available for inclusion in the nascent IFPE data collection. The report focused on data from the OECD Halden Reactor Project (HRP), while also touching upon data available from other sources, in particular those generated as part of the Studsvik ramp test programmes in Sweden and the Risø fission gas release projects in Denmark. In addition, the need for fuel performance data, and the general requirements to address this need, were briefly discussed.

In terms of the need for fuel performance data, it was recognised that computer modelling of fuel behaviour had moved away from bounding (intentionally conservative, deterministic) calculations towards calculations of best-estimate behaviour and of the associated uncertainties. This requires validation against good quality data; the main data sources were judged to be well-instrumented experiments in research reactors, where irradiation histories are accurately known and both normal and off-normal conditions can be investigated.

With respect to the data requirements, it was observed that data could be divided into three broad categories:

- data useful for model development and validation, e.g. radial depression of neutron flux, fuel creep, fuel densification and swelling, and clad creep-down;
- data of direct relevance to fuel licensing requirements, e.g. fuel temperatures, stored heat, fission product release and rod internal pressure, waterside corrosion;
- data for fuel development and optimisation, e.g. performance of fuel variants and new cladding materials, effects of changes in fuel design.

However, for the purposes of the report, a phenomenological categorisation of data was instead adopted. The phenomenological categories are listed below, together with (in brackets) the HRP data, which were identified in these categories:

- radial flux depression (transverse gamma-scanning and micro-coring data for irradiated fuel from IFAs 180, 211, 411 and 431);
- thermal performance (steady-state, beginning-of-life fuel thermocouple data from 54 IFAs; steady-state, through-life fuel thermocouple data from IFAs 410, 411, 432, 504, 505, 507, 513, 552, 562.1 and 562.2; transient fuel thermocouple data from IFAs 429, 507 and 509 and from HRP reports HPR-276 and HWR-249; and stored heat data – evaluated from clad extensometer measurements – from HRP report HWR-300);
- fuel densification and swelling (fuel extensometer data from IFAs 401.2, 408.2, 409.2 and 515.4);
- UO_2 grain growth (ceramographic image analysis data from IFAs 223, 224, 410, 411, 505, 522.1 and 562.1);
- fission product release (radioactive fission gas release data from the IFAs 430, 504, 558 and 563 gas flow rigs; and stable fission gas release data from IFAs 402.3, 429, 432, 505, 512, 513, 517, 518, 519, 520, 523, 524, 534.2, 535 and 553.1 and the 70 rods in the 1% FGR threshold database);
- clad properties (irradiation growth data from HRP reports HPR-246 and HPR-221.07; creep data from IFAs 414, 523, 524 and 585; stress-corrosion cracking data from IFAs 516 and 567; and waterside corrosion data from IFAs 560 and 568);
- pellet-clad mechanical interaction (cladding axial elongation data from IFAs 118, 215, 227, 402, 404, 410, 411, 413, 414, 420, 436, 507, 508, 509, 512, 519.8, 520, 522, 525 and 550.2, cladding diametral strain data from IFAs 227, 404, 414, 420, 436, 508, 509, 512, 520, 522 and 525, and cladding ridge formation data from IFAs 227, 402.2, 404, 414.5, 436, 508, 512.7, 520 and 525);
- integral behaviour (IFAs 429, 431, 432, 504, 505, 509, 513, 522, 552 and 562.1);
- high burn-up effects (IFAs 408, 409, 416, 429.7, 504, 507.2, 512, 515, 518.6, 519.9, 533.2, 535, 550, 562, 565, 566 and 569).

HRP was seen as a major source of good quality data, and the NEA report looked in detail at what was available. The HRP data covered in-reactor measurements of a wide range of measured quantities (e.g. fuel temperature, fuel elongation, clad elongation, clad diameter and rod internal pressure) gathered by an assortment of dedicated instruments (e.g. fuel thermocouples/expansion thermometers, fuel extensometers, clad extensometers, clad diameter gauges and pressure transducers). The in-pile data were supplemented by PIE data on quantities such as FGR (from rod puncturing), clad oxide thickness (from eddy current

measurements and/or metallographic image analysis) and fuel grain size (from ceramographic image analysis). Irradiation can be either in the Halden Boiling Water Reactor (HBWR) moderator, where coolant conditions are limited to those associated with pool boiling at 32 bar, or in dedicated loops where power reactor coolant conditions can be simulated. The development of techniques for instrumentation of segments from parent rods pre-irradiated in commercial reactors (usually referred to as "re-instrumentation") has also facilitated experiments at high burn-up without the need for long burn-up accumulation times.

The HRP data listed in Table 2 were identified as high priority with respect to their usefulness for fuel performance modelling. The emphasis was placed on thermal performance data, since thermal performance was recognised as the most important aspect of fuel behaviour to predict accurately.

Table 2. Summary of high priority data from the OECD Halden Reactor Project as of January 1995

Thermal performance	Fission gas release	Dimensional changes	Integral effects
IFA-410	IFA-429	IFA-227	IFA-429
IFA-411	IFA-432	IFA-404	IFA-432
IFA-418	IFA-504	IFA-414	IFA-504
IFA-429	IFA-505	IFA-509	IFA-505
IFA-431	IFA-513	IFA-512	IFA-509
IFA-432	IFA-518	IFA-519	IFA-513
IFA-504	IFA-519	IFA-520	IFA-522
IFA-505		IFA-525	IFA-552
IFA-509			IFA-562.1
IFA-513			
IFA-522			
IFA-562			

The 1995 NEA report concluded by recommending that a comprehensive fuel performance database was set up, which included the data reviewed in the report and datasets that were available from other sources (in particular those from the Risø fission gas release projects and the Studsvik ramp test programmes). It was noted that attention should be paid to data for fuel in reactor systems other than PWRs and BWRs – for example, VVER, CAGR and CANDU fuel (since the 1995 NEA report almost exclusively focused on PWR and BWR data). It was also recognised that there would be a continuing need to identify and add further data, in particular those associated with high burn-up operation.

3. Definition and scope of fuel performance modelling

To determine the experimental data needs for fuel performance modelling, such modelling must first be defined and its scope qualified. The definition made here is "calculation of the thermo-mechanical and thermo-chemical behaviour of a single fuel rod during its irradiation in a reactor core" (fuel types not in rod form, such as pebble bed reactor fuel, are not considered in this working paper). Or, in slightly more detail, "Given the coolant pressure and the irradiation history, i.e. the evolution of the axial distributions of pin power, bulk coolant temperature and fast neutron flux with time, a fuel performance code calculates the evolution of the thermo-mechanical state of a fuel pin [in a reactor core]" (Rossiter, 2012). By thermo-mechanical state we mean the 3D distributions of temperature, stress, strain and displacement within the fuel rod; the thermo-mechanical state is, in turn, determined by the thermo-mechanical (e.g. heat conduction, fuel and cladding elasticity and plasticity) and thermo-chemical (e.g. cladding corrosion) behaviour.

Fuel performance modelling can be (and has been in recent years) extended to post-irradiation storage (both wet and dry), transport, and final disposal scenarios. However, this is outside the meaning of fuel performance modelling as it is defined here.

In terms of the scope of fuel performance modelling, we restrict ourselves here to UO_2, UO_2 with burnable absorbers (in particular $(U,Gd)O_2$) and MOX in zirconium alloy cladding in PWRs, BWRs, VVERs and PHWRs – that is, to the existing fuel types in the majority of commercial reactors that are operating today. However, much of the discussion will also be relevant to advanced fuel types and other reactor types. The focus is on fuel in PWRs and BWRs, since these represent the largest proportion of currently operating plants, and since the primary experience of the authors is in these reactor systems. Fuel behaviour during normal operation (both steady-state operation and operational transients), anticipated operational occurrences (AOOs) and design-basis accidents (DBAs) must all be simulated (but not necessarily by the same computer codes). Beyond-design-basis accidents (BDBAs) are currently outside the scope of fuel performance modelling, but, after the Fukushima Daiichi accident, this is likely to change in the near future, at least for the initial stages of such accidents where the fuel rods retain some semblance of their pre-accident geometry. All modelling is in general restricted to situations where the fuel and cladding temperatures remain below the melting points of the materials: the post-melting fuel behaviour is effectively covered by severe accident reactor/core modelling, which is outside the scope of fuel performance modelling as it is defined here.

Fuel performance simulation involves modelling many phenomena. These include, for the reactor types considered here:

- standard phenomena associated with thermo-mechanical behaviour of the fuel and cladding materials, in particular heat transfer by conduction, convection and radiation, thermal expansion, creep, elasticity, plasticity, fatigue, phase changes and melting;
- phenomena related to the presence of a neutron flux, in particular cladding hardening, embrittlement and axial growth;
- phenomena related to fissioning, neutron capture and the generation of fission products, in particular (non-uniform) heat generation (leading to a radial power distribution in the fuel pellets), the generation and release of fission gas (xenon and krypton) and helium, and fuel densification and swelling;

- phenomena related to microstructural changes in the fuel, in particular grain growth and formation of high burn-up structure (HBS);
- phenomena related to radial temperature gradients in the fuel pellets, in particular pellet cracking and fuel fragment relocation, pellet hour glassing and clad ridging, and axial elongation;
- chemical phenomena, in particular fuel-clad bonding, stress-corrosion cracking and cladding oxidation (Rossiter, 2012).

Additional phenomena that are modelled for fast reactor fuel pins include plutonium redistribution and oxygen migration in the fuel, and void swelling, dissolution and erosion of the cladding; these are outside the scope of this working paper.

The models for the phenomena described above vary from fuel performance code to fuel performance code (and may be entirely absent in some cases). The complexity of the models also varies, with simple, purely empirical models at one end of the scale, complex, highly mechanistic models at the other end, and intermediate-complexity semi-empirical models somewhere in between. Many of the models are underpinned by sub-models for material properties (which, like the models themselves, can have a range of complexity); these sub-models must take account of the evolution of material properties under irradiation (that is, their dependence on burn-up, fast neutron fluence, etc). For example, the thermal expansion model is underpinned by sub-models for the thermal expansion coefficients of the fuel and cladding materials.

For the purposes of modelling, the fuel rod, or at least the fuel stack region of the rod, must be discretised (or meshed) into volume elements. The form of these volume elements depends upon the type of code, that is $1^1/_2$D, 2D or 3D (Rossiter, 2012). In a "traditional" $1^1/_2$D code, the fuel stack region is divided into a number of axial zones (also known as regions, segments or slices), each of which is, in turn, subdivided into a number of radial annuli (or rings). Nodes are then defined on the volume element boundaries at which the thermo-mechanical equations (the conservation of energy equation and the equilibrium equation, or force balance) are solved (making use of defined stress-strain and strain-displacement relationships), typically using finite difference or finite element techniques. Confusingly, the volume elements, axial zones and/or radial annuli are often themselves described as nodes, but this terminology will not be used here.

Some details of thermal and mechanical modelling are provided in Sections 3.1 and 3.2, respectively. Similarly, some details of fission product behaviour modelling are provided in Section 3.3. Van Uffelen and Suzuki (Van Uffelen and Suzuki, 2012) give further details of fuel performance modelling, and the current state of the art is covered in a more recent review of fuel performance modelling by Hales, Li, Rossiter, Van Uffelen and Williamson (Hales et al., 2019). The thermo-mechanical equations and their typical solution for a $1^1/_2$D code are described in depth by Olander (Olander, 1976).

3.1. Thermal modelling

Thermal modelling as it is performed in a typical $1^1/_2$D code – i.e. assuming only radial heat transfer – is described in this section. In a 2D code axial heat transfer is also modelled, and in a 3D code both axial and circumferential heat transfer are also simulated.

The axial distribution of rod linear heat rate and its evolution with time is provided as an input (see Section 4.1). For each timestep, a radial heat transfer calculation is then performed for each axial zone as follows:

- the radial distribution of the heat generation rate in the fuel pellets is calculated using the radial power distribution model (which is usually underpinned by calculation of the radial distributions of fissile isotope concentrations in the fuel; this, in turn, requires calculation of the radial distributions of non-fissile actinide and burnable absorber isotope concentrations);
- the heat diffusion equation is solved to calculate the cladding inner surface temperature using the cladding outer surface temperature as a boundary condition, taking into account any crud layer and any inner and outer clad oxide layers (with possible intermediate calculations of annulus inner temperatures if the cladding is discretised into rings); the required cladding material properties (thermal conductivity, specific heat capacity and density) are evaluated by sub-models;
- the fuel pellet outer surface temperature is calculated from the cladding inner surface temperature and the heat flux using a model for the gap heat transfer coefficient (see below);
- the heat diffusion equation is solved to calculate the inner temperature of each fuel annulus, starting with the outermost, using the fuel pellet outer surface temperature as a boundary condition; the radial distribution of the heat generation rate in the fuel pellets calculated by the radial power distribution model is used as the "source term" and the required fuel material properties (thermal conductivity, specific heat capacity and density) are evaluated by sub-models.

The cladding oxide has a significantly lower thermal conductivity than the cladding metal. The thickness of the outer (waterside) clad oxide layer is therefore important (the inner oxide layer is generally much thinner) in calculating the cladding temperature distribution. Cladding oxidation is also important, because: (a) the thickness of metal that can sustain stresses is reduced; (b) the ductility of the remaining metal is reduced by the hydrogen absorption that accompanies oxidation (via hydride precipitation, which can have a significant impact on cladding mechanical properties if the hydrides tend to be oriented in the radial direction); and (c) the exothermic reaction can contribute substantially to the local heating of the cladding in the event of an accident when high clad temperatures are achieved (e.g. a LOCA).

The gap heat transfer coefficient gives the temperature drop across the gap as a function of the gap heat flux. This quantity is usually converted to a gap thermal conductance by taking account of the cross-sectional area and the distance (gap size) associated with the heat transfer. The heat transfer corresponds to a combination of three different heat transfer mechanisms: convection and conduction in the (gas-filled) gap; conduction across points of fuel-clad contact when the gap is closed; and radiation (when the temperatures are high enough). Accurate determination of the local gap heat transfer coefficient therefore requires accurate knowledge of the local heat flux, the local temperature drop across the gap and the (azimuthally averaged) local gap size. Unfortunately, the gap size is affected by most phenomena occurring in the fuel (including thermal expansion, swelling, densification, creep, cracking and relocation) and cladding (including thermal expansion and creep-down), has a stochastic component (due to the stochastic nature of pellet cracking and fuel fragment relocation) and cannot be measured accurately in-pile. The determination of the gap size, and hence also of the gap heat transfer coefficient, is therefore subject to large uncertainties.

An accurate prediction of the temperature distribution in the fuel pellets is not only important in order to predict under what conditions melting might occur – it is also essential for the simulation of all other phenomena that depend (exponentially) on the local fuel temperature (e.g. fission gas release). In the case of the UO_2 and MOX fuels considered in this working paper, the heat transfer in the fuel can be treated purely by conduction. Heat transfer by radiation can be important in other semi-transparent fuel types, e.g. $(Th,U)O_2$, $(Th,Pu)O_2$ and zirconia-based inert matrix fuel.

3.2. Mechanical modelling

The general approach in the mechanical modelling is to solve the equilibrium (force balance) equation at the end of each timestep using calculated stress-strain relationships and strain-displacement relationships. The various strain components (in the radial, axial and circumferential directions) that contribute to the fuel and cladding strains are computed as part of the evaluation of the stress-strain relationships; these typically include elastic, thermal, creep, densification, swelling and relocation components for the fuel, and elastic, thermal, creep, instantaneous plastic and growth components for the cladding. The fuel swelling strain components are typically subdivided into fuel matrix (inexorable swelling), fission gas bubble, and HBS pore contributions.

Further details of the mechanical modelling approach are code dependent, with solution schemes closely correlated to the type of code (i.e. $1^1/_2$D, 2D or 3D). In a typical $1^1/_2$D code, the fuel annuli are all considered to be subject to the same axial strain (the so-called plane strain assumption). This, in conjunction with an assumption of axi-symmetry, allows fuel shear stresses to be ignored, such that only the principal stresses along the radial, circumferential and axial directions are non-zero. The mechanical (equilibrium, stress-strain relationship and strain-displacement relationship) equations are then solved by a finite difference scheme.

As a fuel rod is first brought to power, the fuel-clad gap decreases somewhat due to the effects of thermal expansion (the fuel pellets being significantly hotter than the cladding) and poor thermal conductivity of the fuel (cracking and relocation). The gap then increases due to fuel densification. Once densification is effectively complete, the gap gradually decreases due to the combined effects of fuel swelling and clad creep-down (the latter due to the coolant pressure exceeding the rod internal pressure). Upon closure, the clad starts to creep outwards, following fuel swelling, while also exerting a "back-stress" on the fuel which allows some fuel creep. In the "open gap" conditions prior to fuel-clad contact, the mechanical behaviour of the fuel and cladding can be modelled separately (the pellets and cladding can both be treated as free-standing), thereby simplifying the calculations. In contrast, in post-contact "closed gap" conditions, the calculation of the stress and strain distributions in the fuel and cladding is more complicated, since the combined fuel-cladding system must then be considered. An assumption about the potential for relative movement of fuel pellets and cladding must be made; the options can generally be classified as no sliding (fuel and cladding are stuck together), friction-governed sliding, or frictionless sliding, and can potentially vary both spatially and temporally (via dependencies on burn-up, fuel-clad contact pressure, etc).

3.3. Fission product behaviour modelling

The models for fission product behaviour aim at describing their amounts and accommodation (in the fuel matrix, in bubbles and pores, at grain boundaries, in the rod free volume, etc) during irradiation. Fission products in UO_2 and MOX are known to fall into four broad categories:

- soluble (e.g. Zr, La) ;

- metallic precipitates (e.g. Mo);
- second-phase oxide precipitates (e.g. Sr);
- volatile (e.g. Xe, Kr).

The computation of their amount is usually included in the model for the radial power distribution, since this generally relies on prediction of the underlying distributions of the fissile isotope concentrations; the radial power distribution model is therefore important for two reasons.

Special attention is given to those fission products that significantly affect the thermal or mechanical aspects of the fuel behaviour. This includes fission gas (Xe and Kr) and helium, since they can cause significant swelling in the fuel due to the formation of gas bubbles, and can both pressurise the fuel rod and significantly reduce the fuel-clad gap heat transfer coefficient when released into the rod free volume. The behaviour of other fission products is generally restricted to an implicit consideration in the fuel matrix swelling model, although the behaviour of I and Cs is sometimes explicitly modelled (especially as part of the simulation of accidents) due to the dose implications and the nature of free iodine as one of the principal corrosive species in stress-corrosion cracking (in some countries, explicit calculation of the gap inventory of ^{131}I is required).

4. Data requirements

This section describes the data that are needed for fuel performance modelling and why they are needed.

4.1. Code input data

Input data for fuel performance codes must cover all necessary details of the fuel rod being modelled. The data can be divided into three categories:

(1) initial (fresh fuel) conditions at start of irradiation;

(2) irradiation conditions as a function of time (which may include irradiation in two or more reactors);

(3) details of any re-fabrication.

The first category can be further subdivided into fuel-related, cladding-related and fuel rod-related conditions (with any relevant fuel assembly related conditions, e.g. rod pitch, included under the final sub-category). For the common situation where a full-length commercial fuel rod is irradiated, and a segment is cut from this rod and re-fabricated for further irradiation in a test reactor, the first category would correspond to the initial conditions for the full-length (parent) rod, the second category would include the commercial reactor irradiation conditions for the full-length rod and the test reactor irradiation conditions for the re-fabricated rodlet, and the third category would include details of the cutting scheme and of any re-orientation of the segment in the re-irradiation.

The coarseness of the discretisation in the axial and radial directions must be carefully optimised on a case-by-case basis, as it is dependent upon (a) the variations of the input data in the radial and axial dimensions, and (b) the required resolution of the output data. The input data must reflect any variations in conditions from axial zone to axial zone, and, for some quantities, also from radial annulus to radial annulus.

The axial discretisation of a given simulation – that is, the lower boundary elevations and the heights of the axial zones – is in practice determined by any step changes in input conditions (for example, a change in pellet diameter or fuel material) and by the axial gradients of continuously varying quantities (in particular, linear heat rating and fast neutron flux).

The radial discretisation is primarily driven by the required resolution of the calculation results, since the radial gradients of the simulated physical quantities strongly evolve in the course of the irradiation. However, in some cases the radial discretisation is also driven by radial variations in the initial conditions, e.g. in the initial fuel porosity volume fraction, or in the fuel grain size.

The specific fuel performance code input data that are generally required (“R”) in categories 1 and 2 are listed below in Tables 3 to 4. Also included in the tables are optional (“O”) input data, which, although not essential, can allow the analysis of complementary output and/or reduce simulation uncertainties. Input quantities in *italics* are only relevant for certain cases (e.g. fuel pellet inner radius). Each quantity is complemented by a “flag” indicating where the use of generic (“G”) data (which may be from the open literature) is generally acceptable. The issue of generic data is important because for many fuel rods detailed measurements of the fuel and cladding are not performed as part of routine manufacturing. No general rule for acceptance or not of generic data can be given. It must

thus be checked case by case whether the generic information available is sufficiently accurate and comprehensive for replacing specific data, which, in turn, depends upon the objectives of the simulation. Finally, "U" in the table means the principle necessity to consider uncertainty of experimental data. It should be noted, however, that uncertainty quantities or even generic uncertainties would not be available for many of the items marked in the U-column, (see [IAEA, 2019]).

Table 3. Data for initial fuel rod conditions at start of irradiation

Data item			R	O	G	U
Fuel	Composition (e.g. ^{235}U enrichment, Pu enrichment, additive concentration (e.g. Gd_2O_3, Cr_2O_3), stoichiometry)		✓			✓
	Pellet geometry	Outer and/or inner radius/diameter	✓			✓
		Height	✓			✓
		Chamfer height and width (with the information about one-sided or two-sided)	✓			✓
		Dish geometry (e.g. volume, diameter, depth, shape (spherical, conical; provide drawing), one-sided or two-sided)	✓			✓
		Surface roughness (with related information on contact conductance, if any)	✓		✓	
	Pellet physical characteristics	Density	✓			✓
		Porosity (e.g. volume fraction, fractions of open and closed porosity, inter- and intra-granular pore size distributions)	✓			✓
		Thermal stability (e.g. re-sintering test results at 24 hours at 1 700 °C in reducing atmosphere)	✓		✓	✓
		Grain size (2- and/or 3-dimensional value together with its evaluation method) (For MOX fuel, values in UO_2 matrix and Pu-spot, if possible)	✓			✓
		Surface roughness		✓		
		Volume fraction and size distribution of Pu-rich particles (for MOX fuel)	✓			✓
		Typical ceramographs (e.g. polished and etched to reveal grain and pore morphology, identification of any dopant phases, if exist)		✓		
Cladding	Composition/material type (e.g. ^{2}Zry, ^{4}Zry; including the amount of key alloy elements and impurities, if possible)		✓			✓
	Outer and/or inner radius/diameter		✓			✓
	Zr-liner/barrier thickness		✓			✓
	Surface roughness			✓	✓	
	Yield stress (0.2% proof stress)		✓		✓	✓
	Ultimate tensile stress		✓		✓	✓
	Final heat treatment condition (e.g. stress-relieved, recrystallised)		✓		✓	
	Grain size			✓		
	Percent of recrystallisation			✓		
	Typical metallographs (e.g. polished and etched to reveal grain structure)			✓		
	Fabrication condition	Percentage of cold work		✓		
		Accumulated annealing parameter		✓		
		Volume fraction and size distribution of second-phase particles		✓		
		Q value[1]		✓		
		Texture factors (in radial, tangential and axial directions)		✓		
Fuel rod	Fuel assembly type (e.g. 17x17 PWR)		✓			
	Rod-to-rod pitch		✓		✓	
	Overall as-fabricated fuel rod length		✓			✓
	Fuel stack length		✓			✓
	Height of upper and lower depleted blankets		✓			

[1] Q-value is a parameter at cladding fabrication: a ratio of the decrease of wall thickness to the decrease of tube diameter before and after rolling of cladding tube.

Table 3. Data for initial fuel rod conditions at start of irradiation (Continued)

Data item		R	O	G	U
	Upper plenum volume	✓			✓
	Lower plenum volume	✓			✓
	Total free volume in fuel rod	✓			✓
	Dimensions of miscellaneous parts (e.g. plenum spring, clip, support tube, getter)	✓			
	Volume and strength of plenum spring		✓		
	Information about hafnium (Hf) pellets (e.g. presence, dimension)		✓		
	Initial fill gas composition and pressure (e.g. at 20^0C)	✓			✓

Note: The fuel related data must be given as a function of the axial co-ordinate (e.g. zone by zone) whenever it is required by the specific case. For some quantities, input can moreover be required as a function of the radial position.

Table 4. Data for irradiation conditions

Data item		R	O	G	U
Reactor type		✓			
Configuration of reactor core			✓		
Hydraulic diameter			✓		
Coolant chemistry and pH (including details of any coolant additives, e.g. Zn)			✓		
Conversion factor between fission number in rod and thermal power of reactor		✓		✓	
Data collected as a function of time	Linear heat rate (thermal power) history of fuel rod (e.g. average linear heat rate and axial shape factors)	✓			✓
	Fast neutron flux with an axial shape factor (e.g. E > 1 MeV)	✓		✓	✓
	Coolant temperature (at the inlet and outlet of reactor core)	✓		✓	✓
	Coolant pressure	✓		✓	
	Coolant mass flux	✓		✓	✓
	Rod surface temperatures and their axial variation (if available from thermal-hydraulic simulations)		✓		

Note: All irradiation conditions must be given as a function of time and, where indicated, as a function of the axial co-ordinate. For an irradiation in multiple reactors, data must also be provided for each reactor. In order to facilitate modelling of the decay of radionuclides, all times should be given relative to a specified reference date. Providing irradiation conditions as a function of burn-up is not supported here, since: (a) this gives no information on shutdown periods; and (b) in most fuel performance codes burn-up is an output quantity that is dependent on further input parameters (e.g. fuel density).

A typical application of fuel performance codes is the simulation of the irradiation of an experimental fuel rod in a test reactor. For experimental fuel rods that have been pre-irradiated, or that are segments of pre-irradiated "parent" rods, the pre-irradiation must also be simulated, and any modifications due to re-fabrication and instrumentation must be taken into account. The OECD Halden Reactor Project (HRP) has compiled the information required for such simulations (HRP, 2012) in line with the general requirements of input data for fuel performance simulations. Given also the status of HRP as a major contributor of data to the IFPE data collection, the HRP compilation was used as a basis for the present compilation, but with HRP's distinction between "mandatory" and "highly desirable" data ignored.

The data required under category (3) would include any modifications to those parameters for the fuel rod geometry data listed in Table 3, together with details of any drilling of pellets for insertion of instrumentation (such as fuel thermocouples) and, when a segment is cut from a pre-irradiated parent rod, the cutting scheme and any re-orientation of the segment. All this information should ideally be provided in a re-fabrication report.

Where available, any measurements taken in-pile or during PIE (including poolside inspections) can potentially be used to improve the fidelity of the simulation through the "tuning" of input parameters or of model parameters to better match measurements and predictions. In the situation where a re-irradiation is due to take place (which may be of an experimental rodlet containing a segment cut from a pre-irradiated parent rod) any available data from characterisation of the pre-irradiated fuel rod, or from the pre-irradiation itself, can be used to improve the accuracy of the predictions for the re-irradiation.

4.2. Material properties data

As explained in Section 3, fuel performance simulation involves the use of models for the large number of phenomena that need to be taken into account, and many of these models are underpinned by sub-models for material properties. These sub-models must, in turn, be developed using measured data for material properties of both fresh and irradiated material.

"Material properties" here means the mechanical and thermo-physical properties of the various materials in a fuel rod (primarily the fuel and cladding). "Material-based phenomena" (e.g. fuel swelling) and "fuel rod component properties" (e.g. fuel pellet surface roughness) are excluded.

The fuel and cladding material properties data that are required for fuel performance modelling are listed in Table 5. Also included in the table are the primary dependencies of the material property data; unless otherwise stated, these dependencies are those included in the correlations in the MATPRO material properties library (Siefken et al., 2001). Where properties are dependent upon stress, σ, and/or strain, ε, the resultant property values are directionally dependent, with the values in the three principal directions (radial, circumferential and axial) dependent upon the stress and/or strain components in the same directions. Each different fuel or cladding material nominally requires a different dataset; in this sense, cladding materials are judged to be "different" if there are differences in material composition (e.g. Zircaloy-4 is different to Zircaloy-2) and/or differences in manufacturing process (in particular, amount of cold work and percentage recrystallisation). In practice, however, certain material property data can be taken as common to two or more materials (e.g. the thermal expansion coefficients of Zircaloy-4 and Zircaloy-2 can be assumed to be identical).

In addition to the fuel and cladding material properties data, thermal conductivity data are needed for the various gases that can be present in the rod free volume (usually restricted to He, Xe, Kr, N_2 and Ar). The properties of the other fuel rod components (usually restricted to a plenum spring and upper and lower end caps) are generally not required because they are usually not modelled explicitly. However, this may change in the future with advancements in the complexity of fuel performance simulation.

Table 5. Fuel and cladding material properties data required for fuel performance modelling

Property	Fuel	Clad	Primary dependencies*†
Thermal expansion coefficient	✓	✓	T
Creep rate (irradiation-induced and irradiation-enhanced thermal)	✓	✓	T, σ, F, V, g, S, t (fuel); T, σ, ε, Φ, φ, t (clad)
Instantaneous plasticity		✓	T, σ, ε, Φ, O
Specific heat capacity and/or enthalpy	✓	✓	T, S (fuel); T (clad)
Elastic moduli (generally Young's modulus and Poisson's ratio)	✓	✓	T, V, S (fuel); T, Φ, O (clad)
Yield stress (generally 0.2% proof stress)		✓	T, Φ, O, dε/dt
Ultimate tensile stress / failure stress / failure strain		✓	T, Φ, O, dε/dt
Fracture stress	✓		T, V
Thermal conductivity	✓	✓	T, B, V, S (fuel)#; T (clad)
Hardness	✓	✓	T, V, B (fuel)$; T (clad)
Fission gas (Xe and Kr) diffusion coefficient	✓		T, F, S+
Helium diffusion coefficient	✓		T, F, S‡
Oxygen diffusion coefficient (high T only)	✓	✓	T, F, S (fuel)‡; T, Φ (clad)**
Hydrogen diffusion coefficient		✓	T, Φ**
Melting temperature (solidus and liquidus)	✓	✓	B (fuel); O (clad)
Density	✓	✓	T, V, B (fuel); T (clad)††
Phase transition (α→β) temperature##		✓	O$$
Emissivity	✓	✓	T

* All are local quantities for undoped UO_2 fuel; for doped fuel there will generally also be a dependency on local dopant concentrations, while for MOX fuel there will generally also be a dependency on local Pu concentration.

†T = temperature; σ = stress; F = fission rate; V = porosity volume fraction; g = grain size; S = stoichiometry (i.e. O/M ratio); t = time; ε = strain; Φ = fast neutron fluence; φ = fast neutron flux; O = oxygen content; B = burn-up.

\# See Gates et al. (1998).

$ See Spino et al. (2003).

\+ See Lösönen (2000).

‡ Assuming the same primary dependencies as the fission gas diffusion coefficient.

** Expected dependencies.

†† The densities of the fuel and cladding are dependent on temperature due to thermal expansion effects; the density of the fuel is also dependent on porosity volume fraction, fuel swelling and burn-up. In practice: (a) the "clad density" used by fuel performance codes as a material property is simply the room temperature density; the variation with temperature is implicit in the clad thermal expansion model; (b) the "fuel density" used by fuel performance codes as a material property is simply the room temperature density of 100% dense material; the fuel density at start of irradiation is then set to this value multiplied by one minus the input porosity volume fraction, and subsequent variations with temperature, porosity volume fraction and burn-up are taken into consideration in the fuel thermal expansion, densification and swelling models.

In practice, there is not a sharp transition temperature from the α-phase to the β-phase, so the fraction of material in the β-phase is instead required with primary dependencies on T and O (Van Uffelen and Suzuki, 2012).

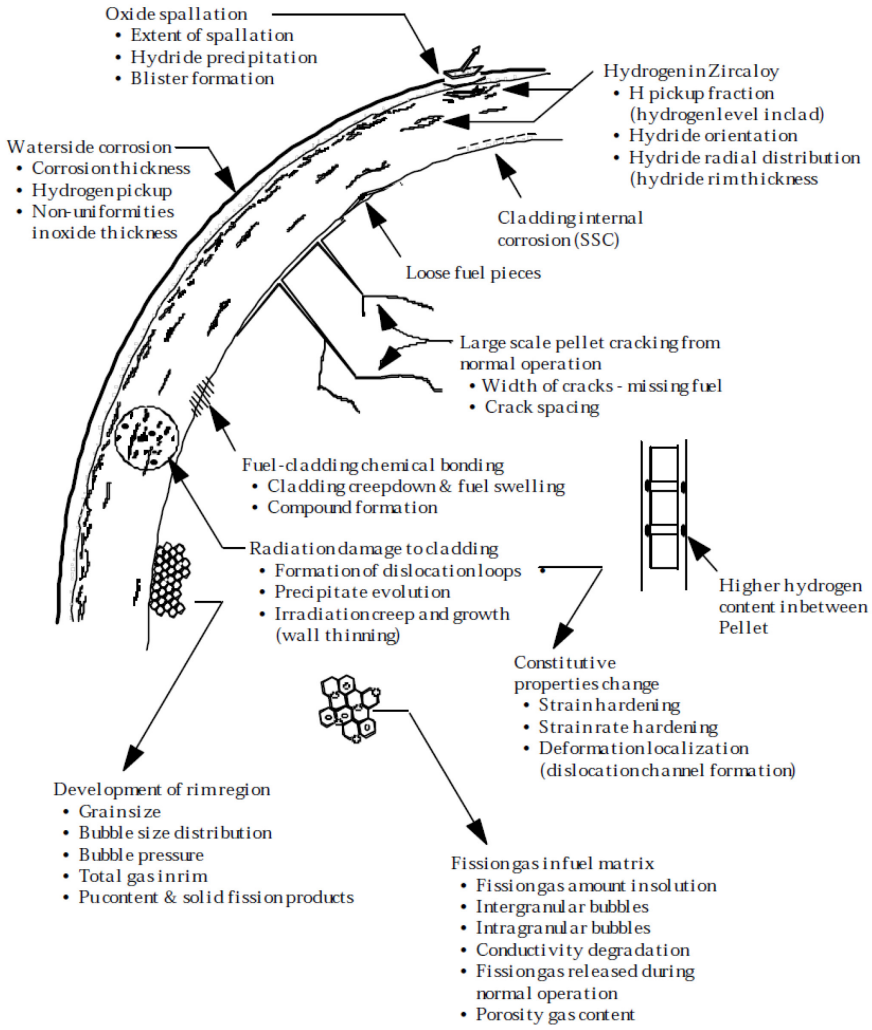
$$ See (Van Uffelen and Suzuki, 2012).

The material properties data listed in Table 5 are those required for modelling at fuel and cladding temperatures up to the melting points of the materials. For post-melting behaviour simulation, additional properties would be required – including the viscosity of the molten fuel – and data for the properties listed would also be required for molten material. However, as noted in Section 3, such post-melting simulation is outside the scope of fuel performance modelling as was defined here.

It is implicitly assumed above that the fuel is a brittle material and that the cladding is a ductile material. Although this is generally true for the materials covered in this working paper (see Section 3), it should be noted that zirconium alloy cladding can become brittle at low temperatures when the fuel is out of reactor. If this low temperature behaviour needs to be simulated, material property data are also required for the ductile-to-brittle transition temperature (DBTT) and for the fracture stress of the resultant brittle material.

Table 5 suggests that the fuel and cladding are isotropic and homogeneous materials. However, this is not the case: mechanical properties of both fuel and cladding are anisotropic (the former "effectively" due to cracking, and the latter due to the anisotropy of the zirconium crystal structure) and inhomogeneities are introduced by radiation damage, the formation of fission products, evolution of the fuel microstructure, chemical reactions between the fuel and cladding and between the cladding and coolant, and phase changes in the cladding. In addition, composite fuel pellets and cladding tubes are possible, e.g. duplex or liner cladding. This means that material properties vary locally within both the fuel and cladding due to local variations in both the state variables (temperature, stress, etc.) and the materials themselves. The complicating factors summarised in Figure 1, which apply to high burn-up PWR fuel in normal operation conditions, are discussed further below. However, inhomogeneity at the microscopic level due to fission products and radiation damage is not included in this discussion, since this is in general not simulated given that material properties are only meaningful at the macroscopic scale.

Figure 1. State of PWR high burn-up fuel and cladding in normal operation conditions



Source: NEA, 2009.

Due to the shear stresses induced by differential thermal expansion, fuel pellets crack as soon as the fuel rods are first brought to power. This introduces an anisotropy into the pellet mechanical behaviour, which is typically modelled (in a simplified manner) by utilising directionally-dependent fuel elastic constants (Olander, 1976). However, this anisotropy is not an intrinsic anisotropy in the material itself, and so is not strictly a feature of the material properties of the fuel. In contrast, zirconium alloy cladding is intrinsically anisotropic due to the hexagonal close-packed (HCP) structure of zirconium. Thus, data on material properties for cladding must ideally cover the behaviour in all three principal directions (radial, circumferential and axial). In practice, it is only the mechanical properties that are affected, and it is the diametral strains of the cladding that are most important, so property data used in fuel performance modelling often relate to the circumferential direction only.

The cladding is oxidised at both the inner and outer surfaces during irradiation. Similarly, a fuel-clad bonding layer can form at higher burn-ups when the fuel and cladding are in strong contact, and the fuel rod can become coated with crud on its outer surface. In theory, then, the fuel property data described in Table 5 must be complemented by corresponding property data on the fuel-clad bonding material, and the cladding property data described in Table 5 must be complemented by corresponding property data on the oxide and crud

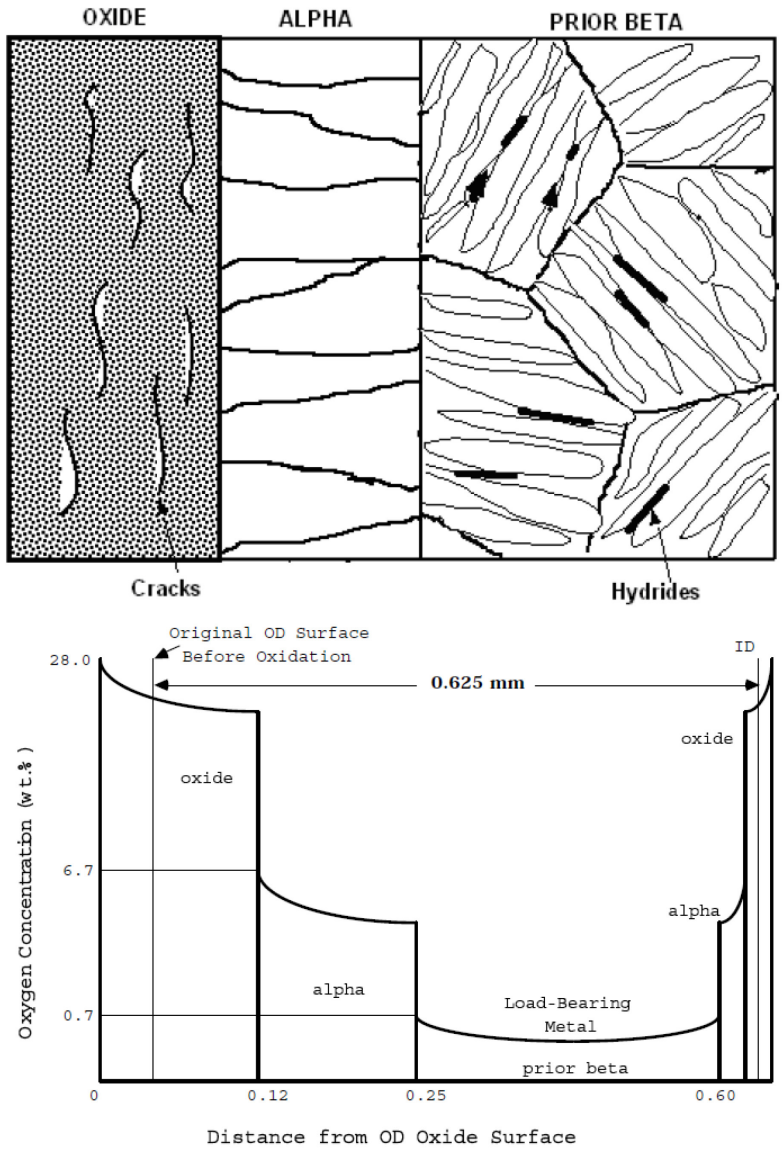
material (which may be subject to inhomogeneities and/or cracking). However, in practice, the fuel-clad bonding and cladding inner oxide layers are usually not modelled explicitly, and explicit modelling of the cladding outer oxide and crud layers is usually restricted to heat transfer calculations (since their mechanical strength is negligible). Thus, in general, only data for the thermal conductivities of the cladding oxide and crud materials and the density and emissivity of the cladding oxide material are needed (the density is needed to enable the volume of the cladding outer oxide layer to be calculated as a function of the volume of clad metal lost due to oxidation).

The fuel microstructure evolves with irradiation, with fission products accumulating (in both solid solution and in separate phases), columnar grains forming in the hot, inner regions of the fuel pellets at high temperature (Olander, 1976) and the so-called "high burn-up structure" (HBS) forming near the pellet rim at high burn-up (Lassmann et al., 1995). This affects the bulk material properties due to the evolution of the distributions of both fission products and porosity (pores migrate up the radial temperature gradient during columnar grain formation, and coarsened spherical bubbles of fission gas form as the HBS develops). Thus, fuel properties must in general be a function of the local burn-up, the local porosity volume fraction and the pore morphology and orientation. This is complicated by the fact that the HBS develops over a significant burn-up interval, so that there is a "transition zone" between the original microstructure and the fully developed HBS.

The determination of the material properties of the cladding is complicated by the uptake of hydrogen into the cladding metal as part of the cladding waterside corrosion, and the subsequent diffusion and precipitation of this hydrogen within the cladding (with the extent of these effects being primarily dependent upon the temperature distribution in the cladding). In particular, the precipitation of the hydrogen produces brittle zirconium hydride platelets (with the orientation of these platelets dependent upon the texture of the cladding and the stress distribution), which then form a second phase within the clad metal. At low burn-ups and/or high temperatures, the concentration of hydrides is insignificant, but at higher burn-ups and low temperatures, (during reactor shutdowns) the concentration of hydrides can become significant. The diffusion and precipitation behaviour of hydrogen means that there is a strong spatial variation in hydride concentration within the cladding, with local concentrations highest close to the cladding outer surface and adjacent to pellet-pellet interfaces.

At the high cladding temperatures which can occur during some design-basis accident scenarios (most notably LOCAs), the as-manufactured α-phase structure of the cladding can be wholly or partly transformed into the β-phase (NEA, 2009), which, upon subsequent cooling, transforms back into the α-phase, but with some characteristics of the β-phase retained (hence it is usually referred to as the "prior-β" phase). Oxidation of the cladding complicates matters, since oxygen – which helps to stabilise the α-phase – can diffuse from the oxide layers into the clad metal. The result is a "radial zoning" of the cladding, as illustrated in Figure 2. In the scenarios where such a situation arises, the material properties data used for fuel performance modelling must reflect any differences in cladding properties between the α, β and prior-β phases, including any effects of oxygen stabilisation.

Figure 2. Schematic illustration of microstructure (top) and oxygen distribution (bottom) in oxide, stabilised-α and prior-β layers in Zircaloy cladding after oxidation at ~ 1 200°C



Source: NEA, 2009.

4.3. Code calibration and validation data

In general, the verification and validation (V&V) of a fuel performance code requires three basic steps. The first step consists of verifying the correctness and numerical stability (or robustness) of the models – for example, whether the numerical solution of fission product diffusion in an idealised spherical grain is correctly programmed (i.e. fast, accurate and robust). This step relies on either analytical solutions – when available – or more detailed numerical solutions that cannot be implemented in a fuel performance code for the sake of computation time. In the second step, the predictions of the models are compared with measured data; this comparison then allows the models to be either validated – that is for the predictions to be shown to agree with measurements – or calibrated – that is for the values of empirical parameters in the models to be tuned to give optimal agreement between measurements and predictions. This is usually performed in two stages: firstly, a subset of

the available measured data is used to calibrate the models, and secondly, the remaining subset is used to validate the calibrated models. In the third and final step, the predictions of the fuel performance code as a whole (i.e. the predictions of the integrated models) are validated against measured data. In the case of some models, either measured data for the second step are unavailable, or the model cannot be assessed independently in a stand-alone manner; the validation of such models is then implicit in the third step.

Given that only the second and third steps of the code V&V process rely on experimental data, we will limit ourselves in the remainder of this section to discussing the experimental data requirements for these two steps.

4.3.1. Data needs for the calibration and validation of models

The data needs for the calibration and validation of fuel performance code models are summarised in Table 6. The models listed are those described in Section 3. Additional models, e.g. absorption and desorption (in/from fuel pellets) of helium fill gas, or fuel hot pressing, are included in some fuel performance codes. Similarly, not all models listed are present in all codes. All fuel-related data are needed for all fuel type/reactor type combinations that are supported. Similarly, all clad-related data are needed for all clad type/reactor type combinations that are supported. The calculations of stresses, strains and displacements in the fuel and cladding are treated as features of the integrated fuel performance code, and so are considered in Section 4.3.2.

Table 6. Data needed for the calibration and validation of fuel performance models

Model	Data needed
Heat transfer within fuel pellets	Fuel centre and surface temperatures; Fuel thermal conductivity (for thermal conductivity sub-model); Fuel specific heat capacity and/or enthalphy (for heat capacity and/or enthalphy sub-models); Fuel density (for density sub-model).
Heat transfer within cladding	Clad inner and outer surface temperatures; Clad thermal conductivity (for thermal conductivity sub-model); Clad specific heat capacity and/or enthalphy (for heat capacity and/or enthalphy sub-models); Clad density (for density sub-model).
Heat transfer between fuel and clad	Cladding inner surface temperature and either fuel outer surface temperature or gap heat transfer coefficient; Fuel and cladding hardnesses (for hardness sub-models); Fuel and cladding emissivities (for emissivity sub-models).
Heat transfer between clad and coolant	Bulk coolant temperature and either cladding outer surface temperature or clad-to-coolant convection heat transfer coefficient*
Thermal expansion of fuel	Fuel stack diameter, length or volume change during power increase or decrease with cladding free-standing; Fuel thermal expansion coefficient (for thermal expansion coefficient sub-model).
Thermal expansion of cladding	Cladding diameter, length or volume change during power increase or decrease with cladding free-standing; Clad thermal expansion coefficient (for thermal expansion coefficient sub-model).
Fuel creep	Fuel creep rate (for creep rate sub-model).
Cladding creep	Clad creep rate (for creep rate sub-model).
Fuel elasticity	Fuel elastic moduli (for elastic moduli sub-model).
Cladding elasticity	Clad elastic moduli (for elastic moduli sub-model).

Model	Data needed
Cladding plasticity	Effects of temperature and irradiation on clad yield stress (for yield stress sub-model)†; Clad instantaneous plasticity (for instantaneous plasticity sub-model); Clad ultimate tensile stress, failure stress or failure strain (for relevant sub-model).
Cladding fatigue	Number of cycles to failure as a function of amplitude of clad stress variations.
Cladding phase changes	Phase transformation temperature / fraction of material in β-phase (for material property sub-model); Oxygen diffusion coefficient in clad (for diffusion coefficient sub-model).
Fuel melting	Fuel melting temperature (for melting temperature sub-model).
Cladding melting	Clad melting temperature (for melting temperature sub-model).
Cladding hardening	As per cladding creep and cladding plasticity.
Cladding embrittlement	Ductile-to-brittle transition temperature (for DBTT sub-model); Hydrogen diffusion coefficient in clad (for diffusion coefficient sub-model); Hydride precipitation and dissolution temperatures.
Cladding axial growth	Elongation of irradiated cladding tubes (either unfuelled or with no pellet-clad mechanical interaction).
Radial power distribution in fuel pellets	Radial concentration profiles of solid fission product, actinide and burnable absorber elements (or their isotopes), in particular Nd, Pu and Gd.
Fission gas generation#	Nuclear data for Xe and Kr isotopic fission yields, decay constants and neutron capture cross-sections.
Fission gas release (from fuel to rod free volume)#	Rod free volume Xe and Kr contents (in addition to fission gas generation data)$; Radial distributions of Xe and Kr concentrations in fuel matrix; Xe and Kr (volume) diffusion coefficients in fuel (for diffusion coefficients sub-model); Partial pressures of Xe and Kr in as-manufactured pores; Fission fragment range in fuel.
Helium generation	Nuclear data for He fission yields, ^{16}O neutron capture cross-section, and decay constants and cross-sections associated with generation and α-decay of the transuranic isotopes (principally ^{238}Pu, ^{242}Cm and ^{244}Cm).
Helium release (from fuel to rod free volume)	Rod free volume He content (in addition to He generation data); Radial distribution of He concentration in fuel matrix; He (volume) diffusion coefficient in fuel (for diffusion coefficient sub-model); Partial pressure of He in as-manufactured pores; Helium solubility in fuel.
Fuel densification	Decrease in fuel stack diameter, length or volume (taking into account the effects of fuel swelling and thermal expansion); Decrease in pore size and/or concentration with burn-up.
Fuel matrix swelling	Increase in fuel stack diameter, length or volume (taking into account the effects of fuel densification, gas bubble swelling and thermal expansion); Nuclear data for elemental fission yields (mechanistic models only); Gibbs free energies of various fission product compounds (mechanistic models only); Destruction rates of fission product compounds due to radiolysis and fission fragments (mechanistic models only); Densities of various fission product phases (mechanistic models only); Occupation sites of fission product atoms in solution and the associated lattice parameter changes (mechanistic models only).

Model	Data needed
Fuel gas bubble swelling	Increase in fuel stack diameter, length or volume (taking into account the effects of fuel densification, matrix swelling and thermal expansion); Radial distributions of intra-granular bubble concentration, radius and pressure; Morphology, volume, areal density and pressure of grain face bubbles, and their variation across the fuel radius; Morphology, volume, linear density and pressure of grain edge bubbles, and their variation across the fuel radius; Xe and Kr (surface and grain boundary) diffusion coefficients in fuel (for diffusion coefficients sub-model); Vacancy (volume, surface and grain boundary) diffusion coefficients in fuel; Fuel surface energy; Fuel grain boundary energy; Dislocation density, and its variation across the fuel radius.
Equi-axed grain growth	Mean grain diameters
Columnar grain growth	Mean grain diameters and aspect ratios; Mean pore diameters and aspect ratios; Pore migration velocities.
Fuel restructuring at high burn-up	Thicknesses of fully-restructured and transition zones; Mean grain diameters and their radial variation; Rim pore concentrations, sizes and pressures and their radial variation; Fission gas release from the HBS; Mechanical properties of the HBS.
Pellet cracking	Images of irradiated fuel (normal operation); Fuel fragment size fractions (extensive fragmentation during severe accidents); Fuel fracture stress (for fracture stress sub-model).
Fuel fragment relocation	As per heat transfer between fuel and clad (normal operation); Change in axial distribution of fuel mass, incl. mass of fuel lost through any clad rupture (extensive fragmentation during severe accidents).
Pellet hourglass deformation	Deformation of non-uniformly heated pellet fragments; Otherwise as per clad ridging.
Clad ridging	Deformation of cladding at pellet-pellet interfaces, especially during and after power ramping.
Axial extrusion of fuel	Extent of pellet dish filling.
Fuel-clad bonding	Images of bonded fuel and clad; Chemical analysis of bonding layer; Oxygen diffusion coefficient in fuel (for diffusion coefficient sub-model).
Stress-corrosion cracking (SCC) of clad	SCC failure statistics as a function of clad stress, time-at-stress and corrosive species (principally free iodine) concentrations.
Cladding inner wall oxidation	Inner wall oxide thickness.
Cladding outer wall oxidation	Outer wall oxide thickness.

* Heat transfer coefficient data must themselves be based on, or validated against, cladding surface and bulk coolant temperature measurements. The temperature measurements must be made at the same axial position, or there must be a method of determining the bulk coolant temperature at the same axial position as the clad surface temperature measurement using the measured coolant temperatures at other axial elevations, the coolant flow rate and the dissipated power in the corresponding "subchannel" around the rod. The heat transfer coefficient during normal operation conditions will be significantly different to that during design-basis accidents such as a LOCA.

† The as-manufactured clad yield stress at room temperature is usually a code input parameter: (see Table 3).

Some fuel performance codes also have models for generation and release of other fission products, most commonly iodine and caesium (see Section 3.3). Equivalent data to those listed for xenon and krypton are then also needed for these other fission products.

$ Often in the form of fission gas release fraction (commonly denoted FGR) values, i.e. the fraction of the generated fission gas that has been released.

4.3.2. Data needs for the validation of an integrated fuel performance code

The data needs for the validation of an integrated fuel performance code model are summarised in Table 7. All data are generally needed for all fuel rod types that are supported.

Table 7. Data needed for the validation of an integrated fuel performance code

Integral phenomenon	Data needed
Rod pressurisation (primarily due to fuel and clad dimensional changes, fission gas release and helium release and gas temperature effects*).	Rod internal pressure
Fuel elongation (primarily due to fuel thermal expansion, densification and swelling, together with fuel elasticity and creep when there is fuel-clad contact).	Fuel stack length
Clad elongation (primarily due to clad thermal expansion and axial growth, together with clad creep, plasticity and elasticity when there is fuel-clad contact).	Cladding elongation
Fuel radial deformation (primarily due to fuel thermal expansion, densification and swelling, together with fuel elasticity and creep when there is fuel-clad contact).	Fuel pellet outer diameter
Clad radial deformation (primarily due to clad thermal expansion and creep, together with clad creep, plasticity and elasticity when there is fuel-clad contact).	Cladding outer diameter
Fuel-to-coolant heat transfer (primarily due to conduction through pellets, conduction, convection and radiation across the fuel-clad gap, conduction through the clad, and convection between the clad and coolant).	Fuel centreline temperature
Fuel volume change (primarily due to fuel thermal expansion, densification, swelling and elasticity).	Fuel density
Rod free volume change (due to fuel and clad dimensional changes).	Rod free volume

* If modelled, absorption and desorption (in/from fuel pellets) of helium fill gas will also have an effect on rod pressurisation.

5. Conclusions

This working paper emphasises the importance of experimental data in developing fuel models and updating validation data collections, thereby ensuring that the codes remain fit for purpose. It also outlines the data requirements for fuel performance modelling and identifies the types of experimental data that can meet these requirements. The results will also guide the continuous development of the NEA International Fuel Performance Experiments (IFPE) data collection.

It is important to note that while this paper provides the data requirements of the current state of fuel performance codes, it does not extend to addressing the complete requirements of 3D code modelling. Future work in this area is encouraged to continue advancing the understanding of fuel behaviour and improving the safety and efficiency of nuclear power operations.

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