IAEA Databases and Tools for Fuels and FCFs

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NFCMS /NEFW, IAEA

IAEA Sub-Programme 1.2.2:
Nuclear Power Reactor Fuel and Fuel Cycle Facilities

Objectives:

- Support Member States (MSs) to understand and address factors affecting the design, fabrication and in-pile behaviour of currently operating and innovative nuclear fuels and materials for power reactors.
- Support MSs to technically implement IAEA Safety Standards when operating or upgrading existing nuclear fuel cycle facilities, and to understand and address factors affecting the ageing of these facilities.

Through:

- Organizing IAEA meetings and developing IAEA publications
- Coordinating research activities (CRPs)
- Maintaining databases (NFCFs, PIE) Integrated Nuclear Fuel Cycle Information System - IAEA INFCIS and NFC simulation tools (NFCSS) Nuclear Fuel Cycle Simulation System (NFCSS) (iaea.org)
- Developing e-Learning Materials on nuclear fuel OPEN-LMS: All courses (iaea.org)
- Building up Networks among practitioners (NFE-Net) Pages - NFE Net (iaea.org)
- Supporting the IAEA Technical Cooperation Programme

To foster collaboration and information exchange , provide reference data, preserve knowledge, and capacity building

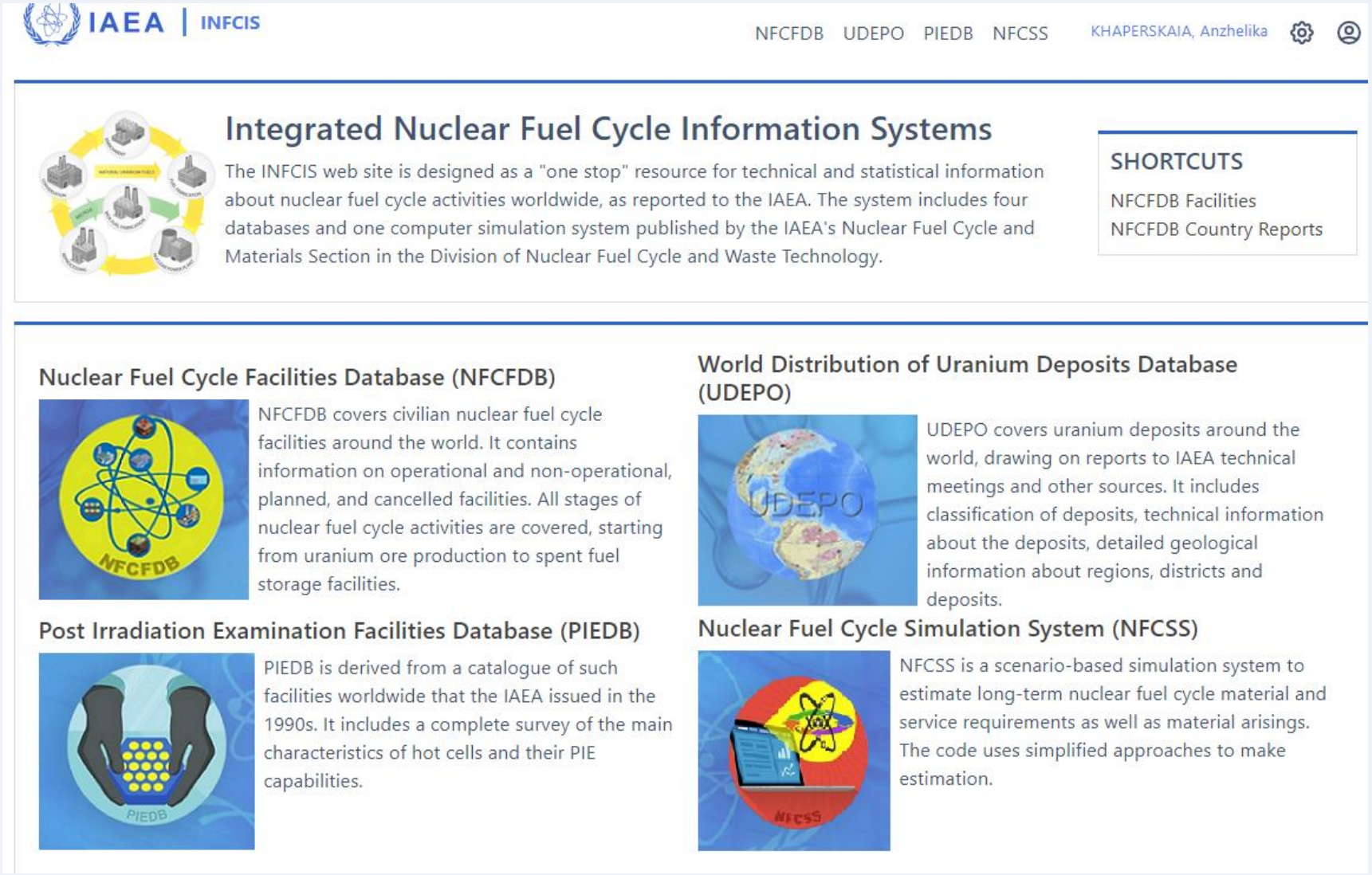
Advised by the TWG FPT:

The Technical Working Group on Fuel Performance and Technology (TWG FPT) is a group of recognized experts from MSs providing advice to DDG-NE and supporting programme implementation, reflecting a global network of excellence and expertise in nuclear power reactor fuel engineering (20 Members + Observers)

Advised by the TWG NFC:

The Technical Working Group on Fuel Cycle Facilities (TWG FCF) is a group of recognized experts from MSs to provide advice to DDG-NE and DDG-NS and support programme implementation, reflecting a global network of excellence and expertise in NFCFs operation areas (20 Members + Observers)

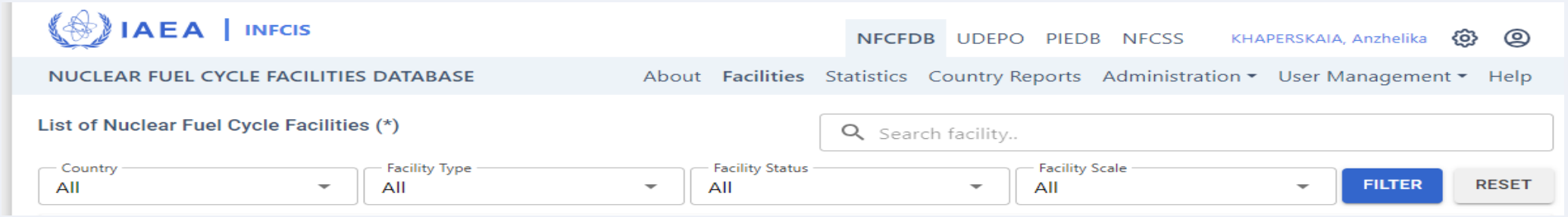
Integrated Nuclear Fuel Cycle Information System

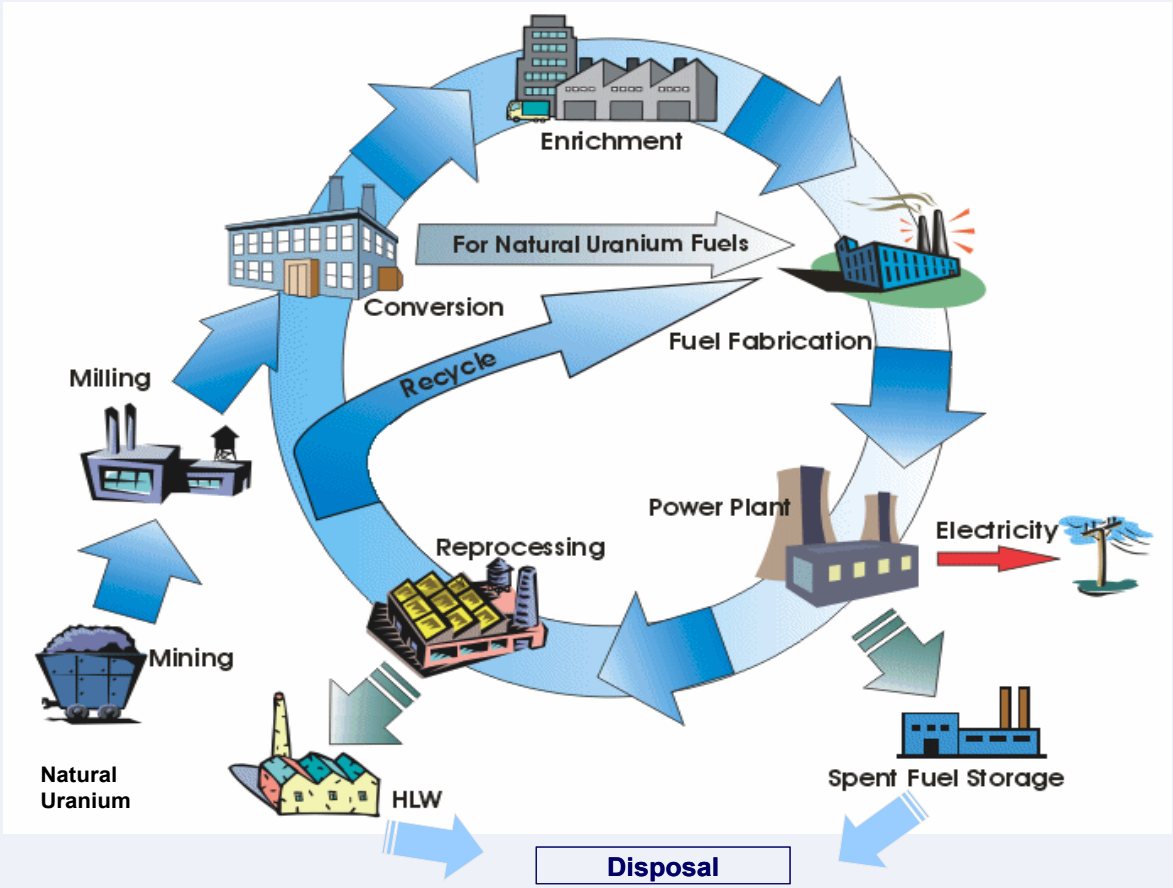


Landing Home page of iNFCIS www.iaea.org/resources/databases/

NFCIS.Contact-Point@iaea.org

Nuclear Fuel Cycle Facilities Database





NFCFDB Caters to

- Uranium Mining and Milling Facilities
- Conversion Facilities
- Enrichment Facilities
- Fuel Fabrication Facilities
- Spent Fuel storage Facilities
- Fuel Reprocessing Facilities
- Waste Management Facilities
- Allied Industries

Facility Details (NFCIS)

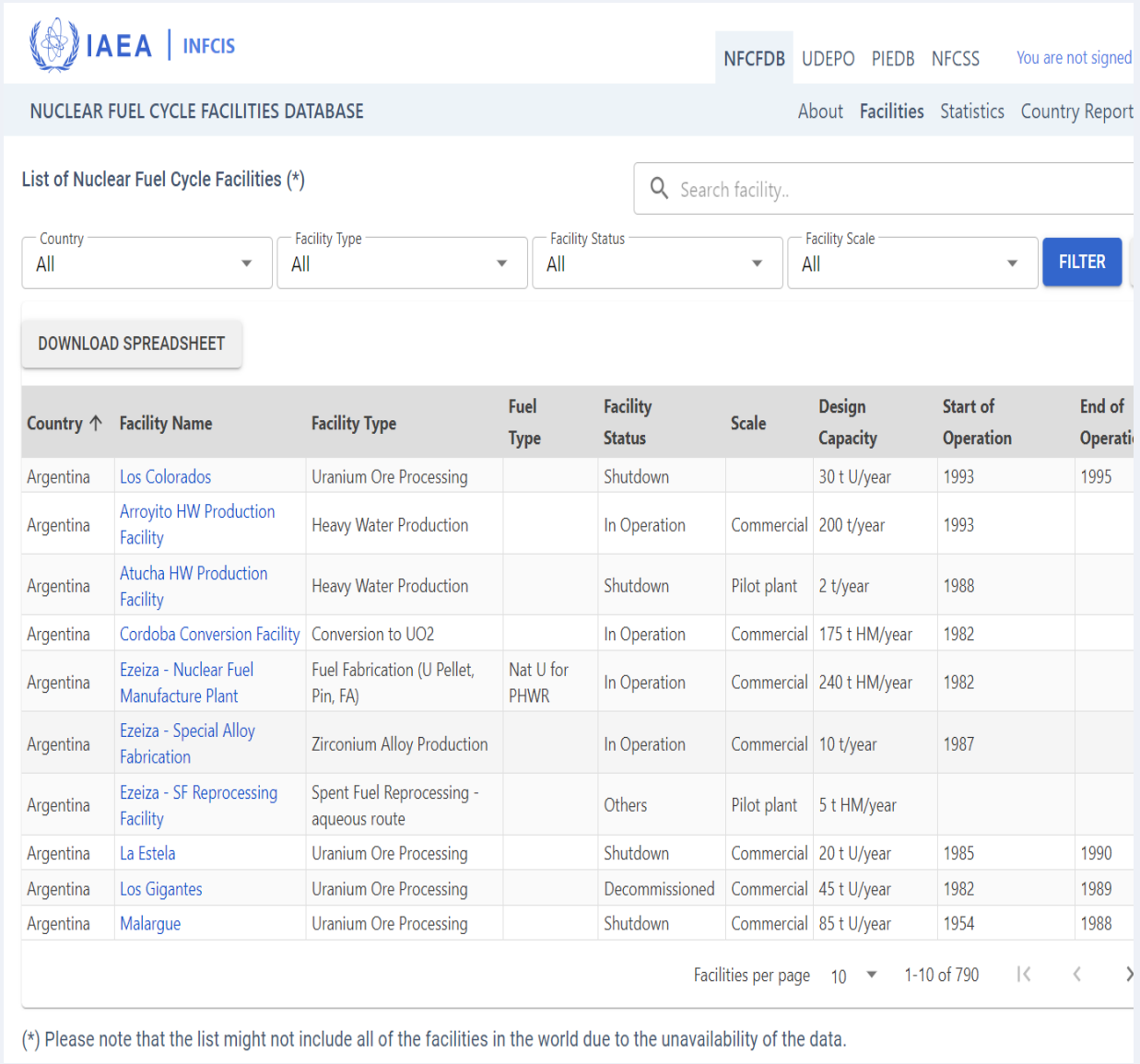
Nuclear Fuel Cycle Facilities Database. General Information (cont.)

NFCF Data (Mandatory)

- Type of facility
- Scale (Lab/pilot plant/ commercial)
- Status (Construction, Commissioning, operation, shutdown, decommissioning, decommissioned, planned, refurbishment)
- Start/close of operation of facility
- Fuel types
- Design Capacity
- Process
- Feed Material
- Product Material
- Contact details
- Operator & Owners

Optional data

- Annual production/Inventory of SF
- Decommissioning Stage
- References
- Location
- Ageing management programmes
- Licensing period

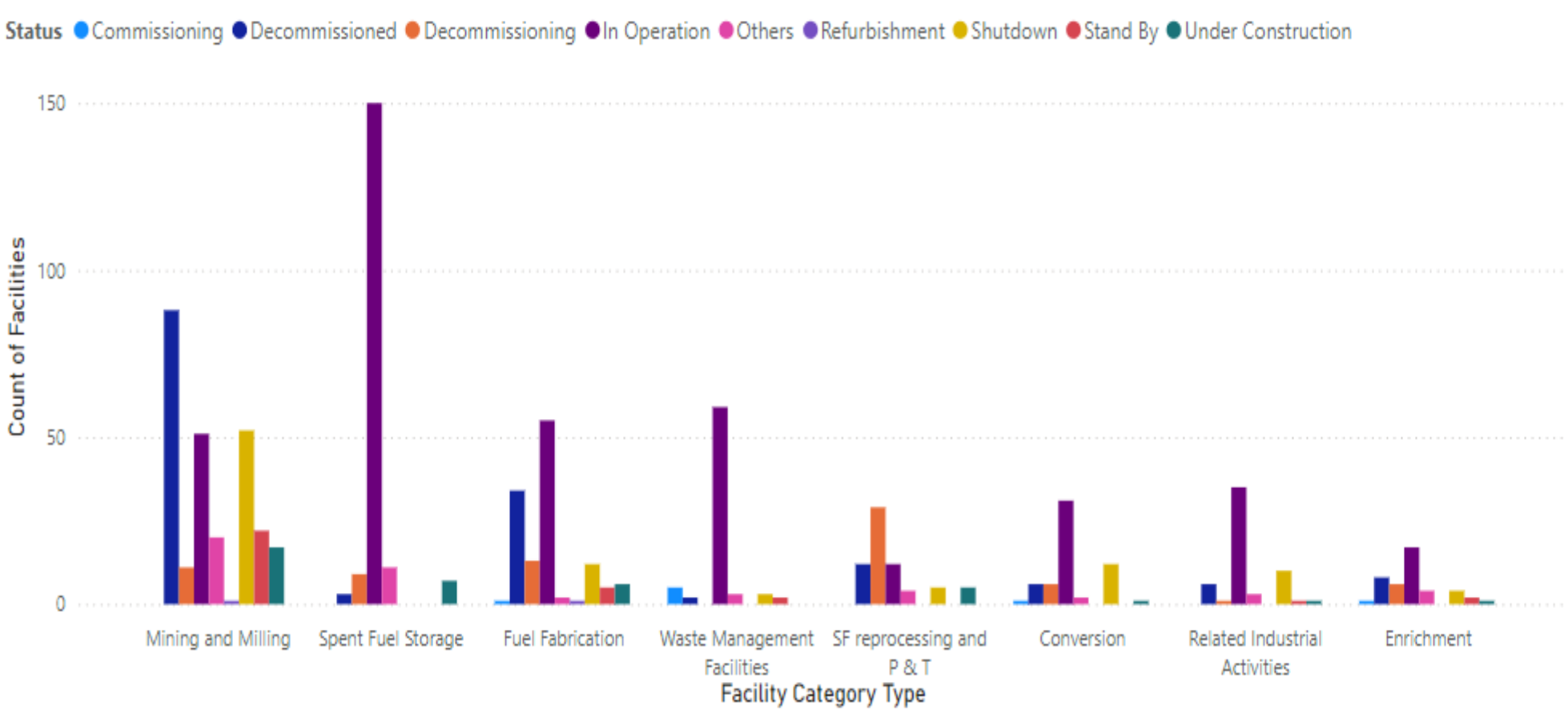


Country	Facility Name	Facility Type	Fuel Type	Facility Status	Scale	Design Capacity	Start of Operation	End of Operati
Argentina	Los Colorados	Uranium Ore Processing		Shutdown		30 t U/year	1993	1995
Argentina	Arroyito HW Production Facility	Heavy Water Production		In Operation	Commercial	200 t/year	1993	
Argentina	Atucha HW Production Facility	Heavy Water Production		Shutdown	Pilot plant	2 t/year	1988	
Argentina	Cordoba Conversion Facility	Conversion to UO2		In Operation	Commercial	175 t HM/year	1982	
Argentina	Ezeiza - Nuclear Fuel Manufacture Plant	Fuel Fabrication (U Pellet, Pin, FA)	Nat U for PHWR	In Operation	Commercial	240 t HM/year	1982	
Argentina	Ezeiza - Special Alloy Fabrication	Zirconium Alloy Production		In Operation	Commercial	10 t/year	1987	
Argentina	Ezeiza - SF Reprocessing Facility	Spent Fuel Reprocessing - aqueous route		Others	Pilot plant	5 t HM/year		
Argentina	La Estela	Uranium Ore Processing		Shutdown	Commercial	20 t U/year	1985	1990
Argentina	Los Gigantes	Uranium Ore Processing		Decommissioned	Commercial	45 t U/year	1982	1989
Argentina	Malargue	Uranium Ore Processing		Shutdown	Commercial	85 t U/year	1954	1988

(*) Please note that the list might not include all of the facilities in the world due to the unavailability of the data.

Nuclear Fuel Cycle Facilities Database. Statistics capabilities

Facility Numbers by Type and Status | Facility Numbers by Country and Status | Facility Numbers by Country and Type



Country Name: All

Facility Category Type: All

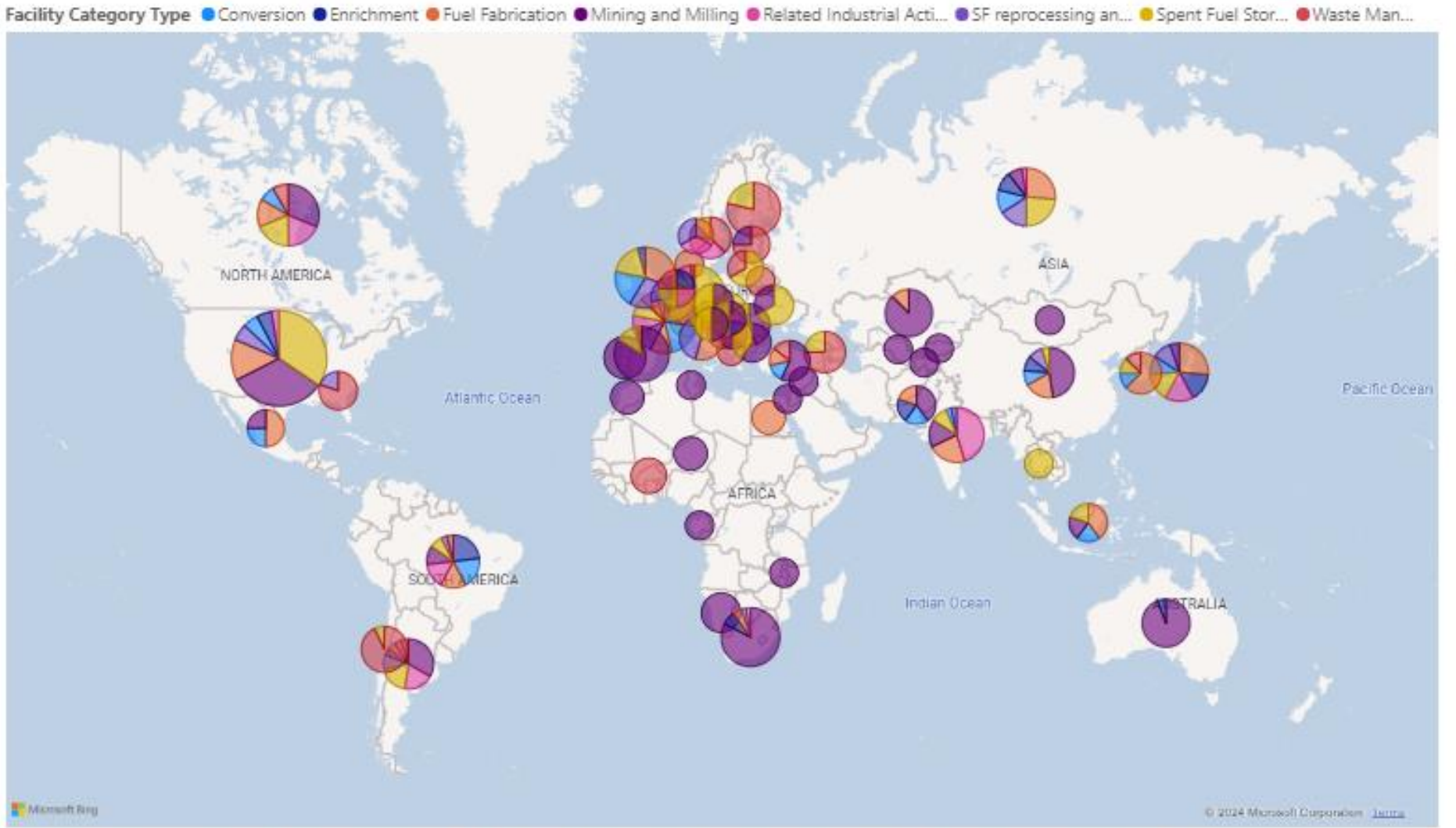
Facility Status: All

Start of Operation: All

	Commissioning	Decommissioned	Decommissioning	In Operation	Others	Refurbishment	Shutdown	Stand By	Under Construction	Total
Total	**8**	**159**	**75**	**410**	**49**	**2**	**98**	**32**	**38**	**871**

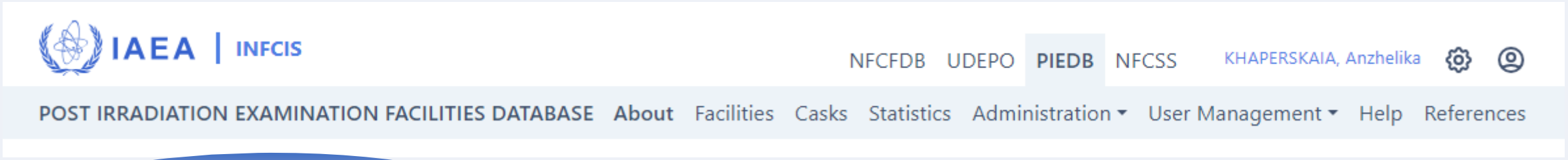
[Facility Details (NFCIS)]

NFCFDB. Mapping the locations of facilities.

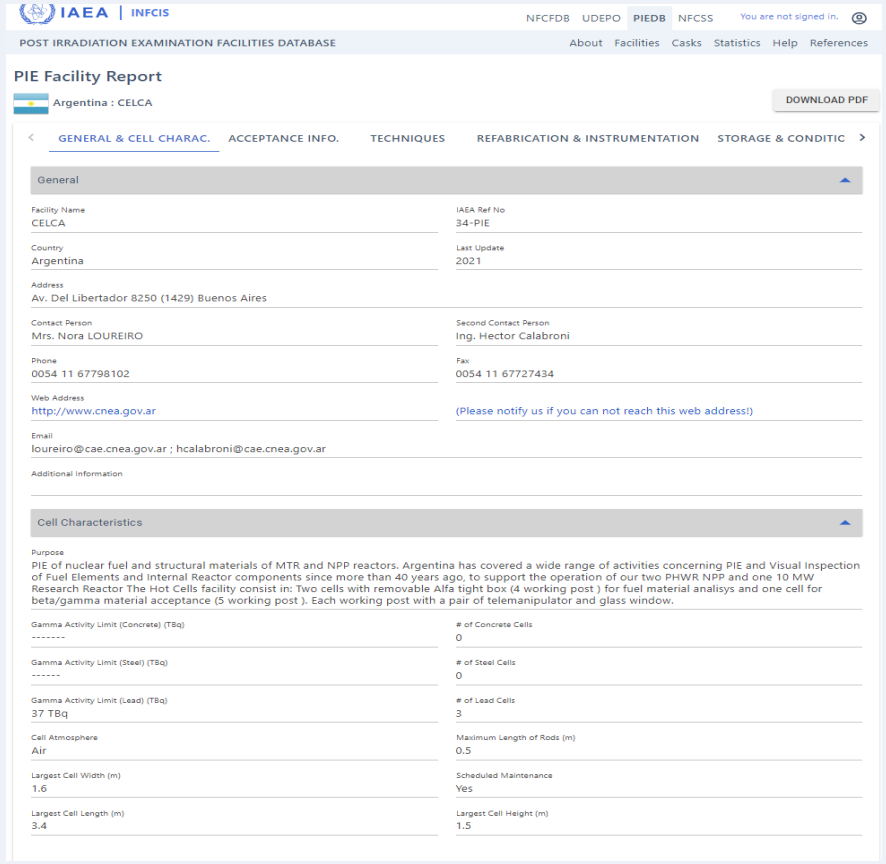


[Facility Details (NFCIS)]

IAEA-EC (HOTLAB) Post Irradiation Examination Facilities Database



52 facilities from 24 countries



The PIEDB consists of 5 main areas describing PIE facilities:

- Acceptance criteria for irradiated components
- Cell characteristics
- PIE techniques (DE, NDE)
- Refabrication/instrumentation capabilities
- Storage and conditioning capabilities

And it has major technical and licensing data of Transport casks for irradiated samples

Post Irradiation Examination Facilities Database (PIE)

IAEA-EC Post Irradiation Examination Facilities Database

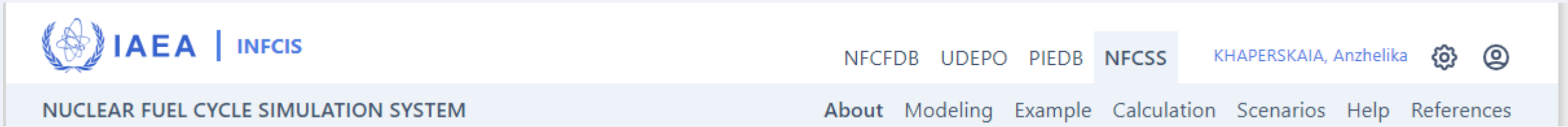
Country Wise Facilities Number of Cells by Country

The boundaries and names shown and the designations used on the map do not imply official endorsement or acceptance by the United Nations nor the IAEA

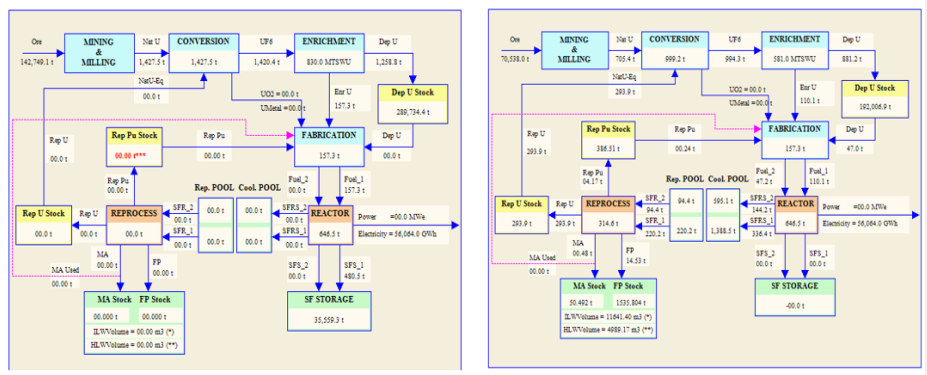
Country	Facility Name	Number of Concrete Cells	Number of Other Cells	Number of Steel Cell
Argentina	CELCA	0	3	0
Belgium	LHMA - Laboratory for High and Medium Activity - SCK-CEN, Belgium	10	20	0
Belgium	SCK•CEN - Chemical and Radiochemical Measurements	0	1	0
Brazil	CTMSP - Hot Cell Pilot Laboratory	0	3	0
Canada	Canadian Nuclear Laboratories	11	4	0
Canada	Centre for Advanced Nuclear Systems, McMaster University	5		
Czech Republic	Hot Cells Facility Research Center Rez			10

[Post Irradiation Examination Facilities Database (PIE)]

Simulation tool (NFCSS)



- NFCSS is a scenario-based publicly available computer model (web-based tool) for the estimation of nuclear fuel cycle material and service requirements
- Reactors types: PWRS, BWRs, PHWRs, RMBKs, AGRs, GCRs, WWERs , FRs
- UOX, MOX and ThOX fuel cycles
- Calculates the requirements for Nat U resources, enrichment and fuel fabrication services, etc. SF inventory, Minor Actinide Inventory, FP inventory, Decay Heat and Radio-toxicity with material Flow Diagrams up to 200 years



Technical Features

- Only long-term actinides are calculated UOX, MOX fuel
- ORIGEN II (PWR-UO2-33G, PWR-MOX and BWR-UO2-27.5G, BWR-Pu) fuel libraries

PWR (Pu-Th MOX) : ORIGEN II library (211) Pu-Th fuel (with modifications)

BWR (Th fuel cycles) : ORIGEN II PWR Th- library (214)

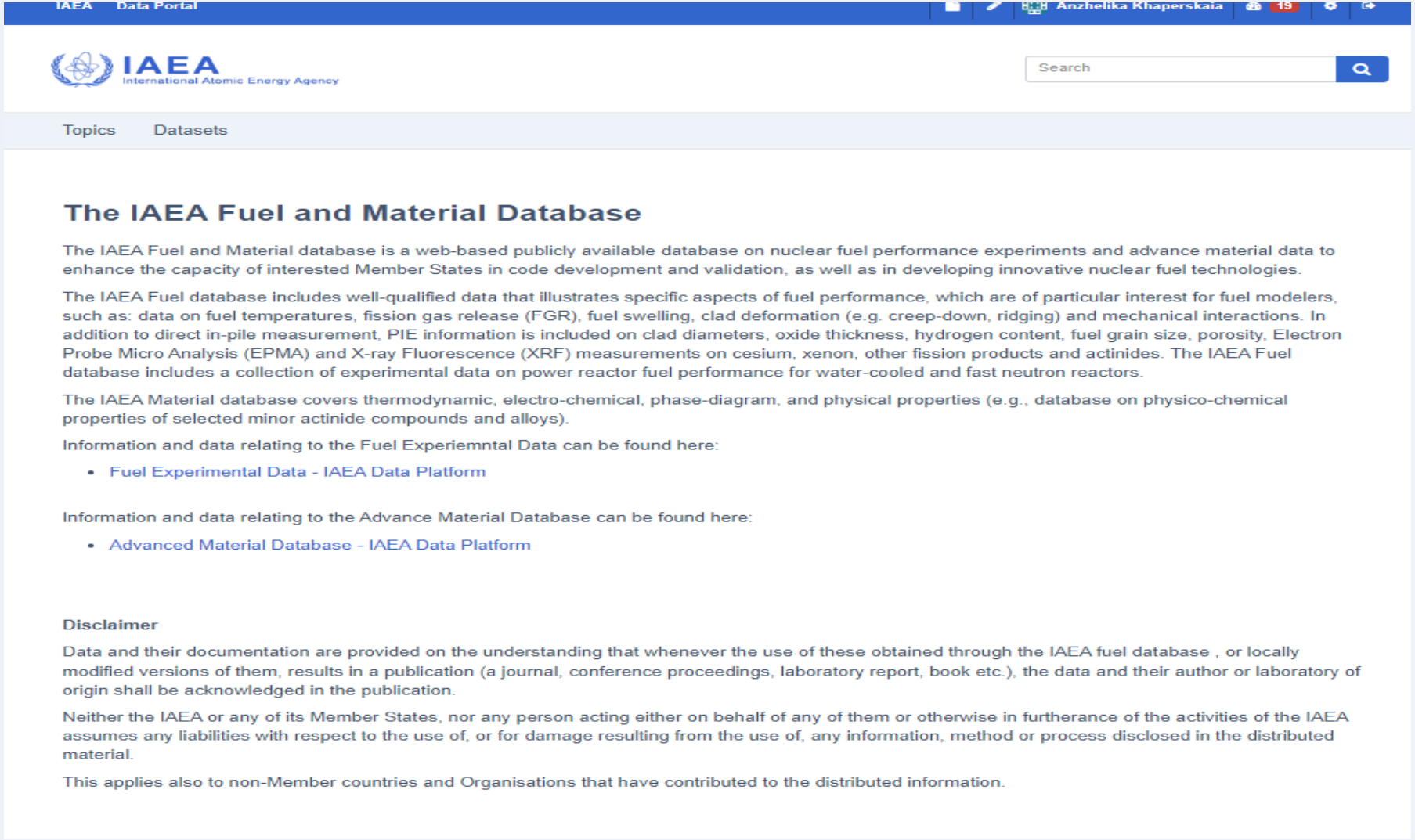
Other reactors : Libraries provided by Consultant experts

Nuclear Fuel Cycle Simulation System (NFCSS)

	Open cycle	Closed Cycle	
The total natural uranium (2025 to 2110)*	337831 tones	202973 tones	
The total spent fuel (or HLW) accumulated in the end of life cycle	37228 tones of SNF	High-level waste	5405 tones
		Plutonium	478 tones
		Minor actinides	54.7 tones

* Example for calculation for NES with 10 PWRs (45GWd/t burn up) having a 60-years lifespan

IAEA Fuel and material database



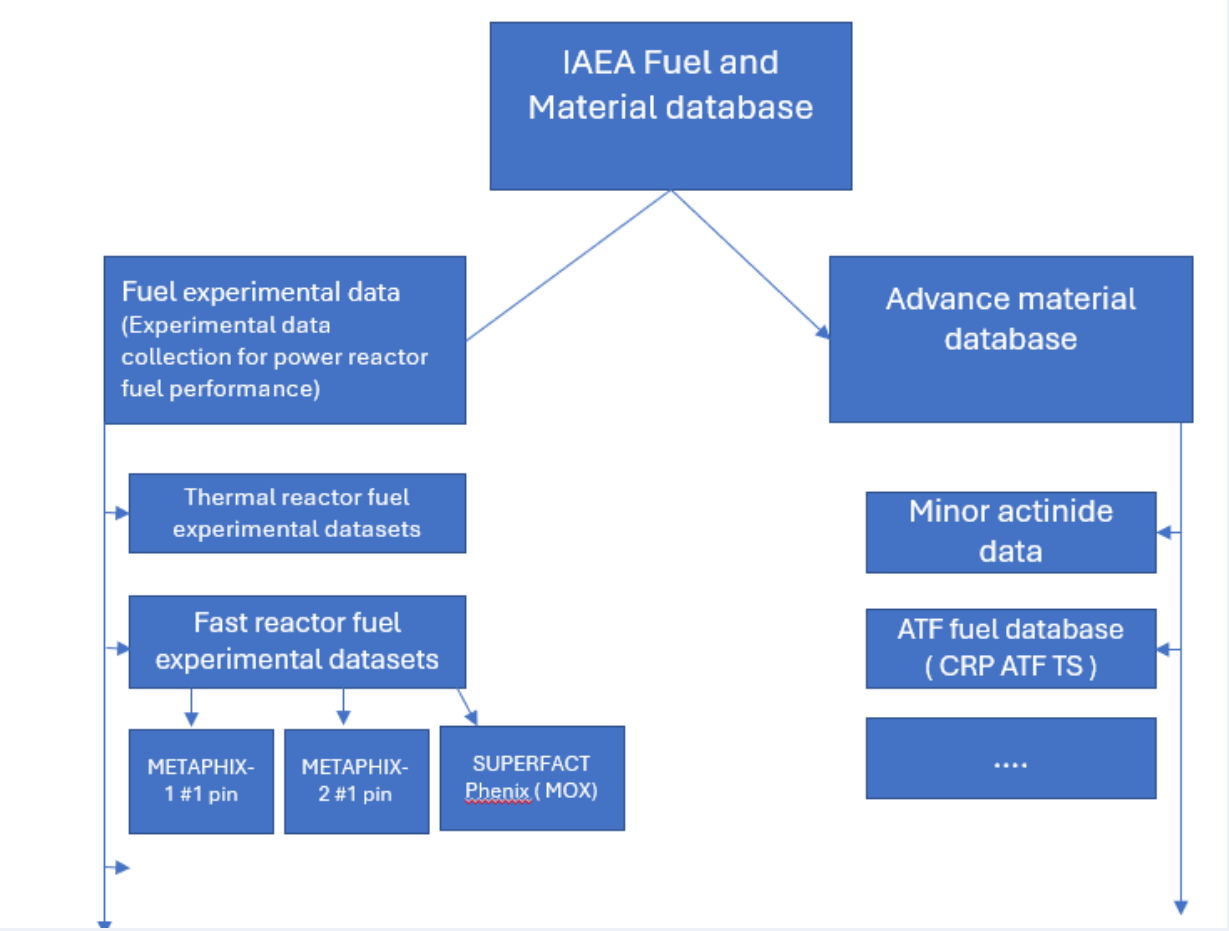
The IAEA Fuel and Material Database - IAEA Data Platform;

IAEA Fuel and material database (cont.)

The **IAEA Fuel database** includes a collection of experimental data on power reactor fuel performance (water-cooled and fast neutron reactors)
Well-qualified data that illustrates specific aspects of fuel performance, which are of particular interest for fuel modelers, such as:

- Data on fuel temperatures
- Fission gas release (FGR)
- Fuel swelling
- Clad deformation (e.g., creep-down, ridging)
- Mechanical interactions

PIE information is included on clad diameters, oxide thickness, hydrogen content, fuel grain size, porosity, Electron Probe Micro Analysis (EPMA) and X-ray Fluorescence (XRF) measurements on cesium, xenon, other fission products and actinides



The **IAEA Material database** covers thermodynamic, electro-chemical, phase-diagram, and physical properties (e.g., database on physico-chemical properties of selected minor actinide compounds and alloys)

IAEA fuel experimental database in IAEA Open Data platform

Fuel Experimental Data

Improved understanding of fuel performance can lead to a reduction in operating margins, increased flexibility in fuel management and improved operating economics.

To better understand fuel performance, the IAEA has addressed different aspects of fuel behaviour modelling in a series of coordinated research projects aimed at:

- assessing fuel performance codes and supporting countries with code development and application needs,
- building a database of well-defined experiments suitable for code validation,
- transferring a mature fuel modelling code to developing countries and supporting its adaptation to the requirements of reactors,
- providing guidance on applying that code to reactor operation and safety assessments, and
- providing guidelines for code quality assurance, code licensing and code application to fuel licensing.

The IAEA Fuel database is a collection of fuel performance experimental data on power reactor fuel performance (water-cooled and fast neutron reactors) derived from IAEA coordinated research projects.

The purpose of this database is to preserve information from fuel performance experiments, such that the data is useful and in a form that can be used for code development and validation. Therefore, each dataset includes sufficient information to assemble an input file to run a code and information against which to compare predictions (data on fuel temperatures, fission gas release (FGR), fuel swelling, clad deformation (e.g., creep-down, ridging) and mechanical interactions. In addition to direct in-pile measurement, PIE information is included on clad diameters, oxide thickness, hydrogen content, fuel grain size, porosity, Electron Probe Micro Analysis (EPMA) and X-ray Fluorescence (XRF) measurements on cesium, xenon, other fission products and actinides.

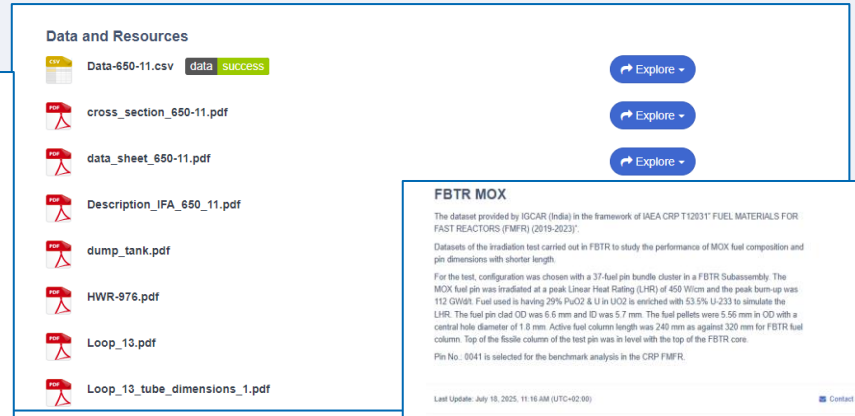
The IAEA would like to thank the OECD/NEA Halden Reactor Project for providing data for some of the IAEA coordinated research projects.

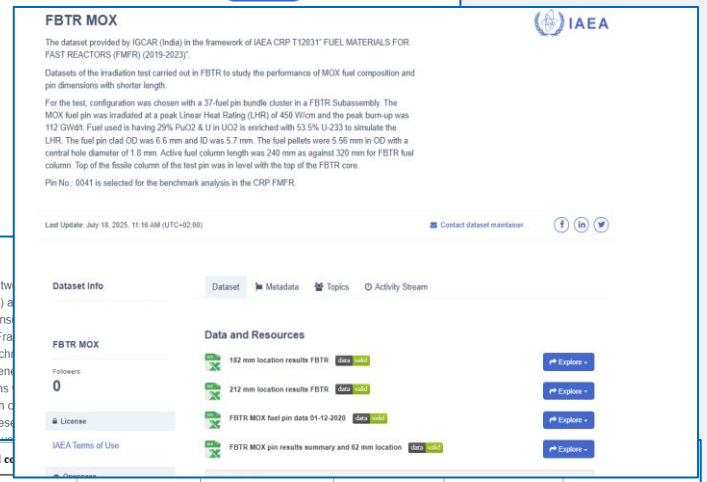
Datasets relating to fuels for thermal reactors can be found here:

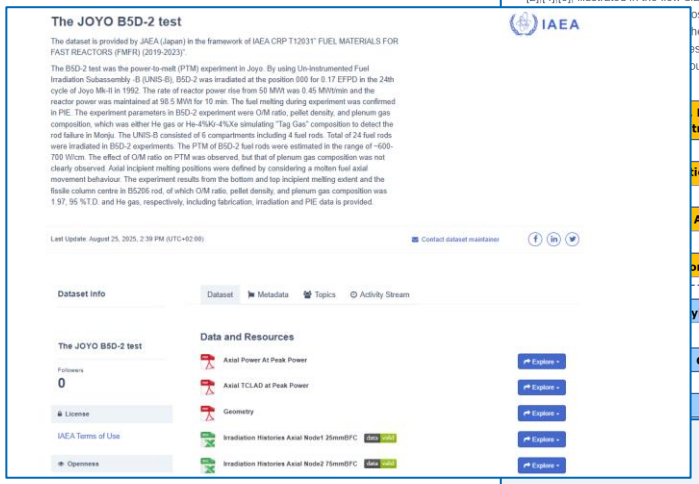
- AEKI Burst Tests
- Halden Reactor Test IFA-650.9 (PWR)
- Halden Reactor Test IFA-650.10 (PWR)
- Halden Reactor Test IFA-650.11 (PWR)
- KIT CORA-15
- KIT QUENCH-L0 Bundle Test
- KIT QUENCH-L1 Bundle Test
- Studsvik LOCA test (NRC-192)
- CRIEPI - DEGREE Single Rod and Bundle Tests

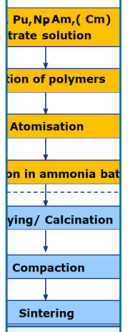
Datsets relating to fuel for fast reactors can be found here:

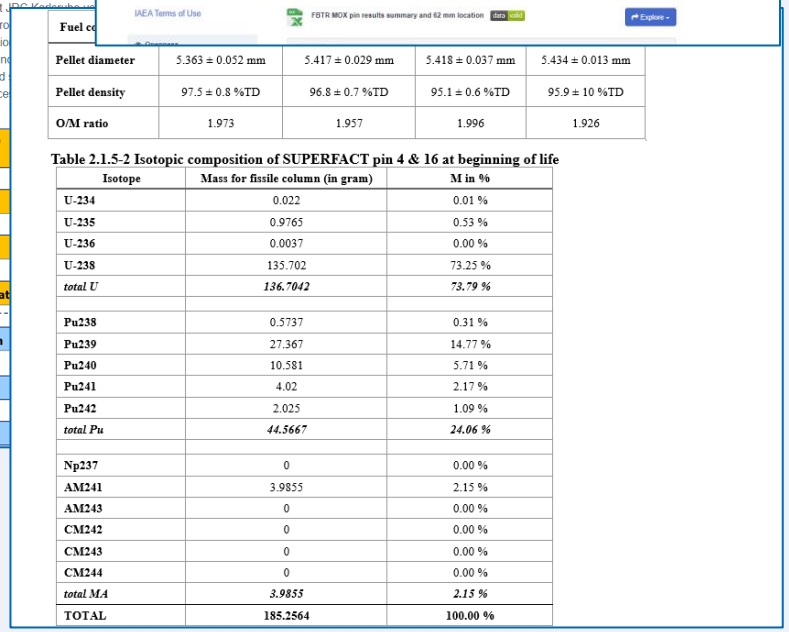
- METAPHIX-1#1
- METAPHIX-2#1
- Sentenay, Phenix (MOX fuel)
- SUPERFACT (MOX fuel)
- FBTR MOX
- The JOYO B5D-2 test











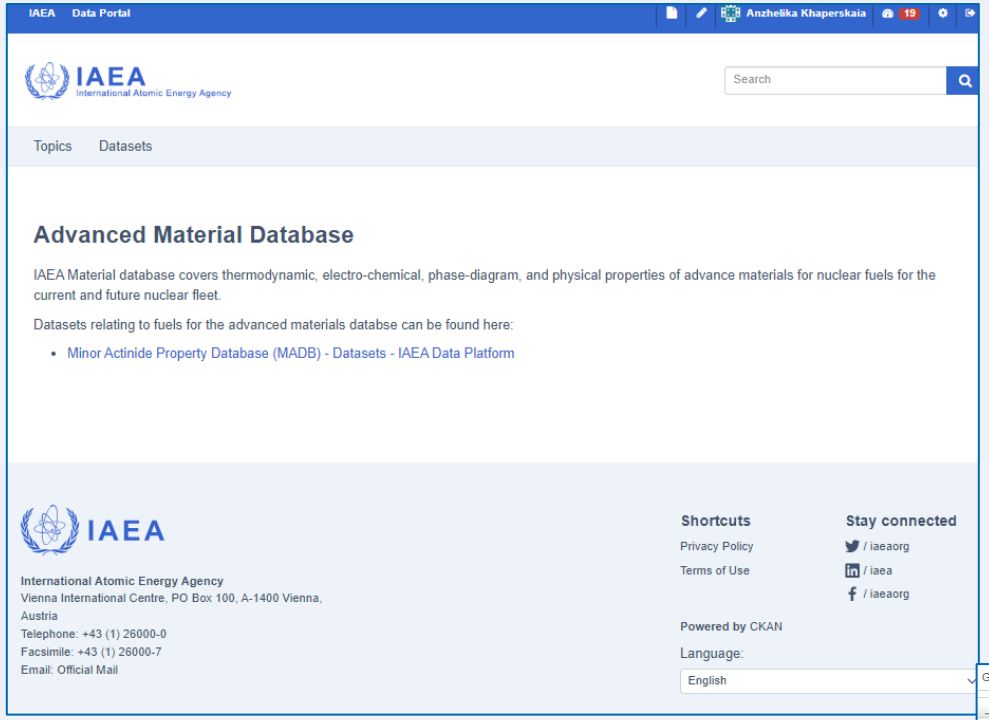
Pellet diameter	5.363 ± 0.052 mm	5.417 ± 0.029 mm	5.418 ± 0.037 mm	5.434 ± 0.013 mm
Pellet density	97.5 ± 0.8 %TD	96.8 ± 0.7 %TD	95.1 ± 0.6 %TD	95.9 ± 10 %TD
O/M ratio	1.973	1.957	1.996	1.926

Table 2.1.5-2 Isotopic composition of SUPERFACT pin 4 & 16 at beginning of life

Isotope	Mass for fissile column (in gram)	M in %
U-234	0.022	0.01 %
U-235	0.9765	0.53 %
U-236	0.0037	0.00 %
U-238	135.702	73.25 %
total U	***136.7042***	***73.79 %***
Pu238	0.5737	0.31 %
Pu239	27.367	14.77 %
Pu240	10.581	5.71 %
Pu241	4.02	2.17 %
Pu242	2.025	1.09 %
total Pu	***44.5667***	***24.06 %***
Np237	0	0.00 %
AM241	3.9855	2.15 %
AM243	0	0.00 %
CM242	0	0.00 %
CM243	0	0.00 %
CM244	0	0.00 %
total MA	***3.9855***	***2.15 %***
TOTAL	**185.2564**	**100.00 %**

[Fuel Experimental Data - IAEA Data Platform]

IAEA Advance Material database in Open Data platform



Minor Actinide Property Database (MADB)

MADB is a bibliographic database on physico-chemical properties of selected Minor Actinide compounds and alloys. The materials and properties are selected based on their importance in the advanced nuclear fuel cycle options.

[Minor Actinide Property Database (MADB) - Minor Actinide Property Database (MADB) - Datasets - IAEA Data Platform]

Grid | Graph | Map | 416 records | « 1 – 100 » | Search data ... | Go »

_id	Publicat...	PublicationTitle	Publicat...	Publicat...	Publicat...	Publicat...	Publicat...	Explana...	Publicat...	Combin...	inTECD...	OrgTitle	Hardcop...	TitleforT...	Abstract	Pub
1	1	Chapter 11. Neptunium Group 15 Compounds and Complexes - Neptunium Nitrides and Other Pnicti...	1	1	Vol 4 - C...	2001	187-189			Robert J....	Yes		Yes	Chapter ...		yes
2	2	V6. Group 15 Compounds and Complexes - Americium Nitrides	2	1	Vol 2 - C...	1995	134-135	NULL	NULL	Robert J....	Yes	NULL	Yes	V6. Grou...	NULL	yes
3	3	A note on AmN and AmO	3	3	Vol. 29	1967	2650-2652	NULL	NULL	Y. Akimoto	Yes	NULL	Yes	A note o...	NULL	yes
4	4	Preparation and Lattice Parameters of Actinide Monochalcogenides and Monopnictides	4	4	NULL	1976	79-84	NULL	NULL	J.P. Char...	Yes	NULL	Yes	Preparati...	NULL	yes
5	5	Synthesis and Study of the Binary Compounds of the Actinides and Lanthanides (III Preparation of A...	10	5	Vol. 247	1982	144-146	NULL	NULL	V.M. Ra...	Yes	NULL	Yes	Synthesi...	NULL	yes
6	6	Techniques of Preparation and Crystal Chemistry of Transuranic Chalcogenides and Pnictides	6	6	NULL	1978	NULL	NULL	NULL	D. A. Da...	Yes	NULL	Yes	Techniqu...	NULL	yes
7	7	Lattice Parameter Expansion by Self-Irradiation Damage of ^{244}Cm-^{240}Pu Oxide and Mononitride	19	7	Suppl. 3	2002	842-845	NULL	NULL	M. Takan...	Yes	NULL	Yes	Lattice P...	NULL	yes
8	8	Fabrication of Americium-based Nitrides by Carbothermic Reduction Method	20	7	Suppl. 3	2002	737-740	NULL	NULL	A. Itoh, ...	Yes	NULL	Yes	Fabricati...	NULL	yes
9	9	Actinide Nitrides and Nitride-halides in High-temperature Systems	22	8	Vol. 271-...	1998	347-354	NULL	NULL	T. Ogaw...	Yes	NULL	Yes	Actinide ...	NULL	yes
10	10	Carbothermic Synthesis of (Cm,Pu)N	19	9	Vol. 294	2001	24-27	NULL	NULL	M. Takan...	Yes	NULL	Yes	Carboth...	NULL	yes
11	11	Fabrication of nitride fuels for transmutation of minor actinides	28	9	Vol. 320	2003	18-24	NULL	NULL	K. Minat...	Yes	NULL	Yes	Fabricati...	NULL	yes
12	12	Synthesis of americium mononitride by carbothermic reduction method	19	10	CD-ROM	1999	CD-ROM	NULL	NULL	M. Takan...	Yes	NULL	Yes	Synthesi...	NULL	yes
13	13	Research on nitride fuel for transmutation of minor actinides	28	11	CD-ROM	2002	CD-ROM	NULL	NULL	K. Minat...	Yes	NULL	Yes	Researc...	NULL	yes
14	14	Fabrication and property measurements of MA nitride fuels and LLFP targets	21	12	CD-ROM	2003	CD-ROM	NULL	NULL	M. Akab...	Yes	NULL	Yes	Fabricati...	NULL	yes
15	15	Preparation of neptunium mononitride by carbothermic reduction	34	7	Vol. 31	1994	677-680	NULL	NULL	Y. Suzuk...	Yes	NULL	Yes	Preparati...	NULL	yes
16	16	Research on actinide mononitride fuel	29	13	Vol. 1	1995	538-545	NULL	NULL	Y. Arai, Y...	Yes	NULL	Yes	Researc...	NULL	yes
17	17	Recent progress of nitride fuel development in JAERI - fuel property, irradiation behavior and applicat...	29	14	Vol. 1	1997	664-669	NULL	NULL	Y. Arai, T...	Yes	NULL	Yes	Recent p...	NULL	yes
18	18	Vaporization behavior of neptunium mononitride	30	9	Vol. 247	1997	33-36	NULL	NULL	K. Nakaji...	Yes	NULL	Yes	Vaporiza...	NULL	yes
19	19	Vaporization behavior of (Np,Pu)N	30	8	Vol. 271-...	1998	666-669	NULL	NULL	K. Nakaji...	Yes	NULL	Yes	Vaporiza...	NULL	yes
20	20	Vaporization behavior of (Pu,Am)N	22	8	Vol. 224	1995	55-59	NULL	NULL	T. Ogaw...	Yes	NULL	Yes	Vaporiza...	NULL	yes
21	21	Vaporization behavior of NpN coloaded with PuN	30	9	Vol. 275	1999	332-335	NULL	NULL	K. Nakaji...	Yes	NULL	Yes	Vaporiza...	NULL	yes
22	22	Heat capacity of neptunium mononitride	30	7	Suppl. 3	2002	620-623	NULL	NULL	K. Nakaji...	Yes	NULL	Yes	Heat cap...	NULL	yes

Open-source ATF material properties database

- ._Thermo-mechanical assessment of full SiC composite cladding for LWR applications with sensitivity analysis.pdf
- A review on thermohydraulic and mechanical physical properties of SiC_FeCrAl and Ti3SiC2 for ATF cladding.pdf
- Accident tolerant clad material modeling by MELCOR- Benchmark for SURRY Short Term Station Black Out.pdf
- Advanced Doped Pellet Technology (ADOPT) Fuel.pdf
- Alloy Selection for Accident Tolerant Fuel Cladding in Commercial Light Water Reactors.pdf
- An investigation of FeCrAl cladding behavior under normal operating and loss of coolant conditions.pdf
- Analysis of Options and Experimental Examination of Fuels for Water Cooled Reactors with Increased Accident Tolerance (
- Analysis of Pellet-to-Cladding Gap Closure for a Metallic Micro-cell Pellet under Normal Operating Conditions.pdf
- ASSESSMENT AT CEA OF COATED NUCLEAR FUEL CLADDING FOR LWRS WITH INCREASED MARGINS IN LOCA AND BEYO
- Assessment of swelling and constituent redistribution in uranium- zirconium fuel using phenomena identification and ranl
- Atomic-scale interface structure of a Cr-coated Zircaloy-4 material.pdf
- Comparison Pictures igh-Temperature Steam Oxidation Kinetics of Advanced Cladding Materials.pdf
- Contribution to the description of the absorber rod behavior in severe accident conditions- An experimental investigation
- Corrosion behavior of Fe-Cr-Si alloys in simulated PWR primary water environment.pdf
- Cracking of Cr-coated accident-tolerant fuel during normal operation and under power-ramping conditions.pdf
- Development and Assessment of an Extended MATPRO Materials Property Library for Accident Tolerant Fuel Materials.pdf
- Early studies on Cr-Coated Zircaloy-4 as enhanced accident tolerant nuclear fuel claddings for light water reactors.pdf
- Effect of Al and Cr Content on Air and Steam Oxidation of FeCrAl Alloys and Commercial APMT Alloy.pdf
- EFFECT OF CHROMIUM COATINGS ON THE MECHANICAL PROPERTIES OF Zr1Nb FUEL CLADDINGS IN LONGITUDINAL AN
- Effect of water vapor on the oxide growth in FeCrAl-based oxide dispersion-strengthened fuel cladding material at 1100 C
- Effectiveness of Cr-Coated Zr-Alloy Clad in Delaying Fuel Degradation for a PWR During a Station Blackout Event.pdf

- Measurements of the thermal conductivity of uranium dioxide by the Flashing Method.pdf
- Mechanical Behavior of FeCrAl and Other Alloys Following Exposure to LOCA Conditions Plus Quen
- Mechanical performance of SiC three-layer cladding in PWRs.pdf
- Mechanical property and damping capacity of ultrafine-grained Fe–13Cr–2Al–1Si alloy produced by
- Microstructural characterization of accident tolerant fuel cladding with Cr–Al alloy coating layer aft
- Microstructure dependent thermal conductivity measurement of Zircaloy-4 using an extended Ram
- Mid-Term Review of Predictive Modelling Activities.pdf
- Modeling QUENCH-14 with RELAP5SCDAP MOD3.5.pdf
- Modelling of Fuel Behaviour in Design Basis Accidents and Design Extension Conditions _IAEA-TE-1
- Modification of MELCOR for severe accident analysis of candidate.pdf
- Morphology_and_Phase_Distributions_of_Molten_Core_in a reactor vessel.pdf
- Neutronic performance of fully ceramic microencapsulated of uranium oxycarbide and uranium nit
- Nitriding and Re-oxidation Behavior of Zircaloy-4 at High Temperatures- PhD_Thesis_Park_Official_
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- On the oxidation mechanism of U3Si2 accident tolerant nuclear fuel.pdf
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- Environmental behavior of light water reactor accident tolerant candidate cladding materials under design condi
- Evaluation of Accident Tolerant FeCrAl Coating for PWR Cladding under Normal Operating Conditions with Coup
- Evaluation of Equivalent Cladding Reacted parameters of Cr-coated claddings oxidized in steam at 1200C in relat
- EVALUATION_METRICS_APPLIED_TO_ACCIDENT_TOLERANT_FU.pdf
- Experimental Results of Reflood Bundle Test QUENCH-15 with ZIRLO™ Cladding Tubes.pdf
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- FEM MODELLING OF THE EXPANDING MANDREL TEST SIMULATING OUT-OF-PILE PCI SCC OF FUEL CLADDING.p
- Fission gas diffusion and release for Cr2O3-doped UO2_ From the atomic to the engineering scale.pdf
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- High temperature steam oxidation of chromium-coated zirconium-based Alloys- Kinetics and process.pdf
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- Large eddy simulation on turbulent heat transfer in reactor vessel lower head corium pools.pdf
- Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant.pdf

These databases will be incorporated into NEXSHARE platform :

- Nuclear Fuel Cycle Facilities Database
- Post-Irradiation Examination Facilities
- Simulation Tool (NFCSS)
- IAEA Fuel and Material Database

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