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Das Bundesamt für Energie (BFE) betreibt ein Monitoring über die Entwicklung der Kernenergie im Rahmen der Berichterstattung des Bundesrats an die Bundesversammlung. Rechtliche Grundlage bildet dafür Artikel 74a des Kernenergiegesetzes vom 21. März 2003 (KEG; SR 732.1). Letztmals wurde das Monitoring zur Kernenergie 2017 aktualisiert[1], weshalb das Bundesamt für Energie im Frühling 2023 eine separate Studie zum Stand der Entwicklung der Kernenergietechnologie in Auftrag gegeben hat.

Für den Inhalt des Berichts sind ausschliesslich die Autorinnen und Autoren aus dem ETH-Bereich verantwortlich.

[1] https://www.bfe.admin.ch/bfe/de/home/versorgung/stromversorgung/bundesgesetz-erneuerbare-stromversorgung.exturl.html/aHR0cHM6Ly9wdWJkYi5iZmUuYWRtaW4uY2gvZGUvcHVibGljYX/Rpb24vZG93bmxvYWQvODg2NA==.html

Contents

Contents 4
List of abbreviations 8
Zusammenfassung 10
Résumé 25
Sintesi 40
Executive summary 53
1 Introduction 65
1.1 Current role of nuclear energy in the Swiss energy mix 65
1.2 Nuclear Energy Worldwide 67
1.3 Technological development of nuclear energy 73
1.4 Economics 75
1.4.1 Factors affecting the economics of nuclear power plants 81
1.4.2 Government subsidies 85
1.5 Deep geological repository 86
1.6 References 87
2 Large-scale WCRs Gen-III/III+ – state of technology and main actors 89
2.1 Main actors – large Gen-III/III+ NPP available on the market 92
2.2 Safety philosophy of Gen-III/III+ NPPs 103
2.2.1 Passive Safety Systems – Main concepts 107
2.3 Conclusions 111
2.4 References 112
3 SMRs – state of technology and main actors 113
3.1 Main SMR designs available on the market 115
3.2 Safety philosophy of LWR SMRs 126
3.3 Emergency planning zone for SMRs 129
3.4 SMRs challenges 130
3.4.1 SMRs fuel 131
3.4.2 SMRs licensing 131
3.5 Conclusions 134
3.6 References 134
4 Microreactors 136
4.1 Microreactors challenges and opportunities 143
4.2 Conclusions 144
4.3 References 145
5 State of GEN IV technologies including SMRs 146
5.1 Generation-IV International Forum (GIF) 146
5.2 Overview of GIF reactor systems 147

5.3 Supercritical Water Reactor (SCWR) ... 150
5.3.1 Overview of SCWR ... 150
5.3.2 Examples of SCWR projects including SMRs ... 150
5.4 Very High Temperature Reactor (VHTR) ... 151
5.4.1 Overview of VHTR ... 151
5.4.2 Examples of HTR projects including SMRs ... 152
5.5 Gas Fast Reactor (GFR) ... 154
5.5.1 Overview of GFR ... 154
5.5.2 Examples of GFR projects including SMRs ... 154
5.6 Sodium Fast Reactor (SFR) ... 155
5.6.1 Overview of SFR ... 155
5.6.2 Examples of SFR projects including SMRs ... 156
5.7 Lead Fast Reactor (LFR) ... 159
5.7.1 Overview of LFR ... 159
5.7.2 Examples of LFR projects including SMRs ... 160
5.8 Molten Salt Reactor (MSR) ... 162
5.8.1 Overview of MSR ... 162
5.8.2 Examples of MSR projects including SMRs ... 166
5.9 Questions and Concluding remarks ... 169
5.10 References ... 174
6 Review of fuel availability ... 177
6.1 Introduction ... 177
6.2 Some Fundamentals of the Nuclear Fuel Supply Chain ... 177
6.3 Sources of Information and Assumptions ... 179
6.4 Current and Future Demand for Nuclear Fuel ... 179
6.4.1 Uranium ... 180
6.4.2 Conversion ... 181
6.4.3 Enrichment ... 182
6.4.4 Fabrication ... 184
6.5 Current and Future Fuel Supply ... 185
6.5.1 Uranium ... 185
6.5.2 Conversion ... 189
6.5.3 Enrichment ... 192
6.5.4 Fabrication ... 194
6.6 Secondary Supply ... 197
6.7 Transport and Logistics ... 199
6.8 Nuclear Fuel Markets – The Cost of Nuclear Fuel ... 199
6.9 Geopolitical Trends and Consequences ... 201
6.10 Fuel Supply Beyond 2040 ... 205
6.10.1 Conventional Uranium Supply ... 205
6.10.2 Unconventional Sources of Uranium ... 205

6.10.3 Alternative Sources of Nuclear Fuel 206
6.10.4 Summary 208
6.11 Summary and Conclusions 208
6.12 References 210
7 Alternative fuel cycles **212**
7.1 General background for alternative fuel cycles 212
7.1.1 Characterization of natural resources 212
7.1.2 ^{235}U as a part of natural uranium 213
7.1.3 Fuel burnup 213
7.1.4 Synthetic actinides 214
7.1.5 Induced radiotoxicity 215
7.2 Alternative fuel cycles 217
7.2.1 List of possible fuels 218
7.2.2 Natural uranium burning in heavy water moderated reactors 219
7.2.3 LEU burning in water moderated reactors in once through cycle 220
7.2.4 HALEU burning in non-water moderated reactors in once through cycle 220
7.2.5 HALEU burning with thorium in once through cycle 221
7.2.6 Fuel reprocessing 221
7.2.7 Pu and RepU single- or multi- recycling in LWR 223
7.2.8 MOX as initial fuel for fast breeder reactor operated in closed U-Pu cycle 224
7.2.9 Thorium utilization in closed Th-U fuel cycle 225
7.2.10 Breeding in open fuel cycle without reprocessing 225
7.2.11 Dedicated MAs burning 225
7.3 Conclusions 227
7.4 References 228
8 Nuclear energy life cycle analysis **229**
8.1 Methodology – Environmental Life Cycle Assessment (LCA) 229
8.1.1 System boundaries and functional unit (FU) 229
8.1.2 Life cycle inventory data 230
8.2 Literature review – life-cycle GHG emissions 230
8.3 Parametrization of a generic nuclear LCA model 231
8.4 Swiss-specific nuclear LCA 233
8.4.1 Swiss-specific LCA: modeling and results 234
8.5 Comparison of nuclear power with other electricity generation technologies 236
8.6 Main uncertainties, data gaps and areas of further research 240
8.7 References 241
9 Legal framework for the construction of new NPPs in Switzerland **243**
9.1 Introduction 243
9.2 Legal restrictions 244
9.2.1 Legal restrictions 244

9.2.2 Level of safety 244
9.2.3 Licensing procedure 244
9.2.4 Stability of the regulatory framework 246
9.3 Options to accelerate the licensing procedures of new NPPs 247
9.3.1 Waiver of the general licence 247
9.3.2 Streamlining of the licensing procedures 248
9.3.2.1. Requirements for the granting of a general licence 248
9.3.2.2. Content of the general licence 248
9.3.2.3. Construction licence 249
10 State of fusion technology and main actors **252**
10.1 Introduction 252
10.2 Fusion reactions of interest and fuel resources 254
10.3 Figure of merit for fusion performance – the triple product 256
10.4 Brief overview of different fusion concepts 259
10.5 Overview of current fusion research and R&D activities and breakthroughs 260
10.5.1 Public programs 260
10.5.2 Private initiatives 262
10.6 Key challenges (approach-independent) 266
10.7 Key challenges (approach-dependent) 267
10.8 Superconducting magnet technology and prospects of HTS 268
10.8.1 LTS magnets 269
10.8.2 HTS magnets 269
10.8.3 Conclusion of this section 271
10.9 Fusion Science and Technology in Switzerland and upcoming opportunities 271
10.9.1 Upcoming opportunities for public and private Swiss institutions in the next 20 years 273
10.10 Conclusions and timeline towards a power plant 275
10.11 References 276

List of abbreviations

BDBA	Beyond Design Basis Accident
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
COLA	Combined License Application
CES	Containment Enclosure Structure
CS	Containment Structure
DBA	Design Basis Accident
DEC	Design Extension Condition
DOE	Department of Energy
EPZ	Emergency Planning Zone
EUR	European Utility Requirements
FIA	Fusion Industry Association
FOAK	First-Of-A-Kind
FRC	Field-Reversed Configuration
FU	Functional Unit
GCR	Gas-Cooled Reactor
GFR	Gas Fast Reactor
GHG	Greenhouse Gas
HALEU	High-Assay Low-Enriched Uranium
HTGR	High Temperature Gas Reactor
HTS	High-Temperature Superconductor
IAEA	International Atomic Energy Agency
IEA	International Energy Agency
IFE	Inertial Fusion Energy
iPWR	Integral Pressurized Water Reactor
LCA	Life Cycle Assessment
LCOE	Levelized Cost Of Energy
LERF	Large Early Release Frequency
LEU	Low-Enriched Uranium
LFR	Lead Fast Reactor
LFSCOE	Levelized Full System Costs of Electricity
LPP	Licensing Project Plan
LTS	Low Temperature Superconductor
LWR	Light Water Reactor

MA	Minor Actinide
MCF	Magnetic-Confinement Fusion
MIF	Magneto-Inertial Fusion
MMR	Micro Modular Reactor
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
NEA	Nuclear Energy Agency
NIF	National Ignition Facility
NSHI	Nuclear Harmonization and Standardization Initiative
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
RepU	Reprocessed Uranium
SCWR	Supercritical Water Reactor
SFOE	Swiss Federal Office of Energy
SFR	Sodium Fast Reactors
SG	Steam Generator
SMR	Small Modular Reactor
VHTR	Very High Temperature Reactor
VRE	Variable Renewable Energy
WCR	Water-Cooled Reactor
WNA	World Nuclear Association

Zusammenfassung

Annalisa Manera und Andreas Pautz (Paul Scherrer Institut)

Kernenergie in der Schweiz

Im Jahr 2023 erzeugte die Schweiz Kernenergie aus vier in Betrieb befindlichen Reaktoreinheiten (Beznau 1 und 2, Gösgen und Leibstadt) mit einer Gesamtkapazität von rund 3 Gigawatt elektrisch (GWe). Die Kernenergieerzeugung spielt im Schweizer Strommix nach wie vor eine wichtige Rolle: Im Jahr 2022 produzierten die vier Kernkraftwerke 23,1 TWh, was rund 36 % der gesamten Stromproduktion entspricht und den zweitgrößten Beitrag zur inländischen Stromerzeugung darstellt. Wasserkraft bleibt die dominierende Quelle und trägt fast 53 % zur Schweizer Stromerzeugung bei. Kernenergie ist besonders in den Wintermonaten wichtig (im Jahr 2022 trug sie über einen Zeitraum von fünf Monaten zu mehr als 40 % des inländischen Produktionsmix bei). Aufgrund der Dominanz von Wasserkraft und Kernenergie (89,2 % des Strommix im Jahr 2022) weist die Schweiz bei ihrer Stromproduktion derzeit nahezu Netto-Null-CO2-Emissionen auf (Tabelle 1.3 im Hauptbericht).

Kernenergie weltweit

Auch in den OECD-Ländern ist Kernenergie nach wie vor die größte einzelne Quelle für kohlenstoffarmen Strom (Stromanteil im Jahr 2022): 15,8 % Kernenergie, 12,6 % Wasserkraft, 9,9 % Wind, 5,9 % Solarenergie). Der größte Anteil der Stromerzeugung sowohl in der OECD als auch weltweit stammt jedoch aus der Verbrennung fossiler Brennstoffe (fast 50 % in den OECD-Ländern und mehr als 60 % weltweit). Das Bild wird noch düsterer, wenn man den weltweiten Primärenergieverbrauch betrachtet, wo mehr als 80 % der Energie immer noch aus fossilen Energieträgern erzeugt werden, während Wasserkraft nur 7 %, Kernenergie nur 4 % und Wind- und Solarenergie zusammen lediglich 5 % ausmachen.

Weltweit nutzen 32 Länder Kernenergie, 13 neue Länder befinden sich in einem fortgeschrittenen Planungs- oder Baustadium zur Aufnahme von Kernenergie in ihren Strommix (in drei dieser Länder befinden sich bereits Kernkraftwerke im Bau) und 17 weitere Länder befinden sich in der Entscheidungsphase. Vier Länder sehen einen Ausstieg aus der Kernenergie vor, aber nur Deutschland hat die Stromerzeugung aus Kernenergie im Jahr 2023 endgültig eingestellt. Spanien plant einen Ausstieg bis 2035, während Belgien trotz seiner Ausstiegsentscheidung aufgrund der jüngsten Energiekrise die Lebensdauer von zwei seiner sieben Kernreaktoren verlängert hat und die Schweiz einen Langzeitbetrieb ihrer bestehenden Kernkraftwerke von möglicherweise bis zu 80 Jahren plant, wobei mit der Umsetzung der Energiestrategie 2050 ein schrittweiser Ausstieg aus der Kernenergie erfolgen soll.

Insgesamt sind im März 2024 weltweit 415 Kernkraftwerke mit einer installierten Leistung von insgesamt 373.257 GWe in Betrieb. Darüber hinaus befinden sich 57 Kernkraftwerke im Bau, die eine zusätzliche Kapazität von 59,22 GWe bereitstellen. In Europa sind 167 Kernkraftwerke in Betrieb (148 GWe) und 9 im Bau (10,1 GWe).

Die Länder mit den meisten in Betrieb befindlichen Kernkraftwerken sind die USA, Frankreich, China und Russland. Mit Stand März 2024 ist China das Land mit dem höchsten Wachstum der Kernenergie, wo derzeit 27 Kernkraftwerke gebaut werden, gefolgt von Indien (7 im Bau befindliche Anlagen), der Türkei (4), Ägypten (4), Südkorea (2) und Russland (4). China hat im Jahr 2023 bereits 53,3 GWe installierte Kernenergiekapazität erreicht (mit fast 400 produzierten TWh im Jahr 2022), weitere 30,9 GWe sind im Bau und es gibt erhebliche Wachstumspläne (mit dem Ziel, bis 2030 eine installierte Kernenergiekapazität von bis zu 150 GWe zu erreichen). In Europa bauen oder planen derzeit folgende Länder neue Kraftwerke in naher Zukunft: Frankreich (1 EPR-Anlage im Bau, 6 EPR-2-Anlagen genehmigt, weitere geplant), Vereinigtes Königreich (2 EPR-Anlagen im Bau, 2 weitere EPR-Anlagen geplant), Slowakei (1 Anlage im Bau, weitere vorgeschlagen), Bulgarien (2 AP1000-Anlagen geplant), Tschechische Republik (4 Anlagen geplant, zusätzlich 3 Standorte für mehrere SMRs identifiziert), Niederlande (2 Anlagen geplant), Rumänien (2 CANDU-Anlagen geplant, 6 NuScale-SMR-Module vorgeschlagen), Ungarn (2 VVER-Anlagen genehmigt), Slowenien (1 Anlage vorgeschlagen), Schweden (2 Anlagen bis 2035 geplant, 10 weitere Anlagen nach 2035 geplant), Estland und Polen (3 AP1000 genehmigt, 2 APR1400-Anlagen geplant, 24 BWR-300-SMR-Anlagen geplant). Darüber hinaus wurden 2023 in Weißrussland und Finnland zwei neue Kernkraftwerke ans Netz angeschlossen. Zu den jüngsten Entwicklungen gehört der Bau des weltweit ersten geologischen Tiefenlagers für hochradioaktive Abfälle in Finnland, dessen Bau Mitte der 2020er Jahre abgeschlossen sein soll. In Schweden wurde ebenfalls eine Baugenehmigung für ein geologisches Tiefenlager erteilt, mit dem Bau soll in den nächsten Jahren begonnen werden, während in Frankreich derzeit ein Antrag auf den Bau eines geologischen Tiefenlagers von der Aufsichtsbehörde geprüft wird. Eine Entscheidung über den endgültigen Lagerort wird für 2025 erwartet, und der Betrieb soll etwa 2040 beginnen. In Kanada soll die Auswahl des Abfalllagerstandorts im Jahr 2024 bekannt gegeben werden. In der Schweiz wird mit der Genehmigung eines geologischen Tiefenlager um 2030 gerechnet (vorbehaltlich einer positiven Beurteilung durch die Aufsichtsbehörden und ggf. eines fakultativen Referendums), die Inbetriebnahme ist für das Jahr 2050 geplant.

In den letzten Jahren haben mehrere Länder, insbesondere im Hinblick auf die durch den Ukraine-Krieg verursachten Veränderungen der geopolitischen Landschaft, ihre Pläne zur Kernenergie überarbeitet. Dies gipfelte in:

● der Gründung der EU-Kernenergieallianz im Jahr 2023, bei der 16 Länder (Frankreich, Belgien, Bulgarien, Kroatien, Tschechische Republik, Finnland, Ungarn, Niederlande, Polen, Rumänien, Slowenien, Slowakei, Estland, Schweden, Italien, Vereinigtes Königreich) den Aufbau einer integrierten europäischen Kernenergieindustrie planen und sich verpflichten, bis 2050 einen Anteil von 150 GWe Kernenergie am EU-Strommix zu erreichen (eine Steigerung von 50 % im Vergleich zum heutigen Anteil der Kernenergie);

● der Gründung der Allianz für kleine modulare Reaktoren der Europäischen Kommission im Jahr 2024 mit dem Ziel, „die technologische und industrielle Führungsrolle Europas im Bereich der Kernenergie aufrechtzuerhalten“;

● die Erklärung zur Kernenergie auf der Klimakonferenz der Vereinten Nationen (COP28) im Dezember 2023, die von 22 Ländern abgegeben wurde, die sich zum Ziel gesetzt haben, die Kernenergie bis 2050 zu verdreifachen, um das neue Nullziel zu erreichen, „in Anerkennung der Schlüsselrolle der Kernenergie bei der Erreichung globaler Netto-Null-Treibhausgasemissionen/Kohlenstoffneutralität bis etwa Mitte des Jahrhunderts“. Zu diesen Ländern gehören die Vereinigten Staaten, Bulgarien, Kanada, die Tschechische Republik, Finnland, Frankreich, Ghana, Ungarn, Japan, Südkorea, Moldawien, die Mongolei, Marokko, die Niederlande, Polen, Rumänien, die Slowakei, Slowenien, Schweden, die Ukraine, die Vereinigten Arabischen Emirate und das Vereinigte Königreich;

● die Einführung eines Investitionsplans in den USA zur Förderung der Entwicklung von SMRs und Mikroreaktoren und deren Einsatz in den USA sowie im Ausland. Das 2022 unterzeichnete Gesetz zur Reduktion der Inflation zielt darauf ab, bestehende und neue Kernkraftwerke durch Investitionshilfen und Steueranreize sowohl für große bestehende Kernkraftwerke als auch für neuere fortschrittliche Reaktoren zur Uranbrennstoff- und Wasserstoffproduktion zu unterstützen. Die Laufzeit mehrerer Kernkraftwerke wurde verlängert (z. B. Diablo Canyon in Kalifornien, dessen Stilllegung 2022 geplant war. Bei sechs Reaktoren wurde die Laufzeit auf

80 Jahre verlängert. Mehrere andere warten auf eine Entscheidung der Aufsichtsbehörde). Bemerkenswert ist, dass der Bundesstaat Michigan das seit 2022 stillgelegte Kernkraftwerk Palisades wieder in Betrieb nimmt. In bestehenden Kernkraftwerken wurden einige Pilotprojekte erfolgreich gestartet, um Kernenergie für die Wasserstoffproduktion zu nutzen.

Aufgrund der zunehmenden Anerkennung der Bedeutung einer zuverlässigen Bandlastproduktion haben mehrere US-Unternehmen wie Amazon, Google, Microsoft und energieintensive Industrien wie Nucor (Stahlproduktion) und Dow Chemicals Vereinbarungen mit Kernenergieanbietern oder -versorgungsunternehmen für die zukünftige Versorgung mit Kernenergie unterzeichnet.

Status der Leichtwasserreaktoren (LWR) der Generation III/III+ und Bauzeit

Reaktoren der Generation III/III+ sind eine neue Generation von Kernkraftwerken, die auf der gleichen Leichtwasserreaktortechnologie (LWR) wie die derzeit in Betrieb befindlichen Anlagen basieren, sich jedoch durch deutlich verbesserte Sicherheitsmerkmale auszeichnen, die in ihren Konstruktionsmerkmalen die Lehren aus den drei größten Reaktorunfällen der Geschichte berücksichtigen. Stand Dezember 2023 sind 38 große LWR-Einheiten der Generation III/III+ in Betrieb, und von den 60 derzeit im Bau befindlichen Reaktoren sind 51 große LWR der Generation III/III+. Weitere Einheiten wurden bestellt oder Ausschreibungen sind im Gange (z. B. drei Einheiten in Polen, zwei Einheiten in Großbritannien, eine in der Tschechischen Republik usw.), und mehrere weitere sind geplant.

Die durchschnittliche Bauzeit der 38 in Betrieb befindlichen Reaktoren der Generation III/III+ beträgt 7,7 Jahre, der Median liegt bei 8 Jahren (siehe Abbildung 2.5 und Abbildung 2.6 im Hauptbericht). Im Vergleich dazu beträgt die durchschnittliche Bauzeit für die weltweit 413 Reaktoren der Generation II und III insgesamt 7,5 Jahre, der Medianwert 6,3 Jahre. Diese Zahlen bestätigen nicht die allgemeine Meinung, dass die Bauzeiten für neuere Kernkraftwerke drastisch zugenommen haben. Sie stützen jedoch eher eine moderate Erhöhung der Bauzeiten mit einigen bemerkenswerten Ausnahmeprojekten, vor allem bei den ersten Kernkraftwerken in Europa und den USA, deren Bauzeiten überproportional angestiegen sind. Andererseits wurde wiederholt nachgewiesen, dass es technisch machbar ist, ein schlüsselfertiges System in weniger als sechs Jahren Bauzeit bereitzustellen, vorausgesetzt, es wird eine funktionierende Lieferkette für die Schlüsselkomponenten aufgebaut.

Insbesondere die ABWR-Anlagen (GE Hitachi/Toshiba) in Japan zeichnen sich durch ihre kurze Bauzeit aus, da sie alle in weniger als vier Jahren fertiggestellt wurden. Der Westinghouse AP-1000 in den USA (Vogtle-3) und die beiden EPR-Anlagen in Olkiluoto und Flamanville (Frankreich) liegen mit Bauzeiten von 10 bzw. 16,5 Jahren am entgegengesetzten Ende des Spektrums.

Diese Projekte waren von Anfang an besonderen Herausforderungen ausgesetzt, da sie Pionierarbeit beim Bau des ersten Großkraftwerks seiner Art in Europa und den USA nach einer jahrzehntelangen Pause von Neubauprojekten leisteten und die Fertigungskapazitäten und Lieferketten wiederaufgebaut werden mussten. Darüber hinaus verlangten die Aufsichtsbehörden sowohl in Finnland als auch in den USA noch bis weit in die Bauphase der Kraftwerke hinein erhebliche Designänderungen. Bei den beiden im Bau befindlichen EPR-Anlagen in Hinkley Point im Vereinigten Königreich kam es zwar nicht in demselben Ausmaß wie in Finnland und Flamanville, aber dennoch zu erheblichen Verzögerungen. Die Verzögerungen waren teilweise auf fehlende Elemente in der britischen Lieferkette, die Notwendigkeit der Schulung britischer Arbeitskräfte (wobei es vor allem beim Hochbau zu

Verzögerungen kam) und eine große Zahl von Designänderungen (über 7.000) zurückzuführen, die von der britischen Aufsichtsbehörde verlangt wurden. Trotz dieser Rückschläge hat die britische Regierung den Bau von zwei weiteren EPR-Anlagen am Standort Sizewell bestätigt.

Der Grad der Vollständigkeit des Detaildesigns bei Baubeginn und der Aufbau einer funktionierenden Lieferkette und Fertigungskapazitäten sind daher wichtige Faktoren bei der Bestimmung der Bauzeit; die Erfahrung mit mehreren aufeinanderfolgenden Anlagen und die Verlässlichkeit des finanziellen und rechtlichen Rahmens sind ebenfalls wichtige Faktoren. China konnte die Bauzeiten für Anlagen kontinuierlich verkürzen; die letzten neun Anlagen (mit standardisiertem HPR1000- und ACPR-1000-Design) wurden alle in 5 bis 7 Jahren gebaut. Bemerkenswert ist auch das jüngste Beispiel der Vereinigten Arabischen Emirate, wo das südkoreanische Unternehmen KHNP innerhalb von neun Jahren und zu Gesamtkosten von nur 24 Milliarden US-Dollar 5,2 GWe Kernenergiekapazität (4 APR1400-Einheiten) errichtet hat.

Wirtschaftlichkeit von LWRs der Generation III/III+

Schätzungen auf der Grundlage seriöser wissenschaftlicher Quellen (PSI 2019) beziffern die Stromgestehungskosten (levelized cost of electricity, LCOE) neuer Kernkraftwerke auf 7 bis 12 Rp/kWh. Solange die Bauzeit unter 8 Jahren bleibt (der Median der 38 Gen-III/III+-Bauten beträgt 7,7 Jahre), sind LCOE von 7 Rappen erreichbar, was mit früheren PSI-Studien aus dem Jahr 2019 übereinstimmt. Dies liegt gut im Bereich der aktuellen und zukünftigen LCOE für erneuerbare Energiequellen in der Schweiz und bestehende Wasserkraftwerke und würde Bandlaststrom liefern. Die PSI-Ergebnisse von 2019 stimmen mit anderen von Experten überprüften Studien überein, die in der frei zugänglichen Literatur veröffentlicht wurden. Die aktuellen LCOE für den Betrieb bestehender Schweizer Kernkraftwerke liegen bei 4,0 – 5,5 Rp/kWh (darin sind bereits die vollen Kosten für die Abfallentsorgung enthalten). Ein Langzeitbetrieb dieser Anlagen von bis zu 60 Jahren würde die Stromgestehungskosten um 1–2 Rappen erhöhen. Dabei ist allerdings zu beachten, dass das Konzept der Stromgestehungskosten LCOE ursprünglich zum Vergleich regelbarer Energiequellen eingeführt wurde, in einem zunehmend komplexen Energiesystem mit einem immer stärkeren Anteil fluktuierender erneuerbarer Energien jedoch nur von begrenztem Wert ist. In zunehmendem Maße wird anerkannt, dass in solchen Fällen nicht nur die Stromgestehungskosten, sondern die gesamten Systemkosten (Ausgleichskosten, Kosten für Netzausbau, Backup-Kosten usw.) berücksichtigt werden müssen. Die OECD hat kürzlich den Versuch einer solchen Studie für das Schweizer Energiesystem veröffentlicht, doch ein grosses, umfassendes Modell des Schweizer Energiesystems, das auch verschiedene Szenarien für den Einsatz von Kernenergie einbezieht, wurde nie erstellt.

Die ersten 1600-MWe-EPR-Anlagen in Olkiluoto und Flamanville waren erheblich teurer als die südkoreanischen APR1400-Anlagen mit 1400 MWe, die in den VAE gebaut wurden. Während der APR1400 6 Milliarden USD pro Einheit kostet, kosten die beiden EPR in Olkiluoto und Flamanville jeweils rund 11 bzw. 13,2 Milliarden Euro. Diese hohen Kapitalkosten müssen jedoch im Verhältnis zur produzierten Energie gesehen werden. Eine einzelne EPR-Einheit würde mehr als 12 TWh/Jahr produzieren. Zum Vergleich: Um die gleiche Jahresleistung eines EPR mit Alpine-Solaranlagen zu produzieren, bräuchte man das Äquivalent von mehr als 3800 „Alpin Solar“ (Anlage Muttsee-Staumauer) zu Kosten von mehr als 30 Milliarden CHF (ohne Berücksichtigung der zusätzlichen Kosten für Backup, Speicherung und Netzerweiterung). Nimmt man stattdessen die Gondosolar-Anlage als Referenz, bräuchte man mehr als 780 solcher Anlagen zu Kosten von rund 29 Milliarden CHF.

Die hohen Kapitalkosten großer Kernkraftwerke sind eine der größten wirtschaftlichen Herausforderungen der Kernenergie, da sie die Zahl potenzieller privater Investoren verringern. Diese Herausforderung wird durch SMRs etwas gemildert und könnte durch Mikroreaktoren weitgehend eliminiert werden, deren Gesamtkapitalkosten mit denen alpiner Solarkraftwerke vergleichbar sind, die aber eine viel höhere und gleichmäßigere Energieabgabe bieten.

Um die hohen Kapitalkosten für große Kernkraftwerke zu bewältigen, wurden in der Vergangenheit verschiedene Modelle umgesetzt, an denen sich Regierungen als Eigenkapitalgeber, als Kreditgeber oder durch politische Maßnahmen wie Kreditbürgschaften oder Contracts for Differences (CfD, Mindestvergütungen) beteiligten. Entgegen der allgemeinen öffentlichen Wahrnehmung wird die Kernenergie von allen Energiequellen am wenigsten subventioniert, wie in Abschnitt 1.4.2 dieses Berichts ausführlich erörtert wird. In der EU erreichten die Subventionen für Kernenergie im Zeitraum 2015–2022 im Jahr 2021 einen Höchstwert von 7,6 Milliarden EUR, verglichen mit 88 Milliarden EUR für erneuerbare Energien und 123 Milliarden EUR für fossile Brennstoffe. In den USA betrugen die maximalen Subventionen für Kernenergie zwischen 2016 und 2022 weniger als 600 Millionen USD im Vergleich zu mehr als 17 Milliarden USD für erneuerbare Energien, mehr als 2,5 Milliarden USD für Kohle und etwa 3 Milliarden USD für Gas.

Die Zinssätze für das Kapital werden maßgeblich durch das Finanzierungssystem (z. B. staatliche Kreditgarantie) und den regulatorischen Rahmen beeinflusst. Eine erfolgreiche Möglichkeit zur Kostensenkung besteht darin, mehrere Einheiten am selben Standort zu bauen. Beim Barakah-Projekt in den VAE beispielsweise konnten die Kosten zwischen dem Bau der Einheiten 1 und 4 um 40 % gesenkt werden. Weitere Faktoren, die sich positiv auf den Erfolg eines Neubaus auswirken, sind die Fertigstellung relevanter Teile des Entwurfs vor Baubeginn, das Vorhandensein einer gut etablierten Lieferkette, der Zugang zu qualifizierten Arbeitskräften und ein stabiler regulatorischer Rahmen.

Die Stromgestehungskosten für SMRs werden voraussichtlich ein Niveau erreichen, das dem großer Kernkraftwerke ähnelt, indem die Skaleneffekte durch die Fertigungseffekte (Massenproduktion) kompensiert werden, obwohl für die ersten Einheiten höhere Kosten zu erwarten sind. Für das erste NuScale-SMR-Projekt, das in den kommenden Jahren in Utah gebaut werden sollte, wurden Kosten von etwa 5,8 ct/kWh prognostiziert (Schätzung 2020). Aufgrund eines Anstiegs der Zinssätze um 150 % und eines erheblichen Anstiegs der Materialkosten (z. B. 40 % Kostensteigerung für Stahl) in den letzten 1,5 Jahren wurden die Kosten voraussichtlich bis 2023 auf 8,9 ct/kWh steigen. Dies war nicht wettbewerbsfähig mit der Stromproduktion aus Gas und der Kohle, die den lokalen Versorgungsunternehmen in Utah zur Verfügung standen, und aus diesem Grund wurde das NuScale-Projekt zu Gunsten eines Gaskraftwerks sistiert.

Sicherheit bei Gen-III/III+

Bei den Kraftwerken der Gen-III/III+ wurden wesentliche Änderungen an den Sicherheitsanforderungen vorgenommen. Systeme zum Management schwerer Unfälle sind nun integraler Bestandteil des Designs und die Unabhängigkeit der verschiedenen sich überlagernden Sicherheitsebenen (defense in depth) wurde gestärkt. Die Umsetzung der neuen Sicherheitsphilosophie hat zu einer ganzen Reihe neuer passiver Sicherheitssysteme geführt (die für ihre Funktion nicht auf externe Energie wie Dieselgeneratoren oder Eingriffe durch den Betreiber angewiesen sind) und verlängerten Karenzzeiten (grace period), mit dem Ziel, das Auftreten schwerer Unfallfolgen mit Kernschmelze und anschließendem Versagen des Sicherheitsbehälters (Containment), die zu frühzeitigen oder großen Freisetzungen radioaktiver Substanzen führen könnten, praktisch auszuschließen. Insbesondere haben diese neuen Sicherheitsansätze zu Folgendem geführt:

● einer Verlängerung der Karenzzeit (in der selbst unter den widrigsten Unfallumständen kein menschliches Eingreifen erforderlich ist) von 30 Minuten, typisch für Gen-II-Designs, auf mindestens 3 Tage, häufiger über eine Woche;

● Kernschadenshäufigkeiten unter 10^{-6}/Jahr (d. h. probabilistisch gesehen weniger als einmal in einer Million Jahren);

● eine Wahrscheinlichkeit eines Versagens des Containments nach dem Kernschaden mit Freisetzungen in die Umwelt von unter 10^{-7}/Jahr (d. h. weniger als einmal alle zehn Millionen Jahre).

Die Wahrscheinlichkeit für einen Kernschaden und die anschließende Freisetzung erheblicher Mengen Radioaktivität ist daher um ein bis zwei Größenordnungen geringer als bei aktuellen, ordnungsgemäß nachgerüsteten Kraftwerken der Generation II, die aufgrund von Nachrüstmaßnahmen und den Post-Fukushima-Stresstests bereits ein ausgezeichnetes Sicherheitsniveau erreicht haben.

Status kleiner modularer Reaktoren (SMRs)

SMRs sind moderne Reaktoren mit einer Nennleistung von bis zu 300 MW(e) pro Einheit. Sie sind für den Bau in Fabriken und den Transport zum Einsatzort konzipiert. Normalerweise werden sie unterirdisch installiert. Die Kernenergieagentur der OECD (NEA) geht davon aus, dass SMRs bis 2035 bis zu 9 % der gesamten neuen Kernkraftwerkskapazität ausmachen werden. Derzeit sind 10 SMRs in Russland und China in Betrieb, und mehrere befinden sich derzeit im Bau oder warten auf die Genehmigung (USA, Kanada, Frankreich), siehe Tabellen 1 und 2.

Unter den wassergekühlten SMRs sind NUWARD (EDF), Roll-Royce (Großbritannien), BWXR-300 (USA), Holtec-180 (USA), AP300 (USA) und VOYGR (NuScale) die für den Einsatz in Europa bis 2030 am weitesten entwickelten Bautypen. Letzterer ist in den USA bereits zugelassen, während sich die anderen Designs in unterschiedlichen Stadien der Vorzertifizierung in den USA, Kanada und einigen europäischen Ländern befinden. Einige SMR-Anbieter haben Bestellungen erhalten (z. B. BWRX-300 in Kanada). NUWARD erhielt 2022 500 Millionen Euro von der französischen Regierung und der Baubeginn eines ersten Reaktors ist für 2030 geplant.

Das allgemeine Interesse an SMRs entstand ursprünglich aus der Notwendigkeit, abgelegene Regionen oder netzunabhängige Gebiete mit Strom zu versorgen, die derzeit auf Gas, Öl oder Diesel angewiesen sind, und veraltete fossilbefeuerte Kraftwerke im Bereich von 300-400 MWe zu ersetzen. Für Länder mit kleinen Stromnetzen, in denen der Einsatz großer Kernkraftwerke nicht möglich wäre oder Investoren und Betreiber nicht in der Lage oder willens wären, viel Kapital zu investieren, werden SMR von manchen als wirtschaftliche Option angesehen. Darüber hinaus bieten SMR möglicherweise Möglichkeiten, energieintensive Industriestandorte (z. B. Beton- oder Stahlindustrie) zu versorgen oder nicht-elektrische Anwendungen der Kernenergie bereitzustellen, z. B. Fernwärme, Meerwasserentsalzung oder sogar Wasserstoffproduktion.

Aufgrund ihrer geringeren Größe verfügen die meisten SMRs über verbesserte Sicherheitsmerkmale, die vollständig auf passiver Sicherheit basieren. Aus diesem Grund hat die Nuclear Regulatory Commission in den USA eine neue Regel zur Dimensionierung von Notfallplanungszonen (emergency planning zone, EPZ) genehmigt, die sich an den möglichen Störfallszenarien und deren Auswirkungen orientiert. Infolgedessen wurde der NuScale SMR mit einer EPZ lizenziert, die auf den Umkreis des Anlagengeländes beschränkt ist (d. h. es ist keine Evakuierungszone erforderlich). Es wird erwartet, dass andere SMRs in den USA eine ähnliche Regelung erhalten werden.

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Tabelle 1 LWR SMRs in fortgeschrittenem Entwicklungsstadium

Name	Thermal power [MWth (MWe)]	Type	Design organisation	Country	Status
CAREM	100 (30)	Integral PWR	CNEA	Argentina	Under construction
ACPR50S	200 (60)	Floating PWR	CGNCP	China	Under construction
ACP100	385 (125)	Integral PWR	CNNC and NPIC	China	Construction started in 2021
KLT-40S	150 (35)	Floating PWR	OKBM	Russia	2 units in operation
VOYGR	250 (77)	Integral PWR	NuScale Power	USA	Shortlisted in USA and Europe
AP300	900 (300)	One-loop PWR	Westinghouse	USA	Shortlisted in UK
UK SMR	1,358 (470)	Integral PWR	Rolls-Royce	UK	Short-listed in Estonia and UK
NUWARD	540 (170)	Integral PWR	EDF	France	FOAK in France by 2030.
BWRX-300	870 (290)	Integral BWR	GE-Hitachi	USA	Several units to be built in Canada and USA. Shortlisted in Europe.
SMR-160	525 (160)	PWR	Holtec	USA	Shortlisted in various countries
SMART	365 (107)	PWR	KAERI	Korea	Licensed in Korea
RITM-200	175 (55)	Floating PWR	OKBM	Russia	Six units in operation. More under construction.
RITM-200N	190 (55)	On-shore PWR	OKBM	Russia	First concrete planned for 2024.
RITM-200S	198	Floating PWR	OKBM	Russia	To be built at Baimskaya copper mine site, deployment by 2027.
RITM-200M	175 (50)	Floating PWR	OKBM	Russia	MOU signed for deployment in Philippines and Myanmar.

Die Hauptvorteile von SMRs sind die deutlich niedrigeren anfänglichen Kapitalkosten aufgrund der geringeren Größe der Anlage, kürzere Bauzeiten aufgrund der Umstellung auf Fabrikproduktion, die erhöhte Flexibilität bezüglich Lastregelung, wodurch sich SMRs leichter mit intermittierenden erneuerbaren Energiequellen integrieren lassen, und die verbesserten Sicherheitskonzepte. Die Skaleneffekte großer Kernkraftwerke werden vermutlich durch Produktionseffekte (fabrikgefertigte Module) und einfachere Baustellen ersetzt, wobei die Kosten pro kWh im gleichen Bereich liegen wie bei großen Kernkraftwerken. Die IAEA und die US-amerikanischen/europäischen Kernenergieaufsichtsbehörden harmonisieren derzeit die Lizenzierung von SMRs, um ein stabiles und transparentes Lizenzierungsumfeld zu schaffen, das unvorhersehbare Änderungen der nationalen Lizenzierungsregelungen vermeidet.

Die meisten kurzfristigen SMRs gehören wie die Großreaktoren zur Kategorie der Leichtwasserreaktoren der Generation III/III+, mit einer plausiblen Perspektive für den kommerziellen Betrieb der ersten Demonstrationsanlagen in westlichen Ländern bis 2030 oder sogar früher (siehe Tabelle 1 oben). Fortschrittliche SMRs mit anderen Kühlmitteln als Wasser (z. B. flüssiges Metall, Helium, geschmolzenes Salz) gehören zu Kernkraftwerken der Generation IV und deren Entwicklung wird von einer Vielzahl von Start-up-Unternehmen verfolgt. Der Zeithorizont für die kommerzielle Nutzung einiger dieser Designs (z. B. geschmolzenes Salz) hinkt jedoch der Entwicklung von LWRs um mehrere Jahre hinterher. Während nicht wassergekühlte SMRs in China und Russland bereits in Betrieb sind, wird voraussichtlich der erste in den westlichen Ländern der natriumgekühlte SMR von Terrapower sein, der in Wyoming (USA) gebaut werden soll. Der Bauantrag für den Terrapower SMR wurde im März 2024 eingereicht und im Mai 2024 von der US-amerikanischen Aufsichtsbehörde zur Prüfung angenommen.

Tabelle 2 Nicht-LWR SMRs in fortgeschrittenem Entwicklungsstadium

Name	Thermal power (MWth)	Type	Design organisation	Country	Status
Thermal spectrum					
HTR-PM	500	HTGR	INET	China	2 units in operation in China since Dec 2021 Additional 18 HTR-PM units proposed.
KP-FHR	311	MSR / solid fuel	Kairos Power	USA	Construction permit for demo unit received in Dec 2023.
XE-100	200	HTGR	X-energy	USA	Completed pre-certification in Canada. Pre-licensing in US. Selected by Dow Chemical (USA)
IMSR	884	Integral MSR	Terrestrial Energy	Canada	Pre-licensing in USA, and Canada.
Fast spectrum					
ARC-100	286	SFR	ARC Clean Tech.	Canada	Pre-licensing in Canada.
Wasteburner	750	MSR	Moltex Energy	Canada	Pre-licensing in Canada
Natrium	840	SFR	TerraPower	USA	Pre-licensing in USA. To be built in Wyoming (USA)
BREST-OD-300	700	LFR	NIKIET	Russia	Under construction in Russia. Completion is planned for 2026.
CFR-600	1500	SFR	CNNC	China	2 units under construction in China Connection to the grid in 2024 - 2025.

Stand der Mikroreaktortechnologie

In den letzten sieben Jahren hat sich ein interessanter Trend zu sogenannten Mikroreaktoren herausgebildet, die elektrische Leistungen im Bereich von bis zu etwa 10 MWe erzeugen sollen (mehrere davon werden in den USA entwickelt, siehe Tabelle 4.1 im Hauptbericht). Dabei handelt es sich um Reaktoren, die vollständig fabrikgefertigt werden, in einen ISO-Container passen, um problemlos (auf einem Schiff, per LKW oder mit der Eisenbahn) von der Fabrik zum Einsatzort transportiert werden zu können (keine Baustelle erforderlich) und 5-10 Jahre oder länger ohne Erneuerung des Brennstoffs funktionieren. Sie können unabhängig, als Teil des Stromnetzes oder innerhalb eines Mikronetzes betrieben werden. Sie sollen in abgelegenen Gebieten (z. B. Bergbaustandorten) eingesetzt werden oder energieintensive Industrien (z. B. Wasserentsalzung, Wasserstoffproduktion usw.) mit Strom und Wärme versorgen. Sie sind aber auch für Industrien interessant, die ein gewisses Maß an Unabhängigkeit vom Stromnetz im Sinne der Versorgungssicherheit benötigen. Die Kühlung erfolgt über Gas (Helium), flüssiges Metall, geschmolzenes Salz oder (Natrium-) Wärmerohre (heat pipes).

Aufgrund der sehr geringen Größe und der Einfachheit des Designs schreitet ihre Entwicklung extrem schnell voran. Die erste Demonstrationseinheit (Heatpipe-Design) wurde innerhalb von 3 Jahren von der NASA und dem Los Alamos National Laboratory zu Kosten von weniger als 20 Millionen Dollar entworfen, gebaut und getestet. Eine zweite Einheit (flüssigmetallgekühlt) wird derzeit in Idaho (USA) gebaut und soll Anfang 2025 in Betrieb gehen. Eine fluoridgekühlte Einheit erhielt im Dezember 2023 die Baugenehmigung, die Inbetriebnahme ist für 2026 geplant. Drei weitere Designs befinden sich in verschiedenen Phasen der Lizenzierung in den USA und Kanada.

Da Mikroreaktoren vollständig in Fabriken gebaut werden und daher voraussichtlich von Serienfertigungseffekten profitieren werden, wird wie in anderen Branchen eine positive Lernkurve erwartet. Weitere vermutete Vorteile sind die sehr geringen Kapitalkosten (in der

Größenordnung von etwa 100 Millionen US-Dollar oder weniger), die sie für einen größeren Kreis von Investoren erschwinglich machen könnten, und die niedrigen Stromgestehungskosten (LCOE) im Vergleich zu den Backup-Alternativen in abgelegenen Gebieten oder Großindustrien aufgrund der vollständigen Fabrikfertigung, der sehr geringen Grundfläche (etwa 15 m^2 für die Anlage und weniger als 2000 m^2 für das Anlagengelände), des vorhersehbaren Bauzeitplans und des geringeren Strahlenrisikos. Aufgrund der sehr geringen Brennstoffmenge und der einfachen Konstruktion ähneln sie eher Forschungsreaktoren, und daher wird mit einer viel schnelleren Genehmigung gerechnet als bei SMRs oder großen Kernkraftwerken. Die Verwendung von höher angereichertem TRISO-Brennstoff in Mikroreaktoren erfordert die Entwicklung von entsprechenden Brennstoffherstellungskapazitäten, was derzeit in den USA und Frankreich geschieht. Dies ist jedoch kein Showstopper, da die Technologie bekannt ist (derselbe Brennstoff wird in den in China betriebenen HTR-PM-SMRs verwendet).

Status von Gen-IV- und Nicht-Leichtwasserreaktoren

Nicht-wassergekühlte Reaktoren (z. B. gekühlt mit Gas, Blei, Natrium, geschmolzenem Salz) werden mit dem Ziel entwickelt, die Effizienz entweder durch eine Erhöhung der thermodynamischen Effizienz und/oder durch eine verbesserte Brennstoffnutzung und eine weitere Reduzierung der Menge an hochradioaktivem Abfall (bei Fast Spectrum Reactors) zu steigern und so den Kreislauf für Kernbrennstoff zu schliessen[2]. Es gibt mehrere Designs, von denen die vielversprechendsten hier erwähnt werden sollen:

• Gasgekühlte thermische Reaktoren, die Helium als Kühlmittel verwenden (HTGR). Sie haben eine höhere thermodynamische Effizienz bei der Umwandlung der im Reaktor erzeugten thermischen Leistung in Elektrizität und da sie bei viel höheren Temperaturen als LWRs arbeiten, eignen sie sich auch zur Wärmeversorgung für energieintensive Industrieprozesse mit hohen Temperaturen (siehe Abbildung 1 für eine Darstellung der für verschiedene Industrieprozesse erforderlichen Temperaturen und der entsprechenden Reaktordesigns, die solche Temperaturen bereitstellen können). Zwei gasgekühlte thermische Reaktoren (HTR-PM-Design) sind in China bereits seit 2021 in Betrieb. Der Xe-100 (X-energy, USA) hat gerade die Vorzertifizierung in Kanada erfolgreich abgeschlossen (d. h. die Aufsichtsbehörde hat keine Probleme festgestellt, die einer Lizenzierung entgegenstehen würden).

• Mit flüssigen Metallen (Natrium oder Blei/Blei-Wismut) gekühlte schnelle Reaktoren (SFR/LFR). Sie arbeiten bei hohen Temperaturen zwischen jenen von LWRs und

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[2] Mehr als 90 % des verbrauchten Kernbrennstoffs sind wiederverwendbar. In einem geschlossenen Brennstoffkreislauf wird der verbrauchte Kernbrennstoff wiederaufbereitet, um das wiederverwendbare Material (meistens Uran) zu extrahieren, das dann zur Herstellung neuer Brennelemente verwendet wird. Ein geschlossener Brennstoffkreislauf kann beispielsweise durch eine Kombination aus LWRs und Gen-IV-Schnellreaktoren erreicht werden. In Schnellreaktoren werden hochenergetische („schnelle") Neutronen zur Spaltung des Kernbrennstoffs verwendet, während in LWRs hauptsächlich thermische (niedrigenergetische) Neutronen zur Spaltung verwendet werden. Ein geschlossener Brennstoffkreislauf ermöglicht eine Verbesserung der Nachhaltigkeit durch Erhöhung der Energieabgabe pro Brennstoffmasseneinheit und durch Verringerung der Menge an hochradioaktivem Abfall pro Energieeinheit.

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gasgekühlten Reaktoren und bei nahezu Umgebungsdruck (Atmosphärendruck). Es liegen erhebliche Betriebserfahrungen vor (Frankreich, Japan, Russland usw.), und mehrere Anlagen sind heute in Betrieb (siehe Tabelle 3). Terrapower (US-Unternehmen) wird einen natriumgekühlten schnellen SMR auf dem Markt anbieten, wobei die erste Anlage vor 2030 in Wyoming (USA) gebaut werden soll.

• Flüssigsalzreaktoren (MSRs), in denen geschmolzenes Salz als Kühlmittel, Brennstoff und/oder Moderator verwendet wird. Diese Reaktoren arbeiten bei hohen Temperaturen, und es gibt sowohl thermische als auch schnelle Designs. Eine erhebliche verbleibende betriebliche Herausforderung dieser Reaktoren ist die stark korrosive Natur der Salze. Im Dezember 2023 erhielt ein thermischer Reaktorentwurf von KAIROS (USA), bei dem geschmolzenes Salz nur als Kühlmittel verwendet wird (mit HALEU TRISO-Brennstoff und Graphit als Moderator), eine Baugenehmigung für eine erste Demonstrationsanlage in Tennessee. Ein integraler MSR (Terrestrial Energy) befindet sich derzeit in den USA und Kanada in der Vorlizenzierungsphase. Ein experimenteller MSR, der ein Thorium-basiertes geschmolzenes Salz als Brennstoff verwendet, erhielt im Juni 2023 in China die Betriebsgenehmigung. Der Bau dieses TMSR-LF1-Reaktors begann im September 2018 und sollte 2024 abgeschlossen sein. Berichten zufolge wurde er jedoch im August 2021 fertiggestellt, nachdem die Arbeiten beschleunigt wurden.

Relevante Gen IV-Reaktoren für den westlichen Markt sind:

• KAIROS, Terrestrial, X-energy (alles thermische Reaktoren, siehe Tabelle 2)

• Terrapower, IMSR (Moltex), ARC-100 (alles schnellen Reaktoren, siehe Tabelle 2)

• die Mikroreaktordesigns, die derzeit in den USA und Kanada lizenziert werden (Tabelle 4.1 im Hauptbericht).

Tabelle 3 Schnelle Reaktoren in Betrieb (alle vom Typ Sodium Fast Reactor SFR)

Country	Reactor name	Operation years	Current status
China	CEFR	2010-present	Active
India	FBTR	1985-present	Active
Russia	BOR-60	1969-present	Active
India	PFBR	Scheduled for 2024	Under construction
Russia	BN-600	1980-present	Active
Russia	BN-800	2014-present	Active

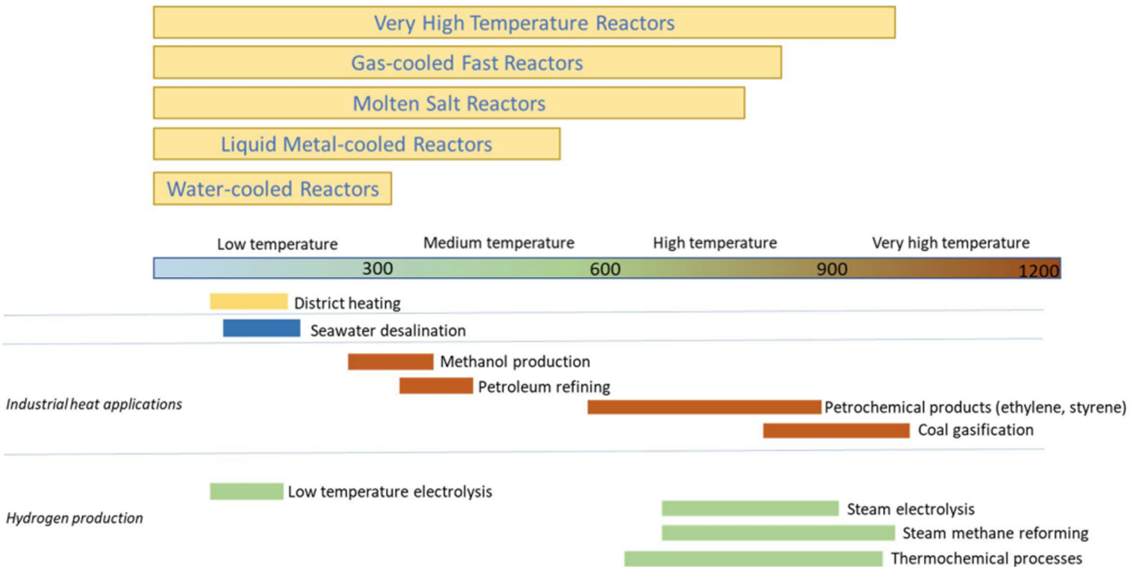

Abbildung 1: Ausgangstemperatur von KKW-Technologien und entsprechenden nichtelektrischen Anwendungen

Verfügbarkeit von Uranbrennstoff und alternative Brennstoffkreisläufe

Natürliche Uranreserven sind eine weit verbreitete Ressource (siehe Abbildung 2) und reichen für die nächsten Jahrhunderte. Wie bei anderen Ressourcen hängen sie auch hier vom Marktpreis ab.

Brennstoff mit schwacher Anreicherung (low enriched Uranium LEU), der in Leichtwasserreaktoren (z. B. in Schweizer Anlagen) verwendet wird, wird in mehreren Anreicherungsanlagen hergestellt und es gibt genügend Vielfalt und verschiedene Quellen, um seine Versorgung sicherzustellen. Es werden keine langfristigen Risiken für die Versorgungssicherheit der Schweiz mit Kernbrennstoff erwartet.

Mit Blick auf die zweite Hälfte dieses Jahrhunderts kann man davon ausgehen, dass ein erhöhter Bedarf an Kernenergie zu verstärkten Explorationsaktivitäten und damit zu erhöhten Uranreserven führen wird. Darüber hinaus wird sich in diesem Zeitraum die Reaktortechnologie mit einem geschlossenen Brennstoffkreislauf so weit entwickeln (z. B. schnelle Reaktoren), dass andere Brennstoffe mit einem viel größeren Energiepotenzial als U-235 verwendet werden können, wodurch die Verfügbarkeit von Kernbrennstoff von Hunderten auf viele Tausende von Jahren verlängert wird. Weltweit sind bereits einige schnelle Reaktoren in Betrieb oder im Bau (siehe Tabelle 2 und Tabelle 3) oder befinden sich in einem fortgeschrittenen Planungsstadium wie der Terrapower-Reaktor, wobei die erste Anlage in Wyoming in diesem Jahrzehnt gebaut werden soll. Ein wichtiger Brennstoff für nicht wassergekühlte moderne Reaktoren, einschließlich des Terrapower-Designs, ist hoch angereichertes Uran (HALEU), das durch eine Anreicherung zwischen 5 und 20 % gekennzeichnet ist. Folglich wird die Produktion von HALEU-Brennstoff erhöht. Ironischerweise werden die durch den Krieg in der Ukraine ausgelösten Unsicherheiten der nuklearen Brennstoffversorgungskette in den nächsten 5-10 Jahren dazu führen, dass die westlichen Brennstoffversorgungskapazitäten und -resilienz gestärkt und zu erweitert werden. In den USA und in Frankreich werden derzeit neue Lieferketten für HALEU-Brennstoff aufgebaut, während die Kapazität bestehender Anlagen für niedrig angereichertes Uran (LEU) sowohl in Europa als auch in den USA erhöht wurde. LEU wird typischerweise in konventionellen wassergekühlten Reaktoren verwendet. Bis Ende 2023 haben die USA außerdem drei neue Uranminen eröffnet, um die Unabhängigkeit bei der Uranversorgung zu erhöhen.

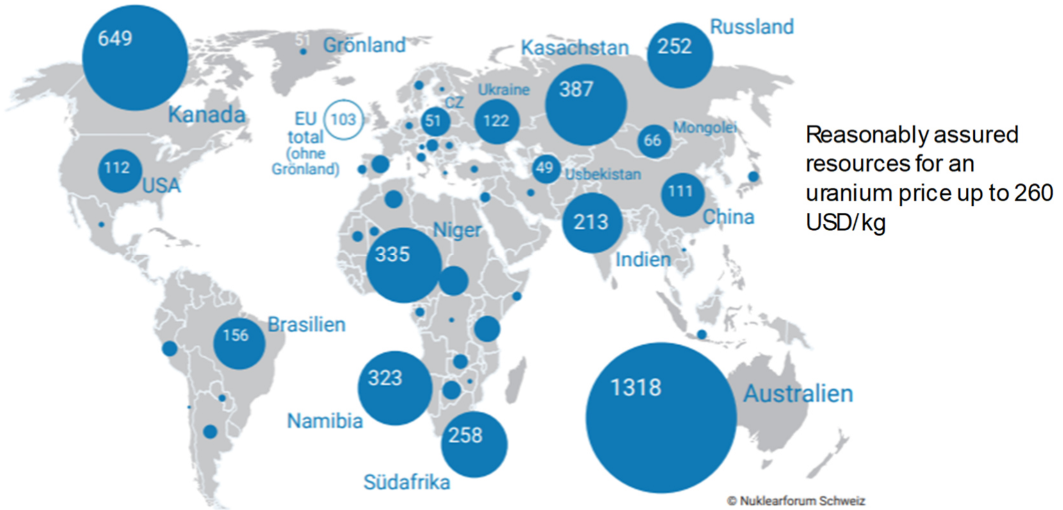

Abbildung 2 Weltweite Verteilung der Uranreserven zum 01.01.2021 (in 1000 Tonnen Uran) bei einem Preis von bis zu 260 USD/kg. Daten aus dem IAEA Redbook 2022.

Fortschrittlichere Brennstoffkreisläufe, die auf eine effizientere Brennstoffnutzung und die Reduzierung hochradioaktiver Abfälle abzielen, basieren auf der Wiederaufbereitung von Brennstoffen und umfassen:

● Herstellung neuer Brennelemente (sogenannte MOX) für LWR unter Verwendung des Pu und des wiederaufbereiteten U, das aus dem verbrauchten Brennstoff der LWR gewonnen wird. Normalerweise kann aus 4 verbrauchten Brennelementen (Fuel Assemblies FA) ein neues, frisches Brennelement hergestellt werden. Dies ist eine bewährte Technologie. MOX-Brennstoff wurde in Schweizer Anlagen erfolgreich eingesetzt und mehrere Jahre lang verwendet. Mit dem Kernenergiegesetz von 2003 wurde die Wiederaufarbeitung per Moratorium untersagt. In anderen Ländern (z. B. Frankreich) wird die Wiederaufbereitung noch immer routinemäßig durchgeführt;

● Schnelle Brüter-Reaktoren mit U-Pu (Breed-and-Burn-Brennstoffzyklus). In diesem Fall wird während des Anlagenbetriebs zusätzlicher Brennstoff im Reaktorkern erzeugt, sodass aus derselben Brennstoffmenge mehr Energie erzeugt werden kann. Es wird HALEU-Brennstoff benötigt;

● Verbrauch von minoren Aktiniden (MAs) in speziellen MA-„Brennern“ (z. B. MYRRHA- und Transmutex-Reaktordesigns), um die Radiotoxizität des vorhandenen abgebrannten Brennstoffs zu verringern. Dies erfordert die Wiederaufarbeitung der Minoren Aktinide, was zwar bereits im Labor erprobt ist, aber noch nicht im industriellen Maßstab existiert. Mit speziellen Brennern ist möglicherweise eine höhere Transmutationsrate pro Energieeinheit als in herkömmlichen schnellen Reaktoren erreichbar, der Betrieb spezieller Transmutationsreaktoren erhöht jedoch die technologische Komplexität und damit auch die Betriebsrisiken. Zudem wäre der überwiegende Teil der aufbereiteten Menge immer noch wiederaufbereitetes Uran (RepU), dessen weitere Handhabung durch den Transmuter nicht berücksichtigt wird. Wenn die Wiederaufbereitung von MAs dennoch umgesetzt würde, könnte ein spezieller Transmuter ein relevanter Bestandteil des Kernbrennstoffkreislaufs sein und zur Minimierung des Abfallstroms beitragen;

● Thorium (Th)-U-Zyklus: Th-232 ist ein fruchtbares Isotop analog zu U-238, das natürlich vorkommt und in der Erdkruste dreimal häufiger ist als U. Mit Thorium kann U-233 gebrütet werden, so wie U-238 zum Brüten von Pu-239 verwendet wird. Die Verwendung von Thorium erfordert eine Wiederaufbereitung und einige Anpassungen der heutigen Technologie. Würde dies jedoch umgesetzt, könnte es bei den aktuellen Anforderungen Hunderte bis Tausende von Jahren zusätzlichen Brennstoff liefern. Thorium-Testreaktoren wurden in der Vergangenheit erfolgreich betrieben. Ein Thorium-Demonstrationsreaktor mit geschmolzenem Salz hat 2023 in China den Betrieb aufgenommen und ein entsprechender 373-MWt-Reaktor soll bis 2030 folgen. Ein 40-MWth-Thorium-Schnellbrüter ist in Indien in Betrieb, da das Land über große Mengen an Thorium-Ressourcen verfügt.

Kommerzielle Wiederaufbereitungsanlagen für Brennelemente gibt es in Frankreich, Russland, Indien, Japan (letztere befindet sich im Bau und soll 2024 fertiggestellt werden) und China (zwei im Bau, die erste soll 2025 in Betrieb gehen). Eine Wiederaufbereitungsanlage in Großbritannien wurde 2022 nach 58 Betriebsjahren geschlossen. Auch die USA arbeiten daran, die Wiederaufbereitungskapazitäten im Land wiederherzustellen, nachdem sie die Wiederaufbereitung in den 1970er Jahren aus politischen Gründen eingestellt hatten. Die Vereinigten Staaten haben dem Antrag Südkoreas, eigene Wiederaufbereitungskapazitäten aufzubauen, bisher nicht zugestimmt.

Lizenzierung neuer Kernkraftwerke in der Schweiz
Das Schweizer Kernenergiegesetz verbietet ausdrücklich die Einreichung von Rahmenbewilligungsgesuchen für neue Kernkraftwerke. Von diesem Verbot nicht betroffen sind Lager-, Entsorgungs- und Forschungseinrichtungen sowie Kernanlagen mit geringem Risiko (ein Begriff, der in der Schweizer Kernenergieverordnung näher erläutert wird). Für letztere ist kein Rahmenbewilligungsgesuchen erforderlich. Mikroreaktoren und SMRs haben aufgrund ihrer hohen passiven Sicherheit und des geringen radioaktiven Inventars das Potenzial, gemäß Schweizer Kernenergiegesetz als Anlagen mit geringem Risiko eingestuft zu werden.

Ökobilanz (life cycle assessment LCA)
Die durch die Ökobilanz (LCA) quantifizierten Umweltbelastungen umfassen Auswirkungen auf den Klimawandel, Emissionen von Luftschadstoffen und giftigen Substanzen sowie Land-, Wasser- und sonstigen Ressourcenverbrauch. Die Ergebnisse der Ökobilanz können verwendet werden, um die Umweltbelastung verschiedener Stromerzeugungstechnologien zu vergleichen. Es gibt mehrere internationale Studien, darunter Analysen, die das Paul Scherrer Institut speziell für die Schweizer Kernkraftwerke durchgeführt hat. Die Umweltbelastung der Schweizer Kernkraftwerke wird zu einem großen Teil durch die Herkunft des Urans bestimmt, wobei die gesamten Treibhausgasemissionen bei etwa 6 g CO2eq/kWh für Schweizer Druckwasserreaktoren und 9 g CO2eq/kWh für Schweizer Siedewasserreaktoren liegen, was im Vergleich zu anderen Formen der Energieerzeugung sehr niedrig ist. Vergleiche für die anderen Kennzahlen zeigen durchweg, dass die Technologien mit den geringsten Umweltauswirkungen (gemäss den meisten Indikatoren) Wind-, Kern- und Wasserkraft sind (siehe Abbildung 8.9 und Abbildung 8.10 im Hauptbericht).
Die Tiefenlagerung radioaktiver Abfälle ist in den meisten Kernenergieländern die Methode der Wahl und wird international als die wichtigste Methode zur Entsorgung abgebrannter Brennelemente angesehen, die keine nennenswerten Schäden für künftige Generationen darstellt. In der Schweiz wird die Nagra im November 2024 ihr Rahmenbewilligungsgesuch für ein Lager in Nördlich Lägern einreichen, das dann insbesondere von der Schweizer Aufsichtsbehörde ENSI geprüft wird. Die prognostizierten Kosten für den gesamten Entsorgungspfad werden alle fünf Jahre neu bewertet und betragen derzeit ca. 17.171 Milliarden CHF (gemäß Kostenstudie 2021, ohne Kosten für Nachbetrieb, Stilllegung und Bundesabfallströme). Dies entspricht etwa 1 Cent pro erzeugter kWh und ist bereits in den oben genannten Stromgestehungskosten für Schweizer KKW enthalten. Das Schweizer

Kernenergiegesetz schreibt vor, dass die Rückstellungen für die Entsorgung der Abfälle während des Anlagenbetriebs den Kernkraftwerksbetreibern belastet und in zwei speziellen Fonds, dem Entsorgungs- und Stilllegungsfonds, geäufnet werden.

Stand der Fusionstechnologie

Die Kernfusion birgt zwar ein enormes Potenzial als zukünftige Energiequelle, befindet sich jedoch noch in der Forschungsphase, und eine funktionierende Demonstrationsanlage zur Stromerzeugung muss noch erprobt werden. Aus diesem Grund ist die Fusion derzeit noch weit von kommerziellen Anwendungen entfernt, was es schwierig macht, einen genauen Zeitplan vorherzusagen. Es wird daher nicht erwartet, dass die Technologie in Energieszenarien im Zeitraum bis 2050 eine Rolle spielen wird. Obwohl in den letzten Jahrzehnten erhebliche Fortschritte erzielt wurden, müssen noch einige Herausforderungen durch gezielte Forschungs- und Entwicklungsaktivitäten bewältigt werden. Zu den wichtigsten Bereichen gehören die Optimierung von Plasmaszenarien, die Ableitung der Wärme, die Kontrolle von Plasmatransienten und die Weiterentwicklung der Materialforschung für plasmaseitige Komponenten (insbesondere unter Bedingungen mit hohem Neutronenfluss) sowie die Entwicklungen von Technologien für das sog. «Blanket», in welchem Tritium erbrütet werden soll.

Tabelle 4 Zusammenfassung des erreichten vs. erforderlichen Dreifachprodukts für verschiedene Fusionskonzepte

	Achieved Triple Product $[10^{21}\text{keV}\cdot\text{s/m}^3]$	Triple Product required for reactor $[10^{21}\text{keV}\cdot\text{s/m}^3]$
Inertial fusion energy (NIF)	≈ 10	$\approx 50-500$
Tokamak (JET, JT-60U,...)	≈ 1	≈ 5
Stellarator (W7-X)	≈ 0.1	≈ 5
Others: FRC, Z-pinch,...	≈ 0.0001	≈ 5
Private sector, tokamak	≈ 0.01	≈ 5
Private sector, other approaches	≈ 0.0005	≈ 5 for steady-state DT ≈ 50-1000 for steady-state alternative fuels

Wichtige Gütezahlen zur Beurteilung der Leistung einer Fusionsanlage sind das Dreifachprodukt (triple product) und der wissenschaftliche Leistungsmultiplikationsfaktor QSci, mit denen sich die technologische Reife eines Fusionskonzepts hin zu einem Fusionskraftwerk beurteilen lässt, das eine Nettoenergieproduktion und eine rentable kommerzielle Nutzung ermöglicht. Betrachtet man Tabelle 4, ist klar, dass der Tokamak (Ansatz der magnetischen Einschlussfusion) das vielversprechendste Konzept ist, aber noch weit von den Anforderungen für ein rentables Fusionskraftwerk entfernt ist. Darüber hinaus beträgt der höchste in einem Tokamak erreichte QSci derzeit ≈ 0,67, während ein QSci deutlich größer als 1 erforderlich ist (QSci bezieht sich nur auf die Verstärkung der Heizleistung und berücksichtigt weder die zusätzliche Leistung, die zum Betrieb der Anlage erforderlich ist, noch die Effizienz der Umwandlung von Wärmeenergie in Elektrizität). Auch bei der Trägheitsfusionsenergie wurden

Fortschritte erzielt, und es werden Anstrengungen unternommen, sie reaktorrelevanter zu machen.

Aufgrund sowohl physikalischer als auch technischer Herausforderungen ist der Zeitplan für ein erstes Fusionskraftwerk derzeit mit erheblichen Unsicherheiten verbunden. Obwohl viele private Unternehmen versprechen, dass der Strom bereits 2035 oder sogar früher ans Netz gehen wird, sollten diese Aussagen mit großer Vorsicht betrachtet werden. Angesichts der vielen Herausforderungen sollten solche Aussagen eher als motivierende Bestrebungen und im Kontext der Notwendigkeit gesehen werden, private Investoren anzuziehen.

Über ITER hinaus zielt die europäische Fusions-Roadmap auf einen ersten Demonstrationsreaktor DEMO bis 2045 ab. Fortschritte hängen von der Höhe der Finanzierung und den heute getroffenen Entscheidungen ab.

Résumé

Annalisa Manera and Andreas Pautz (Paul Scherrer Institute)

Énergie nucléaire en Suisse

En 2023, la Suisse a produit de l'énergie nucléaire au moyen de quatre réacteurs en exploitation (Beznau 1 et 2, Gösgen et Leibstadt) d'une capacité totale d'environ 3 gigawatts électriques (GWe). La production d'énergie nucléaire continue de jouer un rôle important dans le mix d'électricité en Suisse : en 2022, les quatre centrales nucléaires ont produit 23,1 TWh, ce qui représente près de 36 % de la production totale d'électricité et la deuxième plus grande contribution à la production nationale d'électricité. L'énergie hydraulique, qui contribue à presque 53 % de la production d'électricité en Suisse, reste la source d'énergie dominante. L'énergie nucléaire revêt une importance particulière pendant les mois d'hiver (en 2022, elle a contribué à plus de 40 % du mix de production national sur une période de cinq mois). Grâce à la prédominance de l'énergie hydraulique et de l'énergie nucléaire (89,2 % du mix d'électricité en 2022), le bilan net des émissions de CO2 de la production d'électricité en Suisse est aujourd'hui quasiment nul (tableau 1.3 du rapport principal).

Énergie nucléaire dans le monde

Dans les pays de l'OCDE également, l'énergie nucléaire reste la principale source d'électricité à faible émission de carbone (part de l'électricité en 2022 : 15,8 % de nucléaire, 12,6 % d'hydraulique, 9,9 % d'éolien, 5,9 % de solaire). Que ce soit dans les pays de l'OCDE ou dans le reste du monde, la majeure partie de la production d'électricité provient toutefois de la combustion de carburants fossiles (près de 50 % dans l'OCDE et plus de 60 % dans le monde). Le tableau s'assombrit encore si l'on considère la consommation mondiale d'énergie primaire, avec plus de 80 % de l'énergie toujours produite à partir de sources fossiles, contre seulement 7 % pour l'énergie hydraulique, 4 % pour l'énergie nucléaire et 5 % pour l'énergie éolienne et solaire prises ensemble.

32 pays du monde recourent à l'énergie nucléaire, 13 autres sont à un stade avancé de planification ou de construction pour inclure le nucléaire dans leur mix d'électricité (trois de ces pays sont déjà en train de construire des centrales nucléaires) et 17 autres sont en phase de décision. Quatre pays prévoient de sortir du nucléaire, mais seule l'Allemagne a définitivement cessé de produire de l'électricité à partir de l'énergie nucléaire en 2023. L'Espagne prévoit une sortie du nucléaire d'ici 2035, tandis que la Belgique, malgré sa décision de sortir du nucléaire en raison de la récente crise énergétique, a prolongé la durée de vie de deux de ses sept réacteurs, et la Suisse projette une exploitation à long terme de ses centrales existantes pouvant aller jusqu'à 80 ans, avec une sortie progressive dans le cadre de la mise en œuvre de la stratégie énergétique 2050.

En mars 2024, au total 415 centrales nucléaires, pour une puissance installée totale de 373.257 GWe, étaient en service dans le monde. De plus, 57 centrales sont en cours de construction, pour une capacité supplémentaire de 59,22 GWe. En Europe, 167 centrales nucléaires sont en service (148 GWe) et 9 en construction (10,1 GWe).

Les pays dans lesquels se trouve le plus grand nombre de centrales nucléaires en service sont les États-Unis, la France, la Chine et la Russie. En mars 2024, la Chine était le pays où l'énergie nucléaire connaissait la plus forte croissance, avec 27 centrales en construction, suivie par l'Inde (sept centrales en construction), la Turquie (4), l'Egypte (4), la Corée du Sud (2) et la Russie (4). La Chine a déjà atteint une puissance nucléaire installée de 53,3 GWe en 2023 (avec près de 400 TWh produits en 2022), 30,9 GWe supplémentaires sont en cours

de construction et ses plans de croissance sont considérables (avec l'objectif d'atteindre jusqu'à 150 GWe de capacité nucléaire installée d'ici 2030). En Europe, les pays suivants construisent ou prévoient de construire de nouvelles centrales dans un proche avenir : France (1 centrale EPR en construction, 6 centrales EPR 2 autorisées, d'autres planifiées), Royaume-Uni (2 centrales EPR en construction, 2 autres centrales EPR prévues), Slovaquie (1 centrale en construction, d'autres proposées), Bulgarie (2 centrales AP1000 prévues), République tchèque (4 centrales prévues, plus 3 sites identifiés pour plusieurs SMR), Pays-Bas (2 centrales prévues), Roumanie (2 centrales CANDU prévues, 6 modules SMR de NuScale proposés), Hongrie (2 centrales VVER approuvées), Slovénie (1 centrale proposée), Suède (2 centrales prévues d'ici 2035, 10 installations supplémentaires prévues après 2035), Estonie et Pologne (3 centrales AP1000 approuvées, 2 centrales APR1400 prévues, 24 installations SMR BWR-300 prévues). Deux nouvelles centrales ont en outre été raccordées au réseau en Biélorussie et en Finlande en 2023. Parmi les développements récents, il convient de mentionner la construction en Finland du premier dépôt en couches géologiques profondes au monde pour les déchets hautement radioactifs, dont la construction doit aboutir au milieu des années 2020. La Suède a aussi octroyé un permis de construire pour un tel site, dont la construction devrait commencer ces prochaines années, tandis qu'en France, une demande de construction est actuellement examinée par l'autorité de surveillance. Une décision concernant le site définitif est attendue pour 2025, et l'exploitation devrait débuter en 2040. Au Canada, la sélection du site d'entreposage des déchets devrait être connue en 2024. En Suisse, l'autorisation d'un dépôt en couches géologiques profondes est attendue vers 2030 (sous réserve d'une évaluation positive par les autorités de surveillance et, le cas échéant, d'un référendum facultatif), la mise en service prévue pour 2050.

Ces dernières années, notamment dans le contexte des changements géopolitiques induits par la guerre en Ukraine, plusieurs pays ont revu leurs plans en matière nucléaire. Ce qui a abouti à :

● la création en 2023 de l'Alliance européenne du nucléaire, dans le cadre de laquelle 16 pays (France, Belgique, Bulgarie, Croatie, République tchèque, Finlande, Hongrie, Pays-Bas, Pologne, Roumanie, Slovénie, Slovaquie, Estonie, Suède, Italie, Royaume-Uni) prévoient de mettre en place une industrie nucléaire européenne intégrée et s'engagent à atteindre une part de 150 GWe d'énergie nucléaire dans le mix d'électricité de l'UE d'ici 2050 (soit une augmentation de 50 % par rapport à la part actuelle) ;

● la création en 2024 de l'Alliance des petits réacteurs nucléaires modulaires de la Commission européenne dont l'objectif est de « maintenir le leadership technologique et industriel de l'Europe dans le domaine de l'énergie nucléaire » ;

● la déclaration sur l'énergie nucléaire lors de la Conférence des Nations unies sur le climat (COP28) en décembre 2023, adoptée par 22 pays qui se sont fixés pour objectif de tripler l'énergie nucléaire d'ici 2050 afin d'atteindre le nouvel objectif zéro, « en reconnaissant le rôle clé de l'énergie nucléaire dans l'atteinte d'émissions nettes de gaz à effet de serre/neutralité carbone à l'échelle mondiale d'ici le milieu du siècle environ ». Figurent parmi ces pays les États-Unis, la Bulgarie, le Canada, la République tchèque, la Finlande, la France, le Ghana, la Hongrie, le Japon, la Corée du Sud, la Moldavie, la Mongolie, le Maroc, les Pays-Bas, la Pologne, la Roumanie, la Slovaquie, la Slovénie, la Suède, l'Ukraine, les Émirats arabes unis et le Royaume-Uni ;

● l'introduction aux États-Unis d'un plan d'investissement pour promouvoir le développement de SMR et de micro-réacteurs et leur utilisation aux États-Unis ainsi qu'à l'étranger. La loi sur la réduction de l'inflation signée en 2022 vise à soutenir les centrales existantes et les nouvelles centrales par le biais d'aides à l'investissement et d'incitations fiscales, tant pour les grandes centrales existantes que pour les réacteurs de pointe plus récents destinés à la production d'uranium et d'hydrogène. La durée de vie de plusieurs centrales a été prolongée (p. ex. Diablo Canyon en Californie, dont la désaffectation été prévue en 2022. Six réacteurs ont vu leur durée de vie prolongée de 80 ans. Plusieurs autres attendent une décision de l'autorité de surveillance.) À noter que l'État du Michigan remet en service la centrale de Palisades, à l'arrêt depuis 2022. Certains projets pilotes ont été lancés avec succès dans des centrales nucléaires existantes afin d'utiliser l'énergie nucléaire pour la production d'hydrogène.

Compte tenu de la reconnaissance croissante de l'importance d'une production fiable en ruban, plusieurs entreprises américaines telles qu'Amazon, Google, Microsoft et des industries à forte consommation d'énergie comme Nucor (production d'acier) et Dow Chemicals ont signé des accords avec des fournisseurs ou des distributeurs d'énergie nucléaire pour leur approvisionnement futur.

Statut des réacteurs à eau légère (LWR) de génération III/III+ et durée de construction

Les réacteurs de génération III/III+ sont une nouvelle génération de centrales nucléaires basées sur la même technologie de réacteurs à eau légère (LWR) que celles des centrales en service actuellement, mais qui se distinguent par une nette amélioration des dispositifs de sécurité, dont les caractéristiques de conception tiennent compte des enseignements tirés des trois plus grands accidents nucléaires de l'histoire. En décembre 2023, 38 grandes unités LWR de génération III/III+ sont en service, et sur les 60 réacteurs actuellement en construction, 51 sont de grandes LWR de génération III/III+. D'autres unités ont été commandées ou des appels d'offre sont en cours (p. ex. trois unités en Pologne, deux unités en Grande-Bretagne, une unité en République tchèque, etc.), et plusieurs autres sont prévues.

La durée moyenne de construction des 38 réacteurs de génération III/III+ en service est de 7,7 ans, la médiane étant de 8 ans (voir figures 2.5 et 2.6 du rapport principal). En comparaison, la durée moyenne de construction des 413 réacteurs de génération II et II dans le monde est globalement de 7,5 ans, la valeur médiane de 6,3 ans. Ces chiffres ne tendent pas confirmer l'avis général selon lequel la durée de construction des nouvelles centrales a fortement augmenté. Ils corroborent plutôt un allongement modéré des travaux de construction, avec quelques projets d'exception notoires, notamment pour les premières centrales nucléaires en Europe et aux États-Unis, dont les durées de construction ont connu une hausse disproportionnée. D'autre part, il a été prouvé à moult reprises qu'il est techniquement possible de livrer un système clé en main en moins de six ans, pour autant qu'une chaîne d'approvisionnement performante soit mise en place pour les composants clés.

En particulier les centrales ABWR (GE Hitachi/Toshiba) au Japon se caractérisent par leur courte durée de construction, puisqu'elles ont toutes été achevées en moins de quatre ans. Le réacteur AP-1000 de Westinghouse aux États-Unis (Vogtle 3) et les deux centrales EPR à Olkiluoto (Finlande) et Flamanville (France) se situent à l'extrémité opposée du spectre, avec des durées de construction respectives de 10 et 16 ans et demi.

Dès le départ, ces projets ont dû faire face à des défis particuliers, car il s'agissait de chantiers pionniers pour la construction des premières grandes centrales de ce type en Europe et aux États-Unis après une pause de plusieurs décennies dans les projets de construction de nouvelles centrales, et il fallait relancer les capacités de fabrication et les chaînes d'approvisionnement. De plus, les autorités de surveillance, tant en Finlande qu'aux États-Unis, ont exigé d'importantes modifications de la conception des centrales pendant une bonne partie de la phase de construction. Les deux centrales EPR en construction à Hinkley Point au Royaume-Uni ont aussi subi des retards importants, bien que de moindre ampleur qu'en Finlande et à Flamanville. Ces retards étaient en partie dû à l'absence de certains éléments dans la chaîne d'approvisionnement britannique, à la nécessité de former la main-d'œuvre (avec des retards principalement dans la construction des bâtiments) et à un grand nombre de changements de conception (plus de 7000) imposés par l'autorité de surveillance. Malgré ces écueils, le gouvernement britannique a confirmé la construction de deux nouvelles centrales EPR sur le site de Sizewell.

Le degré d'exactitude de la conception en début de construction et la mise en place d'une chaîne d'approvisionnement et de capacités de fabrication opérationnelles sont donc des facteurs importants dans la détermination de la durée de construction ; l'expérience acquise sur plusieurs sites consécutifs et la fiabilité du cadre financier et juridique sont également des facteurs importants. La Chine n'a cessé de réduire les délais de construction de ses centrales, ses neufs dernières installations (de conception standardisée HPR1000 et ACPR-1000) ayant toutes été construites en 5 à 7 ans. Le récent exemple des Émirats arabes unis, où la société sud-coréenne KHNP a établi une capacité nucléaire de 5,2 GWe (4 unités APR1400) en neuf ans pour un coût total de seulement 24 milliards de dollars, est également remarquable.

Rentabilité des LWR de génération III/III+

Des estimations basées sur des sources scientifiques sérieuses (PSI 2019) chiffrent le coût de revient de l'électricité (levelized cost of electricity, LCOE) des nouvelles centrales nucléaires entre 7 et 12 centimes/kWh. Tant que la durée de construction reste inférieure à 8 ans (la médiane des 38 constructions de génération III/III+ est de 7,7 ans), il est possible d'atteindre des LCOE de 7 centimes, ce qui est conforme aux études PSI réalisées en 2019. Ce chiffre se situe bien dans la fourchette des LCOE actuels et futurs pour les sources d'énergie renouvelables en Suisse et les centrales hydroélectriques existantes, et fournirait de l'électricité en ruban. Les résultats PSI de 2019 sont conformes à d'autres études validées par des experts et publiées dans la littérature en libre accès. Les LCOE actuels pour l'exploitation des centrales nucléaires suisses existantes se situent à 4,0-5,5 cts/kWh (y compris l'intégralité des coûts de gestion des déchets). Une exploitation à long terme de ces centrales jusqu'à 60 ans entraînerait une augmentation de 1 à 2 centimes du coût de production de l'électricité. Il convient toutefois de noter que le concept de coût de revient de l'électricité LCOE a été introduit à l'origine pour comparer des sources d'énergie réglables, mais que sa valeur est limitée dans un système énergétique de plus en plus complexe, avec une proportion de plus en plus importante d'énergies renouvelables fluctuantes. Il est de plus en plus reconnu qu'il est nécessaire, en pareil cas, de tenir compte non seulement des coûts de production de l'électricité, mais également de l'ensemble des coûts des systèmes (coûts d'équilibrage, coûts de développement du réseau, coûts de réserve, etc.). L'OCDE a récemment publié la tentative d'une étude de ce type pour le système énergétique suisse, mais aucun grand modèle complet incluant différents scénarios d'utilisation de l'énergie nucléaire n'a jamais été réalisé.

Les premières centrales EPR à 1600 MWe à Olkiluoto et Flamanville ont été bien plus coûteuses que les centrales APR1400 sud-coréennes à 1400 MWe construites aux Émirats arabes unis. Alors que l'APR1400 coûte 6 milliards de dollars par unité, les deux EPR à Olkiluoto et Flamanville coûtent respectivement près de 11 et 13,2 milliards d'euros. Ce coût élevé du capital doit toutefois être rapporté à l'énergie produite. Une seule unité EPR produirait plus de 12 TWh/an. À titre de comparaison, pour produire la capacité annuelle d'une centrale EPR avec des installations solaires alpines, il faudrait plus de 3800 « Alpin Solar » (installation du barrage de Muttsee) pour un coût supérieur à 30 milliards de francs (sans compter les coûts supplémentaires de réserve, de stockage et de développement du réseau), ou plus de 780 installations du type de l'installation de Gondosolar pour un coût d'env. 29 milliards de francs.

Le coûts élevé du capital d'une grande centrale nucléaire constitue l'un des principaux défis économiques de l'énergie nucléaire, car il réduit le nombre potentiel d'investisseurs privés. Ce risque est légèrement atténué par les SMR et pourrait être totalement éliminé par les microréacteurs dont le coût total du capital est comparable aux centrales solaires alpines, tout en fournissant une production d'énergie beaucoup plus élevée et plus régulière.

Plusieurs modèles impliquant la participation des gouvernements en tant que bailleurs de fonds propres ou fournisseurs de crédit ou par le biais de mesures politiques telles que garanties de prêt ou « contracts of différences » (CfD, rémunérations minimales) ont été mis en œuvre par le passé pour maîtriser le coût élevé du capital des grandes centrales. Contrairement à la perception générale du public, l'énergie nucléaire est la moins subventionnée de toutes les sources d'énergie, comme nous le verrons en détail à la section 1.4.2 du présent rapport. Dans l'Union européenne, sur la période 2015-2022, les subventions pour l'énergie nucléaire ont atteint un record de 7,6 milliards d'euros en 2021, comparé à 88 milliards d'euros pour les énergies renouvelables et 123 milliards d'euros pour les combustibles fossiles. Aux États-Unis, le montant maximum des subventions pour l'énergie nucléaire était inférieur à 600 millions de dollars entre 2016 et 2022, contre plus de 17 milliards pour les énergies renouvelables, plus de 2,5 milliards pour le charbons et près de 3 milliards pour le gaz.

Les taux d'intérêt du capital ont été fortement influencés par le système financier (p. ex. la garantie de crédit de l'État) et le cadre réglementaire. Une méthode efficace pour réduire les coûts consiste à construire plusieurs unités sur le même site. Lors du projet Barakah aux Émirats arabe unis par exemple, les coûts ont pu être réduits de 40 % entre la construction des unités 1 et 4. Au nombre des facteurs ayant un impact positif sur la réussite d'une nouvelle construction figurent la réalisation des parties pertinentes de la conception avant le début de la construction, l'existence d'une chaîne d'approvisionnement bien établie, l'accès à une main-d'œuvre qualifiée et un cadre réglementaire stable.

Le coût de production de l'électricité pour les SMR atteindront vraisemblablement un niveau similaire aux grandes centrales nucléaires, les économies d'échelle étant compensées par les retombées de la fabrication (production de masse), bien que des coûts plus élevés soient attendus pour les premières unités. Pour le premier projet de SMR de NuScale, qui devrait être construit ces prochaines années dans l'Utah, le coût a été estimé à 5,8 cts/kWh (estimation de 2020). En raison d'une hausse des taux d'intérêt de 150 % et d'une augmentation considérable des frais de matériel (p. ex. 40 % d'augmentation du coût de l'acier) au cours des 18 derniers mois, les coûts devraient grimper à 8,9 cts/KWh jusqu'en 2023. Le projet n'étant pas compétitif par rapport à la production d'électricité à partir du gaz et du charbon, auxquels les services publics locaux de l'Utah avaient accès, il a été suspendu au profit d'une centrale à gaz.

Sécurité de la génération III/III+

D'importantes modifications ont été apportées aux exigences de sécurité des centrales de génération III/III+. Les systèmes de gestion des accidents majeurs font désormais partie intégrante de la conception, et l'indépendance des différents niveaux de sécurité superposés (defense in depth) a été renforcée. La mise en œuvre de cette nouvelle philosophie de sécurité a abouti à la mise en place de toute une série de nouveaux systèmes de sécurité passifs (qui ne dépendent pas d'une énergie externe comme les générateurs diesel ou d'une intervention de l'exploitant pour fonctionner) et de délais de carence prolongés (grace period) dont l'objectif est de pratiquement exclure la survenance de conséquences graves telles que cœur en fusion suivi d'une défaillance de l'enceinte de confinement (containment), qui pourraient entraîner des fuites précoces ou importantes de substances radioactives. Ces nouvelles approches ont notamment débouché sur les éléments suivants :

● prolongation du délai de carence (pendant lequel aucune intervention humaine n'est nécessaire, même dans les circonstances les plus extrêmes d'un accident) de 30 minutes pour les conceptions de génération II à minimum 3 jours, plus souvent plus d'une semaine ;

● fréquences de dommages nucléaires inférieures à 10^{-6}/an (c'est-à-dire, en termes de probabilité, moins d'une fois tous les millions d'années) ;

● probabilité d'une défaillance de l'enceinte de confinement après l'accident avec rejets dans l'environnement inférieure à 10^{-7}/an (soit moins d'une fois tous les dix millions d'années).

La probabilité d'un accident nucléaire et d'une très importante émission de radioactivité est donc d'une à deux fois inférieure à celle des centrales actuelles de génération II dûment mises à niveau, lesquelles ont déjà atteint un excellent niveau de sécurité grâce aux mesures et aux tests de résistance post-Fukushima.

Statut des petits réacteurs modulaires (SMR)

Les petits réacteurs modulaires (SMR) sont des réacteurs modernes d'une puissance nominale pouvant atteindre 300 MW(e) par unité. Ils sont conçus pour être construits en usine et transportés sur le lieu où ils seront utilisés. Ils sont généralement installés sous terre. L'Agence de l'OCDE pour l'énergie nucléaire (AEN) estime que les SMR constitueront près de 9 % de la capacité totale des nouvelles centrales nucléaires d'ici 2035. Dix SMR sont actuellement en service en Russie et en Chine, et plusieurs autres sont en construction ou attendent une approbation (États-Unis, Canada, France), voir tableau 1 et 2.

Parmi les SMR refroidis à l'eau, NUWARD (EDF), Roll-Royce (Royaume-Uni), BWXR-300 (États-Unis), Holtec-180 (États-Unis), AP300 (États-Unis) et VOYGR (NuScale) sont les types de construction les plus avancés pour une utilisation en Europe d'ici 2030. VOYGR a déjà été homologué aux États-Unis, tandis que les autres conceptions en sont à différents stades de pré-certification aux États-Unis, au Canada et dans certains pays européens. Certains fournisseurs de SMR ont reçu des commandes (p. ex. BWRX-300 au Canada). En 2022, NUWARD a reçu 500 millions d'euros de la part du gouvernement français en vue de la construction d'un premier réacteur prévu pour 2030.

L'intérêt général pour les SMR est né de la nécessité d'alimenter en électricité des régions isolées ou des zones hors réseau qui dépendent actuellement du gaz, du pétrole ou du diesel, et de remplacer les centrales électriques à combustibles fossiles vieillissantes de l'ordre de 300 à 400 MWe. Pour les pays disposant de réseaux électriques de petite taille, dans lesquels l'utilisation d'une grande centrale nucléaire ne serait pas possible ou dont les investisseurs et exploitants ne seraient pas en mesure ou refusent d'investir des capitaux importants, les SMR sont considérés par certains comme une option économiquement viable. De plus, les SMR peuvent offrir des possibilités d'alimenter des sites industriels gourmands en énergie (p. ex. l'industrie du béton et de l'acier) ou de proposer des applications non électriques de l'énergie nucléaire, telles que le chauffage urbain, le dessalement de l'eau de mer ou même la production d'hydrogène.

En raison de leur taille réduite, la plupart des SMR présente des caractéristiques de sécurité basées entièrement sur la sécurité passive. De ce fait, la Nuclear Regulatory Commission américaine a approuvé une nouvelle règle de dimensionnement des zones de planification d'urgence (emergency planning zone, EPZ) basée sur différents scénarios d'accident et leurs conséquences. Le SMR de NuScale a donc été autorisé avec une EPZ limitée au périmètre du terrain des installations (c'est-à-dire qu'aucune zone d'évacuation n'est requise). D'autres SMR aux États-Unis devraient bénéficier d'une réglementation similaire.

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Tableau 1 LWR SMR à un stade de développement avancé

Nom	Puissance thermique [MWth (MWe)]	Type	Organisation de conception	Pays	Statut
CAREM	100 (30)	Integral PWR	CNEA	Argentine	En construction
ACPR50S	200 (60)	Floating PWR	CGNCP	Chine	En construction
ACP100	385 (125)	Integral PWR	CNNC et NPIC	Chine	Début construction en 2021
KLT-40S	150 (35)	Floating PWR	OKBM	Russie	2 unités en service
VOYGR	250 (77)	Integral PWR	NuScale Power	États-Unis	Sélectionné aux États-Unis et en Europe
AP300	900 (300)	One-loop PWR	Westinghouse	États-Unis	Sélectionné en GB
UK SMR	1,358 (470)	Integral PWR	Rolls-Royce	GB	Sélectionné en Estonie et en GB
NUWARD	540 (170)	Integral PWR	EDF	France	Premier du genre en France d'ici 2030.
BWRX-300	870 (290)	Integral BWR	GE-Hitachi	États-Unis	Plusieurs unités prévues au Canada et aux États-Unis Sélectionné en Europe.
SMR-160	525 (160)	PWR	Holtec	États-Unis	Sélectionné dans divers pays
SMART	365 (107)	PWR	KAERI	Corée	Autorisé en Corée
RITM-200	175 (55)	Floating PWR	OKBM	Russie	6 unités en service. Plus en construction.
RITM-200N	190 (55)	On-shore PWR	OKBM	Russie	Premier concrètement planifié pour 2024.
RITM-200S	198	Floating PWR	OKBM	Russie	Prévu sur le site de la mine de cuivre de Baimskaya, déploiement en 2027.
RITM-200M	175 (50)	Floating PWR	OKBM	Russie	Protocole d'accord signé pour le déploiement aux Philippines et au Myanmar.

Les avantages principaux des SMR sont leurs très faibles coûts d'investissement initiaux dus à la taille réduite de l'installation, les délais de construction réduits résultant du passage à la production en usine, la grande flexibilité de la compensation de charge, ce qui facilite leur intégration avec des sources d'énergie renouvelables intermittentes, et des concepts de sécurité améliorés. Les économies d'échelle des grandes centrales nucléaires seront probablement remplacées par des gains de production (modules fabriqués en usine) et des chantiers plus simples, le coût du kWh étant du même ordre que pour celles-ci. L'AIEA et les autorités de surveillance américaines/européennes sont en train d'harmoniser les procédures d'octroi de licence pour les SMR afin de créer un cadre stable et transparent qui évite les modifications imprévisibles des réglementations d'octroi nationales.

La plupart des SMR à court terme appartiennent, comme les grands réacteurs, à la catégorie des réacteurs à eau légère de génération III/III+, avec une perspective plausible d'exploitation commerciale des premières centrales de démonstration dans les pays occidentaux d'ici 2030, voire avant (voir tableau 1 ci-dessus). Les SMR avancés faisant appel à d'autres agents réfrigérants que l'eau (p. ex. métal liquide, hélium, sel fondu) font partie des centrales nucléaires de génération IV et leur développement est assuré par un grand nombre de start-up. Toutefois, sur le plan commercial, certaines de ces conceptions (le sel fondu p. ex.) accusent un retard de plusieurs années sur le développement des LWR. Alors que des SMR non refroidis à l'eau sont déjà en service en Chine et en Russie, le premier SMR refroidi au sodium devrait être celui de Terrapower, qui doit être construit dans le Wyoming (États-Unis). La demande de permis de construire pour ce SMR a été déposée en mars 2024 et acceptée aux fins d'examen par l'autorité de surveillance américaine en mai 2024.

Tableau 2 SMR non-LWR à un stade de développement avancé

Nom	Puissance thermique (MWth)	Type	Organisation de conception	Pays	Statut
Spectre thermique					
HTR-PM	500	HTGR	INET	Chine	2 unités en service en Chine depuis déc. 2021, 18 unités HTR-PM supplémentaires proposées.
KP-FHR	311	MSR / fuel solide	Kairos Power	États-Unis	Permis de construire pour une unité de démonstration obtenu en déc. 2023.
XE-100	200	HTGR	X-energy	États-Unis	Pré-certification terminée au Canada. Pré-licence aux États-Unis. Sélectionné par Dow Chemical (États-Unis)
IMSR	884	MSR intégral	Terrestrial Energy	Canada	Pré-licence aux États-Unis et au Canada.
Spectre rapide					
ARC-100	286	SFR	ARC Clean Tech.	Canada	Pré-licence aux Canada.
Wasteburner	750	MSR	Moltex Energy	Canada	Pré-licence aux Canada
Sodium	840	SFR	TerraPower	États-Unis	Pré-licence aux États-Unis. À construire dans le Wyoming (États-Unis)
BREST-OD-300	700	LFR	NIKIET	Russie	En construction en Russie. Fin des travaux prévu pour 2026.
CFR-600	1500	SFR	CNNC	Chine	2 unités en construction en Chine. Connexion au réseau en 2024-2025.

État de la technologie des microréacteurs

Au cours des sept dernières années, une tendance intéressante s'est dessinée à propos de ce que l'on appelle les microréacteurs, lesquels sont conçus pour atteindre des puissances électriques allant jusqu'à environ 10 MWe (plusieurs d'entre eux sont en cours de développement aux États-Unis, voir le tableau 4.1 du rapport principal). Il s'agit de réacteurs entièrement assemblés en usine, qui tiennent dans un container ISO pour être aisément acheminés (par bateau, camion ou train) de l'usine au site d'utilisation (aucun chantier n'est nécessaire) et qui fonctionnent pendant 5 à 10 ans ou plus sans renouvellement du combustible. Ils peuvent fonctionner de manière indépendante, comme partie intégrante du réseau électrique ou au sein d'un micro-réseau. Ils sont conçus pour être utilisés dans des zones isolées (p. ex. dans des sites miniers) ou pour approvisionner en électricité et en chaleur des industries à forte consommation d'énergie (dessalement de l'eau, production d'hydrogène, etc.). Ils présentent également un intérêt pour les industries qui ont besoin d'un certain degré d'indépendance vis-à-vis du réseau électrique en termes de sécurité d'approvisionnement. Le refroidissement se fait au gaz (hélium), au métal liquide, au sel fondu ou par des caloducs (sodium) (heat pipes).

Leur développement avance à vitesse grand V en raison de leur très petite taille et de la simplicité de leur conception. La première unité de démonstration (caloducs) a été élaborée, construite et testée en trois ans par la NASA et le laboratoire national Los Alamos pour un coût inférieur à 20 millions de dollars. Une deuxième unité (refroidie au métal liquide) est actuellement en construction dans l'Idaho (États-Unis) et devrait démarrer début 2025. Une unité refroidie au fluorure a obtenu le permis de construire en décembre 2023 et sa mise en service est prévue pour 2026. Trois autres conceptions se situent à différentes phases du processus d'obtention de la licence aux États-Unis et au Canada.

Puisque les microréacteurs sont entièrement construits en usine et qu'ils devraient profiter des effets de la fabrication en série, une courbe d'apprentissage positive est attendue comme dans d'autres secteurs. D'autres avantages présumés sont les très faibles coûts du capital (de l'ordre d'env. 100 millions de dollars ou moins), qui pourraient le rendre accessible à un plus grand nombre d'investisseurs, et le faible coût de production de l'électricité (LCOE) par rapport aux alternatives de secours dans les régions isolées ou les grandes industries, du fait de la fabrication complète en usine, de la très faible surface au sol (environ 15 m^2 pour la centrale et moins de 2'000 m^2 pour le site de la centrale), du planning prévisible de construction et du risque réduit de rayonnements. En raison de la très faible quantité de combustible et de la simplicité de la construction, ils s'apparentent plus à des réacteurs de recherche, leur homologation devrait donc être beaucoup plus rapide que celle des SMR ou des grandes centrales nucléaires. L'utilisation de combustible tri-structurel isotropique à particules (TRISO) plus hautement enrichi dans les microréacteurs nécessite le développement de capacités de production de combustible adaptée, actuellement en cours aux États-Unis et en France. Il ne s'agit toutefois pas d'un frein, car la technologie est connue (le même combustible est utilisé dans les SMR HTR-PM en service en Chine).

État des réacteurs de génération IV et autres qu'à eau légère

Les réacteurs non refroidis à l'eau (refroidis p. ex. au gaz, au plomb, au sodium ou au sel fondu) sont développés dans le but d'augmenter leur rendement, soit en améliorant l'efficacité thermodynamique et/ou l'utilisation du combustible et en réduisant davantage la quantité de déchets hautement radioactifs (dans le cas des réacteurs à spectre rapide), ce qui permet de boucler le cycle du combustible nucléaire[3]. Il existe plusieurs conceptions, dont les plus prometteuses sont mentionnées ici :

• Réacteurs thermiques refroidis au gaz utilisant l'hélium comme agent réfrigérant. Ces réacteurs présentent une meilleure efficacité thermodynamique pour convertir la puissance thermique générée dans le réacteur en électricité et, comme ils fonctionnent à des températures beaucoup plus élevées que les LWR, ils se prêtent également à la production de chaleur pour les processus industriels à haute température et à forte consommation d'énergie (voir figure 1 pour une illustration des températures requises pour différents processus industriels et des conceptions de réacteurs correspondantes susceptibles de

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[3] Plus de 90 % du combustible nucléaire irradié est réutilisable. Dans un cycle du combustible fermé, le combustible nucléaire usé est retraité afin d'en extraire la matière réutilisable (généralement de l'uranium), laquelle est destinée à la fabrication de nouveaux éléments de combustible. Un tel cycle peut par exemple être obtenu en combinant des LWR et des réacteurs rapides de génération IV. Les réacteurs rapides utilisent des neutrons de haute énergie (« rapides ») pour la fission du combustible nucléaire, tandis que les LWR utilisent principalement des neutrons thermiques (de basse énergie). Un cycle du combustible fermé permet d'améliorer la durabilité en augmentant la production d'énergie par unité de masse de combustible et en réduisant la quantité de déchets hautement radioactifs par unité d'énergie.

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fournir de telles températures). Deux réacteurs thermiques refroidis au gaz (conception HRT-PM) sont déjà opérationnels depuis 2021 en Chine. Le Xe-100 (X-energy, États-Unis) vient de passer avec succès la pré-certification au Canada (c'est-à-dire que l'autorité de surveillance n'a identifié aucun problème qui empêcherait l'octroi d'une licence).

• Réacteurs rapides refroidis au moyen de métaux liquides (sodium ou plomb/plomb-bismuth). Ce type de réacteur travaillent à des températures élevées, entre celles des LWR et des réacteurs refroidis au gaz, et à une pression proche de la pression ambiante (pression atmosphérique). Il existe une vaste expérience pratique (France, Japon, Russie, etc.), et plusieurs centrales sont actuellement en service (voir tableau 3). Terrapower (entreprise américaine) mettra sur le marché un SMR rapide refroidi au sodium, dont la première installation devrait être construite dans le Wyoming (États-Unis) avant 2030.

• Réacteurs à sels fondus (MSR), utilisant du sel fondu comme agent réfrigérant, combustible et/ou modérateur. Ces réacteurs fonctionnent à des températures élevées, et il existe des conceptions thermiques et rapides. L'un des principaux défis opérationnels subsistant avec ce type de réacteurs est la nature hautement corrosive des sels. En décembre 2023, un projet de réacteur thermique de KAIROS (États-Unis) utilisant du sel fondu uniquement comme agent réfrigérant (avec du combustible HALEU TRISO et du graphite comme modérateur) a obtenu un permis de construire pour une première centrale de démonstration dans le Tennessee. Un MRS intégral (Terrestrial Energy) est actuellement en phase de pré-licence aux États-Unis et au Canada. Un MSR expérimental, utilisant du sel fondu à base de thorium comme combustible, a obtenu sa licence d'exploitation en juin 2023 en Chine. La construction de ce réacteur TMSR-LF1 a débuté en septembre 2018 et devrait se terminer en 2024. Cependant, selon les rapports, il a été achevé en août 2021, à la suite d'un coup d'accélérateur des travaux.

Réacteurs de génération IV pertinents pour le marché occidental :

• KAIROS, Terrestrial, X-energy (tous des réacteurs thermiques, voir tableau 2),

• Terrapower, IMSR (Moltex), ARC-100 (tous des réacteurs rapides, voir tableau 2),

• les conceptions de microréacteurs actuellement en phase d'octroi de licence aux États-Unis et au Canada (voir tableau 4.1 du rapport principal).

Tableau 3 Réacteurs rapides en service (tous du type SFR)

Pays	Nom du réacteur	Années en opération	État actuel
Chine	CEFR	2010-actuellement	Actif
Inde	FBTR	1985-actuellement	Actif
Russie	BOR-60	1969-actuellement	Actif
Inde	PFBR	Planifié pour 2024	En construction
Russie	BN-600	1980-actuellement	Actif
Russie	BN-800	2014-actuellement	Actif

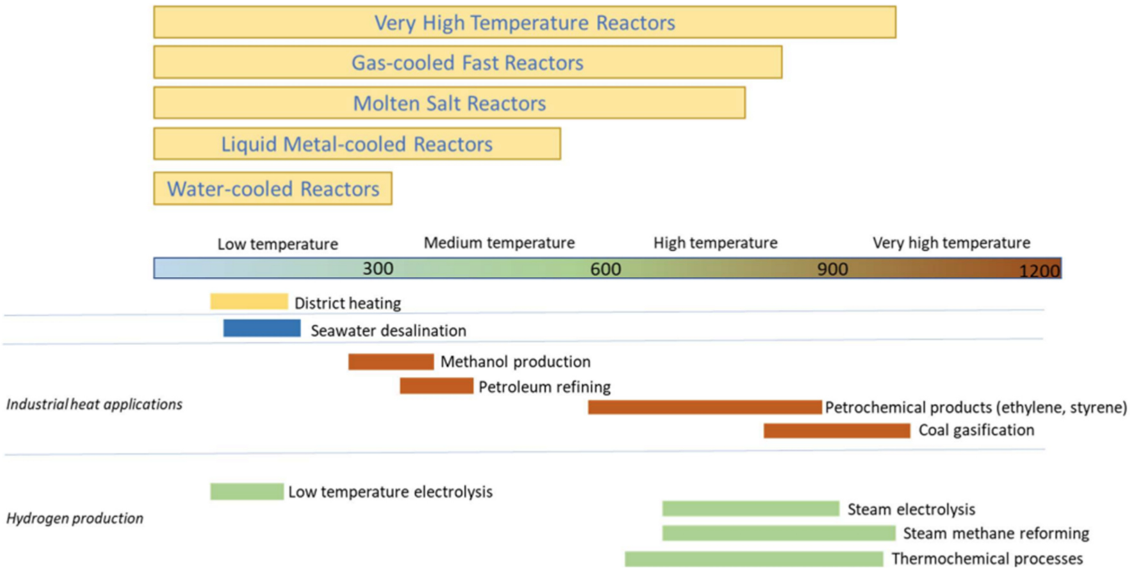

Figure 1 : Température de départ des technologies nucléaires et utilisations non électriques correspondantes

Disponibilité de l'uranium et cycles du combustible alternatifs

Les réserves naturelles d'uranium sont une ressource très répandue (voir figure 2) et suffisent pour les siècles à venir. Elles dépendent du prix du marché, tout comme les autres ressources.

Le combustible faiblement enrichi (LEU) utilisé dans les réacteurs à eau légère (p. ex. dans les centrales suisses) est produit dans plusieurs installations d'enrichissement, et la diversité et les différentes sources sont suffisantes pour assurer son approvisionnement. La sécurité de l'approvisionnement en combustible nucléaire de la Suisse ne devrait pas être menacée à long terme.

En ce qui concerne la seconde moitié de ce siècle, il est raisonnable de penser qu'un besoin accru d'énergie nucléaire entraînera une augmentation des activités d'exploration et donc des réserves d'uranium. De plus, durant cette période, le technologie des réacteurs à cycle fermé se développera de telle sorte qu'il sera possible d'utiliser d'autres combustibles ayant un potentiel énergétique beaucoup plus important que l'U-235, ce qui prolongera la disponibilité du combustible nucléaire de centaines à plusieurs milliers d'années. À l'échelle mondiale, plusieurs réacteurs rapides sont déjà opérationnels ou en construction (voir tableaux 2 et 3) ou en sont à un stade de planification avancé, tel que le réacteur de Terrapower dont la première centrale devrait être construite dans le Wyoming au cours de cette décennie. Un combustible important pour les réacteurs modernes non refroidis à l'eau, incluant la conception de Terrapower, est l'uranium hautement enrichi (HALEU), caractérisé par un enrichissement compris entre 5 et 20 %. La production de combustible HALEU devrait donc augmenter. Ironiquement, les incertitudes quant à la chaîne d'approvisionnement du combustible nucléaire provoquées par la guerre en Ukraine favoriseront le renforcement et l'expansion des capacités et de la résilience de l'approvisionnement en combustible de l'Occident ces 5 à 10 prochaines années. De nouvelles chaînes d'approvisionnement du combustible HALEU sont actuellement développées aux États-Unis et en France, tandis que la capacité des installations existantes d'uranium faiblement enrichi (LEU) a été augmentée tant en Europe qu'aux États-Unis. Le LEU est généralement utilisé dans les réacteurs conventionnels refroidis à l’eau. À fin 2023, les États-Unis ont en outre ouvert trois nouvelles mines d’uranium pour gagner en indépendance dans leur approvisionnement.

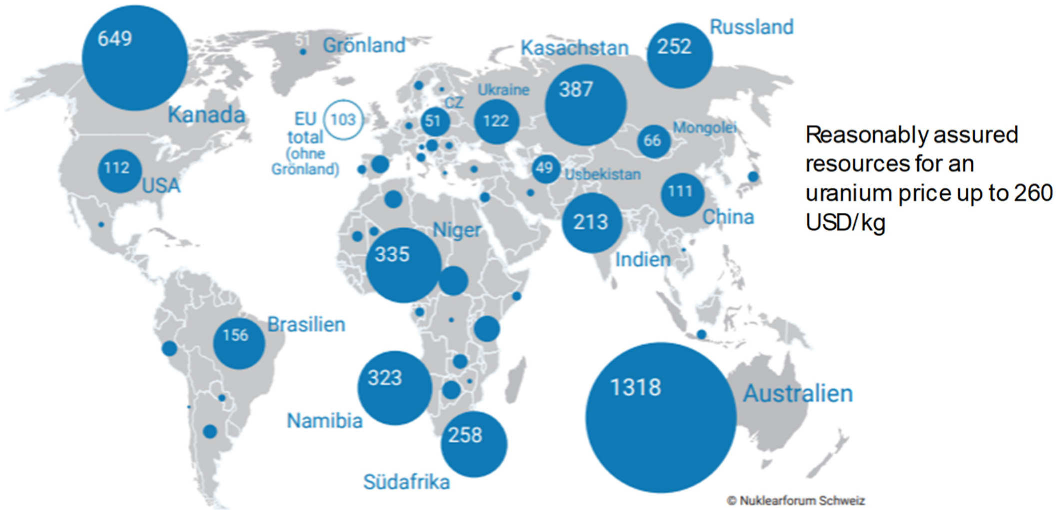

Figure 2 Répartition mondiale des réserves d’uranium au 1er janvier 2021 (en milliers de tonnes) à un prix allant jusqu’à 260 USD/kg. Données tirées du Redbook 2022 de l’AIEA.

Les cycles du combustible les plus avancés, visant une utilisation plus efficace du combustible et une réduction des déchets hautement radioactifs, se basent sur le retraitement du combustible et comprennent :
● la production de nouveaux éléments de combustible (appelés MOX) pour les LWR en utilisant le plutonium (Pu) et l’uranium (U) retraité issu du combustible usé des LWR. Normalement, quatre éléments de combustible irradiés (Fuel Assemblies FA) peuvent donner naissance à un nouvel élément de combustible. Il s’agit d’une technologie éprouvée. Le combustible MOX a été utilisé avec succès pendant plusieurs années dans des centrales suisses. La loi sur l’énergie nucléaire de 2003 a interdit le retraitement par moratoire. Dans d’autres pays (p. ex. en France), le retraitement reste une pratique de routine ;
● Réacteurs surgénérateurs rapides uranium-plutonium (cycle du combustible Breed-and-Burn). Dans ce cas, du combustible supplémentaire est produit dans le cœur du réacteur pendant le fonctionnement de la centrale, ce qui permet de produire plus d’énergie à partir de la même quantité de combustible. Ce processus nécessite du combustible HALEU ;
● Consommation d’actinides mineurs dans des « brûleurs » spécifiques (p. ex. les conceptions de réacteurs MYRRHA et Transmutex) afin de réduire la radiotoxicité du combustible usé existant. Ce processus nécessite le retraitement des actinides mineurs, léquel a déjà été testé en laboratoire, mais n’existe pas encore à l’échelle industrielle. Des brûleurs spécifiques peuvent permettre d’atteindre un taux de transmutation par unité d’énergie plus élevé que dans les réacteurs rapides conventionnels, mais l’exploitation de réacteurs de transmutation spéciaux accentue la complexité technologique et donc les risques opérationnels. De plus, la majeure partie de la quantité retraitée serait toujours de l’uranium de retraitement (RepU), dont la manipulation ultérieure par le transmuter n’est pas prise en compte. Si toutefois les actinides mineurs étaient retraités, un transmuter spécial pourrait constituer un élément pertinent du cycle du combustible et contribuer à minimiser le flux de déchets ;
● Cycle thorium (Th)-uranium : Le Th-232 est un isotope fertile naturel analogue à l’U-238, qui trois fois plus abondant dans la croûte terrestre que l’uranium. Le thorium peut se transformer en U-233 tout comme l’U-238 peut être utilisé pour générer du Pu-239. L’utilisation du thorium nécessite un retraitement et quelques adaptations de la technologie actuelle. Cependant, si

cette solution était mise en œuvre, elle pourrait fournir du combustible supplémentaire pendant des centaines ou des milliers d'années, compte tenu des exigences actuelles. Des réacteurs expérimentaux au thorium ont été exploités avec succès par le passé. Un réacteur de démonstration au thorium utilisant du sel fondu a été mis en service en Chine en 2023 et un réacteur équivalent de 373 MWt devrait suivre d'ici 2030. Un surgénérateur rapide au thorium de 40 MWth est opérationnel en Inde, le pays disposant de grandes quantités de cette ressource.

Il existe des installations commerciales de retraitement de combustible en France, en Russie, en Inde, au Japon (cette dernière est en construction et devrait être terminée en 2024) et en Chine (deux sont en construction, la première devrait être opérationnelle en 2025). En Grande-Bretagne, une usine de retraitement a été fermée en 2022 après 58 ans d'exploitation. Les États-Unis s'emploient à reconstituer leurs capacités de retraitement après l'arrêt du processus dans les années 1970 pour des raisons politiques. Ils n'ont pas encore accepté la demande de la Corée du Sud de développer ses propres capacités de retraitement.

Homologation de nouvelles centrales nucléaires en Suisse

La loi suisse sur l'énergie nucléaire interdit expressément le dépôt de demandes d'autorisation générale pour de nouvelles centrales nucléaires. Ne sont pas concernées par cette interdiction les installations de stockage, d'élimination des déchets et de recherche ainsi que les centrales nucléaires à faible risque (un terme qui est expliqué plus en détail dans l'ordonnance sur l'énergie nucléaire). Ces dernières ne requièrent pas de demande d'autorisation générale. En raison de leur niveau élevé de sécurité passive et leur faible inventaire radioactif, les microréacteurs et les SMR peuvent potentiellement être classés parmi les installations à faible risque selon la loi suisse sur l'énergie nucléaire.

Écobilan (life cycle assessment, LCA)

Les impacts environnementaux quantifiés par la LCA englobent les effets sur le changement climatique, les émissions de polluants atmosphériques et de substances toxiques ainsi que l'utilisation du sol, de l'eau et d'autres ressources. Les résultats de l'écobilan peuvent être utilisés pour comparer l'impact environnemental de différentes technologies de production d'électricité. Il existe plusieurs études internationales, et notamment des analyses de l'institut Paul Scherrer réalisées spécialement pour les centrales nucléaires suisses. L'impact environnemental des centrales nucléaires suisses est en grande partie déterminé par l'origine de l'uranium, dont les émissions totales de gaz à effet de serre sont d'environ 6 g CO2eq/kWh pour les réacteurs à eau pressurisée et 9 g CO2eq/kWh pour les réacteurs à eau bouillante, ce qui est très faible par rapport aux autres formes de production d'énergie. Les comparaisons avec les autres indicateurs montrent invariablement que les technologies présentant les impacts environnementaux les plus faibles (selon la plupart des indicateurs) sont l'éolien, le nucléaire et l'hydroélectricité (voir figures 8.9 et 8.10 du rapport principal).

Le dépôt en couches géologiques profondes des déchets radioactifs est la méthode de choix dans la plupart des pays producteurs d'énergie nucléaire et est considérée comme la principale méthode de gestion du combustible irradié ne représentant aucun danger majeur pour les générations futures. En Suisse, la Nagra déposera en novembre 2024 une demande d'autorisation générale pour un stockage à Nord des Lägern, qui sera ensuite notamment examinée par l'autorité de surveillance suisse ENSI. Les coûts prévus pour l'ensemble de la voie d'évacuation sont évalués tous les cinq ans et s'élèvent actuellement à env. 17,171 milliards de francs (selon l'étude de coûts de 2021, sans les coûts de post-exploitation, de désaffectation et des flux des déchets fédéraux). Ce montant équivaut à 1 centime par kWh produit et est déjà compris dans les coûts de production d'électricité des centrales suisses mentionnés plus haut. La loi suisse sur l'énergie nucléaire prévoit que les provisions pour la gestion des déchets pendant l'exploitation des installations sont à la charge des exploitants des centrales nucléaires et alimentent deux fonds spéciaux (le fonds d'évacuation des déchets et le fonds de désaffectation).

État de la technologie de fusion

Si la fusion nucléaire recèle un énorme potentiel en tant que source d'énergie future, elle n'en est encore qu'à la phase de recherche, et une installation de démonstration capable de produire de l'électricité doit encore être testée. Cette technologie est donc encore loin des applications commerciales, ce qui complique la prévision d'un calendrier précis. Il n'est d'ailleurs pas prévu que la fusion joue un rôle dans les scénarios énergétique avant 2050. Malgré les progrès considérables réalisés ces dernières décennies, certains défis doivent encore être relevés grâce à des activités ciblées de recherche et développement. Au nombre des principaux domaines figurent l'amélioration des scénarios de plasma, la propagation de la chaleur, le contrôle des transitoires du plasma et le développement de la recherche sur les matériaux exposés au plasma (en particulier dans des conditions de flux neutronique élevé), ainsi que le développement de technologies pour le « blanket », dans lequel le tritium doit être transformé.

Tableau 4 Résumé du triple produit atteint par rapport au triple produit requis pour différents concepts de fusion

	Achieved Triple Product $[10^{21}\text{keV}\cdot\text{s/m}^3]$	Triple Product required for reactor $[10^{21}\text{keV}\cdot\text{s/m}^3]$
Inertial fusion energy (NIF)	≈ 10	$\approx 50 - 500$
Tokamak (JET, JT-60U,...)	≈ 1	≈ 5
Stellarator (W7-X)	≈ 0.1	≈ 5
Others: FRC, Z-pinch,...	≈ 0.0001	≈ 5
Private sector, tokamak	≈ 0.01	≈ 5
Private sector, other approaches	≈ 0.0005	≈ 5 for steady-state DT ≈ 50-1000 for steady-state alternative fuels

Le triple produit et le facteur scientifique de multiplication de puissance Qsci sont des indices importants pour évaluer la performance d'une installation de fusion. Ils permettent d'évaluer la maturité technologique d'un concept de fusion vers une centrale à fusion permettant une production nette d'énergie et une exploitation commerciale rentable. L'examen du tableau 4 montre clairement que le tokamak (approche de la fusion par confinement magnétique) constitue le concept le plus prometteur, mais qu'il est encore loin de répondre aux exigences d'une centrale à fusion rentable. De plus, le Qsci le plus élevé enregistré dans un tokamak est actuellement d'env. 0,67, alors qu'un Qsci nettement supérieur à 1 est nécessaire (le Qsci se réfère uniquement à l'accroissement de la puissance de chauffe et ne tient compte ni de la puissance supplémentaire nécessaire au fonctionnement de l'installation, ni de l'efficacité de la conversion de l'énergie thermique en électricité). Le domaine de l'énergie de fusion inertielle a aussi connu des progrès, et des recherches sont en cours pour la rendre plus pertinente pour les réacteurs.

En raison de défis à la fois physiques et techniques, de grandes incertitudes entourent actuellement le calendrier de réalisation d'une première centrale à fusion. Bien que de nombreuses entreprises privées promettent que l'électricité pourra déjà être introduite dans le réseau en 2035 voire avant, ces déclarations sont à prendre avec beaucoup de prudence. Devant les multiples défis à relever, ces annonces devraient plutôt être considérées comme des velléités de motivation et placées dans le contexte de la nécessité d'attirer des investisseurs privés.

Au-delà du projet ITER, la feuille de route européenne pour la fusion prévoit un premier réacteur de démonstration DEMO d'ici 2045. Les progrès dépendent du niveau de financement et des décisions prises actuellement.

Sintesi

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L'energia nucleare in Svizzera

Nel 2023, la Svizzera ha prodotto energia nucleare dai quattro reattori in funzione (Beznau 1 e 2, Gösgen e Leibstadt), aventi una capacità totale di circa 3 gigawatt elettrici (GWe). La produzione di energia nucleare continua a essere importante nel mix elettrico del Paese: nel 2022, le quattro centrali nucleari hanno prodotto 23,1 TWh, che corrispondono a circa il 36 % dell'intera produzione di energia elettrica e rappresentano il secondo contributo, in termini di grandezza, alla produzione di corrente elettrica nazionale. L'energia idroelettrica rimane la fonte dominante e contribuisce a quasi il 53 % della produzione di energia elettrica svizzera. L'energia nucleare è particolarmente importante nei mesi invernali (nel 2022 ha rappresentato oltre il 40 % del mix di produzione nazionale nell'arco di cinque mesi). Grazie alla predominanza dell'energia idroelettrica e nucleare (pari al 89,2 % del mix elettrico del 2022), la produzione di energia elettrica svizzera presenta attualmente un saldo netto delle emissioni di CO2 quasi pari a zero (tabella 1.3 del rapporto principale).

L'energia nucleare nel mondo

Anche nei Paesi OCSE l'energia nucleare rappresenta ancora la maggiore fonte singola di energia elettrica a basse emissioni di carbonio (quote dell'elettricità nel 2022: 15,8 % energia nucleare, 12,6 % energia idroelettrica, 9,9 % eolico, 5,9 % energia solare). Tuttavia, nei Paesi OCSE e in tutto il mondo, la quota maggiore dell'elettricità prodotta si ottiene dalla combustione di combustibili fossili (quasi il 50 % nei Paesi OCSE e oltre il 60 % in tutto il mondo). Il quadro peggiora ulteriormente se si prende in considerazione il consumo mondiale di energia primaria: oltre l'80 % dell'energia continua a essere prodotta da vettori energetici fossili, mentre l'energia idroelettrica rappresenta solo il 7 %, l'energia nucleare il 4 % e l'energia solare ed eolica, sommate, il 5 %.

Nel mondo sono 32 i Paesi a utilizzare l'energia nucleare, in 13 nuovi Paesi i lavori per l'integrazione dell'energia nucleare nel proprio mix energetico si trovano in uno stadio avanzato di pianificazione e realizzazione (in tre di questi Paesi vi sono già centrali nucleari in costruzione) e altri 17 Paesi si trovano nella fase decisionale. Quattro Paesi prevedono l'abbandono del nucleare, ma solo la Germania, nel 2023, ha definitivamente cessato la produzione di corrente elettrica da questa fonte di energia. La Spagna prevede l'abbandono del nucleare entro il 2035, mentre il Belgio, nonostante una decisione analoga, ha prolungato la durata di vita di due dei suoi sette reattori nucleari a causa della recente crisi energetica. La Svizzera pianifica un esercizio a lungo termine delle proprie centrali attuali probabilmente fino a 80 anni, benché con l'attuazione della Strategia energetica 2050 si prospetti un abbandono graduale.

A marzo 2024, nel mondo erano attive 415 centrali nucleari con una potenza installata totale di 373,257 GWe. Inoltre, 57 centrali sono in fase di costruzione, per una capacità aggiuntiva di 59,22 GWe. In Europa ci sono 167 centrali nucleari in attività (148 GWe) e 9 in costruzione (10,1 GWe).

I Paesi con il numero maggiore di centrali in attività sono gli Stati Uniti, la Francia, la Cina e la Russia. A marzo 2024, la Cina era il Paese con la crescita maggiore nel settore dell'energia nucleare con 27 centrali attualmente in fase di costruzione, seguita da India (7 impianti in costruzione), Turchia (4), Egitto (4), Corea del Sud (2) e Russia (4). Nel 2023 la Cina ha già raggiunto i 53,3 GWe di capacità nucleare installata (con quasi 400 TWh prodotti nel 2022), a cui si aggiungono altri 30,9 GWe da impianti in costruzione nonché importanti piani di potenziamento (con l'obiettivo di arrivare fino a 150 GWe di capacità nucleare installata entro il 2030). In Europa, i seguenti Paesi stanno costruendo o hanno pianificato nuove centrali in un prossimo futuro: Francia (1 impianto EPR in costruzione, 6 impianti EPR-2 autorizzati, altri pianificati), Regno Unito (2 impianti EPR in costruzione, 2 ulteriori impianti EPR pianificati), Slovacchia (1 impianto in costruzione, altri proposti), Bulgaria (2 impianti AP1000 pianificati), Repubblica Ceca (4 impianti pianificati, altre 3 sedi individuate per diversi SMR), Paesi Bassi (2 impianti pianificati), Romania (2 impianti CANDU pianificati, 6 moduli SMR NuScale proposti), Ungheria (2 impianti VVER autorizzati), Slovenia (1 impianto proposto), Svezia (2 impianti pianificati entro il 2035, 10 ulteriori impianti dopo il 2035), Estonia e Polonia (3 AP1000 autorizzati, 2 impianti APR1400 pianificati, 24 impianti SMR BWR-300 pianificati). Inoltre, nel 2023, in Bielorussia e Finlandia sono state collegate alla rete due nuove centrali nucleari. Tra gli sviluppi più recenti si segnala la costruzione del primo deposito al mondo in strati geologici profondi per i rifiuti radioattivi in Finlandia, la cui realizzazione dovrebbe terminare a metà degli anni 2020. Anche in Svezia è stata concessa l'autorizzazione edilizia per un deposito in strati geologici profondi, la cui costruzione dovrebbe iniziare nei prossimi anni, mentre in Francia le autorità di vigilanza stanno attualmente valutando una domanda per la costruzione di un deposito analogo. La decisione sull'ubicazione definitiva di tale deposito è attesa per il 2025, con la messa in funzione intorno al 2040. In Canada, la scelta per l'ubicazione del deposito delle scorie verrà comunicata nel 2024. In Svizzera, l'autorizzazione per un deposito in strati geologici profondi è attesa intorno al 2030 (previa valutazione positiva delle autorità di vigilanza e riuscita di un eventuale referendum facoltativo); la messa in funzione è prevista per il 2050.

Negli ultimi anni molti Paesi, a causa in particolare delle modifiche all'assetto geopolitico causate dalla guerra in Ucraina, hanno rivisto i propri piani relativi all'energia nucleare. I risultati sono stati i seguenti:

● la nascita dell'Alleanza europea per il nucleare nel 2023, con cui 16 Paesi (Francia, Belgio, Bulgaria, Croazia, Repubblica Ceca, Finlandia, Ungheria, Paesi Bassi, Polonia, Romania, Slovenia, Slovacchia, Estonia, Svezia, Italia, Regno Unito) pianificano la costruzione di un'industria nucleare europea e si impegnano a raggiungere entro il 2050 una quota di 150 GWe di energia nucleare nel mix energetico europeo (un aumento del 50 % in confronto alla quota odierna);

● la nascita dell'Alleanza per i piccoli reattori modulari della Commissione europea nel 2024, con l'obiettivo di preservare il ruolo di guida tecnologica e industriale dell'Europa nel settore dell'energia nucleare;

● la dichiarazione sull'energia nucleare della Conferenza sul clima delle Nazioni Unite (COP28) del dicembre 2023, sottoscritta da 22 Paesi, in cui si mira a triplicare l'energia nucleare entro il 2050 per raggiungere il nuovo obiettivo delle emissioni nette pari a zero, riconoscendo il ruolo chiave dell'energia nucleare nel raggiungimento dell'azzeramento globale delle emissioni nette di gas serra/della neutralità carbonica entro la metà del secolo. Tra i Paesi in questione si annoverano Stati Uniti, Bulgaria, Canada, Repubblica Ceca, Finlandia, Francia, Ghana, Ungheria, Giappone, Corea del Sud, Moldavia, Mongolia, Marocco, Paesi Bassi, Polonia, Romania, Slovacchia, Slovenia, Svezia, Ucraina, Emirati Arabi Uniti e il Regno Unito;

● l'introduzione di un piano di investimenti negli Stati Uniti per la promozione dello sviluppo di SMR e microreattori e il relativo impiego negli Stati Uniti e all'estero. La legge sulla riduzione dell'inflazione, firmata nel 2022, ha come obiettivo la promozione delle centrali nucleari nuove e di quelle esistenti tramite aiuti agli investimenti e incentivi fiscali sia per le grandi centrali nucleari già presenti sia per i nuovi reattori più avanzati per la produzione di combustibile all’uranio e all’idrogeno. La durata di diverse centrali nucleari è stata prolungata (p. es. Diablo Canyon in California, la cui disattivazione era pianificata per il 2022. Per sei reattori la durata è stata prolungata a 80 anni; molti altri sono in attesa di una decisione dalle autorità di vigilanza). È interessante notare che lo Stato del Michigan riprenderà l’esercizio della centrale nucleare di Palisades, disattivata nel 2022. Nelle centrali nucleari esistenti sono stati avviati con successo alcuni progetti pilota per sfruttare l’energia nucleare per la produzione di idrogeno.

Dato il crescente riconoscimento dell’importanza di una produzione affidabile per il carico di base, diverse aziende statunitensi come Amazon, Google, Microsoft e aziende attive in settori energivori come Nucor (produzione siderurgica) e Dow Chemicals hanno sottoscritto degli accordi con fornitori o imprese di servizi per l’approvvigionamento futuro di energia nucleare.

Stato dei reattori ad acqua leggera (LWR) della generazione III/III+ e tempi di costruzione

I reattori della generazione III/III+ rappresentano una nuova generazione di centrali nucleari, che si basano sulla stessa tecnologia dei reattori ad acqua leggera (LWR) degli impianti attualmente in uso; tuttavia, si differenziano per le caratteristiche di sicurezza sensibilmente migliorate e per le caratteristiche strutturali, che prendono in considerazione gli insegnamenti tratti dai tre grandi incidenti nucleari della storia. A dicembre 2023 erano in uso 38 grandi unità LWR della generazione III/III+; dei 60 reattori attualmente in fase di costruzione, 51 sono grandi LWR della generazione III/III+. Ulteriori unità sono già state commissionate oppure sono stati aperti i relativi bandi di concorso (p. es. tre unità in Polonia, due unità in Gran Bretagna, una nella Repubblica Ceca ecc.) e diverse altre sono in fase di pianificazione.

I tempi di costruzione medi per i 38 reattori in uso della generazione III/III+ sono pari a 7,7 anni, la mediana è 8 anni (v. figura 2.5 e 2.6 nel rapporto principale). In confronto, il tempo di costruzione medio per i 413 reattori della generazione II e III in tutto il mondo ammonta in totale a 7,5 anni; il valore mediano è 6,3 anni. Queste cifre smentiscono l’opinione comune secondo cui i tempi di costruzione delle nuove centrali nucleari sono drasticamente aumentati. Dimostrano piuttosto un aumento moderato dei tempi di costruzione, con alcune eccezioni degne di nota, soprattutto tra i primi progetti in Europa e negli Stati Uniti, i cui tempi di costruzione sono aumentati in modo sproporzionato. D’altra parte è stato dimostrato più volte che è tecnicamente possible realizzare un sistema «chiavi in mano» con un tempo di costruzione inferiore a sei anni, purché venga realizzata una catena di fornitura funzionante per i componenti chiave.

In particolare, gli impianti ABWR (GE Hitachi/Toshiba) in Giappone si distinguono per tempi di costruzione brevi, dato che sono stati realizzati tutti in meno di quattro anni. Il reattore AP-1000 di Westinghouse negli Stati Uniti (Vogtle 3) e i due impianti EPR situati rispettivamente a Olkiluoto (Finlandia) e a Flamanville (Francia) si trovano all’altra estremità dello spettro, con tempi di costruzione da 10 a 16,5 anni.

Questi progetti hanno dovuto affrontare sin dall’inizio sfide notevoli, dato che dopo una pausa decennale nella realizzazione di nuovi impianti hanno richiesto un lavoro pionieristico per la costruzione della prima grande centrale di questo tipo in Europa e negli Stati Uniti ed è stato necessario ricostruire le capacità produttive e le catene di fornitura. Inoltre, le autorità di vigilanza sia in Finlandia che negli USA pretesero significative modifiche progettuali fino in una fase avanzata della costruzione delle centrali. I due impianti EPR in costruzione a Hinkley Point, nel Regno Unito, hanno subito ritardi significativi, ma non si è giunti alla stessa situazione della Finlandia o di Flamanville. Tali ritardi sono stati in parte dovuti a elementi mancanti nella catena di fornitura britannica, alla necessità di formazione della forza lavoro locale (i rallentamenti hanno riguardato soprattutto l'edilizia) e a un numero elevato di modifiche alla progettazione (oltre 7000) richieste dalle autorità di vigilanza britanniche. Nonostante tali ripercussioni, il Governo britannico ha confermato la realizzazione di due ulteriori impianti EPR nella sede di Sizewell.

Il grado di completezza del progetto dettagliato all'inizio dei lavori e la creazione di una catena di fornitura funzionante e di capacità produttive sono pertanto fattori importanti nella determinazione dei tempi di costruzione, così come le esperienze raccolte con diversi impianti consecutivi e l'affidabilità del quadro normativo e finanziario. La Cina è stata in grado di ridurre costantemente i tempi di costruzione degli impianti: gli ultimi nove impianti (con un progetto standardizzato HPR1000 e ACPR-1000) sono stati realizzati in 5–7 anni. È degno di nota anche l'ultimo esempio preveniente dagli Emirati Arabi Uniti, dove l'impresa sudcoreana KHNP ha realizzato una capacità nucleare installata di 5,2 GWe (4 unità APR1400) in nove anni e per il costo totale di soli 24 miliardi di dollari.

Efficienza economica degli LWR della generazione III/III+

Le stime basate su fonti scientifiche affidabili (PSI 2019) attestano il prezzo di costo dell'elettricità (levelized cost of electricity, LCOE) delle nuove centrali nucleari tra 7 e 12 ct./kWh. Se il tempo di costruzione rimane al di sotto degli 8 anni (la mediana delle 38 strutture della generazione III/III+ è di 7,7 anni), è possibile raggiungere un LCOE di 7 centesimi, in linea con i precedenti studi PSI del 2019. Il valore è all'interno dell'intervallo degli LCOE attuali e futuri per le fonti di energia rinnovabili in Svizzera e le attuali centrali idroelettriche, e fornirebbe la corrente per il carico di base. I risultati PSI del 2019 corrispondono ad altri studi verificati da esperti, pubblicati nella letteratura liberamente accessibile. Gli LCOE attuali per il funzionamento delle centrali nucleari svizzere esistenti si attestano a 4,0–5,5 ct./kWh (la cifra comprende già la totalità dei costi per lo smaltimento delle scorie). Un esercizio a lungo termine di questi impianti fino a 60 anni incrementerebbe il prezzo di costo dell'elettricità di 1–2 centesimi. Occorre considerare che il concetto del prezzo di costo dell'elettricità (LCOE) era stato originariamente introdotto per confrontare fonti energetiche regolabili, mentre in un sistema energetico sempre più complesso con una quota sempre maggiore di energie rinnovabili fluttuanti ha un valore limitato. Si riconosce sempre di più la necessità di prendere in considerazione, in questi casi, non solo il prezzo di costo dell'elettricità, bensì i costi complessivi del sistema (costi di compensazione, costi per l'ampliamento della rete, costi di backup ecc.). Recentemente, l'OCSE ha pubblicato il tentativo di uno studio simile per il sistema energetico svizzero, tuttavia non è mai stato realizzato un modello ampio e completo del sistema energetico svizzero che prenda in considerazione anche l'impiego dell'energia nucleare.

I primi impianti EPR da 1600 MWe a Olkiluoto e Flamanville sono stati sensibilmente più dispendiosi rispetto agli impianti APR1400 sudcoreani da 1400 MWe, realizzati negli Emirati Arabi Uniti. Mentre gli APR1400 sono costati 6 miliardi di dollari a unità, i due EPR a Olkiluoto e Flamanville sono costati singolarmente circa 11–13,2 miliardi di euro. Questi elevati costi del capitale devono però essere considerati in rapporto all'energia prodotta. Una singola unità EPR produrrebbe oltre 12 TWh/anno. A titolo di confronto, per produrre la stessa potenza annuale di un EPR con un impianto solare alpino sarebbe necessario l'equivalente di oltre 3800 «Alpin Solar» (impianto della diga del Muttsee) al costo di oltre 30 miliardi di franchi (senza prendere in considerazione i costi aggiuntivi per il backup, lo stoccaggio e l'ampliamento della rete). Se invece si prendesse come riferimento l'impianto Gondosolar, sarebbero necessari oltre 780 impianti analoghi al costo di circa 29 miliardi di franchi.

Gli elevati costi di capitale delle grandi centrali nucleari rappresentano una delle principali sfide economiche dell'energia nucleare, poiché riducono il numero dei potenziali investitori privati.

Tali sfide vengono parzialmente mitigate dagli SMR e potrebbero essere eliminate in massima parte dai microreattori, i cui costi complessivi di capitale sono equiparabili a quelli di un impianto solare alpino, ma con una produzione di energia molto più elevata e regolare.

Per superare gli elevati costi di capitale delle grandi centrali nucleari, in passato sono stati applicati diversi modelli che coinvolgevano i governi come fornitori di capitale proprio, come enti finanziatori o tramite misure politiche quali garanzie sui crediti o Contracts for Differences (CfD, remunerazioni minime). Contrariamente alla percezione pubblica generale, l'energia nucleare è la meno sovvenzionata tra le fonti energetiche, come viene spiegato a fondo nella sezione 1.4.2 del presente rapporto. Nell'UE, le sovvenzioni per l'energia nucleare nel periodo 2015–2022 hanno raggiunto nel 2021 un valore massimo di 7,6 miliardi di euro, rispetto agli 88 miliardi per le energie rinnovabili e ai 123 miliardi per i combustibili fossili. Negli Stati Uniti, le sovvenzioni massime per l'energia nucleare tra il 2016 e il 2022 sono state inferiori ai 600 milioni di dollari, rispetto agli oltre 17 miliardi per le energie rinnovabili e ai 2,5 miliardi per il carbone e ai 3 miliardi per il gas.

I tassi di interesse per il capitale vengono influenzati in modo decisivo dal sistema di finanziamento (p. es. garanzia di credito statale) e dal quadro normativo. Un metodo efficace per abbassare i costi consiste nel realizzare diverse unità nella stessa sede. Per esempio, con il progetto Barakah negli Emirati Arabi Uniti è stato possibile ridurre del 40 % i costi complessivi per la costruzione delle unità 1 e 4. Ulteriori fattori che influiscono positivamente sul successo di un nuovo impianto sono la realizzazione di componenti rilevanti del progetto prima dell'inizio dei lavori, la presenza di una catena di fornitura ben consolidata, l'accesso a una forza lavoro qualificata e un quadro normativo stabile.

Si prevede che il prezzo di costo dell'elettricità per gli SMR raggiungerà un livello simile a quello delle grandi centrali nucleari, compensando l'economia di scala con gli effetti dell'economia di produzione (produzione di massa), anche se per le prime unità si attendono costi più elevati. Per il primo progetto SMR NuScale, che avrebbe dovuto essere realizzato nei prossimi anni in Utah, sono stati previsti costi di circa 5,8 ct./kWh (stima 2020). Sulla base dell'aumento dei tassi di interesse del 150 % e di una crescita significativa dei costi del materiale (p. es. 40 % di aumento dei costi per l'acciaio) negli ultimi 1,5 anni, si è previsto un aumento dei costi a 8,9 ct./kWh fino al 2023. Ciò non avrebbe potuto competere con la produzione di energia elettrica dal gas e dal carbone, due materie prime a disposizione dei fornitori locali nello Utah, pertanto il progetto NuScale è stato sospeso a favore di una centrale a gas.

Sicurezza della generazione III/III+

Nelle centrali della generazione III/III+ sono state apportate modifiche significative ai requisiti di sicurezza. I sistemi di gestione degli incidenti gravi ora sono parte integrante della progettazione ed è stata rafforzata l'indipendenza dei vari livelli di sicurezza sovrapposti (defense in depth). L'implementazione della nuova filosofia di sicurezza ha portato a un'intera serie di nuovi sistemi di sicurezza passiva (che non dipendono per il funzionamento da energia esterna come generatori diesel o da interventi del gestore) e a periodi di grazia (grace period) prolungati, con l'obiettivo di escludere a livello pratico il verificarsi di gravi conseguenze in caso di incidente con fusione del nocciolo e conseguente fallimento del contenimento (containment), che potrebbe portare a una liberazione precoce o di grandi dimensioni delle sostanze radioattive. In particolare, i nuovi approcci alla sicurezza hanno determinato quanto segue:

● un prolungamento del periodo di grazia (in cui non è necessario alcun intervento umano anche nelle circostanze più avverse causate da un incidente) da 30 minuti, tipico degli impianti della generazione II, ad almeno 3 giorni, più spesso a una settimana;

● frequenze di danneggiamento del nocciolo inferiori a 10^{-6}/anno (ossia, in termini probabilistici, inferiori a una volta in un milione di anni);

● una probabilità di fallimento del contenimento dopo danni al nocciolo con liberazione di sostanze radioattive nell'ambiente inferiore a 10^{-7}/anno (ossia meno di una volta ogni dieci milioni di anni).

La probabilità di danni al nocciolo, e di una successiva liberazione di quantità significative di radioattività, è quindi inferiore da uno a due ordini di grandezza rispetto alle attuali centrali della generazione II sottoposte a un adeguato retrofitting, che hanno comunque già raggiunto un'eccellente livello di sicurezza in seguito alle misure di retrofitting e agli stress test post Fukushima.

Stato dei piccoli reattori modulari (SMR)

Gli SMR sono reattori moderni con una potenza nominale fino a 300 MW(e) per singola unità. Sono progettati per la costruzione in fabbrica e il trasporto nel luogo di impiego e di norma vengono installati sotto terra. L'Agenzia per l'energia nucleare dell'OCSE (AEN) parte dal presupposto che gli SMR costituiranno fino al 9 % della capacità di tutte le nuove centrali nucleari entro il 2035. Attualmente in Russia e in Cina sono attivi 10 SMR e molti altri si trovano in fase di realizzazione o in attesa di approvazione (Stati Uniti, Canada, Francia) (v tabelle 1 e 2).

Tra gli SMR con raffreddamento ad acqua, NUWARD (EDF), Rolls-Royce (Gran Bretagna), BWXR-300 (USA), Holtec-180 (USA), AP300 (USA) e VOYGR (NuScale) sono le tipologie costruttive con il grado di sviluppo più avanzato per l'impiego in Europa entro il 2030. L'ultimo tipo è già stato approvato negli Stati Uniti, mentre gli altri progetti si trovano in fasi diverse della certificazione preliminare negli Stati Uniti, in Canada e in alcuni Paesi europei. Alcuni fornitori di SMR hanno ricevuto delle ordinazioni (p. es. BWRX-300 in Canada). Nel 2022, NUWARD ha ricevuto 500 milioni di euro dal Governo francese e l'inizio dei lavori per il primo reattore è previsto per il 2030.

L'interesse generale per gli SMR è nato originariamente dalla necessità di approvvigionare di energia elettrica regioni remote o aree non raggiunte dalla rete, al momento alimentate a gas, petrolio o diesel, e per sostituire le centrali a combustibili fossili obsolete nell'ordine di grandezza di 300–400 MWe. Gli SMR vengono considerati da alcuni un'opzione conveniente nei Paesi con reti elettriche di piccole dimensioni, in cui l'impiego di grandi centrali nucleari non sarebbe possibile, oppure in assenza della capacità o della volontà da parte di gestori e investitori di impiegare grandi capitali. Inoltre, gli SMR offrono la possibilità di alimentare sedi industriali energivore (p. es. settore del calcestruzzo e dell'acciaio) o di sfruttare l'energia nucleare per applicazioni diverse dalla produzione di energia elettrica, ad. es. teleriscaldamento, desalinizzazione dell'acqua di mare o persino produzione di idrogeno.

Grazie alle dimensioni ridotte, la maggior parte degli SMR dispone di caratteristiche di sicurezza migliorate, basate completamente sulla sicurezza passiva. Pertanto, la Nuclear Regulatory Commission negli Stati Uniti ha approvato una nuova normativa sul dimensionamento delle zone di pianificazione d'emergenza (emergency planning zone, EPZ), orientata a possibili scenari di incidente e alle relative conseguenze. L'SMR NuScale dispone quindi di una licenza con una EPZ limitata alla vicinanze dell'area dell'impianto (in altre parole, non è necessaria alcuna zona di evacuazione). Si prevede che altri SMR negli Stati Uniti riceveranno una regolamentazione analoga.

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Tabella 1 SMR LWR in stadio di sviluppo avanzato

I vantaggi principali degli SMR sono i costi di capitale iniziali sensibilmente più bassi grazie alle dimensioni ridotte dell'impianto, i tempi di costruzione più brevi grazie alla produzione in fabbrica, la maggiore flessibilità relativa alla regolazione del carico (che rende gli SMR più facili da integrare con le fonti di energia rinnovabili intermittenti) e il miglioramento dei piani di misure di sicurezza. L'economia di scala delle grandi centrali nucleari verrà presumibilmente sostituita dagli effetti dell'economia di produzione (moduli realizzati in fabbrica) e da cantieri semplificati, per cui i costi al kWh saranno nello stesso intervallo delle grandi centrali nucleari. L'AIEA e le autorità di vigilanza sull'energia nucleare statunitensi ed europee stanno armonizzando le concessioni delle licenze per gli SMR, al fine di creare un quadro stabile e trasparente ed evitare modifiche imprevedibili alle normative nazionali per le licenze.

La maggior parte degli SMR a breve termine appartiene, come i grandi reattori, alla categoria dei reattori ad acqua leggera della generazione III/III+, con una prospettiva plausibile per l'utilizzo commerciale dei primi impianti dimostrativi nei Paesi occidentali entro il 2030 o persino prima (v. tabella 1 in alto). Gli SMR più avanzati, con altri refrigeranti rispetto all'acqua (p. es. metallo liquido, elio, sali fusi), appartengono alle centrali nucleari della generazione IV, sviluppate da una varietà di imprese start-up. Tuttavia, l'orizzonte temporale per l'utilizzo commerciale di alcune di queste versioni (p. es. a sali fusi) è rimasto indietro di molti anni rispetto allo sviluppo degli LWR. Mentre gli SMR non raffreddati ad acqua sono già in funzione in Cina e in Russia, si presume che il primo nei Paesi occidentali sarà l'SMR raffreddato a sodio di Terrapower, che dovrebbe essere costruito in Wyoming (USA). La domanda di autorizzazione edilizia per l'SMR di Terrapower è stata presentata a marzo 2024 e a maggio dello stesso anno è stata accettata per la verifica da parte dell'autorità di vigilanza statunitense.

Tabella 2 SMR non LWR in uno stadio di sviluppo avanzato

Stato della tecnologia dei microreattori

Negli ultimi sette anni è emerso un trend interessante nel settore dei cosiddetti microreattori, che dovrebbero produrre energia elettrica intorno ai 10 MWe (alcuni di essi vengono sviluppati negli Stati Uniti, v. tabella 4.1 nel rapporto principale). Si tratta di reattori completamente realizzati in fabbrica: possono essere facilmente trasportati dallo stabilimento di produzione al luogo di impiego in un container ISO (per nave, su gomma o su rotaia), quindi non è necessario alcun cantiere, e funzionano per 5–10 anni o anche più a lungo senza rinnovamento del combustibile. Possono funzionare in modo indipendente, come parte della rete elettrica o all'interno di una microrete. Potranno essere impiegati in aree remote (p. es. in siti minerari) o fornire corrente e calore a stabilimenti industriali enegivori (p. es. desalinizzazione dell'acqua di mare, produzione dell'idrogeno ecc.). Tuttavia, sono interessanti anche per i settori industriali che necessitano di un certo grado di indipendenza dalla rete elettrica per la sicurezza dell'approvvigionamento. Il raffreddamento avviene tramite gas (elio), metallo liquido, sali fusi o tubi di calore (per il sodio) (heat pipes).

Grazie alle dimensioni ridotte e alla semplicità della progettazione, il loro sviluppo avanza a grande velocità. La prima unità dimostrativa (struttura con heat pipe) è stata progettata, costruita e testata nel giro di 3 anni dalla NASA e dal Los Alamos National Laboratory, con costi inferiori ai 20 milioni di dollari. Una seconda unità (raffreddata a metallo liquido) è in fase di costruzione in Idaho (USA) e dovrebbe essere messa in funzione all'inizio del 2025. Un'unità raffreddata a fluoruro ha ricevuto l'autorizzazione edilizia nel dicembre 2023 e la messa in funzione è prevista per il 2026. Tre ulteriori progetti si trovano in diverse fasi di acquisizione della licenza negli Stati Uniti e in Canada.

Poiché i microreattori vengono completamente costruiti in fabbrica, e quindi beneficeranno presumibilmente di una serie di effetti dovuti alla produzione in serie, come in altri settori ci si attende una curva di apprendimento positiva. Altri vantaggi attesi sono i costi di capitale estremamente bassi (nell'ordine di circa 100 milioni di dollari o meno), che potrebbero renderli accessibili a una cerchia più ampia di investitori, il basso prezzo di costo dell'elettricità (LCOE) in confronto alle alternative di backup in aree remote o in industrie di grandi dimensioni grazie alla realizzazione completa in fabbrica, la superficie di base molto limitata (circa 15 m^2 per l'impianto e meno di 2000 m^2 per l'area circostante), la possibilità di prevedere i tempi di costruzione e un rischio da radiazioni ridotto. Grazie alla quantità necessaria di combustibili molto limitata e alla struttura semplificata, i microreattori sono simili ai reattori di ricerca; pertanto, si prevedono tempi di autorizzazione più veloci rispetto agli SMR o alle centrali nucleari di grandi dimensioni. L'utilizzo nei microreattori del combustibile TRISO (TRi-Structural ISOtropic Particle Fuel) a maggiore arricchimento richiede lo sviluppo di corrispondenti capacità di produzione del combustibile, operazione attualmente in corso negli USA e in Francia. Questo aspetto non rappresenta però un ostacolo, dato che la tecnologia è già nota (lo stesso combustibile viene utilizzato negli SMR HTR-PM in funzione in Cina).

Stato della generazione IV e dei reattori non ad acqua leggera

I reattori non raffreddati ad acqua (p. es. raffreddati a gas, piombo, sodio, sali fusi) vengono sviluppati con l'obiettivo di incrementare il rendimento tramite l'aumento dell'efficienza termodinamica e/o tramite un migliore utilizzo del combustibile e un'ulteriore riduzione della quantità di scorie altamente radioattive (nei reattori a spettro veloce), in modo da chiudere il ciclo del combustibile nucleare[4]. Esistono diversi progetti, tra cui vale la pena citare i più promettenti:

• reattori termici raffreddati a gas, che impiegano l'elio come refrigerante. Presentano un'elevata efficienza termodinamica nella trasformazione in elettricità della potenza termica generata nel reattore e, poiché funzionano a temperature molto più elevate degli LWR, si adattano anche alla fornitura di calore a processi industriali energivori con temperature elevate (v. figura 1 per una rappresentazione delle temperature necessarie per diversi processi industriali e il corrispondente progetto di reattore che potrebbe fornire tali temperature). Due reattori termici raffreddati a gas (progetto HTR-PM) sono in funzione dal 2021 in Cina. Lo Xe-100 (X-energy, USA) ha già concluso con successo la certificazione preliminare in Canada (in altre parole, l'autorità di vigilanza non ha rilevato problemi che potessero bloccare la procedura per la concessione della licenza);

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[4] Oltre il 90 % del combustibile nucleare usato è riutilizzabile. In un ciclo chiuso, il combustibile nucleare usato viene ritrattato per estrarre il materiale riutilizzabile (principalmente l'uranio) che poi viene impiegato per la produzione di nuovi elementi di combustibile. Un ciclo del combustibile chiuso può essere raggiunto p. es. con una combinazione di LWR e reattori veloci di generazione IV. Nei reattori veloci vengono impiegati neutroni ad alta energia («veloci») per la fissione del combustibile nucleare, mentre negli LWR per la fissione vengono utilizzati principalmente neutroni termici (a bassa energia). Un ciclo del combustibile chiuso consente un miglioramento della sostenibilità grazie all'aumento della produzione di energia per unità di massa del combustibile e alla diminuzione della quantità di scorie altamente radioattive per unità di energia.

• reattori veloci con raffreddamento a metallo liquido (sodio o piombo/piombo-bismuto). Funzionano ad alte temperature, comprese tra quelle degli LWR e quelle dei reattori raffreddati a gas, e a una pressione vicina a quella ambientale (pressione atmosferica). Sono disponibili considerevoli esperienze riguardo al loro esercizio (Francia, Giappone, Russia ecc.) e oggi sono in funzione diversi impianti (v. tabella 3). Terrapower (impresa statunitense) offrirà sul mercato un SMR veloce raffreddato al sodio; il primo impianto dovrebbe essere realizzato in Wyoming (USA) prima del 2030;

• reattori a sali fusi (MSR), in cui i sali fusi vengono utilizzati come refrigerante, combustibile e/o moderatore. Questi reattori funzionano ad alte temperature e vi sono versioni sia termiche che veloci. La sfida più grande posta all'esercizio di tali reattori è la natura fortemente corrosiva dei sali. Nel dicembre 2023, un progetto di reattore termico di KAIROS (USA), che utilizza sali fusi come refrigerante (con combustibile HALEU TRISO e la grafite come moderatore) ha ottenuto l'autorizzazione edile per un primo impianto dimostrativo in Tennessee. Un MSR integrale (Terrestrial Energy) si trova attualmente nella fase preliminare di concessione della licenza in USA e in Canada. Nel giugno 2023, in Cina, è stata concessa l'autorizzazione di esercizio a un MSR sperimentale, che utilizza come combustibile un sale fuso a base di torio. La costruzione di questo reattore TMSR-LF1 è iniziata nel settembre 2018 e avrebbe dovuto concludersi nel 2024. Tuttavia, secondo i rapporti, il reattore era ultimato già nell'agosto 2021, dopo un'accelerazione dei lavori.

I reattori di generazione IV rilevanti per il mercato occidentale sono:

• KAIROS, Terrestrial, X-energy (tutti reattori termici, v. tabella 2)

• Terrapower, IMSR (Moltex), ARC-100 (tutti reattori veloci, v. tabella 2)

• le versioni di microreattori attualmente in fase di concessione della licenza negli USA e in Canada (tabella 4.1 nel rapporto principale).

Tabella 3 Reattori veloci attualmente in esercizio (tutti del tipo SFR)

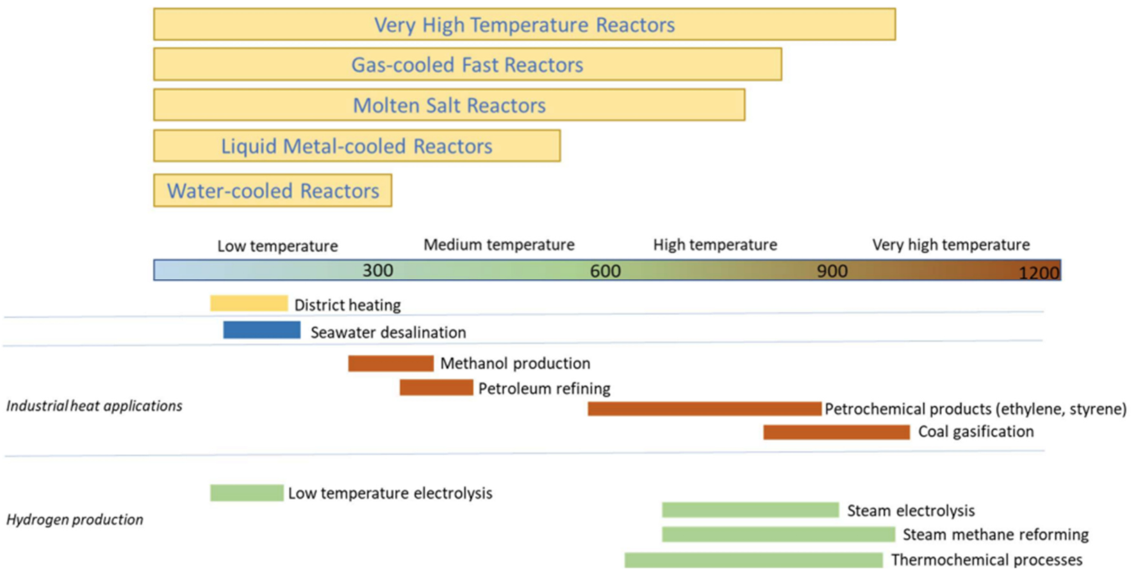

Figura 1: Temperatura iniziale delle tecnologie nucleari e corrispondenti applicazioni non elettriche

Disponibilità dell'uranio e cicli alternativi del combustibile

Le riserve naturali di uranio sono una risorsa con un’ampia distribuzione (v. figura 2) e saranno sufficienti per i prossimi secoli. Come altre risorse, dipendono dal prezzo di mercato.

Il combustibile con arricchimento debole (LEU), che viene utilizzato nei reattori ad acqua leggera (p. es. negli impianti svizzeri), viene prodotto in diversi impianti di arricchimento ed è presente una sufficiente varietà e diversità delle fonti da garantire il suo approvvigionamento. Non si prevedono rischi a lungo termine per la sicurezza di approvvigionamento del combustibile nucleare in Svizzera.

Considerando la seconda metà di questo secolo, si può presupporre che un fabbisogno più elevato di energia nucleare porti a un intensificarsi delle attività di esplorazione e quindi a maggiori riserve di uranio. Inoltre, in questo lasso di tempo, la tecnologia dei reattori con un ciclo di combustibile chiuso si svilupperà a tal punto (p. es. reattori veloci) da poter utilizzare altri combustibili, con un potenziale energetico molto maggiore rispetto all’U-235, prolungando la disponibilità del combustibile nucleare da centinaia a molte migliaia di anni. In tutto il mondo, i reattori veloci sono già in funzione o in costruzione (v. tabella 2 e 3) oppure si trovano in una fase di pianificazione avanzata, come il reattore di Terrapower, il cui primo impianto dovrebbe essere realizzato nel corso di questo decennio in Wyoming. Un importante combustibile per i reattori moderni non raffreddati ad acqua, comprese le versioni di Terrapower, è l’uranio arricchito (HALEU), caratterizzato da un arricchimento tra il 5 e il 20 %. La produzione di combustibile HALEU verrà quindi aumentata. Paradossalmente, l’insicurezza della catena di approvvigionamento dei combustibili nucleari, causata dalla guerra in Ucraina, finirà per rafforzare e ampliare le capacità di approvvigionamento di combustibili e la resilienza di tale approvvigionamento in Occidente nei prossimi 5–10 anni. Al momento, negli Stati Uniti e in Francia, vengono costruite nuove catene di fornitura per il combustibile HALEU, mentre in Europa e negli USA è stata aumentata la capacità degli impianti esistenti per l’uranio a basso arricchimento (LEU), utilizzato di norma nei tradizionali reattori raffreddati ad acqua. Inoltre, alla fine del 2023, gli Stati Uniti hanno inaugurato tre nuove miniere di uranio, per aumentare la propria indipendenza nell’approvvigionamento di questo materiale.

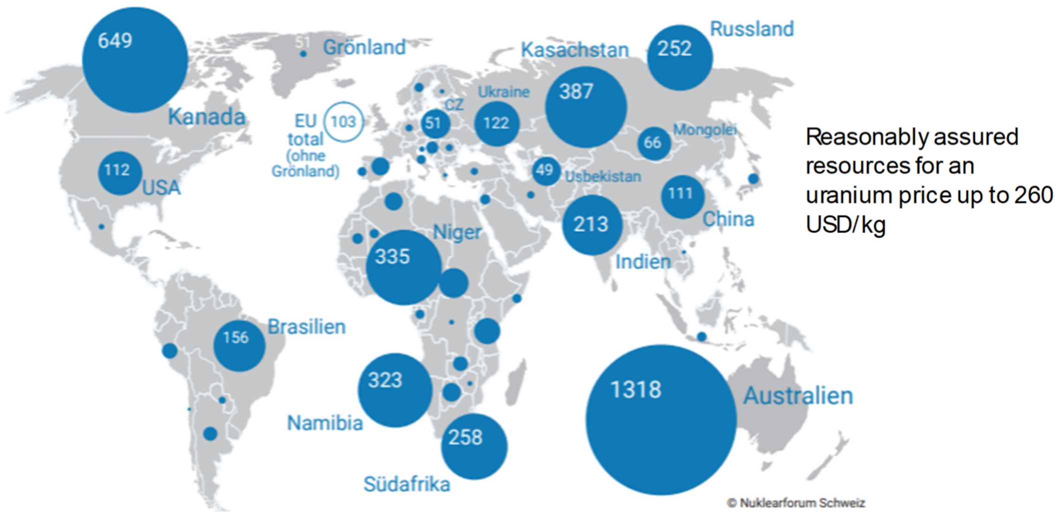

Figura 2 Distribuzione mondiale delle riserve di uranio al 1.1.2021 (in migliaia di tonnellate) a un prezzo di 260 USD/kg. Dati dal Red Book 2022 dell’AIEA.

I cicli di combustibile più avanzati, che mirano a un utilizzo più efficiente del combustibile e alla riduzione delle scorie altamente radioattive, si basano sul ritrattamento dei combustibili e comprendono:

- la produzione di nuovi elementi di combustibile (i cosiddetti MOX) per gli LWR con l'utilizzo del plutonio (Pu) e dell'uranio ritrattato (U), recuperato dal combustibile usato degli LWR. Di norma, da quattro elementi di combustibile usati (Fuel Assemblies, FA) è possibile ricavare un nuovo elemento di combustibile. È una tecnologia affermata: il combustibile MOX era stato introdotto con successo negli impianti svizzeri ed è stato utilizzato per molti anni. Con la legge federale sull'energia nucleare del 2003, il ritrattamento è stato vietato tramite una moratoria. In altri Paesi (p. es. la Francia), il ritrattamento viene eseguito di routine;
- reattori surrigeneratori veloci con U-Pu (ciclo del combustibile Breed-and-Burn). In questo caso, durante il funzionamento dell'impianto, nel nucleo del reattore viene prodotto combustibile aggiuntivo, in modo da ottenere più energia dalla stessa quantità di combustibile. È necessario il combustibile HALEU;
- utilizzo di attinidi minori (MA) in speciali «bruciatori» MA (p. es. versioni di reattori MYRRHA e Transmutex) per ridurre la radiotossicità del combustibile esausto presente. Ciò richiede il ritrattamento degli attinidi minori, che è già stato testato in laboratorio ma che non esiste ancora su scala industriale. Con i bruciatori speciali è possibile ottenere un tasso di trasmutazione per unità di energia più elevato rispetto ai reattori veloci tradizionali; tuttavia, il funzionamento di speciali reattori di trasmutazione aumenta la complessità tecnologica e quindi i rischi operativi. Inoltre, la maggior parte della quantità lavorata sarebbe comunque uranio ritrattato (RepU), la cui ulteriore manipolazione non viene presa in considerazione nel trasmutatore. Se si implementasse comunque il ritrattamento degli MA, un trasmutatore speciale potrebbe rappresentare una componente rilevante del ciclo del combustibile e contribuire alla riduzione del flusso di rifiuti;
- ciclo torio (Th)-uranio: Il Th-232 è un isotopo fertile analogo all'U-238, presente naturalmente nella crosta terrestre con una frequenza tre volte maggiore rispetto all'uranio. Il torio può essere utilizzato per surrigenerare l'U-233, così come l'U-238 viene utilizzato per la surrigenerazione del Pu-239. L'utilizzo del torio richiede un ritrattamento e alcune modifiche alla tecnologia odierna. Nel caso di un'eventuale implementazione potrebbe fornire combustibile aggiuntivo per centinaia o migliaia di anni, tenendo conto delle sfide attuali. I reattori di prova al torio sono stati messi in funzione in passato con successo. Un reattore dimostrativo al torio con sali fusi è stato messo in funzione in Cina nel 2023 e dovrebbe essere seguito da un reattore corrispondente da 373 MWt entro il 2030. Un surrigeneratore veloce al torio da 40 MWth è in funzione in India, dato che il Paese dispone di grandi riserve di questo materiale.

Gli impianti commerciali per il ritrattamento degli elementi di combustibile sono presenti in Francia, Russia, India, Giappone (quest'ultimo è in costruzione e deve essere terminato nel 2024) e in Cina (due in costruzione, il primo inizierà a funzionare nel 2025). Nel 2022, in Gran Bretagna, un impianto di ritrattamento è stato chiuso dopo 58 anni di attività. Anche gli Stati Uniti si stanno impegnando a rispristinare le proprie capacità di ritrattamento, dopo lo stop subito da questa attività per motivi politici negli anni Settanta. Gli USA non hanno ancora accettato la domanda della Corea del Sud di costruire capacità di ritrattamento proprie.

Concessione delle licenze per nuove centrali nucleari in Svizzera

La legge federale sull'energia nucleare vieta espressamente la presentazione di domande di autorizzazione di massima per nuove centrali nucleari. Tale divieto non riguarda le strutture di stoccaggio, smaltimento e ricerca e gli impianti nucleari a basso rischio (una definizione che viene spiegata nell'ordinanza sull'energia nucleare). Per questi ultimi non è necessaria una domanda di autorizzazione di massima. Grazie alla loro elevata sicurezza passiva e al ridotto inventario radioattivo, i microreattori e gli SMR possono essere classificati come impianti a basso rischio ai sensi della legge federale sull'energia nucleare.

Ecobilancio (life cycle assessment, LCA)

I carichi ambientali quantificati tramite l'ecobilancio (LCA) comprendono ripercussioni sul cambiamento climatico, emissioni di inquinanti atmosferici e sostanze tossiche, oltre al consumo di suolo, acqua e altre risorse. I risultati dell'ecobilancio possono essere utilizzati per confrontare l'impatto ambientale di diverse tecnologie di produzione dell'energia elettrica. Vi sono numerosi studi internazionali, e relative analisi, redatte dall'Istituto Paul Scherrer specificamente per le centrali nucleari svizzere. L'impatto ambientale delle centrali nucleari svizzere viene determinato in gran parte dall'origine dell'uranio, per cui le emissioni totali di gas serra in Svizzera si attestano a circa 6 g CO2eq/kWh per i reattori ad acqua pressurizzata e a 9 g CO2eq/kWh per i reattori ad acqua bollente, valori molto bassi rispetto ad altre forme di produzione di energia. I confronti con gli altri indici hanno dimostrato sistematicamente che le tecnologie con gli impatti climatici inferiori (secondo la maggior parte degli indicatori) sono l'eolico, il nucleare e l'energia idroelettrica (v. figure 8.9 e 8.10 nel rapporto principale).
Nella maggior parte dei Paesi che producono energia nucleare il deposito in strati geologici profondi delle scorie radioattive rappresenta il metodo di elezione e viene considerato a livello internazionale il principale sistema di smaltimento degli elementi di combustibile esausti che non causa danni significativi alle generazioni future. In Svizzera, nel novembre 2024, la Nagra presenterà domanda per l'autorizzazione di massima per la costruzione di un deposito presso Lägern Nord, che verrà poi sottoposto a verifica in particolare dall'Ispettorato federale della sicurezza nucleare (IFSN). I costi stimati per l'intero iter di smaltimento verranno rivalutati ogni cinque anni e attualmente ammontano a circa 17,171 miliardi di franchi (secondo lo studio sui costi del 2021, esclusi i costi per la gestione post operativa, la disattivazione e i flussi di scorie federali). Ciò corrisponde a circa 1 centesimo per kWh prodotto ed è già compreso nel prezzo di costo dell'elettricità per le centrali nucleari svizzere menzionato sopra. La legge federale sull'energia nucleare prescrive che durante l'esercizio dell'impianto gli accantonamenti per lo smaltimento delle scorie siano a carico del gestore della centrale e che siano convogliati in due fondi speciali, il fondo di smaltimento e fondo di disattivazione.

Stato della tecnologia di fusione

La fusione nucleare racchiude un enorme potenziale come fonte energetica futura, tuttavia si trova ancora in fase sperimentale e un impianto dimostrativo funzionante per la produzione di energia elettrica deve ancora essere testato. Pertanto, al momento la fusione è ancora lontana dalle applicazioni commerciali, il che rende difficile calcolare tempistiche precise. Si prevede che questa tecnologia non avrà un ruolo negli scenari energetici fino al 2050. Nonostante negli ultimi decenni siano stati fatti progressi significativi, occorre ancora superare alcune sfide tramite attività di ricerca e sviluppo mirate. Tra le aree principali ricordiamo l'ottimizzazione degli scenari del plasma, la dissipazione del calore, il controllo dei transitori del plasma e l'ottimizzazione della ricerca sui materiali per i componenti affacciati al plasma (in particolare in caso di elevato flusso di neutroni), oltre allo sviluppo di tecnologie per la cosiddetta «blanket» per la produzione del trizio.

Tabella 4 Riepilogo del prodotto triplo raggiunto vs. necessario per diversi progetti di fusione

	Achieved Triple Product [10^{21}keV·s/m³]	Triple Product required for reactor [10^{21}keV·s/m³]
Inertial fusion energy (NIF)	≈ 10	$\approx 50 - 500$
Tokamak (JET, JT-60U,...)	≈ 1	≈ 5
Stellarator (W7-X)	≈ 0.1	≈ 5
Others: FRC, Z-pinch,...	≈ 0.0001	≈ 5
Private sector, tokamak	≈ 0.01	≈ 5
Private sector, other approaches	≈ 0.0005	≈ 5 for steady-state DT ≈ 50-1000 for steady-state alternative fuels

Il prodotto triplo (triple product) e il fattore scientifico di moltiplicazione di potenza QSci sono fattori importanti per valutare la potenza di un impianto di fusione, con i quali è possibile giudicare la maturità tecnologica di un progetto di fusione nell'ottica di una centrale a fusione che consenta una produzione netta di energia e un utilizzo commerciale redditizio. Se si osserva la tabella 4 è evidente che il Tokamak (approccio della fusione a confinamento magnetico) è il progetto più promettente, ma è ancora lontano dai requisiti di una centrale a fusione redditizia. Inoltre, attualmente il QSci massimo raggiunto in un Tokamak è ≈ 0,67, quando è necessario un QSci significativamente sopra a 1 (QSci si riferisce solo al rafforzamento della potenza termica e non prende in considerazione la potenza aggiuntiva necessaria per l'esercizio dell'impianto né l'efficienza della trasformazione dell'energia termica in elettricità). Anche nella fusione a confinamento inerziale sono stati compiuti dei progressi e sono in corso delle ricerche per renderla rilevante ai fini dei reattori.

A causa delle sfide di ordine fisico e tecnico, la tempistica per la prima centrale nucleare a fusione al momento è fortemente incerta. Nonostante molte imprese private promettano di portare l'elettricità alla rete già nel 2035 o anche prima, simili affermazioni vanno trattate con grande cautela. Viste le numerose sfide, tali dichiarazioni vanno interpretate come uno sforzo motivazionale e inserite nel contesto della necessità di attrarre investitori privati.

Oltre al progetto ITER, la roadmap europea per la fusione mira a un reattore dimostrativo DEMO entro il 2045. I progressi dipendono dall'entità del finanziamento e dalle decisioni prese oggi.

Executive summary

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Nuclear energy in Switzerland

As of 2023, Switzerland generates nuclear power from four operating reactor units (Beznau 1 and 2, Gösgen and Leibstadt) with a total capacity of approximately 3 Gigawatt electric (GW_e). Nuclear power generation plays still an important role in Switzerland's electricity mix: in the year 2022, the four nuclear units produced 23.1 TWh, accounting for approximately 36% of the total electricity production, and ranking as the second-largest contributor to domestic power generation. Hydropower remains the dominant source, contributing nearly 53% to Swiss electricity generation. Nuclear energy is particularly important in the winter months (in 2022 it contributed to more than 40% of the domestic production mix over a period of five months). Because of the predominance of hydroelectric and nuclear energy sources (89.2% of the electricity mix in the year 2022), Switzerland currently features close to net-zero CO_2-emissions in its electricity production (Table 1.3 in main report).

Nuclear energy worldwide

Nuclear energy also continues to be the largest single source of low-carbon electricity in the OECD countries (electricity share in 2022): 15.8% nuclear, 12.6% hydro, 9.9% wind, 5.9% solar). However, the largest fraction of electricity generation both in the OECD and worldwide stems from burning fossil fuels (almost 50% in OECD countries and more than 60% worldwide). The picture gets even bleaker on consideration of the world's primary energy consumption, where more than 80% of the energy is still being generated from fossil energy carriers, while hydropower accounts for as little as 7%, nuclear only for 4%, and wind and solar power together for merely 5%.

Globally, 32 countries are using nuclear energy, 13 new countries are at an advanced planning or construction stage to introduce nuclear energy in their electricity mix (3 of those countries have NPPs already in construction), and 17 additional countries are in the decision-making phase (summary in

of main report). Four countries foresee a phase-out of nuclear energy, but only Germany ultimately terminated domestic generation in 2023. Spain plans a phase-out by 2035, while Belgium has, despite its phase-out decision, extended the lifetime of two of its seven nuclear reactors due to the recent energy crisis, and Switzerland is planning long-term operation for its existing nuclear power plants potentially up to 80 years of operation, with a gradual phase-out of nuclear following the implementation of the Energy Strategy 2050.

In total, as of March 2024, 415 NPPs are in operation worldwide, for a total of 373.257 GW_e of installed capacity. In addition, 57 NPPs are in construction, providing an additional capacity of 59.22 GW_e. In Europe, 167 NPPs are in operation (148 GW_e) and 9 are under construction (10.1 GW_e).

The countries with the largest number of operating NPPs are the USA, France, China and Russia. As of March 2024, the country with the highest growth of nuclear energy is China, which has 27 NPPs currently under construction, followed by India (7 units under construction), Turkey (4), Egypt (4), South Korea (2) and Russia (4). China has already reached 53.3 GW_e of installed nuclear capacity in 2023 (with close to 400 TWh produced in 2022), with an additional 30.9 GW_e under construction and substantial growth plans (with claims of reaching up to 150 GW_e installed nuclear capacity by 2030). In Europe, the following countries are currently building or planning new power plants in the near future: France (1 EPR unit under construction, 6 EPR-2 units approved, more planned), United Kingdom (2 EPR units under construction, 2 additional EPR units planned), Slovakia (1 unit under construction, more proposed), Bulgaria (2 AP1000 units planned), Czech Republic (4 units planned, in addition 3 sites identified for several SMRs), The Netherlands (2 units planned), Romania (2 CANDU units planned, 6 NuScale SMR modules proposed), Hungary (2 VVER units approved), Slovenia (1 unit proposed), Sweden (2 units planned by 2035, 10 additional units planned beyond 2035), Estonia, and Poland (3 AP1000 approved, 2 APR1400 units planned, 24 BWR-300 SMR units planned). In addition, two new nuclear power plants have been connected to the grid in 2023 in Belarus and Finland.

Recent developments include the construction of the world's first deep geological repository for highly radioactive nuclear waste in Finland, with construction to be completed by the mid 2020s. In Sweden, a construction license for a geological repository was likewise granted, with construction to start in the next few years, while in France an application for the construction of a deep geological repository is currently being evaluated by the nuclear authority, with a decision on the final disposal site expected in 2025 and operation to start around 2040. In Canada, the selection of the waste disposal site is planned to be announced in 2024. In Switzerland, the approval for a deep geological repository is expected around 2030 (subject to a positive assessment by the nuclear safety authorities and potentially an optional referendum), with plans to start operation in 2050.

In the last few years, especially in view of the changes in the geopolitical landscape caused by the Ukraine war, several countries have been revising their plans on nuclear energy. This has culminated in:

- the launch of the EU Nuclear Alliance in 2023, with 16 countries (France, Belgium, Bulgaria, Croatia, Czech Republic, Finland, Hungary, Netherlands, Poland, Romania, Slovenia, Slovakia, Estonia, Sweden, Italy, UK) planning to develop an integrated European nuclear industry, with the commitment of reaching 150 GW_e of nuclear energy in the EU electricity mix by 2050 (an increase of 50% compared to today's nuclear fleet);
- the launch of the European Commission Small Modular Reactor Alliance in 2024, aimed at "maintaining European technological and industrial leadership in nuclear";
- the nuclear declaration at the United Nation Climate Change Conference (COP28) in December 2023, made by 22 countries having as goal to triple nuclear energy by 2050 in order to reach the new zero goal, "recognizing the key role of nuclear energy in achieving global net-zero greenhouse gas emissions / carbon neutrality by or around mid-century". These countries include the United States, Bulgaria, Canada, Czech Republic, Finland, France, Ghana, Hungary, Japan, South Korea, Moldova, Mongolia, Morocco, Netherlands, Poland, Romania, Slovakia, Slovenia, Sweden, Ukraine, United Arab Emirates, and the UK;
- the launch in the US of an investment plan to foster development of SMRs and microreactors and their deployment in US as well as abroad. The Inflation Reduction Act signed in 2022, aims at providing support for existing and new NPPs through investment and tax incentives for both large existing NPPs as well as newer advanced reactors, for uranium fuel and hydrogen production. Several NPPs have had their lifetime extended (e.g. Diablo Canyon in California, which was supposed to be shutdown in 2022. Six reactors have had their lifetime extended to 80 years. Several others are expecting decision from the nuclear authority). Remarkably, the state of Michigan is reopening the Palisades NPP which was shutdown since 2022. A few pilot projects have been successfully started at existing NPPs to use nuclear energy for hydrogen production.

Because of the increased recognition of the importance of reliable baseload power, several US companies such as Amazon, Google, Microsoft and energy-intensive industries like Nucor (steel production) and Dow chemicals have signed agreements with nuclear vendors or utilities for future supply of nuclear energy.

Status of GEN III/III+ Light Water Reactors (LWRs) and construction time

Gen-III/III+ reactors are a new generation of nuclear power plants, based on the same Light Water Reactor (LWR) technology as the currently operating plants, but characterized by significantly improved safety characteristics, which include in their design features the lessons-learnt from the three major reactor accidents in history. As of December 2023, 38 large Gen-III/III+ LWR units are in operation, and out of 60 reactors currently under construction, 51 are large Gen-III/III+ LWRs. Additional units have been ordered or tenders are in progress (e.g. three units in Poland, two units in UK, one in Czech Republic, among others), and several more are planned.

The mean construction time of the operating 38 Gen-III/III+ reactors is 7.7 years, with a median of 8 years (see Figure 2.5 and Figure 2.6 in main report). In comparison, the world's operating fleet of 413 reactors of Generation-II and III altogether features a mean construction time of 7.5 years, and a corresponding median of 6.3 year. These numbers do not corroborate the common notion that construction times of recent nuclear power plants have been drastically increasing, but rather support a modest growth, however, with some notable outlier projects, primarily for first-of-a-kind power plants in Europe and the US, for which construction times have grown out of proportion. On the other hand, it has been demonstrated repeatedly that it is technologically feasible to provide a turnkey system in less than six years construction time, provided a functioning supply chain for key components is established.

Notably, the ABWR (GE Hitachi/Toshiba) units in Japan stand out for their short construction time, being all completed in under 4 years. The Westinghouse AP-1000 in the USA (Vogtle-3) and both EPR in Olkiluoto and Flamanville (France) are located at the opposite end of the spectrum, with construction times of 10 and 16.5 years, respectively.

These projects had been subject to unique challenges from their very inception, as they marked pioneering efforts in building the first-of-its-kind large power plant in Europe and USA after decades of building inactivity, and the need to re-establish manufacturing capabilities and supply chains. Additionally, significant design changes were required by the nuclear authorities well into the construction phase of the power plants, both in Finland and USA. Not at the same severity level as for Finland and Flamanville, but yet rather significant delays have been observed with the two EPR units under construction in the UK at Hinkley Point. The delays were partly associated with missing chain links in the UK supply chain, the need to train UK workforce (with delays encountered mostly with the civil construction), and a large number of design changes (more than 7'000) requested by the UK nuclear authority. Despite these setbacks, the UK government has confirmed the construction of two additional EPR units at the Sizewell site.

The level of completeness of the detailed design at construction start and the establishment of a functioning supply chain and manufacturing capabilities are thus important factors in the determination of the construction length; the experience with multiple successive units and the reliability of the financial and regulatory framework are likewise important contributors. China has been capable to consistently decrease plant construction times, with the latest nine units (of standardized HPR1000 and ACPR-1000 design) all built between 5 and 7 years. Noteworthy is also the recent example of the United Arab Emirates, where the South Korean company KHNP has built 5.2 GW_e of nuclear capacity (4 APR1400 units) within 9 years and for a total cost of $24 billion only.

Gen III/III+ LWRs Economics

Estimates based on reputable scientific sources (PSI 2019) put the levelized cost of electricity (LCOE) of new NPPs between 7 and 12 ct./kWh. As long as the construction time stays below 8 years (the median of the 38 Gen-III/III+ builds is 7.7 years), a LCOE of 7 cents is achievable, in agreement with previous PSI studies dating back to 2019. This is well within the range of current and future LCOE for renewable energy sources in Switzerland and existing hydropower plants and would deliver base-load electricity. The 2019 PSI results are consistent with other peer-reviewed studies reported in the open literature. The current LCOE for the operation of existing Swiss NPPs is at the level of 4.0 – 5.5 ct/kWh (therein already included the full costs of the waste disposal path). Long-term operation of up to 60 years of these plants would increase LCOE by 1-2 cents. It has to be noted, however, that the LCOE metric was first introduced to compare dispatchable energy sources, which has limited value when one speaks about an increasingly complex energy system with higher and higher penetration of renewables. It has been increasingly recognized that in such cases it is crucial that not only the LCOE is taken into account, but that the entire system costs (balancing costs, grid expansion costs, backup costs, etc.) need to be accounted for. An attempt at such a study for the Swiss energy system has been recently published by OECD , but a large, comprehensive Swiss energy system model including different nuclear deployment scenarios has never been created.

The first-of-a-kind 1600 MW_e EPR units in Olkiluoto and Flamanville have been considerably more expensive than the 1400 MW_e South Korean APR1400 built in UAE. While the APR1400 cost 6 billion USD per unit, the two EPR in Olkiluoto and Flamanville cost around 11 and 13.2 billion euros respectively. These high capital costs need however to be considered in relation to the energy produced. A single EPR unit would produce more than 12 TWh/year. In comparison, to produce the same yearly output of an EPR with Alpine solar plants, one would need the equivalent of more than 3800 “Alpin Solar” (Muttsee-Staumauer plant) at a cost of more than 30 billion CHF (without considering the additional costs for backup, storage and extension of the grid). Taking instead the Gondosolar plant as reference, one would need more than 780 of such plants at a cost of about 29 billions CHF.

The high capital costs of large NPP are one of the main economics challenges of nuclear energy, as it reduces the number of potential private investors. This challenge is somewhat alleviated with SMRs and could largely be eliminated with microreactors, which have overall plant capital costs comparable to alpine solar plants, but with a much higher and steadier energy output.

To cope with the high capital costs for large NPPs, different models have been implemented in the past, with governments participating as equity investors, as loan providers or through policy measures such as loan guarantees or contract for difference (CfD). Contrary to common public perception, nuclear energy is the least subsidized among the energy sources, as discussed in detail in section 1.4.2 of this report. In the EU, over the period 2015 – 2022, subsidies for nuclear energy reached a maximum of 7.6 billion EUR in 2021, compared to 88 billion EUR for renewables and 123 billion for fossil fuels. In the US, within 2016 and 2022, the maximum subsidies for nuclear energy were less than 600 million USD compared to more than 17 billion USD for renewables, more than 2.5 billion USD for coal and about 3 billion USD for gas.

The interest rates on the capital are significantly affected by the financing scheme (e.g. state loan guarantee) and regulatory framework. A successful way to reduce costs is by building multiple units at the same site. In the Barakah project in the UAE, for example, a 40% reduction in labour costs was experienced between the construction of Units 1 and 4. Other factors that positively impact the success of a new build are the completion of relevant parts of the design before the start of construction, the presence of a well-established supply chain, access to a skilled workforce and a stable regulatory framework.

The LCOE for SMRs are forecasted to reach levels similar to large NPPs, by replacing the economy of scale with the economy of manufacturing (factory-built), though higher costs need to be anticipated for first-of-a-kind units. The first NuScale SMR project that was supposed to be built in Utah in the coming years had forecasted costs of about 5.8 ct/kWh (2020 estimate).

Due to a 150% increase in interest rates and significant increase of material costs (e.g. 40% cost increase for steel) over the past 1.5 years, the costs were forecasted to rise to 8.9 ct/kWh in 2023. This was not competitive with the gas and coal available to the Utah local utilities and for this reason the NuScale project was superseded by a gas plant.

Gen-III/III+ safety

Significant changes to the safety requirements have been adopted with Gen-III/III+ power plants, with severe accident management systems now being integral to the design, and a strengthened independence of the different levels of defence-in-depth. The implementation of the new safety philosophy has resulted in a whole new range of engineered passive safety systems (which do not rely on external power such as Diesel generators, or operator action for their functioning) and extended grace periods, with the goal to practically eliminate the occurrence of severe accidents sequences with core meltdown and subsequent containment failure that could lead to early or large radioactive releases. In particular, these new safety approaches have led to:

- an increase of the grace period (in which no human intervention is needed even under the most adverse accidental circumstances) from 30 minutes, typical of Gen-II designs, to a minimum of 3 days, more commonly beyond one week;
- core damage frequencies below 10^{-6}/year (i.e. probabilistically speaking less than once in a million years);
- a probability of a failure of the containment subsequent to the core damage with releases to the environment below 10^{-7}/year (i.e. less than once every ten million years).

The probability for a core damage and the subsequent release of significant amounts of radioactivity has therefore been reduced by one to two orders of magnitude compared to current, properly backfitted Generation-II power plants, which have already reached excellent safety levels due to retrofitting measures and the Post-Fukushima stress tests.

Status of Small Modular Reactors (SMRs)

SMRs are advanced reactors with nominal power of up to 300 MW(e) per unit. They are designed to be built in factories and shipped to the site of deployment. Usually they are installed below ground level. The OECD Nuclear Energy Agency (NEA) anticipates a share of SMRs as high as 9% of the total new nuclear capacity by 2035. Presently, 10 SMRs are in operation in Russia and China, and several are currently under construction or awaiting licensing (US, Canada, France), see Tables 1 and 2.

Among the water-cooled SMRs currently on the market, the most advanced for deployment in Europe by 2030 are NUWARD (EDF), Roll-Royce (UK), BWXR-300 (USA), Holtec-180 (USA), AP300 (USA) and VOYGR (NuScale). The latter is already licensed in the USA, while the other designs are at different stages of design pre-certification in USA, Canada and in some European countries. Some SMR vendors have received purchase orders (e.g. BWRX-300 in Canada). NUWARD was granted €500 million by the French Government in 2022 and construction start of a first reactor is planned for 2030.

The general interest in SMRs arose originally from the need to power remote regions or off-grid areas, which are currently relying on gas, oil or Diesel, and to replace outdated fossil-fired power plants in the range of 300-400MW_e. For countries with small electricity grids where the deployment of large NPPs would not be possible or investors and operators would not be able or willing to invest large capitals, SMR are seen by some as economic options. Furthermore, SMR may present opportunities to supply energy-intensive industrial sites (concrete or steel industry for instance), or to provide non-electricity applications of nuclear energy, e.g. district heating, sea water desalination, or most notably hydrogen production.

Because of their smaller size, most SMRs have enhanced safety characteristics, fully based on passive safety. This is why the Nuclear Regulatory Commission in the US has approved a new sizing rule for Emergency Planning Zone (EPZ), based on a consequence-oriented approach. As a result, the NuScale SMR has been licensed with an EPZ limited to the plant site perimeter (e.g. no evacuation zone is required). It is expected that other SMRs will receive a similar ruling in the USA.

Table 1 LWR SMRs in advanced stage of development

Name	Thermal power [MWth (MWe)]	Type	Design organisation	Country	Status
CAREM	100 (30)	Integral PWR	CNEA	Argentina	Under construction
ACPR50S	200 (60)	Floating PWR	CGNCP	China	Under construction
ACP100	385 (125)	Integral PWR	CNNC and NPIC	China	Construction started in 2021
KLT-40S	150 (35)	Floating PWR	OKBM	Russia	2 units in operation
VOYGR	250 (77)	Integral PWR	NuScale Power	USA	Shortlisted in USA and Europe
AP300	900 (300)	One-loop PWR	Westinghouse	USA	Shortlisted in UK
UK SMR	1,358 (470)	Integral PWR	Rolls-Royce	UK	Short-listed in Estonia and UK
NUWARD	540 (170)	Integral PWR	EDF	France	FOAK in France by 2030.
BWRX-300	870 (290)	Integral BWR	GE-Hitachi	USA	Several units to be built in Canada and USA. Shortlisted in Europe.
SMR-160	525 (160)	PWR	Holtec	USA	Shortlisted in various countries
SMART	365 (107)	PWR	KAERI	Korea	Licensed in Korea
RITM-200	175 (55)	Floating PWR	OKBM	Russia	Six units in operation. More under construction.
RITM-200N	190 (55)	On-shore PWR	OKBM	Russia	First concrete planned for 2024.
RITM-200S	198	Floating PWR	OKBM	Russia	To be built at Baimskaya copper mine site, deployment by 2027.
RITM-200M	175 (50)	Floating PWR	OKBM	Russia	MOU signed for deployment in Philippines and Myanmar.

Main advantages of SMRs are the significantly lower initial capital costs due to the smaller size of the plant, shorter construction times because of the shift to factory production, the increased flexibility for load-following operation that makes SMRs easier to integrate with intermittent renewables sources, and the enhanced safety concepts. The economy of scale of large NPPs is believed to be replaced by economy of production (factory-built modules) and simpler construction sites, with a cost per kWh in the same range as large NPPs. Harmonization activities in licensing SMR are ongoing between the IAEA and US/European nuclear regulators to create a stable and transparent licensing environment that avoids unpredictable changes to national licensing regimes.

Most near-term SMRs belong, like the large-scale reactors, to the category of Light Water Reactors of Generation-III/III+, with a credible perspective for the commercial operation of the first demonstrator plants in Western countries by 2030 or even earlier (see Table 1 above). Advanced SMRs, with coolants other than water (e.g. liquid metal, helium, molten salt) belong to nuclear plants of Generation-IV and are being pursued by a wide range of start-up companies. However, the time horizon for the commercial deployment of some of these designs (e.g. molten salt) is lagging several years behind LWRs development. While non-water cooled SMRs are already in operation in China and Russia, the first one in the Western countries will be the sodium-cooled SMR by Terrapower, to be built in Wyoming (USA). The construction permit application for the Terrapower SMR was submitted in March 2024 and accepted for review by the US nuclear authority in May 2024.

Table 2 Non-LWR SMRs in advanced stage of development

Name	Thermal power (MWth)	Type	Design organisation	Country	Status
			Thermal spectrum		
HTR-PM	500	HTGR	INET	China	2 units in operation in China since Dec 2021 Additional 18 HTR-PM units proposed.
KP-FHR	311	MSR / solid fuel	Kairos Power	USA	Construction permit for demo unit received in Dec 2023.
XE-100	200	HTGR	X-energy	USA	Completed pre-certification in Canada. Pre-licensing in US. Selected by Dow Chemical (USA)
IMSR	884	Integral MSR	Terrestrial Energy	Canada	Pre-licensing in USA, and Canada.
			Fast spectrum		
ARC-100	286	SFR	ARC Clean Tech.	Canada	Pre-licensing in Canada.
Wasteburner	750	MSR	Moltex Energy	Canada	Pre-licensing in Canada
Natrium	840	SFR	TerraPower	USA	Pre-licensing in USA. To be built in Wyoming (USA)
BREST-OD-300	700	LFR	NIKIET	Russia	Under construction in Russia. Completion is planned for 2026.
CFR-600	1500	SFR	CNNC	China	2 units under construction in China Connection to the grid in 2024 - 2025.

Status of Microreactors technology

Over the past seven years, an interesting trend has emerged on so-called microreactors, designed to produce electrical power in the range of up to about 10 MW_e (with several being proposed in the USA, see Table 4.1 in main report). Those are reactors designed to be fully factory-fabricated, to fit into an ISO container to be easily transported (on ship, track, train) from the factory to the deployment site (no construction site needed), and to operate without refuelling for 5-10 years or more. They can be operated independently, as part of the electric grid, or within a microgrid. Thought to be deployed in remote areas (e.g. mining sites), or to provide electricity and heat to energy-intensive industries (e.g. water desalination, hydrogen production, etc.), they are also of interest for industries requiring a certain level of independence from the electrical grid and a guarantee of security of energy supply. Cooling is through gas (Helium), liquid metal, molten salt, or (sodium) heat pipes.

Because of the very small size and the simplicity of the design, their development is proceeding extremely fast. The first demonstration unit (heat pipe design) was designed, built and tested within 3 years by NASA and Los Alamos National Laboratory at the cost of less than $20 million. A second unit (liquid-metal cooled) is currently being built in Idaho (USA) and is expected to start operation by early 2025. A fluoride-cooled unit was awarded construction license in December 2023, with operation targeted for 2026. Three other designs are at various stages of licensing in the USA and Canada.

Because microreactors will be entirely factory-built and are therefore expected to benefit from manufacturing productivity, a positive learning curve is anticipated, as for other industries. Other claimed advantages are the very small capital costs (in the order of ~$100s million or less), which could make them reasonably affordable for a wider range of investors, and yield small levelized cost of electricity (LCOE) costs compared to the backup alternatives available in remote areas or to large industries because of complete factory fabrication, very small footprint (~15 m^2 for the plant and less than 2000 m^2 for the plant site), a predictable construction schedule, and a reduced radiological risk. Because of the very small amount of fuel and the simplicity of the design, they are more similar to research reactors and therefore a much more expedited licensing is expected than for SMRs or large NPPs. The use of TRISO fuel of higher enrichment requires the development of fuel fabrication capabilities, currently on-going in the USA and France. This is however not a showstopper as the technology is known (same fuel used in the HTR-PM SMRs operating in China).

Status of Gen-IV and non-LWRs

Non-water cooled reactors (e.g. cooled by gas, lead, sodium, molten salt) are designed with the goal of increasing efficiency through either increase of thermodynamic efficiency and/or through improved fuel utilization and further reduction of the amount of highly radioactive waste (fast spectrum reactors), thus establishing a circular economy for nuclear fuel[5]. Several designs exist, of which the most promising one are worth mentioning here:

- Gas-cooled thermal reactors using helium as a coolant. They have a higher thermodynamic efficiency in converting into electricity the thermal power generated in the reactor, and since they operate at much higher temperatures than LWRs, are also suitable to supply heat for high-temperature energy-intensive industrial processes (see Figure 1 for an illustration of the temperatures needed for various industrial processes and the corresponding reactor designs which can provide such temperatures). Two gas-cooled thermal reactors (HTR-PM design) are already in operation in China since 2021. The Xe-100 (X-energy, USA) has just successfully completed pre-certification in Canada (i.e. no showstoppers have been identified by the nuclear authority that would preclude licensing).
- Fast reactors cooled with liquid metals (sodium or lead/lead-bismuth). They operate at high temperatures intermediate between LWRs and gas-cooled reactors, and at almost ambient (atmospheric) pressure. Significant operation experience exists (France, Japan, Russia, etc), with several units in operation today (see Table 3). Terrapower (US company) is offering a sodium-cooled fast SMR on the market, with the first unit foreseen to be built in Wyoming (USA) before 2030.
- Molten Salt reactors (MSRs), in which molten salt is used as coolant, fuel, and/or moderator. These reactors operate at high temperatures, and both thermal and fast designs exist. A significant remaining operational challenge of these reactors is given by the strong corrosive nature of salts. In December 2023, a thermal reactor design by KAIROS (USA) in which molten salt is used only as coolant (with HALEU TRISO fuel, and graphite as moderator) has been granted a construction license for a first demonstration unit in Tennessee. An integral MSR (Terrestrial Energy) is currently undergoing pre-licensing in USA and Canada. An experimental MSR using a Thorium-based molten salt as fuel was granted the operation license in China in June 2023. The construction of this TMSR-LF1 reactor began in

[5] More than 90% of the spent nuclear fuel is reusable. In a closed fuel cycle, the spent nuclear fuel is reprocessed to extract the reusable material (mostly uranium), which is then used to create new fuel assemblies. A closed fuel cycle can be achieved, for example, through a combination of LWRs and Gen-IV fast reactors. In fast reactors, high energy ("fast") neutrons are used to fission the nuclear fuel, while in LWRs it is mostly thermal (low energy) neutrons that are used to cause fissions. A closed fuel cycle allows to improve sustainability by increasing energy output per unit of fuel mass, and by reducing the amount of high level waste produced per unit of energy.

September 2018 and was scheduled to be completed in 2024. However, it was reportedly completed in August 2021 after work was accelerated.

Relevant Gen IV reactors for the Western market are:

- KAIROS, Terrestrial, X-energy (all thermal reactors, see Table 2)
- Terrapower, IMSR (Moltex), ARC-100 (all fast spectrum reactors, see Table 2)
- the microreactor designs currently undergoing licensing in the USA and Canada (Table 4.1 in full report).
-

Table 3 Fast power plants in operation (all of SFR type)

Country	Reactor name	Operation years	Current status
China	CEFR	2010-present	Active
India	FBTR	1985-present	Active
Russia	BOR-60	1969-present	Active
India	PFBR	Scheduled for 2024	Under construction
Russia	BN-600	1980-present	Active
Russia	BN-800	2014-present	Active

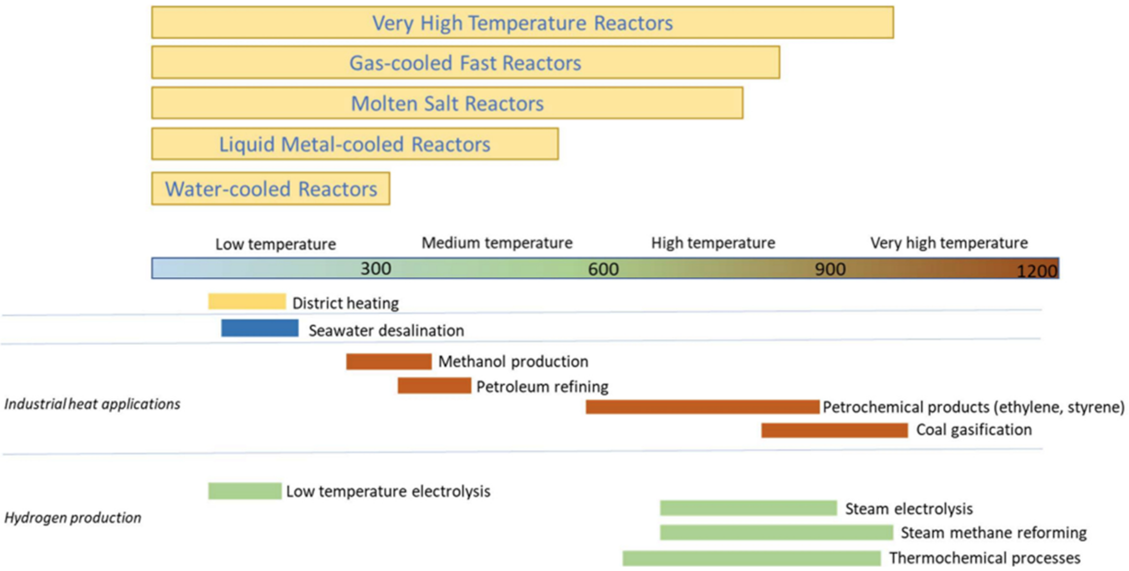

Figure 1 Output temperature of NPP technologies and corresponding non-electric applications

Uranium Fuel availability and alternative fuel cycles

Natural uranium reserves are a widely distributed resource (see Figure 2) and are sufficient for the next few centuries. As with other resources, they depend also on the market price.
Low Enriched Uranium (LEU) fuel used in LWRs (e.g. Swiss plants) is produced in several enrichment plants and there is sufficient variety and diverse sources to assure its supply. No long-term risks for the security of supply of nuclear fuel to Switzerland is anticipated.
Looking further ahead into the second half of this century, it is reasonable to assume that an increased need for nuclear power will lead to increased exploration activities and therefore to

increased uranium reserves. In addition, on this timescale, reactor technology will develop to such an extent (e.g. fast reactors) with a closed fuel cycle that other fuels with much greater energy potential than U-235 can be used, thus extending the availability of nuclear fuel from hundreds to many thousands of years. A few fast reactors are already in operation or in construction worldwide (see Table 2 and Table 3), or at an advanced planning stage like the Terrapower reactor, with the first unit planned for construction in Wyoming in the present decade. A major fuel for non-water-cooled advanced reactors, including the Terrapower design, is high-assay low-enriched Uranium (HALEU), characterized by enrichment between 5 and 20%. Consequently, HALEU fuel production is being increased. Ironically, the disruptions to the nuclear fuel supply chain by the war in Ukraine will in the next 5-10 years serve to strengthen and enlarge western fuel supply capabilities and resilience. New supply chains for HALEU fuel are currently being built in the US and in France, while the capacity of existing low enriched Uranium (LEU) plants both in Europe and US has been increased. LEU is typically used in conventional water-cooled reactors. By the end of 2023, US has also opened three new uranium mines to increase independence for Uranium supply.

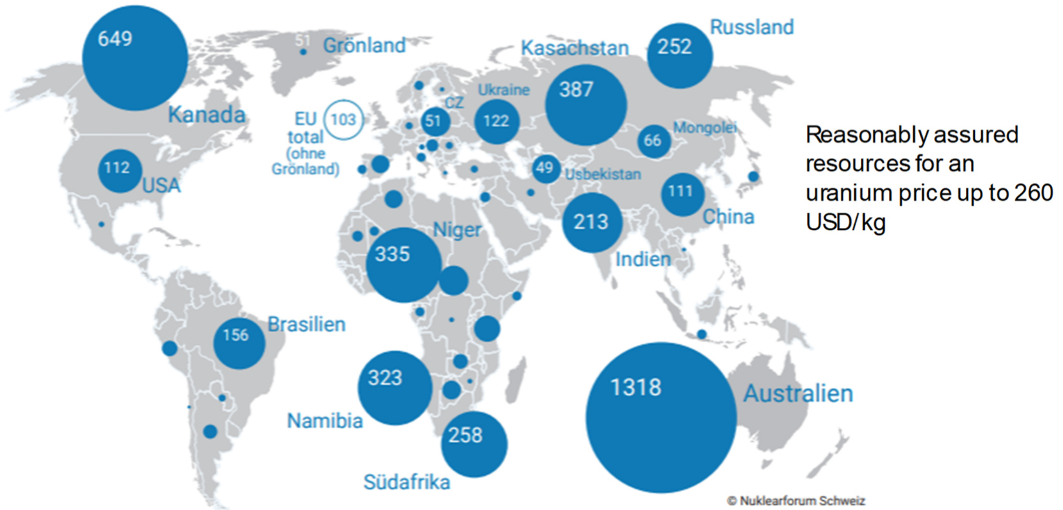

Figure 2 Worldwide distribution of Uranium reserves as per 01.01.2021 (in 1000 tons of Uranium) for a price up to 260 USD/kg. Data from IAEA Redbook 2022.

More advanced fuel cycles aimed at more efficient fuel use and reduction of high-radioactive waste rely on fuel reprocessing and include:

- fabrication of new fuel elements (so-called MOX) for LWRs using the Pu and reprocessed U extracted from LWRs spent fuel. Typically out of 4 spent fuel assemblies (FAs) a new fresh FA can be made. This is a well-established technology. Successfully implemented in Swiss plants and used for several years, MOX fuel was later on phased out because the export of spent fuel for reprocessing purposes had been prohibited by law. Reprocessing is still routinely done in other countries (e.g. France);
- breeding with U-Pu in fast reactors (breed-and-burn fuel cycle). In this case, additional fuel is generated in the reactor core during the plant operation so that more energy can be generated from the same fuel load. It requires HALEU fuel;
- consumption of minor actinides (MAs) in dedicated MAs “burners” (e.g. MYRRHA and Transmutex reactor designs), to decrease the radiotoxicity of existing spent fuel. This requires the reprocessing of MAs which, while already proven in the laboratory, does not exist at industrial scale yet. With dedicated burners, a higher transmutation rate per energy unit than in conventional fast reactors is potentially achievable, however, the operation of dedicated transmutation reactors increases the technological complexity and thereby also the operational risks. Moreover, the bulk reprocessed mass would still be Reprocessed Uranium (RepU), whose further handling is not addressed by the transmuter. Nevertheless,

if MAs reprocessing would be implemented, a dedicated transmuter could be a relevant component of the nuclear fuel cycle and support the minimization of the waste stream;

- Thorium (Th)-U cycle: Th-232 is a fertile isotope analogous to U-238, which occurs naturally and is three times more abundant in the earth's crust than is U. Thorium can be deployed to breed U-233 in the same fashion U-238 is used to breed Pu-239. The use of Thorium requires reprocessing and some adaptation of today's technology. However, if this were implemented, it would provide hundreds to thousands of years of additional fuel supply, at the current requirements. Thorium test reactors have been successfully operated in the past. A Thorium molten salt demonstration reactor has started operation in China in 2023, which will be followed by a 373 MWt reactor of the same type by 2030. A 40 MWth Thorium fast breeder reactor is in operation in India, since the country has large amounts of Thorium resources.

Commercial fuel reprocessing plants exist in France, Russia, India, Japan (the latter one being in construction, to be completed in 2024), China (two in construction, the first to become operational in 2025). One reprocessing plant in the UK was closed in 2022 after 58 years of operation. The USA are also working at re-establishing reprocessing capabilities in the country, after phasing out reprocessing in the 1970s for political reasons. United States has so far not consented to the request by South Korea to build own reprocessing capabilities.

Licensing of new nuclear power plants in Switzerland

The Swiss Nuclear Energy Act expressively prohibits the submittal of General License Applications for new nuclear power plants. Not affected by this ban are storage, disposal and research facilities, as well as nuclear facilities of low risk (a term which is further elaborated in the Swiss Nuclear Energy Ordinance). For the latter, no General License application is necessary. Microreactors and SMRs, due to their high level of passive safety and smaller source term, have the potential to qualify as low-risk facilities according to Swiss nuclear law.

Environmental Life Cycle Assessment (LCA)

The environmental burdens quantified by Life Cycle Assessment (LCA) include impacts on climate change, emissions of air pollutants and toxic substances as well as land, water and other resource consumption. LCA results can be used to compare the environmental performance of different power generation technologies. Several international studies exist, including analyses performed by the Paul Scherrer Institut specifically for the Swiss NPPs. The environmental performance of Swiss NPPs is to a large extent determined by the origin of uranium, with total GHG emissions around 6 g CO2eq/kWh for Swiss PWRs and 9 g CO2eq/kWh for Swiss BWRs, which compare very favourably with other forms of energy generation. Comparisons for the other metrics consistently show that the technologies with the lowest environmental impact in most categories are wind, nuclear power, and hydro (see Figure 8.9 and Figure 8.10 in the main report).

The deep geological disposal of radioactive waste is the method of choice in most countries deploying nuclear energy and is being regarded internationally as the principal way of spent fuel disposal, posing no significant harm to future generations. In Switzerland, Nagra will submit its General License application for a disposal site at Nördlich Lägern in November 2024, which will then be in particular scrutinized by the Swiss regulator, ENSI. The projected costs for the entire waste disposal path are re-evaluated every five years, and currently amount to approx. 17,171 Billion CHF (according to the Kostenstudie 2021, excluding the costs of post-operation, decommissioning, and Federal waste streams). This corresponds to roughly 1 cent per kWh generated, and is already included in the LCOE for Swiss NPP cited above. The Swiss Nuclear Law stipulates that the accruals for the deferred liability of the waste disposal are debited to the nuclear utilities during plant operation, and are accumulated in a specific fonds, the "Entsorgungs- and Stillegungsfonds".

State of fusion technology

Nuclear fusion, while holding immense potential as a future energy source, is still at the research stage, and a working demonstration facility for electricity generation has not yet been proven. Because of this, fusion is currently far from commercial uses, making it difficult to forecast a precise timeline. The technology is thus not expected to play a role for energy scenarios on the 2050s timescale. While significant progress has been made in recent decades, several challenges must still be addressed through focused research and development activities. Key areas include optimizing plasma scenarios, managing heat exhaust, controlling plasma transients, and advancing material research for plasma-facing components, high neutron flux conditions, and breeding blanket technologies.

Table 4 Summary of the achieved vs required triple product for various fusion concepts

	Achieved Triple Product [10^{21}keV·s/m³]	Triple Product required for reactor [10^{21}keV·s/m³]
Inertial fusion energy (NIF)	≈ 10	$\approx 50 - 500$
Tokamak (JET, JT-60U,...)	≈ 1	≈ 5
Stellarator (W7-X)	≈ 0.1	≈ 5
Others: FRC, Z-pinch,...	≈ 0.0001	≈ 5
Private sector, tokamak	≈ 0.01	≈ 5
Private sector, other approaches	≈ 0.0005	≈ 5 for steady-state DT ≈ 50-1000 for steady-state alternative fuels

Important figures of merit to assess the performance of a fusion machine are the triple product and the scientific power multiplication factor Q_{Sci}, which allow to assess the proximity of a given fusion concept to a machine capable of net energy production and viable commercial use. Looking at Table 4, it is clear that the Tokamak (magnetic confinement fusion approach) is the most promising concept, but still far from what is required for a viable fusion plant. In addition, the highest Q_{Sci} achieved in a Tokamak is currently ≈ 0.67, while a Q_{Sci} significantly larger than 1 is needed (Q_{Sci} is only concerned with the amplification of the heating power and does not consider the additional power required to operate the machine, nor the efficiency of the conversion of thermal energy into electricity). Inertial fusion energy has also shown progress, and efforts are underway to make it more reactor-relevant.

Because of both physics and engineering challenges, the timeline towards a first fusion power plant is currently associated with significant uncertainties. While many private companies promise power to the grid by as soon as 2035 or even earlier, these statements should be warranted with a lot of care. Considering the many challenges, such statements should rather be seen as motivational aspirations and in the context of the need to attract private investors.

Beyond ITER the European fusion roadmap targets a first demonstration reactor DEMO by 2045. Progress depends on funding levels and decisions taken today.

1 Introduction

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1.1 Current role of nuclear energy in the Swiss energy mix

As of 2023, nuclear power in Switzerland is generated by four units distributed across three separate locations, as summarized in Table 1.1, with a fifth unit (KKM) permanently shutdown since 2019 and currently in decommissioning.

Table 1.1 Summary of Swiss Nuclear Power Plants

Power Plant	Leibstadt (KKL)	Gösgen (KKG)	Beznau (KKB)	Mühleberg (KKM)
Status	Operational	Operational	Operational	In Decommissioning
Reactor type	BWR	PWR	PWR	BWR
Vendor and country of origin	General Electric USA	Siemens Germany	Westinghouse USA	General Electric USA
No. of units and Capacity [MWe]	1 x 1233	1 x 1010	2 x 365	1 x 373
In operation since	1984	1979	1969 / 1972	1972

Table 1.2 Electricity production plants in Switzerland [1.1]

Category	Installed capacity [MW]	Number of plants
Hydropower	16’132.2	1’490
Photovoltaic	4’444	190’911
Nuclear energy	3’014.6	4
Waste	379.9	30
Natural Gas	280.1	188
Biomass	253.2	433
Wind Energy	88.1	68

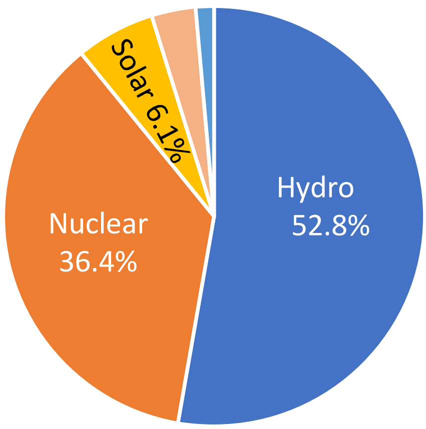


Figure 1.1 Electricity production in Switzerland by source for year 2022. Total of 63.5 TWh [data from BFE].

The electricity production plants present in Switzerland and their corresponding installed capacity is reported in Table 1.2. Their contribution to the overall share of electricity production for 2022 is portrayed in Figure 1.1. Currently, nuclear power plays a substantial role in Switzerland's electricity generation. In year 2022, the four nuclear units produced 23.1 TWh, accounting for more than 36% of the total electricity production, and therefore ranking as the second-largest contributor. Hydropower remained the dominant source, contributing to nearly 53% of the nation's electricity production. Because of the predominance of hydroelectric and nuclear energy sources, which together made up 89.2% of the electricity mix in year 2022, Switzerland belongs to the very few countries in Europe that have already achieved close to zero CO2-emissions in their electricity production.

Particularly noteworthy is the significance of nuclear energy during the winter months. As shown in Figure 1.2, nuclear contributed to more than 40% of the Swiss electricity mix for five months in 2022.

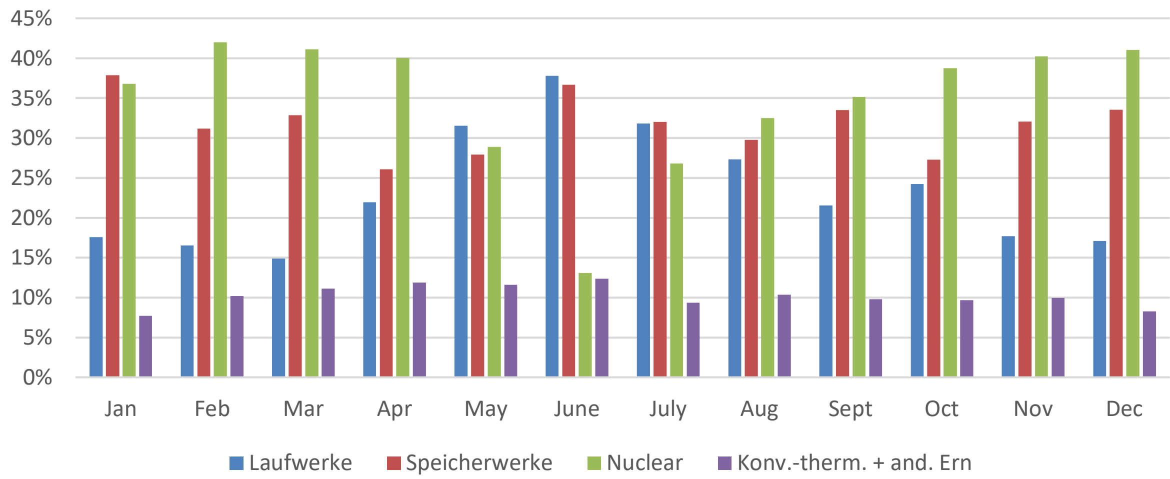


Figure 1.2 Electricity production in Switzerland by source and per month of the year, 2022 (BFE data)

Table 1.3 Electricity mix [1.2] in selected countries and associated carbon intensity [1.3] for year 2022

Country	gCO_2-equiv/kWh	Oil	Gas	Coal	Nuclear	Wind	Solar	Biomass	Hydro	Wind, solar, biomass	Nuclear + fossile
		%	%	%	%	%	%	%	%	%	%
Norway	29	0.5	0.5	0.0	0.0	10.4	0.1	0.1	88.3	10.7	1.0
Sweden	45	1.6	0.2	< 0.01	29.8	19.4	1.4	7.4	40.3	28.1	31.6
Switzerland	46	3.6	0.0	0.0	37.0	0.1	4.3	0.2	54.8	4.6	40.6
France	85	2.1	9.2	0.9	63.3	8.2	4.3	2.1	9.8	14.6	75.5
Finland	131	4.9	1.8	4.1	34.3	16.6	0.4	19.1	18.8	36.1	45.1
Austria	158	5.3	18.2	0.2	0.0	10.7	4.2	5.6	55.8	20.5	23.7
Spain	217	4.0	30.6	2.7	20.5	21.7	11.5	2.4	6.6	35.6	57.8
Netherlands	356	5.0	39.6	12.1	3.4	17.9	13.9	8.0	0.1	39.8	60.1
USA	367	0.9	39.3	19.3	18.0	10.1	4.8	1.2	6.0	16.1	77.5
Italy	372	5.3	50.7	7.6	0.0	7.1	9.9	6.6	10.7	23.7	63.6
Germany	385	3.2	16.5	31.1	6.3	21.7	10.1	8.1	3.0	39.9	57.1
Japan	483	3.9	34.2	32.9	5.4	1.0	10.2	4.5	7.6	15.7	76.4
Poland	635	2.7	7.0	69.2	0.0	11.0	4.6	4.3	1.2	19.9	79.0
World	436	3.1	22.1	35.7	9.2	7.5	4.5	2.4	15.2	14.4	70.1

Switzerland has already achieved an energy mix characterized by low carbon intensity, as shown in Table 1.3, where the 2022 energy mix of selected countries is reported together with the associated carbon intensity. For context, the International Energy Agency (IEA) countries average for 2021 was of 329 g CO2/kWh [1.4]. Countries in Table 1.3 who have achieved low emissions in their electricity generation all have a balanced mix of renewables and nuclear energy. Exception is Norway, whose geographical conditions allow for a very high share (close to 90%) of hydropower. Noteworthy is the last column of Table 1.3: excepting Norway for the reason just mentioned, all other countries have a considerable share of baseload power in their energy mix in the form of either nuclear or fossil fuels. Austria, which has a similar share of hydropower as Switzerland but a higher share of other renewables, still relies for almost 24% on fossil fuels. Germany also still exhibits higher emissions than the IEA countries average because fossil fuels still amount to more than 50% of Germany electricity production.

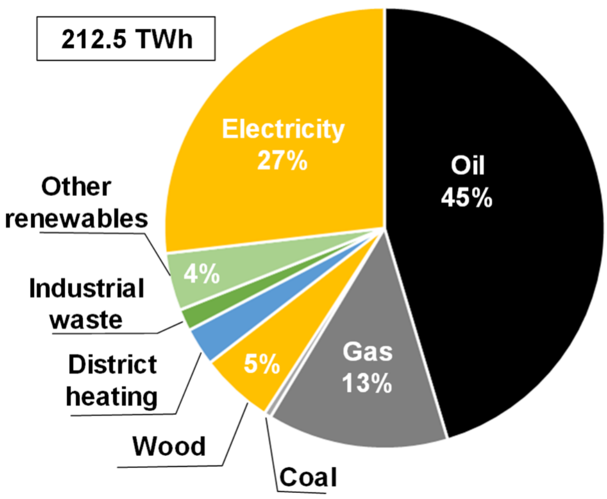


Figure 1.3 Energy mix in Switzerland in 2022 (BFE data)

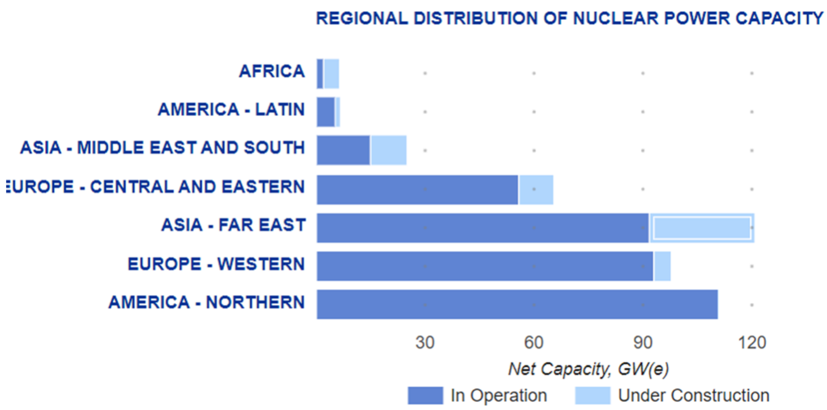
Electricity is only a part of the entire energy consumption.

The Swiss energy consumption for the year 2022 amounted to a total of 212.5 TWh. The share of each energy source is depicted in Figure 1.3. While the Swiss electricity production is already today almost CO2-free, close to 60% of the overall energy mix is still based on fossil fuel. Therefore, a strong increase of the electricity needs is to be expected if fossil fuels are to be replaced by electrification and synthetic fuels.

1.2 Nuclear Energy Worldwide

Nuclear energy remains the largest source of low-carbon electricity in the OECD countries (electricity share in 2022: 15.8% nuclear, 12.6% hydro, 9.9% wind, 5.9% solar) and the second largest source in the world, after hydropower. However, the by far largest fraction of electricity generation both in the OECD and worldwide stems from the burning of fossil fuels (49.7% in OECD countries and more than 60% worldwide). The picture gets even bleaker on consideration of the world's primary energy consumption, where more than 80% of the energy is being generated from fossil energy carriers, while nuclear accounts only for 4%, and wind and solar power together for merely 5%.

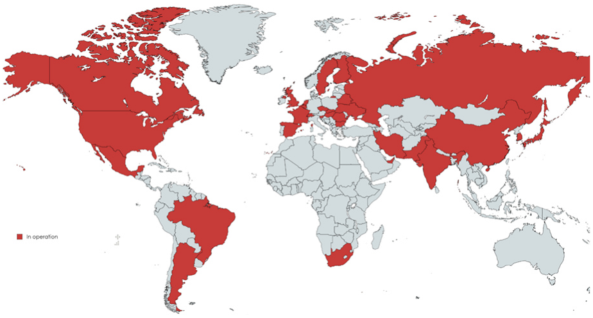
The IAEA's new annual nuclear power outlook high case projection issued in October 2023 projects installed nuclear capacity will more than double to 890 GWe by 2050, compared to today's 369 GWe installed capacity. This represents an almost 25% increase from the Agency's projections of 2020, with its projections revised up for a third consecutive year [1.5]. In the low case scenario, capacity is projected to increase to 458 GWe. The big drivers are climate change as well as security of energy supply.



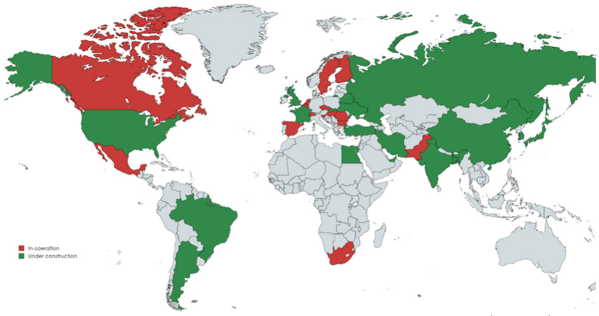
Number of operating NPPs as of March 2024

USA	94
France	56
China	55
Russia	37

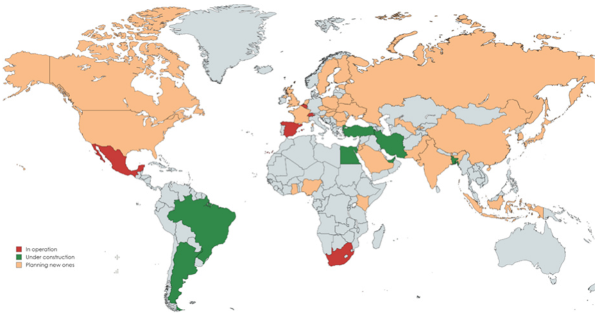
Figure 1.4 Regional distribution of nuclear power capacity (source: IAEA, Pris) and countries with the highest number of operating NPPs



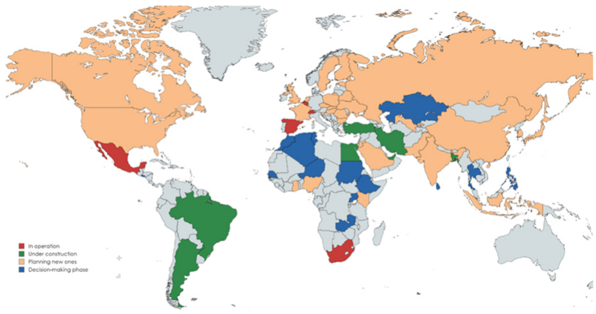
with NPPs in operation (red)



with NPPs in construction (green)



with NPPs planned (yellow)



in decision-making phase (blue)

Figure 1.5 Countries with NPPs in operation (red), in construction (green), planned (yellow). Countries in blue are in the decision-making phase.

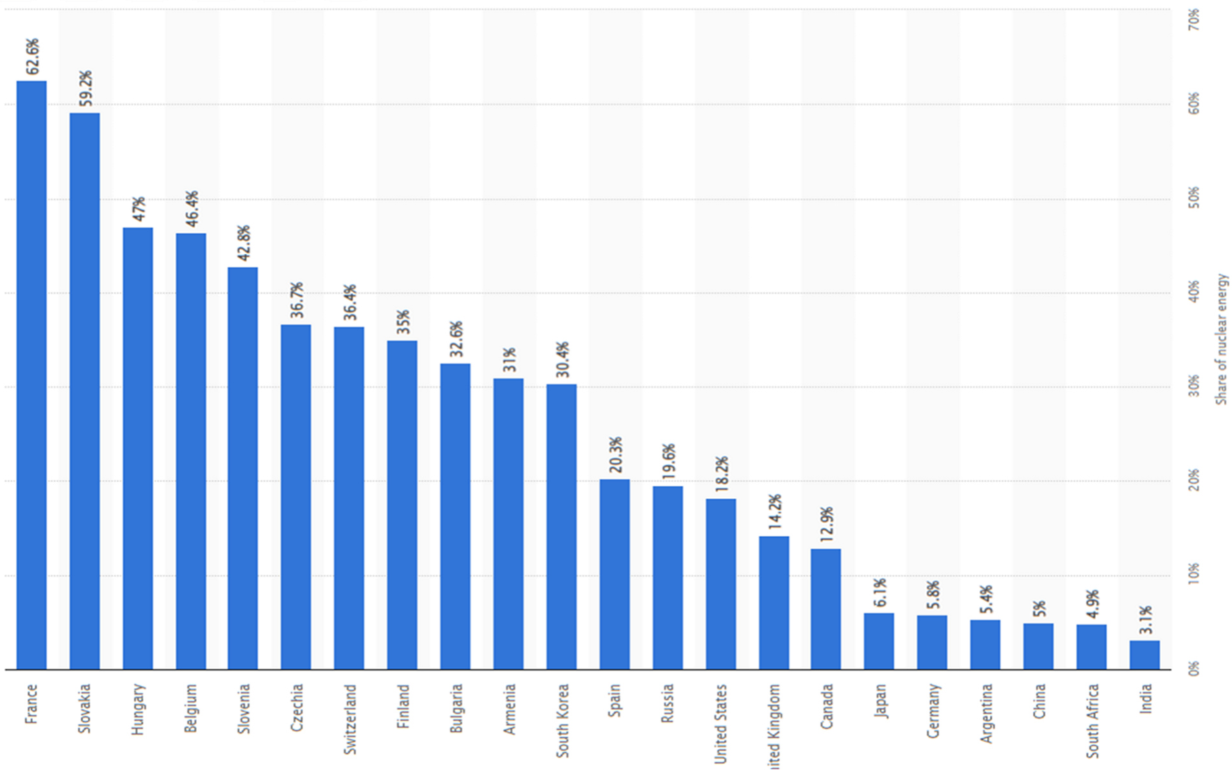


Figure 1.6 Countries with the largest share of nuclear energy in the electricity mix for year 2022 [1.6]

Table 1.4 Countries using or considering use of nuclear energy

Countries already using nuclear energy	
With NPPs in construction (15 countries)	Argentina, Belarus, Brasil, China, France, India, Iran, Japan, Russia, United Kingdom, Slovakia, South Korea, United Arabic Emirates, Ukraine, United States
With plans for construction of new NPPs (10 countries)	Bulgaria, Canada, Czech Republic, Finland, The Netherlands, Romania, Hungary, Pakistan, Slovenia, Sweden
No new NPPs planned currently (4 countries)	Armenia, Mexico, South Africa, Taiwan
With plans to phase out nuclear (4 countries) [Germany has no operating NPPs since April 2023]	Belgium, Germany, Spain, Switzerland
Newcomers	
With NPPs in construction (3 countries)	Bangladesh, Egypt, Turkey
With plans for construction of new NPPs (10 countries) [Poland and Saudi Arabia close to start construction]	Estonia, Ghana, Indonesia, Jordania, Kenya, Nigeria, Poland, Saudi Arabia, Uganda, Uzbekistan
In decision making phase (17 countries)	Algeria, Burkina Faso, El Salvador, Ethiopia, Indonesia, Kazakhstan, Morocco, Niger, Philippines, Rwanda, Senegal, Sri Lanka, Sudan, Thailand, Tunisia, Zambia, Zimbabwe

In total, as of March 2024, 415 NPPs are in operation worldwide, for a total of 373,257 GWe of installed capacity. In addition, 57 NPPs are in construction, providing an additional capacity of 59.22 GWe. In Europe, 167 NPPs are in operation (148 GWe) and 9 are in construction (10.1 GWe).

Globally, 32 countries are using nuclear energy, 13 new countries have plans to introduce nuclear energy in their electricity mix (3 of those countries have NPPs already in construction), and 17 additional countries are in the decision-making phase (see summary in Table 1.4, Figure 1.4 and Figure 1.5). Four countries are planning a phase-out of nuclear energy, with only Germany having ultimately terminated domestic generation in 2023. Spain plans a phase-out by 2035, while Belgium and Switzerland plan long term operation for their existing nuclear fleet.

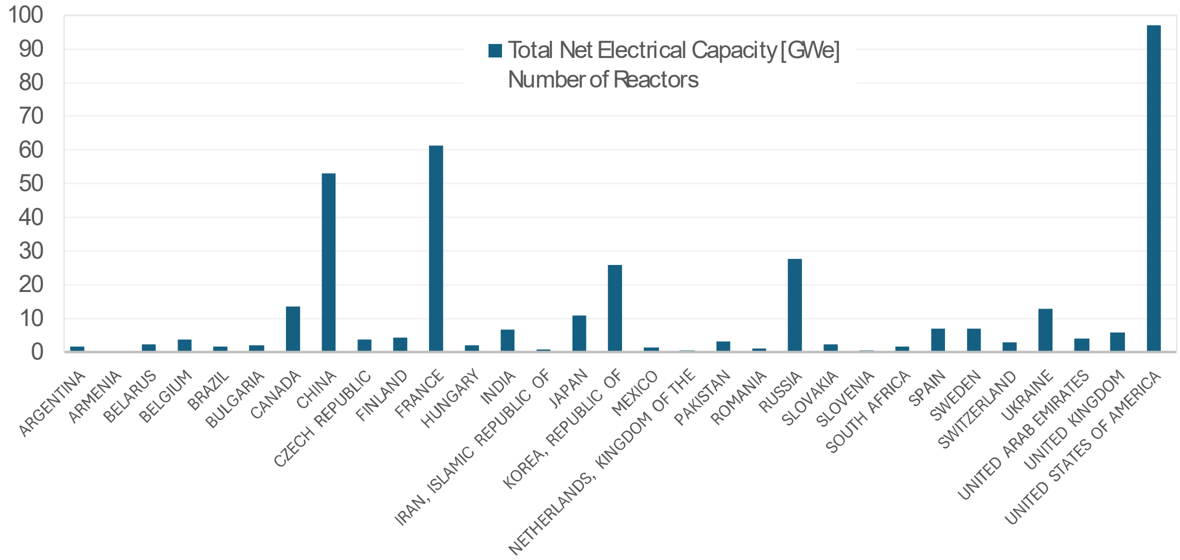


Figure 1.7 Reactors in operation and corresponding installed capacity by country (March 2024, IAEA Pris)

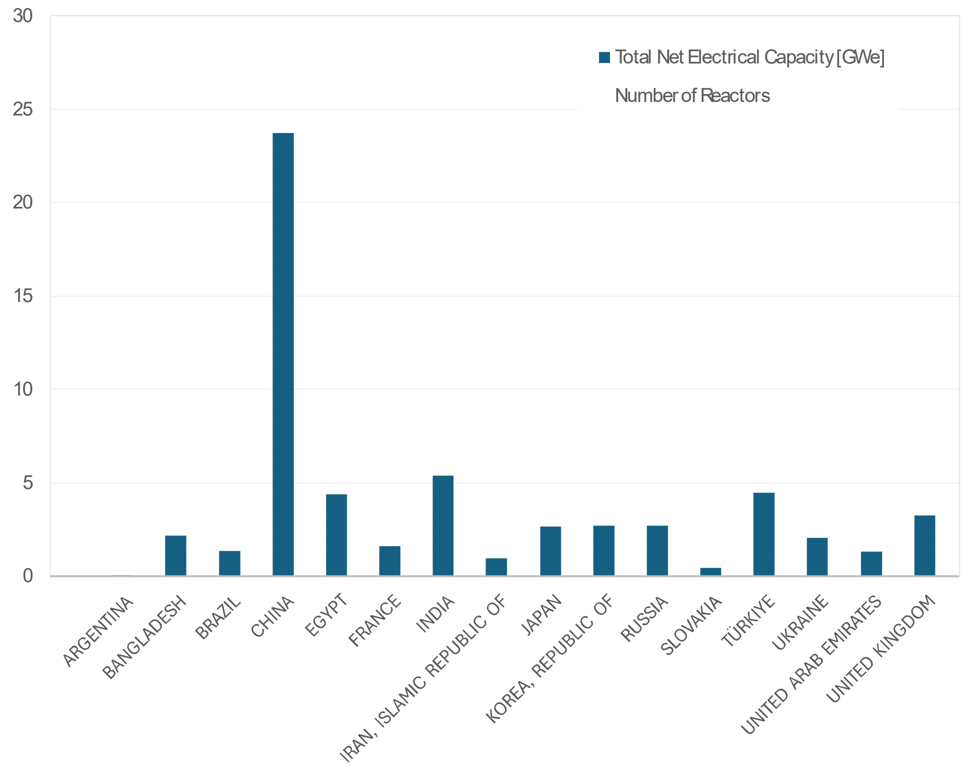


Figure 1.8 Reactors under construction and corresponding capacity by country (March 2024, IAEA Pris)

The regional distribution of countries where NPPs are in operation and under construction is reported in Figure 1.4, while the countries with the largest share of nuclear energy in their electricity mix for year 2022 are shown in Figure 1.6. The countries with the largest number of operating NPPs are USA, France, China and Russia (see Figure 1.7). The countries with the highest growth of nuclear energy as of March 2024 are instead (see also Figure 1.8):

- China: 27 NPP units under construction (30.9 GWe), 41 planned (44.7 GWe), 158 proposed (> 180 GWe);

- India: 7 NPP units under construction (5.9 GWe), 12 planned (8.4 GWe), and more than 50 proposed;
- Turkey: 4 NPP units under construction (4.8 GWe), 8 proposed (9.6 GWe);
- Egypt: 4 NPP units under construction (4.8 GWe);
- South Korea: 2 NPP units under construction (2.7 GWe), 2 planned (2.8 GWe);
- Russia: 4 under construction (4.0 GWe), 14 planned (8.9 GWe), 36 proposed (37.7 GWe).

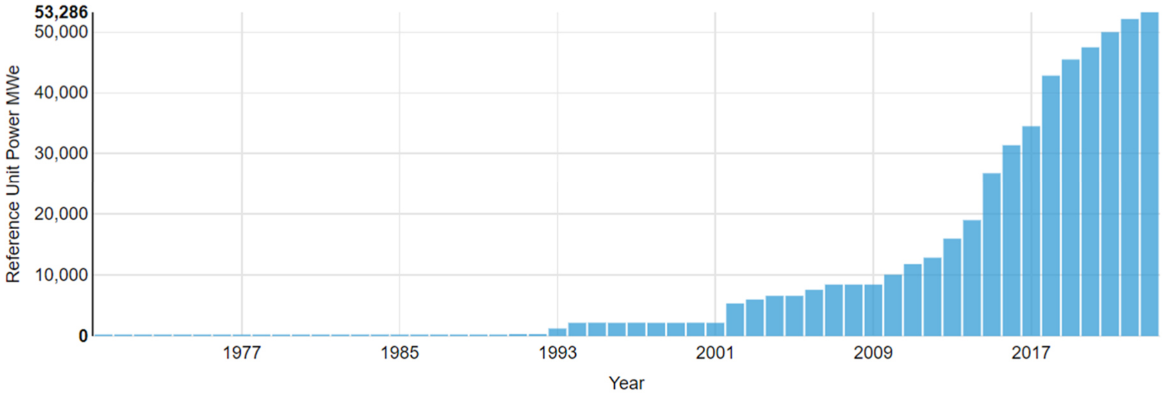


Figure 1.9 Growth of nuclear energy capacity in China [1.7]

Noteworthy is not only the rapid growth in China, shown in Figure 1.9, but also the recent example of the United Arab Emirates (UAE). Here KEPCO (South Korea) has built four large NPPs over the past 9 years, yielding 26 TWh produced in 2022 (double as much as UAE production from solar energy) and on track to produce more than 40 TWh/y with the fourth NPP unit starting operation in 2024.

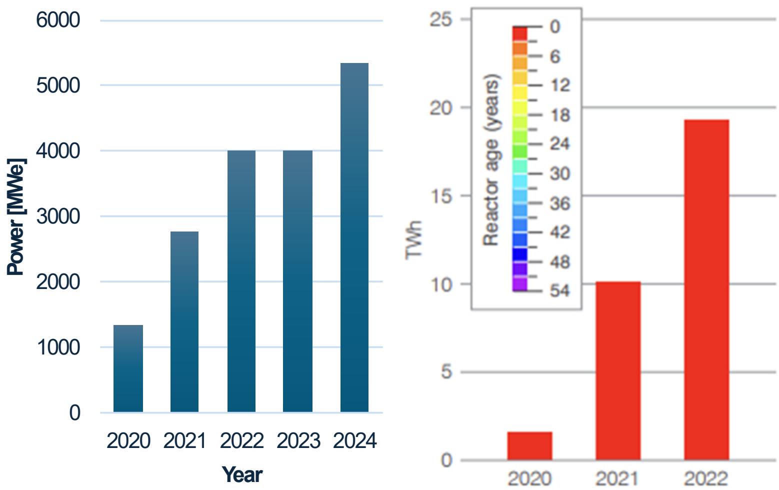


Figure 1.10 Growth of nuclear energy capacity in MWe (left) and generated power in TWh (right) in United Arab Emirates [1.7]

Recently, especially in view of the changes in the geopolitical landscape caused by the Ukraine war, several countries are revising their plans on nuclear energy. Worth mentioning:

- in July 2023, an EU nuclear alliance was launched by France together with 15 other countries (France, Belgium, Bulgaria, Croatia, Czech Republic, Finland, Hungary, Netherlands, Poland, Romania, Slovenia, Slovakia, Estonia, Sweden, Italy, UK) with the plan to develop an integrated European nuclear industry, and the goal of reaching 150 GWe of nuclear energy in the EU electricity mix by 2050.
- at the recent United Nation Climate Change Conference (COP28, United Arab Emirates, November 2023), 22 countries have launched a declaration to triple nuclear energy by 2050 in order to reach the new zero goal. These countries include the United States, Bulgaria, Canada, Czech Republic, Finland, France, Ghana, Hungary, Japan,

Republic of Korea, Moldova, Mongolia, Morocco, Netherlands, Poland, Romania, Slovakia, Slovenia, Sweden, Ukraine, United Arab Emirates, and United Kingdom;

- in February 2024, the European Commission has established a new European Industrial Alliance aiming to accelerate the development, demonstration and deployment of Small Modular Reactors (SMRs) in Europe by the early 2030s.
- US has initiated an aggressive investment plan to foster development of SMRs and microreactors and their deployment in US as well as abroad. The Inflation Reduction Act signed in 2022, aims at providing support for existing and new NPPs through investment and tax incentives for both large existing NPPs as well as newer advanced reactors, for uranium fuel and hydrogen production. Several NPPs have had their lifetime extended (e.g. Diablo Canyon in California, which was supposed to be shut down in 2022. Six reactors have had their lifetime extended to 80 years. Several others are expecting decision from the nuclear authority). Remarkable, the state of Michigan is reopening the Palisades NPP which was shut down since 2022. A few pilot projects have been successfully started at existing NPPs to use nuclear energy for hydrogen production.
- the Japanese government has approved a bill in May 2023 aimed at promoting the “green transformation” that recognizes nuclear as "a power source that contributes to energy security and has a high decarbonisation effect". Accordingly, Japan is planning to extend the lifetime of their current NPP fleet and to build new NPPs to replace the ones being decommissioned;
- The government of South Korea has approved a plan to increase nuclear power electricity share from the current 27.4% to 34.6% by 2036.
- In India, the government has approved 10 NPPs, with a plan for 20 NPPs by 2031.
- Sweden has removed the existing ban to new NPPs and has changed the 2040 electricity targets from 100% renewables to 100% fossil-free, with the plan of building new NPPs;
- France has reverted the goal to decrease the nuclear energy mix share down to 50% and is planning up to 14 new NPPs;
- Poland has started transitioning from coal to nuclear. 6 NPP units have been already ordered, with plans for additional ones under development.
- In March 2024, a parliamentary motion was passed in The Netherlands to support the construction of at least four new large NPPs.
- Tenders are currently on-going for new NPPs in Czech Republic and Saudi Arabia
- A public-private partnership in Romania has completed site selection for 6 SMR units, having received a 3 billion USD loan from USA.
- The UK government has approved the construction of two new NPP units (Sizewell C), additional to the two units currently under construction at Hinkley Point.

In addition, to eliminate the dependence from Russia enriched uranium supply, new supply chains are currently being built in the US and in France, while the capacity of existing enrichment plants both in Europe and US has been increased. US has also opened three new uranium mines in 2023. More details are discussed in chapter 6.

The importance of nuclear energy in the current geopolitical landscape is also evident by the agreements taking place for the financing and construction of new nuclear power plants in emerging countries, as summarized in Table 1.5. State-owned nuclear companies in Russia and China have taken the lead in offering nuclear power plants to emerging countries, usually with finance and fuel services [1.7]. This because Russia and China are not bound by OECD guidelines on minimum interest rates and loan repayment terms [1.9], allowing them to offer more attractive export financing packages and affording a distinct advantage in competing for overseas markets. Moreover, state-owned competitors, such as Russia and China, will make equity investments into nuclear exports, again bestowing them a competitive edge.

Recognizing the current export trends, the US government has launched several activities to support nuclear programs in emerging countries, aimed at fostering the export of US nuclear technology. In particular, US has engaged with Poland on new NPPs construction, has established a strategic joint partnership with Japan and Ghana on SMRs deployment, has enabled potential investment drivers for new nuclear in the Partnership for Global Infrastructure and Investment (PGII) and the US-UAE Partnership to Accelerate Transition to Clean Energy (PACE). Export-Import Bank (EXIM) and DFC have issued Letters of Interest (LOI) pledging potential support ranging in the billions of dollars for nuclear projects in Romania and Poland. EXIM has issued a $3 billion LOI for the construction of the GE-Hitachi BWRX-300 in Poland. The US Trade and Development Agency (USTDA) has been actively involved in project preparation work such as feasibility and front-end engineering and design (FEED) studies, including awarding a grant to Romania's RoPower for a FEED study for the country's first SMR plant in October 2022. The US government has also launched the Foundational Infrastructure for Responsible Use of Small Modular Reactor Technology (FIRST) aim to build long-term relationships and engage in capacity building work.

Table 1.5 Agreements with emerging countries on new NPPs [1.7]

Russia	Jordan, Egypt, Tunisia, Algeria, Morocco, Nigeria, Ghana, Ethiopia, Sudan, Zambia, Kazakhstan, Venezuela, Bolivia, Paraguay, Myanmar, Indonesia, Vietnam, Laos, Cambodia, Philippines, Cuba, Uzbekistan, Rwanda, Burundi, Azerbaijan, Congo, Cuba, Sri Lanka, Uganda
China	Sudan, Kenya, Thailand, Cambodia
Others	Poland, Lithuania, Philippines, Kenya, Uganda, Ghana, Romania

1.3 Technological development of nuclear energy

Nuclear technology for commercial power generation can be classified into four generations, as illustrated in Figure 1.11.

Gen-I: prototype commercial NPPs built between the 1950s and 1960s. As of today, all of the Gen-I NPPs have been decommissioned.

Gen-II: commercial NPPs which started operation in the 1970s and 1980s, designed with an operational lifetime of ~ 40 years. They include Light Water Reactors (LWRs), both Pressurized (PWRs) and Boiling Water Reactors (BWRs), the Advanced Gas Cooled Reactors (AGRs) deployed in UK, the VVER (a type of PWR) and RBMK reactors, both of Russian design, and the CANDU heavy water reactors developed in Canada. Gen-II constitutes the bulk of the NPPs operating today. Following technological advancements, retrofitting and more accurate methodologies for the estimation of safety margins, the lifetime of many Gen-II NPPs operating today has been extended to 60 years and beyond. All Swiss NPPs belong to this generation of power plants.

To note that the Swiss NPPs have been routinely upgraded and retrofitted over the years based on state-of-the-art knowledge, such that several years before the Fukushima accident of 2011, they already featured bunkering of the emergency Diesel and safety systems for enhanced protection from external events, hydrogen recombiners and containment filtered venting.

Also in Japan, noteworthy is the example of the Onagawa NPP (3 units), which was the closest plant to the epicenter of the earthquake that ravaged Japan in 2011 and lead to the Fukushima accident in the Daiichi NPP, but was designed to cope with a tsunami of +14.8m, compared to the 5.6m of Daiichi NPP. Not only the safety of the three units of the Onagawa NPP was not adversely affected by the earthquake and consequent tsunami, but the premises of the power plant were subsequently used as shelter for several hundreds of residents of the neighboring areas, who remained at the plants for several months following the earthquake.

Gen-III/III+: these NPPs are evolutionary designs of Gen-II plant types and have started to be built since the early 1990s, with Gen-III+ plants including further developments compared to Gen-III. While water is still used as coolant, their design is based on a radical change in the safety philosophy, aimed at keeping the effects of any accident within the plant site boundary, with substantial releases beyond the site boundary rendered highly unlikely (probabilities down to 10^{-8} – 10^{-9}/year). Passive safety systems aimed at preventing or mitigating the effects of a core melt are included in Gen-III/III+ reactors. The core damage frequency (CDF) is down to 10^{-6} – 10^{-7}/year. More details on the safety philosophy and design characteristics of Gen-III/III+ reactors are given in chapters 0 and 3.2. These designs deploy the concept of Design Extension Conditions (DECs), according to which severe accidents become integral part of the NPP design and are addressed with a balanced combination of passive and active safety systems. More details are given in Chapter 0.

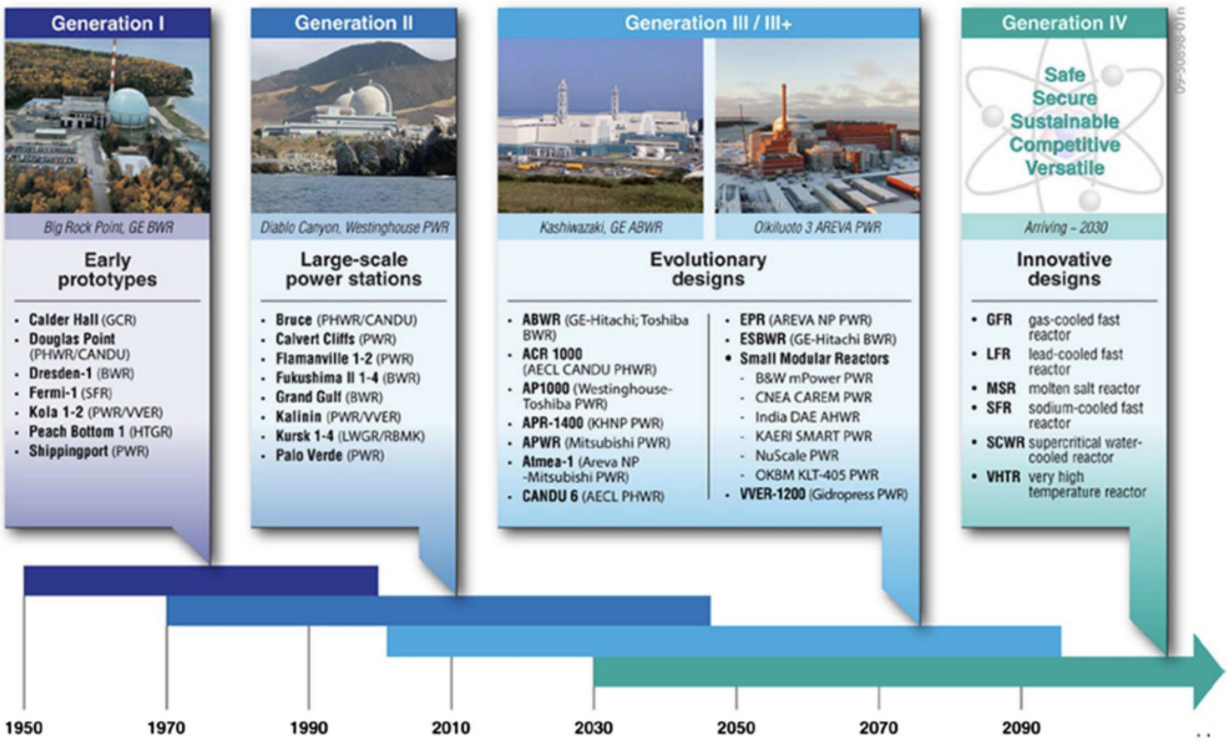


Figure 1.11 Evolution of nuclear reactors technology (www.gen-4.org)

Gen-III/III+ NPPs are based on advanced designs featuring improved safety and economics, whereas Gen-III+ plants include further developments. Distinctive characteristics of Gen-III/III+ designs, aimed at improving safety and economics, include [1.8]:

- a simpler and more robust design, making the reactors easier to operate and less vulnerable to operational disturbances;
- significant use of passive safety features that require no active controls and rely on natural phenomena;

- reduced probability of occurrence of accidents involving core melting;
- new mitigation measures in case of core melt accidents, to reduce significantly the impact of such accidents to the environment and to the public;
- resistance to the impact of a large aircraft;
- longer time interval between refueling, resulting in a higher availability;
- higher burn-up to increase fuel use and reduce the amount of waste produced;
- longer operating lifetime, of 60+ years, already from design.

Gen-IV: these are revolutionary concepts using alternative coolants (gas, liquid metals, salts), typically with fast neutron spectrum. Because of the higher operating temperatures, they have higher thermodynamic efficiency in the conversion of thermal to electric power. They are specifically designed to improve fuel efficiency, acting also as breeders and actinides burners, and tend to have a closed fuel cycle in association with reprocessing and recycling of both plutonium and minor-actinides. This results in a considerable decrease of the amount of produced waste and decay time. Details on open and closed fuel cycles are given in chapters 6 and 7.

An additional classification of the reactors which are currently on the market or in advanced state of development is according to size (see also Figure 1.12):

- Large and intermediate size NPPs with capacity above 300 MWe
- Small Modular Reactors (SMRs) with capacity between 10 and 300 MWe
- Microreactors with capacity below 10 MWe

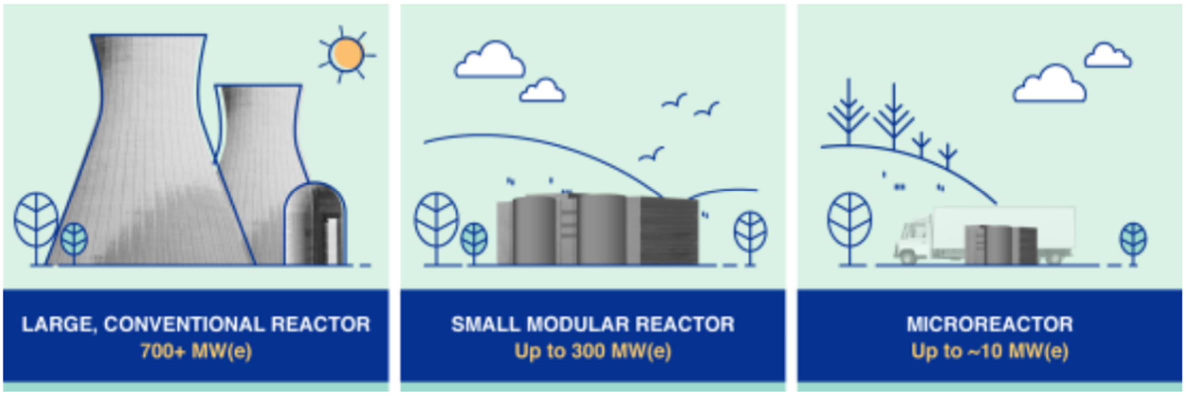


Figure 1.12 NPP classification by power output (Image: A. Vargas/IAEA)

1.4 Economics

Compared to other baseload energy sources, nuclear power is very capital intensive and with low marginal costs (low fuel costs, plus low variable operation and maintenance costs) [1.10]. Therefore, capital costs have a dominant role (60% or more) in the determination of the levelized costs of electricity (LCOE). The cost of fuel per energy generated is very low compared to other thermal plants such as gas and coal and has little impact on the overall LCOE for nuclear power. An updated analysis of the costs of different electricity production technologies for Switzerland, including nuclear, was carried out by PSI in 2019 and published in a report of the Swiss Federal Office of Energy [1.10]. The results are shown in Figure 1.16 and Figure 1.17 and Table 1.6. An average load factor of 85% was assumed for nuclear. To note that in Table 1.6 the overnight capital costs are reported, which do not include interests. The costs reported in the news instead typically include the interests costs as well, which might generate confusion on the economics of NPPs.

Table 1.6 Nuclear power plant costs for Switzerland [1.10]

Nuclear power		Currently operating plants in CH	New plants (hypothetical new, Gen III/III+)	2035 (SMR)
Electricity generation potential	TWh/a	not applicable		
Investment costs	CHF/kW	1'300-6'000	4'000-7'000	3'000-9'000
Electricity generation costs	Rp./kWh	4-6	7.5 (5.1 - 12.5)	7.4 (5.1 - 12.2)

In Table 1.6, a cost of 7.5 Rp/kWh is obtained assuming a EPR with a construction period of 6 years and a 6% interest rate. The costs are strongly affected by the time it takes to build the plant. With a construction time of 9 years, the cost would rise to 8.8 Rp/kWh. The PSI results are in line with the LCOE values published by OECD [1.11] and MIT [1.12]. The overnight costs of recent Gen-III+ NPPs are shown in Figure 1.13 [1.12]. The large variability in the figure for the NPPs built in Europe and USA is mostly associated to first-of-a-kind issues in regions where no NPPs have been built for over three decades. South Korea and China reports much lower costs that only partly can be justified with lower labor costs.

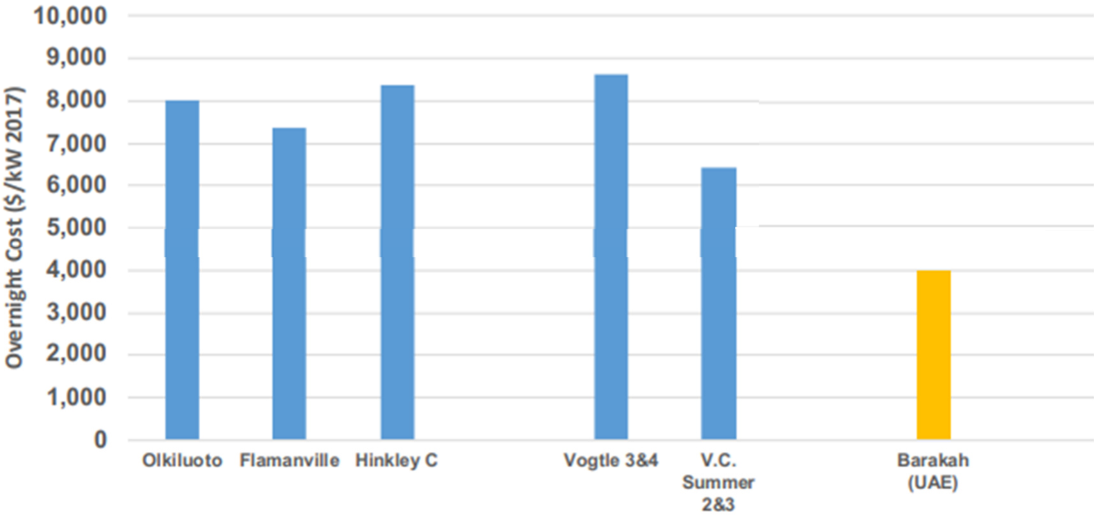


Figure 1.13 Overnight cost of recent Gen-III+ NPPs [1.12]

Because nuclear is capital intensive, the LCOE costs of nuclear will strongly depend on the interest rates on the capital and on the construction time (as loan interests would have to be paid for a longer time before the NPP starts producing electricity and therefore profits). An important reasons for lower costs in China is given by the design standardization and the establishment of a functioning supply chain and trained workforce. The construction times for the NPPs built in China over the last several years are shown in Figure 1.14. Similar results had been obtained in France in the decades between 1970s and 2000, again through a combination of standardized design and supply chain availability (see Figure 1.15).

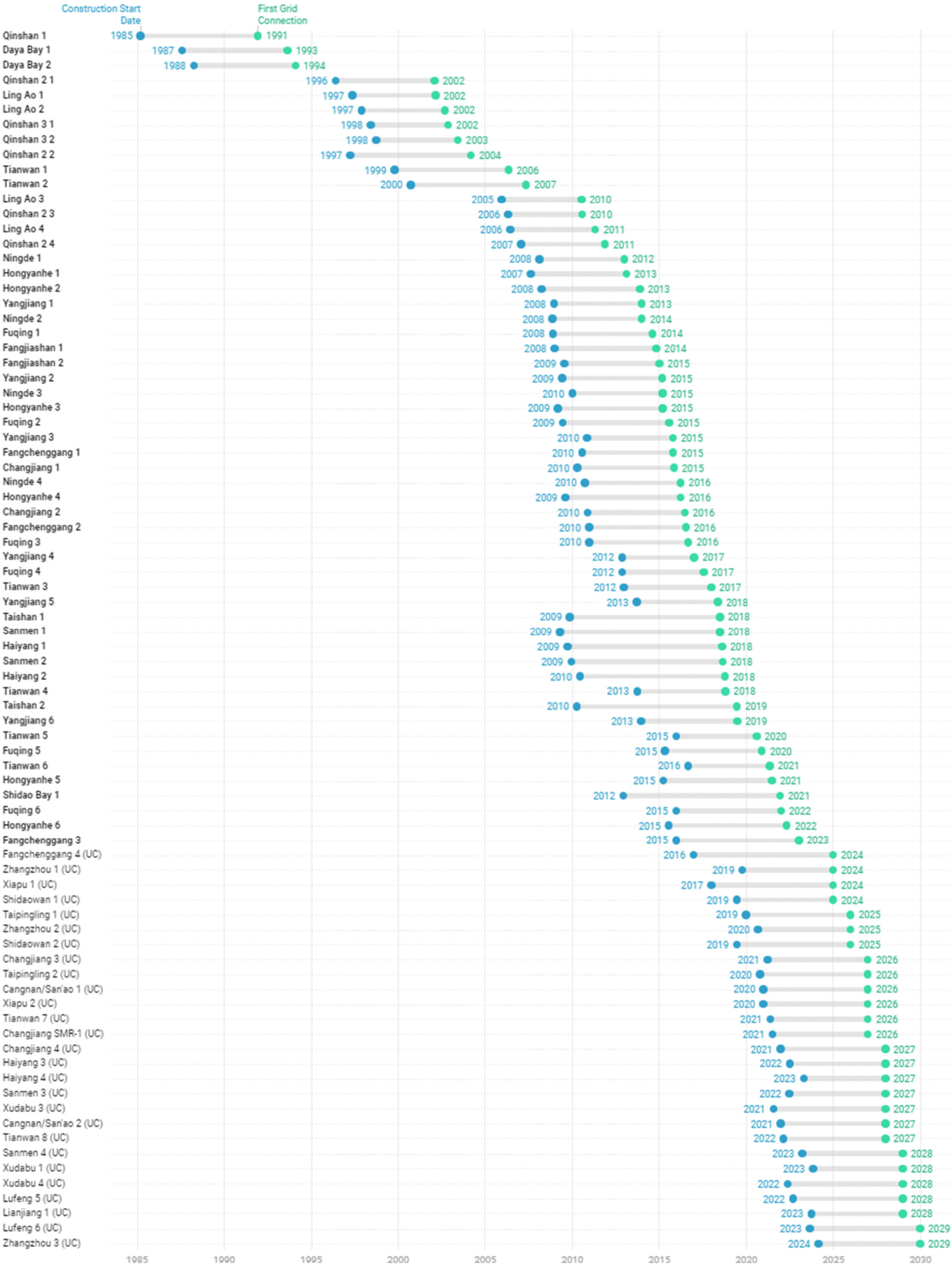


Figure 1.14 Construction time for NPPs in China ([1.7] and IAEA Pris)

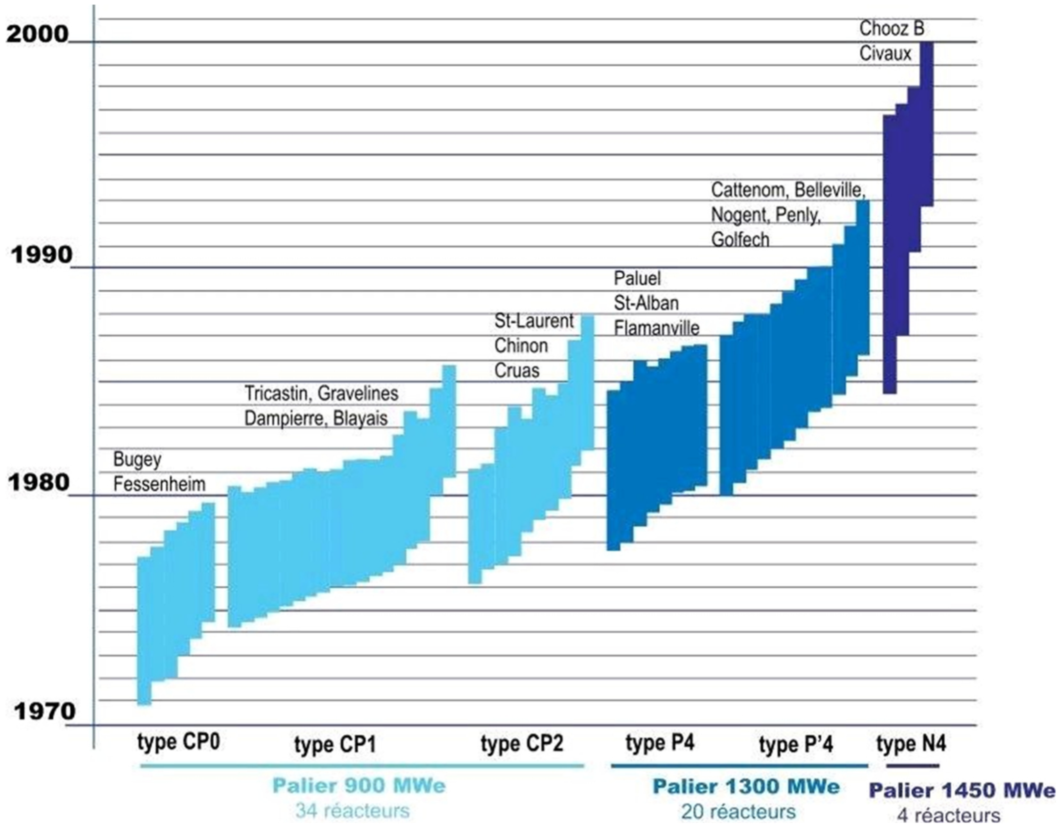


Figure 1.15 Construction time for NPPs in France

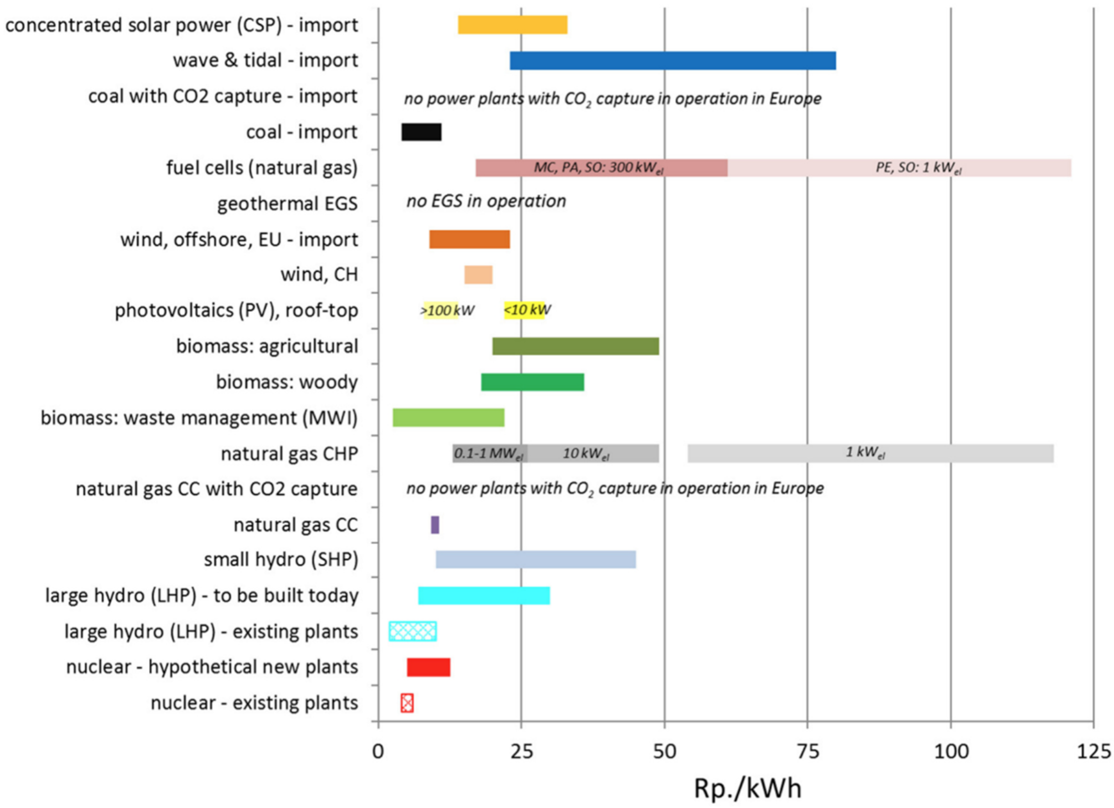


Figure 1.16 : LCOE for different energy sources (year 2018) [1.10]

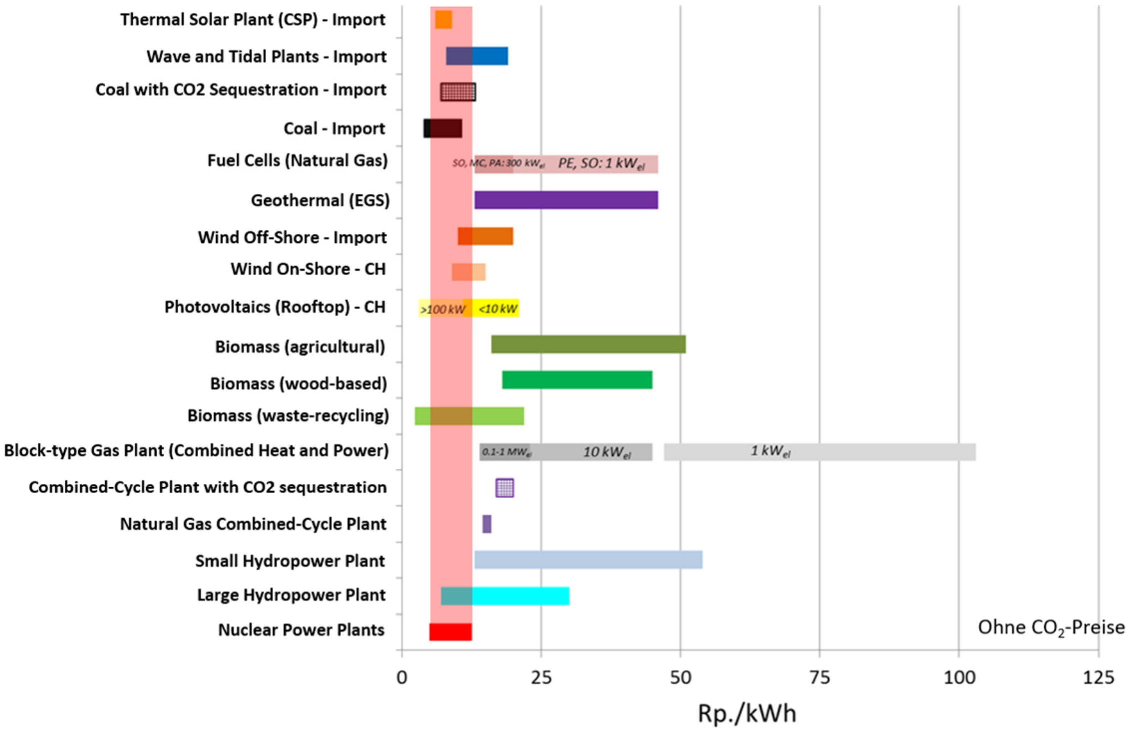


Figure 1.17 : LCOE for different energy estimated for year 2050 [1.10]

The LCOE for SMRs are forecasted to reach levels similar to large NPPs, by replacing the economy of scale with the economy of manufacturing (factory-built), though higher costs need to be anticipated for first-of-a-kind units. The first NuScale SMR project that was supposed to be built in Utah in the coming years had forecasted costs of about 5.8 ct/kWh (2020 estimate). Due to a 150% increase in interest rates and significant increase of material costs (e.g. 40% cost increase for steel) over the past 1.5 years, the costs were forecasted to rise to 8.9 ct/kWh in 2023. The project was subsequently cancelled because the forecasted price was no longer competitive with the cheaper gas and coal available to Utah.

It has to be mentioned that in recent years, several concerns have been raised on the applicability of the LCOE metric when comparing various energy sources (Ref. [1.12] to [1.15]). This because the LCOE metric was initially developed to compare the costs of dispatchable baseload plants (e.g. nuclear, coal, gas, hydro) and it is not appropriate for comparing with intermittent energy sources (e.g. wind, solar). A consensus is emerging for considering other costs components of the overall "energy system", including grid costs (distribution and transmission), balancing costs (both short-term, as well as long-term costs to maintain adequate backup capacity), utilisation costs (profile costs or back-up costs) [1.13]. Often neglected, connection costs may be significant, especially if distant resources have to be connected to the grid [1.13].

Cometto et al. [1.13] published a study on the system costs due to back-up, balancing, and grid connection and extension, averaging data over six countries (Finland, France, Germany, South Korea, United Kingdom, and USA). The results, reported in Table 1.7, show an increase in system costs with increasing renewable penetration. Idel [1.14] has recently introduced an alternative metric to LCOE, the so-called Levelized Full System Costs of Electricity (LFSCOE), which compares the costs of serving the entire market using just one source plus storage. The results from Idel's study [1.14] performed specifically for Germany and the Texas electricity markets (ERCOT) are summarized in Figure 1.18. Similar studies have been also published in Ref. [1.15], with results summarized in Figure 1.19 representing the average grid level costs for various energy sources in the presence of 10% and 30% share of variable renewable energy (solar and wind). These results, though affected by various level of uncertainties and variability depending on the particular country energy mix, all conclude that system costs of dispatchable technologies, such as coal, gas, nuclear power or hydro, are much lower than the ones associated with VRE and increase with increasing share of VRE.

While the LFSCOE metric introduced by Idel might not be the most appropriate, as energy systems are not made out of single energy sources, it demonstrates the current open debate on how to adequately compare different energy sources with very different availability characteristics. A study on energy scenarios specific to Switzerland that attempts to include all system costs has recently been published by OECD [1.16]. The study considered a scenario with 2.2 GWe available through long term operation (LTO), one in which the new demand of energy supply is covered only with variable renewable energy (VRE, with 90% solar and 10% wind), a scenario with 3.2 GWe of new NPP by 2050 with no LTO of existing NPPs, a scenario with 1.6 GWe of new NPP with the rest covered by solar, and a scenario with 2 GWe from new gas plants combined with a carbon price of USD 100 per tonne of CO_2 and the rest provided by VRE. The resulting system costs of the various scenarios for different levels of interconnection with Europe are summarized in Figure 1.20, and indicate that, excluding LTO, replacing the current Swiss nuclear power plants with new ones would result in the lowest system costs and in net revenues from trading (negative orange bars in the figure), while the VRE and gas scenarios would result in higher system costs and net trading costs (positive orange bars). The results are exacerbated with decreasing access to trading with EU.

Table 1.7 System costs (average for 6 countries: Finland, France, Germany, UK, USA, South Korea) [1.13]

	System Costs at the Grid Level [USD/MWh]											
Technology	***Nuclear***		***Coal***		***Gas***		***On-shore wind***		***Off-shore wind***		***Solar***	
Penetration level	*10%*	*30%*	*10%*	*30%*	*10%*	*30%*	*10%*	*30%*	*10%*	*30%*	*10%*	*30%*
Back-up Costs (Adequacy)	0.00	0.00	0.05	0.05	0.00	0.00	6.03	7.38	5.71	7.67	15.88	18.04
Balancing Costs	0.53	0.35	0.00	0.00	0.00	0.00	4.19	8.34	4.19	8.34	4.19	8.34
Grid Connection	1.71	1.71	0.94	0.94	0.51	0.51	6.24	6.24	18.68	18.68	13.71	13.71
Grid Reinforcement and Extension	0.00	0.00	0.00	0.00	0.00	0.00	2.23	6.28	1.51	3.82	4.46	13.55
Total Grid-Level System Costs	***2.24***	***2.05***	***0.99***	***0.99***	***0.51***	***0.51***	***18.69***	***28.24***	***30.11***	***38.51***	***38.25***	***53.64***

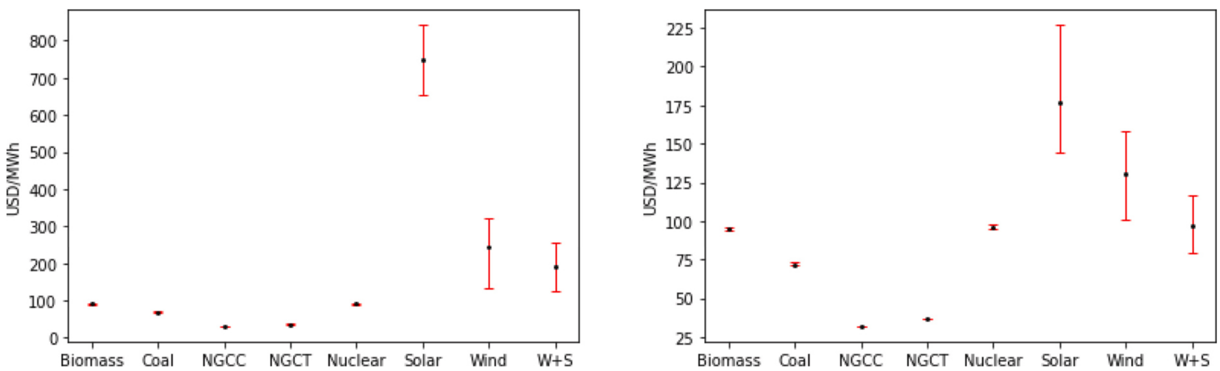


Figure 1.18 NPP LFSCOE95 Mean and Variance in the German (left) and Texas ERCOT market [1.14]

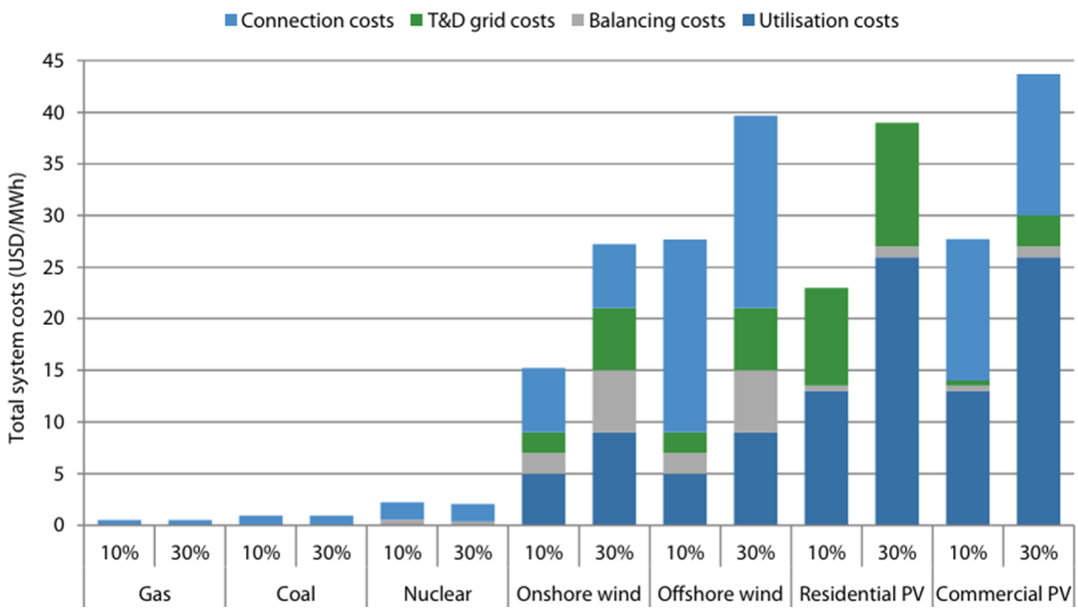


Figure 1.19 Grid-level system costs of various generation technologies for shares of 10% and 30% of variable renewable energy (VRE) generation [1.15]

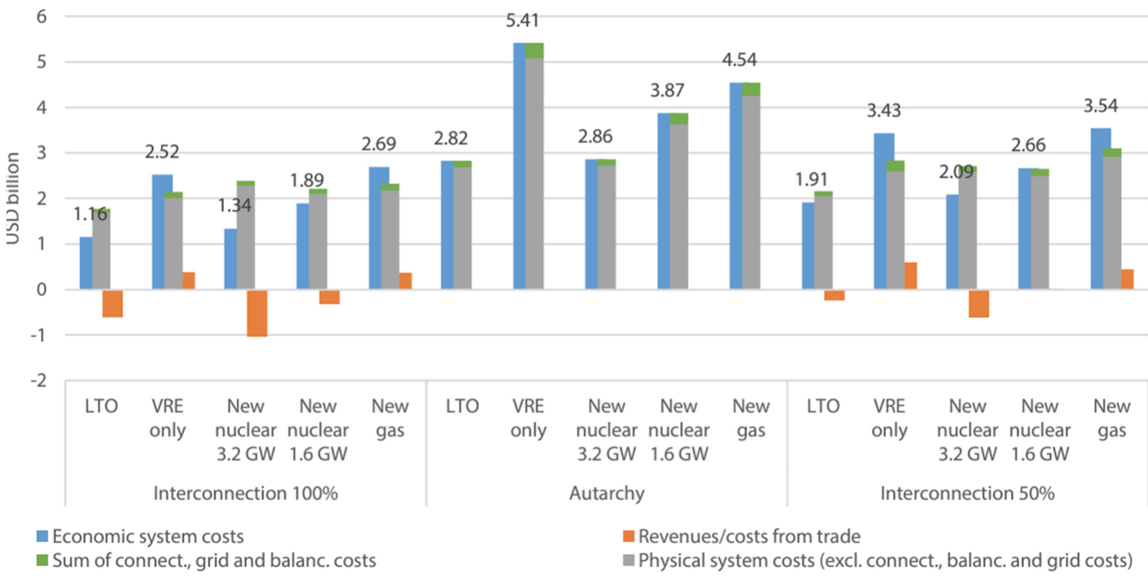


Figure 1.20 System costs of net zero scenarios for Switzerland under different trading with EU [1.16]. 100% interconnection: same level of interconnection capacity with EU as in 2022; autarchy: no interconnection with EU; 50% interconnection: 50% of interconnection capacity with EU compared to 2022 levels.

1.4.1 Factors affecting the economics of nuclear power plants

As mentioned previously, one of the main economics challenges of nuclear energy is associated with the high capital costs, which reduces the number of potential private investors. Therefore, factors that affect capital costs, interest rates on the capital costs loans, and construction duration are going to have a strong impact on the overall LCOE of the particular NPP project. In particular several insights have been drawn from past experience (see Refs. [1.12] and [1.17]):

- **Supply chain:** US and Western Europe did not build NPPs for several decades. The supply chain was lost, together with work force and capabilities to manage large construction sites. This is not the case in countries where NPPs have been consistently built (China, Russia, South Korea). To note that, when a new NPP is built in a country that has no recent experience with such constructions, there are two choices that can be made: rely mostly on skilled workforce from abroad to minimize the risk of delays (as can be seen in construction projects in Egypt and Turkey) or train the local workforce. The latter case is especially advantageous in case more units are planned, such as the case in UK and Poland.

- **Project structure:** the way the project is structured matters. In successful projects there is more vertical integration, i.e. the company that designs and build the plant is often the same company that owns and operates the plant. In this kind of projects the supply chain is very much integrated in the design team. Taking as an example the recent Vogtle3&4 project in the US instead, there was a separation between the vendor (Westinghouse), the constructor and the utility. Several instances were registered in which components were designed by the vendor that the supplier could not deliver, leading to the need of additional design iterations. This generates issues especially if all the parties involved do not have the same incentives to complete the project on time. The situation is aggravated by the nuclear regulatory environment which provides very little flexibility to accommodate small changes during construction.
- **Civil work:** civil and structural design was also found to be a significant cost driver. In current large NPP projects, only 20-25% of the total capital cost is associated with the cost of the equipment (vessels, pipes, etc), while ~50% of the costs are associated with the actual installation (site preparation, civil work, site excavation, site oversight, etc.). The remaining 25% is divided between design and engineering (done almost entirely upfront and most site independently) and owner's costs (cost of land, insurance). While productivity in the manufacturing sector has increased with time, the productivity of the construction sector has instead decreased. Therefore, costs saving and reduction in construction times would be achieved by shifting a larger share of the project from construction to manufacturing. This philosophy is at the basis of SMR designs for which more activities are shifted from the construction site to manufacturing in factories, similar to what successfully done with chemical plants.
- **Design completion:** the level of completion of the detailed design at the time of start of the NPP constructions has a considerable impact both on cost estimations and construction time. Many cost overruns during plant construction occur because the first-of-a-kind (FOAK) plant designs are not fully completed prior to beginning construction. As a matter of fact, there is a strong negative correlation between overnight construction costs (OCC) and percent design completion. The detailed design was not completed in all these plants built recently in Europe and US. In countries with success stories, 90-95% of the detailed design was completed at the time of plant construction start. This is a reason why building multiple units in the same country and on the same site is advantageous. An effective way to reduce costs is by building multiple units at the same site. In the Barakah project in the UAE, for example, a 40% reduction in labour costs was experienced between the construction of Units 1 and 4. In addition, multi-units NPPs tend to have lower operating costs per MWh.
- **Regulatory framework:** the interest rates on the capital are significantly affected by the financing scheme (e.g. state loan guarantee) and the regulatory framework. Licensing can have a significant impact on both costs and construction times of a nuclear power plant. Country-specific regulations that require changes to the standard design or to the manufacturing and verification procedures are unfavourable. Examples are the Olkiluoto EPR in Finland, and the Vogtle-3 AP-1000 in the USA, and Hinkle Point in UK for which changes in the design were required while the reactors were already in construction. In all these cases, the need to adapt the design to fulfil new regulations was a contributing factor to the construction delays and significant costs

increase. Another example of potential adverse effects on costs and construction times is given by the licensing framework existing in Switzerland, where the licensing of a new nuclear power plant could be stopped through appeals also at the last stage of the operation license, when the nuclear plant is already in construction. This on its own constitutes an enormous financial risk. Another financial risk in is due to the fact that the regulatory framework in Switzerland is based on a dynamic definition of regulations, leading to a higher chance of designs changes required during the plant construction.

- **Market:** the market in which the NPP operates is also strongly affecting the economic viability of NPPs. In states with deregulated electricity markets, nuclear power plant operators have found increasing difficulty competing with low-cost gas (e.g. shell gas in USA), and subsidized wind/solar power with priority grid access [1.7]. In Switzerland, state subsidies (up to 60% of investment costs) are available for solar and wind projects, in addition to the communalization of costs for grid expansion.
- **Interest rates:** the most important risks driving interest rates on nuclear investment are construction risks, price risks and political risks. De-risking NPP builds would allow investors both private and public to access capital at lower interest rates. In the past, this has been achieved through direct public financing, or through state loan guarantee programs. This was combined with long-term political commitment to nuclear power and regulated tariffs [1.18].

A summary of the financing models and the sources of financing deployed for recent and planned NPP projects worldwide is reported in Table 1.8 [1.19].

Table 1.8 Financing structures of recent and proposed NPPs [1.19]

Country	Plant Name	Capacity	Construction Start	Financing Model	Financing Sources
Bangladesh	Rooppur 1-2	2x1200MW$_e$	2017	State	Sovereign funding and export credit provided by Russian SOEs
Belarus	Belarusian 1-2	2x1100MW$_e$	2013	State	Sovereign funding and export credit provided by Russian SOEs
Brazil	Angra 3	1x1340MW$_e$	2010	Utility	Eletrobras Elecronuclear S.A. (SOE)
China	Changjiang 3	1x1000MW$_e$	2021	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Fangchenggang 3-4	2x1000MW$_e$	2015	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Fuqing 6	1x1000MW$_e$	2015	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Hongyanhe 5-6	2x1061MW$_e$	2015	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Sanaocun 1	1x1117MW$_e$	2020	State	CGN (CEO) equity, China Development Bank, Bank of China
China	Shidao Bay 1 (HTGR)	1x500MW$_e$	2012	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Taipingling 1-2	2x1116MW$_e$	2019	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Taishan 1-2	2x1750MW$_e$	2010	Utility, JV	EDF and CGN equity (SOEs) , China Development Bank, Bank of China, Société Générale
China	Tianwan 5-6	2x1000MW$_e$	2015	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Xiapu 1 (FBR)	1x642MW$_e$	2017	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Xudabu 3	1x1200MW$_e$	2021	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Zhangzhou 1-2	2x1126MW$_e$	2019	State	Chinese State/SOE equity, China Development Bank, Bank of China
Egypt	El Dabaa 1-4	4x1200MW$_e$*	2021*	State	Sovereign funding and export credit provided by Russian SOEs

Table 1.8 cont.

China	Xudabu 3	1x1200MW$_e$	2021	State	Chinese State/SOE equity, China Development Bank, Bank of China
China	Zhangzhou 1-2	2x1126MW$_e$	2019	State	Chinese State/SOE equity, China Development Bank, Bank of China
Egypt	El Dabaa 1-4	4x1200MW$_e$*	2021*	State	Sovereign funding and export credit provided by Russian SOEs
Finland	Olkiluoto 3	1x1720MW$_e$	2005	Utility (Mankala)	TVO (Cooperative) Equity and Credit, SEK and Coface (BPI France) (ECAs) and Commercial Bank Credit Facilities[67]
France	Flamanville 3	1x1650MW$_e$	2007	Utility	EDF (SOE) equity
Hungary	Paks II 1-2	2x1200MW$_e$	2021	Utility	MVM (SOE) equity funding and export credit provided by Russian SOEs
India	Kakrapar 3-4	2x700MW$_e$	2010	State	Indian State budget
India	Kundankulam 3-4	2x1000MW$_e$	2017	State	Indian State budget
India	Prototype Fast Breeder Reactor	1x500MW$_e$	2009	State	Indian State budget
India	Rajasthan 7-8	2x700MW$_e$	2011	State	Indian State budget
Japan	Ohma	1x1328MW$_e$	2010	Utility	J-Power (IOU) equity
Japan	Shimane 3	1x1325MW$_e$	2006	Utility	Energia (IOU) equity
Pakistan	Chashma 3-4	2x340MW$_e$	2005	State	Host government funding and Chinese (exporter) sovereign and bilateral financing
Pakistan	Kanupp 2-3	2x1100MW$_e$	2015	State	Host government funding and Chinese (exporter) sovereign and bilateral financing
Poland	Zarnowiec 1-6*	6x300MW$_e$*	2025*	Utility	PGE (SOE) and USDFC (ECA) Credit Facility
Romania	Cernavodă 1-4 (LTO & New Build)*	2x650MW$_e$ 2x720MW$_e$*	2022*	Utility	SN Nuclearelectrica (SOE) and USDFC (ECA) Credit Facility
Russia	Akademik Lomonsov 1-2 (Floating)	2x30MW$_e$	2007	State	Rosatom Group (SOE) Russian state funding
Russia	Baltic 1	1x1109MW$_e$	2012	State	Rosatom Group (SOE) Russian state funding
Russia	Kursk II 1-2	2x1175MW$_e$	2018	State	Rosatom Group (SOE) Russian state funding
Russia	Leningrad II 1-2	2x1066MW$_e$	2008	State	Rosatom Group (SOE) Russian state funding
Russia	Novovoronezh II 1-2	2x1100MW$_e$	2008	State	Rosatom Group (SOE) Russian state funding
Russia	Rostov 3-4	2x1000MW$_e$	2008	State	Rosatom Group (SOE) Russian state funding
Saudi Arabia		2x1000MW$_e$*	2025*	State*	TBD – Ongoing competitive process for technology vendor(s) and potential financing sources.
S. Korea	Shin Hanul 1-2	2x1340MW$_e$	2012	Utility	KHNP (KEPCO) (SOE) equity
S. Korea	Shin-Kori 5-6	2x1340MW$_e$	2017	Utility	KHNP (KEPCO) (SOE) equity
S. Korea	Shin-Wolsong 1-2	2x997MW$_e$	2007	Utility	KHNP (KEPCO) (SOE) equity
Turkey	Akkuyu 1-3	3x1200MW$_e$	2018	BOO (PPA)	Rosatom Group (SOE)
United Arab Emirates	Barakah 1-4	4x1345MW$_e$	2012	Utility (PPA), JV with Sovereign and ECA Financing and Support	ENEC & KEPCO Equity (SOEs), Government of Abu Dhabi Direct Loan, First Gulf Bank, National Bank of Abu Dhabi, HSBC, Standard Chartered, KEXIM and USEXIM (ECAs) Loans

Table 1.8 cont.

United Kingdom	Hinkley Point C	2x1630MW_e	2018	Utility (CfD), JV	EDF & CGN equity (SOE)
United States	Vogtle 3-4	2x1250MW_e	2013	Utility	Southern Company (IOU) equity and USDOE Loan Guarantees
Uzbekistan		2x1200MW_e*	2023*	State	Sovereign funding and export credit provided by Russian SOEs

1.4.2 Government subsidies

It often exists the misconception that nuclear energy is heavily subsidized. However, the involvement of the government in the construction of new nuclear power plants is mostly through investments (i.e. government, as an investor, has equity in the power plant and therefore participates in the revenues) or through policies (e.g. a state loan guarantee).

Subsidies can be in the form of [1.20], [1.21]:

- preferential tax treatments (resulting in tax expenditures for the state)
- direct expenditures to recipients (subsidies contributions to produces and consumers) through grants, or low-interest and preferential loans;
- research and development (R&D) grants;
- Income or price supports (most of the mechanisms in this category can be considered as cross-subsidies, as they consist in transferring amounts of money from groups of people / technology / territory to another specific group. Most often, such measures are financed through final consumers' tariffs/prices [1.21]. Examples are capacity payments, biofuels blending mandates, renewable energy quotas with tradable certificates, differentiated grid connection charges, energy efficiency obligations, interruptible load schemes, contract for difference, feed-in premiums, feed-in tariffs, consumer price guarantees (cost support and price regulation) and producer price guarantees /price regulation [1.21]);
- loan guarantee (a pledge by the state to become liable for part of the debt obligation if the borrower defaults. This financial instrument is typically used with large-scale, capital-intensive projects to lower the perceived risk for lenders, therefore making it easier for companies to secure funding from private lenders, and at lower interest rates).

US government energy-specific subsidies are reported in Figure 1.21 and Figure 1.22 [1.20]. The subsidies in EU are instead presented in Figure 1.23 and Figure 1.24.

Both in EU as well as in the US, the subsidies for nuclear energy are significantly lower than the subsidies allocated to other energy sources. In EU, over the period 2015 – 2022, subsidies for nuclear reached a maximum of 7.6 billion EUR in 2021, compared to 88 billion EUR for renewables in the same time period. Three countries contributed to 90% of the EU subsidies for nuclear energy, namely France, Germany and Italy [1.21].

In France (44% of EU subsidies to nuclear), subsidies are dedicated to R&D, funding long-term costs of nuclear energy use such as waste management, and payments related to an early closure of the Fessenheim plant. In Germany (35% of EU subsidies to nuclear) subsidies to nuclear were primarily paid as compensation for early closure of NPPs, including a settlement in 2021 with remaining nuclear operators. Italy (12% of EU subsidies to nuclear) are mostly related to the costs associated to the premature closure of NPPs in the early 1990s (site decommissioning, waste management, and other post-closure costs) [1.21].

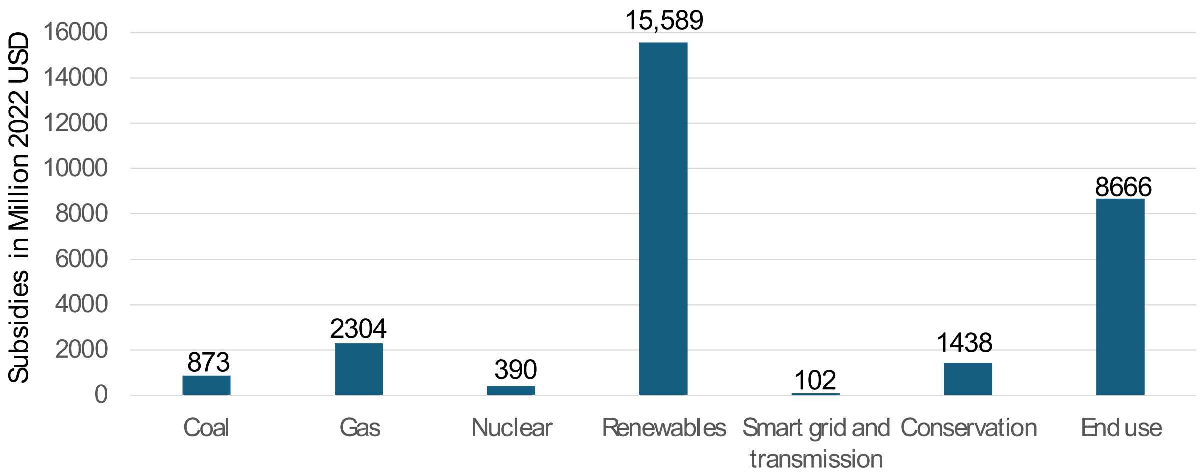


Figure 1.21 Energy-related US government subsidies in 2022 million USD in year 2022 [1.20]

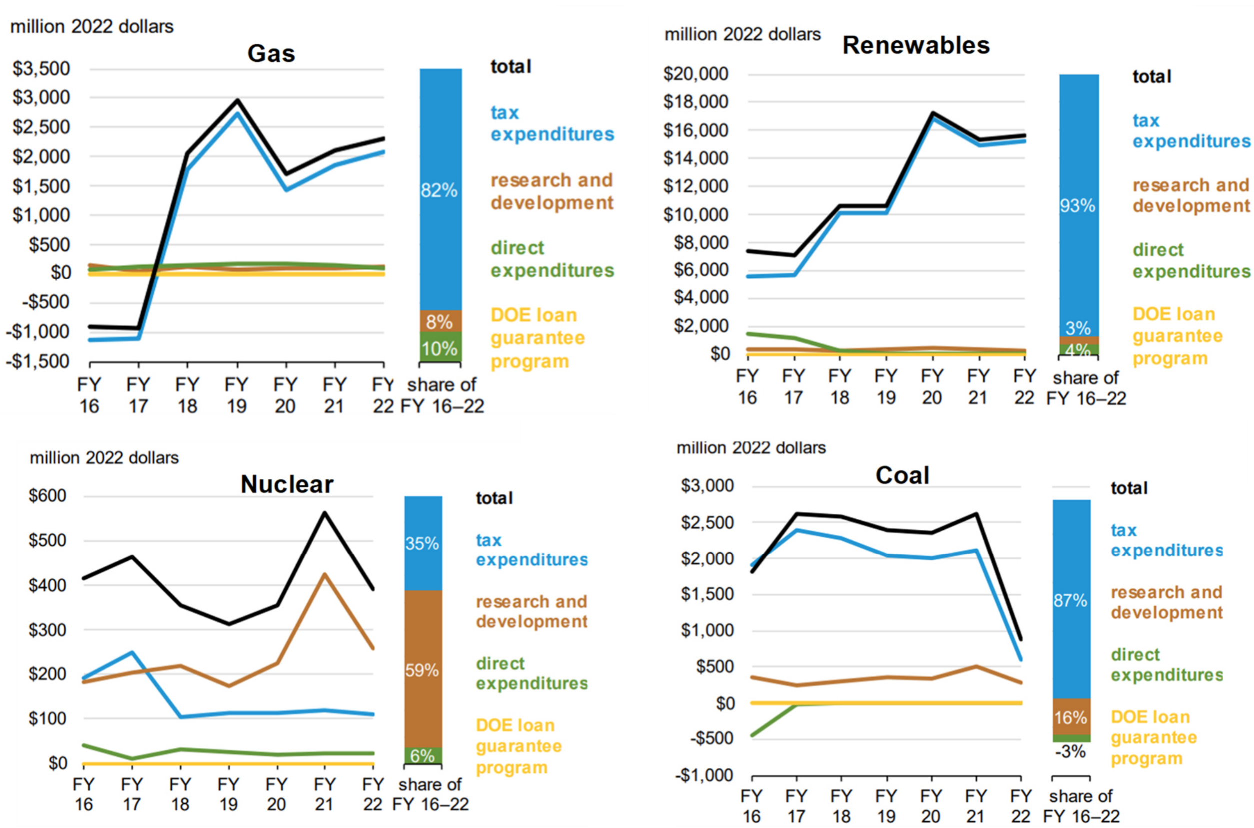


Figure 1.22 US Government Subsidies by energy source over the period 2016 to 2022

1.5 Deep geological repository

In past years, considerable progress has been made in various countries on the development of deep geological repositories for nuclear waste:

- the construction of the deep geological repository in Finland, with construction to be completed by mid 2020s
- In Sweden, a construction license for a geological repository was granted, with construction to start in the next few years,

- while in France an application for the construction of a deep geological repository is currently being evaluated by the nuclear authority, with a decision on the final site expected in 2025 and operation to start around 2040.
- In Canada, the site selection is planned to be announced in 2024.

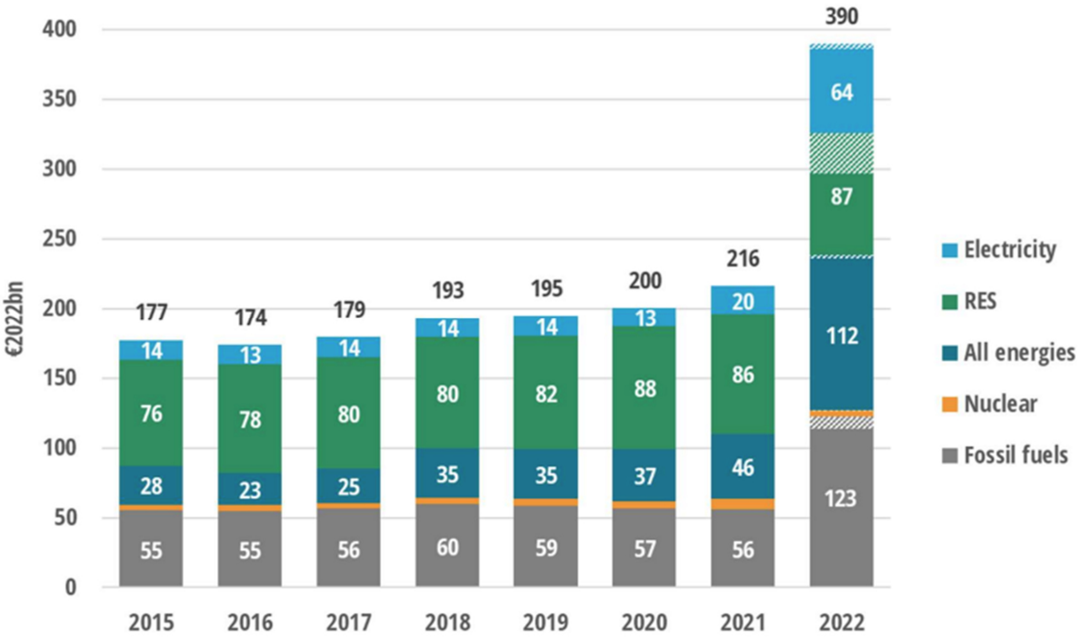


Figure 1.23 Energy-related EU Subsidies in 2022 billion EUR [1.21]

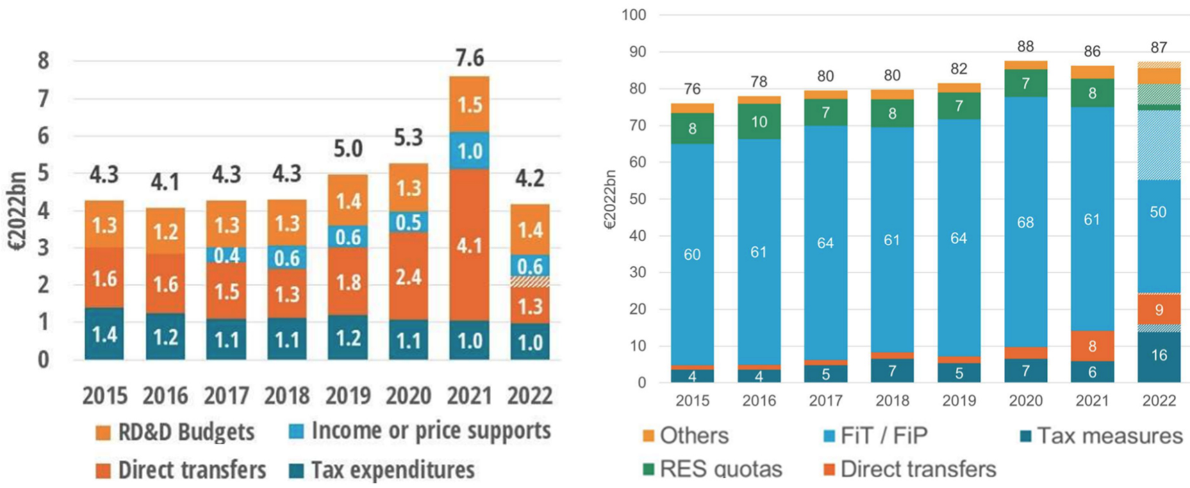


Figure 1.24 Subsidies by category for nuclear (left) and renewables (right) in the period 2015 - 2022 [1.21]

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2 Large-scale WCRs Gen-III/III+ – state of technology and main actors

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Water-cooled reactors (WCRs) have prominently shaped the commercial nuclear industry from its inception point, and presently they represent more than 95% of the total 440 operating civilian power reactors globally, as shown in Figure 2.1. In fact, most of the nuclear reactors currently under construction are water-cooled [2.2] see also

Table **2.2**, Table 3.1 to Table 3.3 and
Table 5.3).

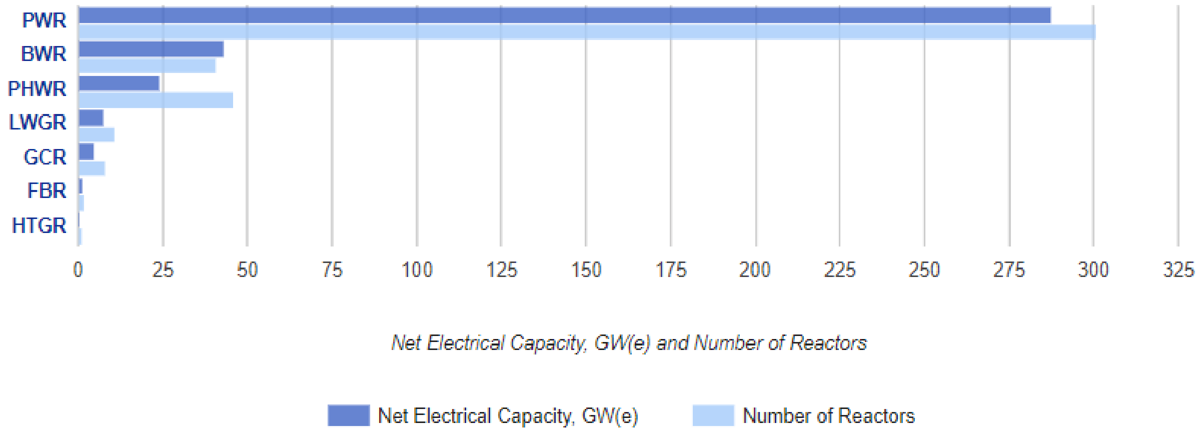

Figure 2.1 Reactors in operation worldwide by type [2.1].

Most of the nuclear power plants operating today were initially intended to function for 40 years. Due to technological advancements, their lifespan is being extended to 60 years, with the possibility of even longer operation. Therefore, it's foreseeable that WCRs will maintain their significant role moving forward into the 21st century [2.2].

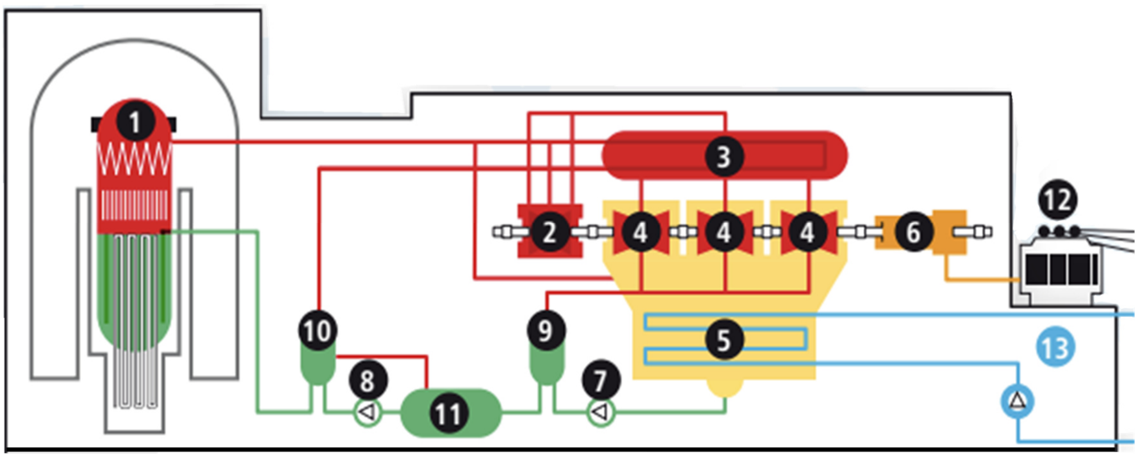

Figure 2.2 Scheme of a BWR (source: KKL). Water (green line) is sent to the core where it boils, producing steam (1) that is sent to the turbine (2, 4) to produce electricity. The steam coming out of the turbine is then condensed in the condenser (5) and sent back to the reactor core (green line).

The most prevalent types worldwide are Light Water Reactors (LWRs), specifically Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). In these types of reactors, water is used both as coolant as well as moderator. In BWRs steam created inside the reactor core is sent directly to the steam turbines to produce electricity (see Figure 2.2). In PWRs instead (Figure 2.3), the core coolant loop (primary loop) is separated from the secondary loop, in which steam for the turbines is generated in steam generators. The Russian VVER design belongs to the broad category of PWRs.

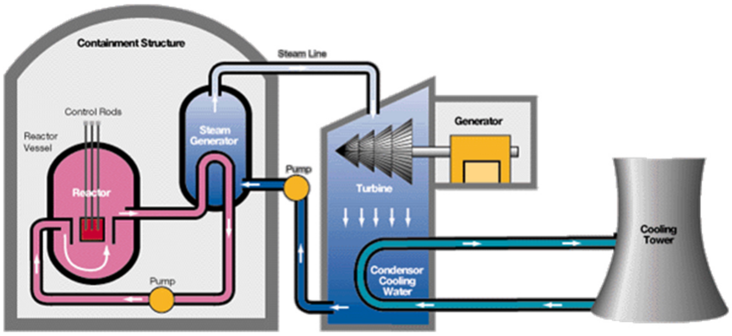

Figure 2.3 Scheme of a PWR. The reactor core is cooled by water (in pink) circulating in a closed loop (primary loop). Steam for the turbine is created in a secondary loop (in blue) through steam generators, in which heat is transferred from the primary circuit to the secondary side of the steam generator. The steam coming out of the turbine is condensed and sent back to the steam generator. The steam out of the turbine is condensed by removing heat through a third loop, which might be connected to a cooling tower, or might take water directly from a nearby water source (river or sea).

Another category of WCRs consists of Pressurized Heavy Water Reactors (PHWRs), which use heavy water as moderator. Light or heavy water in pressure tubes is used to cool the reactor core fuel. Heavy water present in the so-called calandria (see Figure 2.4) is used as moderator. As for PWRs, steam for the turbine is generated through steam generators in a light-water cooled secondary loop, separated from the core coolant primary circuit. An illustration of the working principle of PHWRs is shown in Figure 2.4. The CANDU reactors in operation in Canada are examples of PHWRs. CANDU have been exported to India, China, South Korea, Argentina and Romania. India and Argentina operate also PHWRs of own design.

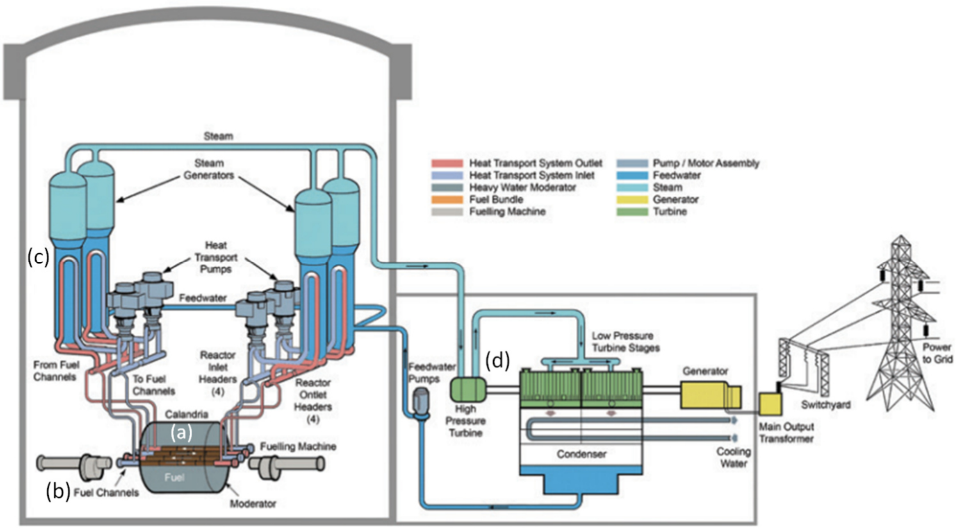

Figure 2.4 Scheme of a PHWR, CANDU type. Heavy water is contained in the calandria vessel (a) and is used as moderator to slow down neutrons for the fission chain reaction. Pressurized light water (primary coolant) circulates in the fuel channels (b) to cool the nuclear fuel. The hot water coming out from the fuel channels is sent to the primary side of the steam generators (U-tubes in the figure). Here, the heat is transferred from the primary core coolant to the secondary side of steam generators (c) to produce steam for the turbine (d), similarly as done for PWRs.

While all LWRs necessitate fuel that is slightly enriched in the fissionable isotope U-235 (up to $\sim 4-5\%$, see chapter 6), the PHWRs that use heavy water as coolant can work with natural uranium, thanks to the better core neutron economy (heavy water has a lower probability to absorb neutrons when compared to light water, leading to more neutrons available for the fission chain reaction). Two Gen-III reactors of the PHWR type are currently on the market: the Canadian Advanced CANDU 6 (EC6) and the Indian Pressurized HWR (IPHWR), which has been developed in India based on country past experience with CANDU reactors.

With few exceptions, practically all near-term new builds are Generation-III/III+ WRCs (i.e. LWRs and HWRs), based on proven water reactor technology, with large reactors dominating the current decade. A growing presence of Small Modular Reactors (SMRs) is anticipated by the year 2030.

Generation-III/III+ power plants represent a significant advancement in nuclear safety principles, further building upon the foundation laid by 2nd-generation of NPPs and integrating lessons learned from post-Fukushima stress tests. These modern designs are distinguished by an increased redundancy in active safety systems and/or the implementation of passive safety systems. Severe accident sequences such as core meltdown are explicitly included in the plant design, with safety systems specifically designed to cope with severe accident conditions. Core damage frequencies are maintained below 10^{-6}/year, together with the "practical elimination" of accident sequences leading to early release of radioactivity (e.g. Fukushima scenarios). More details on the safety aspects are provided in section 0.

2.1 Main actors – large Gen-III/III+ NPP available on the market

The large water-based Gen-III/III+ reactors currently available on the market are summarized in Table 2.1. As of December 2023, 38 such power plant units are in operation, and 51 are in construction.

These designs utilize proven technology, building on 50 years of experience with Gen-II water cooled reactors, but with an emphasis on safety features that rely on natural forces (see discussion in section 0 on passive safety systems for details). To be noted that, among the Chinese Gen-III/III+ designs listed in Table 2.1, currently only the ACPR-1000 is being exported abroad, but the HPR1000 was also developed for export.

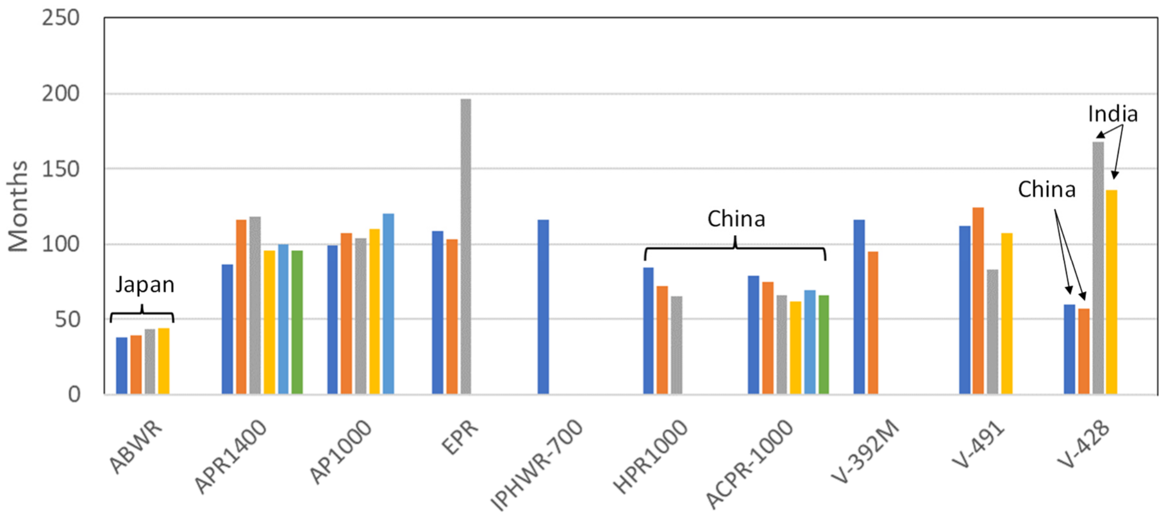

Figure 2.5 Construction times of large Gen-III/III+ NPPs

As already shown in Table 1.5, Russia is dominating the market with emerging countries outside Europe. China has also taken over part of this market. The market in Europe, US and Canada instead is exclusively dominated by Western companies.

Figure 2.5 provides an overview of the construction timelines for various large Gen-III/III+ NPPs constructed thus far. Notably, the ABWR units in Japan stand out for their remarkable short construction time, being all completed in under 4 years. On the contrary, the EPR in Olkiluoto, Finland, and the VVER V-428 in Kudankulam, India, represent the upper end of the spectrum, with construction periods lasting 16 and 14 years, respectively. Overall, the mean construction time is of 7.7 years, with a median of 8 years, as illustrated in Figure 2.6. The establishment of a supply chain is an important factor in the determination of the construction length, as well as the experience with multiple successive units. China, for example, has consistently decreased the plant construction time, with the nine units of HPR1000 and ACPR-1000 designs all built between 5 and 7 years. Noteworthy is also the considerable difference in construction time for the V-428 units (Russian design) built in China and India.

In addition, it is also important to highlight the unique challenges faced during the construction of the Olkiluoto EPR. Firstly, it marked a pioneering effort in establishing the first of its kind large power plant in Europe, after decades of inactivity. This endeavor necessitated the re-establishment of supply chains, adding complexity to the project. Secondly, differences between the French and Finnish regulatory requirements prompted revisions to the original EPR design, which needed to be executed and approved by the Finnish nuclear authority, STUK. This significantly contributed to the delays experienced in the project. Not as severe as for the cases in Finland and Flamaville, delays have been observed with the two EPR units under construction in the UK at Hinkley Point as well. These have been caused partly by the Covid-related interruption of the supply chain and Covid-enforced working restrictions, though additional delays recently announced by EDF in January 2024 were associated with missing supply chain specific to UK, the need to train UK workforce (with delays encountered mostly with the civil construction) and a significant number of design changes required by the UK nuclear authority (7000 changes, resulting in 35% more steel and 25% more concrete compared to the original design).

Overall, a well established supply chain and a trained workforce, as well as a detailed design in advanced stage of completion before construction starts are essential factors determining the plant construction duration and the potential risks for delays.

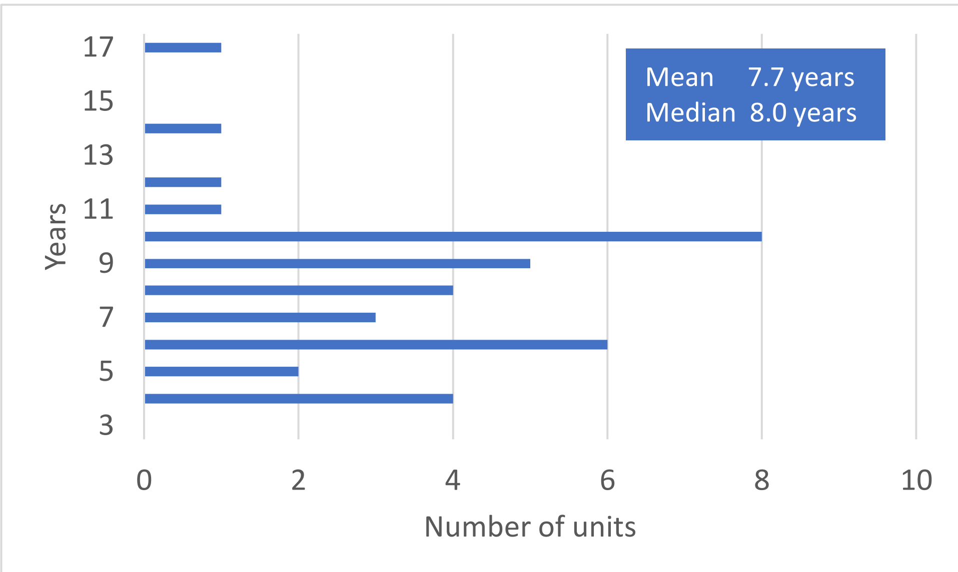

Figure 2.6 Distribution of Gen-III/III+ units construction times

Table 2.1 Large water-cooled Gen-III/III+ NPP available on the market

Name	Type	Developer/vendor (country)	In operation/under construction
ABWR	BWR	General Electric/Hitachi/Toshiba (USA/Japan)	Yes/Yes
AP1000	PWR	Westinghouse (USA)	Yes/Yes
APR1400	PWR	KHNP (South Korea)	Yes/Yes
APWR	PWR	MHI/Westinghouse (Japan/USA)	No/No
EC 6	PHWR	AECL (Canada)	No/No
EPR	PWR	EDF/Framatome (France)	Yes/Yes
EPR1200	PWR	EDF/Framatome (France)	No/No
ESBWR	BWR	General Electric (USA)	No/No
IPHWR-700	PHWR	NPCIL (India)	Yes/Yes
VVER-392M	PWR	Gidropress (Russia)	Yes/Yes
VVER-428	PWR	Gidropress (Russia)	Yes/Yes
VVER-491	PWR	Gidropress (Russia)	Yes/Yes
Hualong-one	PWR	CNNC & CGN (China)	Yes/Yes
ACPR-1000	PWR	CGNPC (China)	Yes/No
ACP-1000	PWR	CNNC (China)	No/No
CAP-1000	PWR	SPIC (China)	No/Yes
CAP-1400	PWR	SPIC (China)	No/Yes

In the following section, a brief overview of the main characteristics of the currently marketed large Gen-III/III+ NPP is presented. More details on the individual plants design can be found in Ref. [2.2]. Common to all designs is an expected lifetime of at least 60 years and significantly enhanced safety (see 0 for details on the safety philosophy and its implementation). To note that all designs are capable of load-following operation.

Table 2.2 Large advanced Gen-III/III+ NPPs in operation [IAEA PRIS]

<table>
<tr><th>Developer</th><th>Reactor</th><th>Power [MWe]</th><th>Details</th></tr>
<tr><td>GE/Hitachi, Toshiba</td><td>ABWR</td><td>1380</td><td>Certified in Japan, USA (1997) and Taiwan.
Commercial operation in Japan since 1996.
<table>
<tr><th>4 operating units in Japan:</th><th>Construction time [months]</th></tr>
<tr><td>• Kashiwazaki-Kariwa 6</td><td>38</td></tr>
<tr><td>• Kashiwazaki-Kariwa 7</td><td>39.5</td></tr>
<tr><td>• Shika-2</td><td>43.5</td></tr>
<tr><td>• Hamaoka-5</td><td>44.5</td></tr>
</table>
2 units under construction
• Shimane-3 (Japan)
• Ohma-1 (Japan)</td></tr>
<tr><td>KHNP</td><td>APR1400 (PWR)</td><td>1450</td><td>Certified in South Korea (2003), USA (2019) and UAE.
<table>
<tr><th>6 units in operation:</th><th>Construction time [months]</th></tr>
<tr><td>• Saeul-1 (South Korea)</td><td>86</td></tr>
<tr><td>• Saeul-2 (South Korea)</td><td>116</td></tr>
<tr><td>• Shin Hanul 1 (S.Korea)</td><td>118</td></tr>
<tr><td>• Barakah-1 (UAE)</td><td>96</td></tr>
<tr><td>• Barakah-2 (UAE)</td><td>100</td></tr>
<tr><td>• Barakah-3 (UAE)</td><td>96</td></tr>
</table>
4 units under construction:
• Shin Hanul 2 (South Korea)
• Saeul-3 (South Korea)
• Saeul-4 (South Korea)
• Barakah-4 (United Arab Emirates)</td></tr>
<tr><td>Westinghouse</td><td>AP1000 (PWR)</td><td>1250</td><td>Four units operating in China; two under construction in the USA.
<table>
<tr><th>5 units in operation:</th><th>Construction time [months]</th></tr>
<tr><td>• Haiyang-2 (China)</td><td>99</td></tr>
<tr><td>• Haiyang-1 (China)</td><td>107</td></tr>
<tr><td>• Sanmen-2 (China)</td><td>104</td></tr>
<tr><td>• Sanmen-1 (China)</td><td>110</td></tr>
<tr><td>• Vogtle -3 (USA)</td><td>120*</td></tr>
</table>
1 unit under construction:
• Vogtle-4 (USA)*

*Due to changes to US NRC regulation on plane crash, the design of Vogtle-3&4 had to be modified and re-approved by US NRC after construction had already started.</td></tr>
</table>

Table 2.2 cont.

<table>
<tr>
<td>Framatome (& EDF)</td>
<td>EPR
(PWR)</td>
<td>1750</td>
<td>3 units in operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Taishan-2 (China)
• Taishan-1 (China)
• Olkiluoto-3 (Finland)</td><td>109
103
196*</td></tr>
</table>
* the construction of this unit was plagued by delays due to the need for EDF to re-establish the supply chain and due to changes to the design after construction had already started in order to adapt EDF design to Finnish regulations.

3 units under construction:
• Flamanville-3 (France)
• Hinkley Point C-1 (UK)
• Hinkley Point C-2 (UK)
• Sizewell C-1 (UK) to start construction in 2024
• Sizewell C-2 (UK) to start construction in 2024</td>
</tr>
<tr>
<td>NPCIL

India</td>
<td>IPHWR-700

(PHWR)</td>
<td>700</td>
<td>In operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Kakrapar-3 (India, 2021)</td><td>116</td></tr>
</table>
5 units under construction:
• Kakrapar-4 (India)
• Rawatbhata-7 (India)
• Rawatbhata-8 (India)
• Gorakhpur-1 (India)
• Gorakhpur-2 (India)
Planned:
10 units (Banswara 1-4; Kaiga 5&6; Chutka 1&2, Gorakhpur 3&4)</td>
</tr>
<tr>
<td>CNNC & CGN</td>
<td>Hualong One

HPR1000

(PWR)</td>
<td>1180</td>
<td>Planned as main Chinese export design.

3 units in operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Fangchenggang-3 (China)
• Fuqing-6 (China)
• Fuqing-5 (China)</td><td>84
72
65</td></tr>
</table>
10 units under construction:
• Sanaocun-1 (China)
• Sanaocun-2 (China)
• Changjiang-3 (China)
• Changjiang-4 (China)
• Fangchenggang-4 (China)
• Lufeng-5 (China)
• Taipingling-1 (China)
• Taipingling-2 (China)
• ZhangZhou-1 (China)
• ZhangZhou-2 (China)</td>
</tr>
</table>

Table 2.2 cont.

<table>
<tr><td>LHNPC</td><td>ACPR-1000
(PWR)</td><td>1119</td><td>6 units in operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Hongyanhe-6 (China)</td><td>79</td></tr>
<tr><td>• Hongyanhe-5 (China)</td><td>75</td></tr>
<tr><td>• Yangjiang-6 (China)</td><td>66</td></tr>
<tr><td>• Yangjiang-5 (China)</td><td>62</td></tr>
<tr><td>• Kanupp-3 (Pakistan)</td><td>69</td></tr>
<tr><td>• Kanupp-2 (Pakistan)</td><td>66</td></tr>
</table></td></tr>
<tr><td>LHNPC</td><td>CAP-1000
(PWR)</td><td>1250</td><td>4 units under construction:
• Haiyang-3 (China)
• Haiyang-4 (China)
• Sanmen-3 (China)
• Sanmen-4 (China)</td></tr>
<tr><td>SPIC</td><td>CAP-1400
(PWR)</td><td></td><td>2 units under construction:
• Shidaowan-1 (China)
• Shidaowan-2 (China)</td></tr>
<tr><td>Gidropress</td><td>VVER
V-491
(PWR)</td><td>1200</td><td>4 units in operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Leningrad 2-1 (Russia)</td><td>112</td></tr>
<tr><td>• Leningrad 2-2 (Russia)</td><td>124</td></tr>
<tr><td>• Belarusian-1/Ostrovets (Belarus)</td><td>83</td></tr>
<tr><td>• Belarusian-2/Ostrovets (Belarus)</td><td>107</td></tr>
</table>
4 units under construction
• Tianwan-7 (China)
• Tianwan-8 (China)
• Xudabu-3 (China)
• Xudabu-4 (China)</td></tr>
<tr><td>Gidropress</td><td>VVER
V-392M
AES-2006
(PWR)</td><td>1181</td><td>Designs meet EUR requirements
2 units in operation:
<table>
<tr><td></td><td>Construction time [months]</td></tr>
<tr><td>• Novovoronezh 2-2 (Russia)</td><td>116</td></tr>
<tr><td>• Novovoronezh 2-1 (Russia)</td><td>95</td></tr>
</table>
4 units under construction:
• Akkuyu-1 (Turkey) V-509
• Akkuyu-2 (Turkey)
• Akkuyu-3 (Turkey)
• Akkuyu-4 (Turkey)</td></tr>
</table>

Table 2.2 cont.

<table>
<tr><td>Gidropress</td><td>VVER-TOI

V-510</td><td>1255</td><td>Latest Gen-III+ design in the VVER series.

7 units under construction:
Kursk 2-1 (Russia)
Kursk 2-2 (Russia)
Rooppur-1 (Bangladesh) V-523
Rooppur-2 (Bangladesh) V-523
Eldabaa-1 (Egypt)
Eldabaa-2 (Egypt)
Eldabaa-3 (Egypt)</td></tr>
<tr><td>Gidropress</td><td>VVER V-428

AES-91

(PWR)

V-412/466

AES-92</td><td>1126</td><td>AES-92 (VVER V-412/466) certified to conform to European Utility Requirements (EUR).

4 units in operation:<table><tr><td></td><td>Construction time [months]</td></tr><tr><td>Tianwan-4 (China)+
Tianwan-3 (China) +
Kudankulam-2 (India)
Kudankulam-1 (India)</td><td>60
57
168
136</td></tr></table>5 units under construction:
Kudankulam-3 (India)
Kudankulam-4 (India)
Kudankulam-5 (India)
Kudankulam-6 (India)
Bushehr-2 (Iran)
+ The Tianwan AES-91 units, commissioned in 2007, were the first reactors in the world to have “core catchers” installed.</td></tr>
</table>

ABWR: the Advanced Boiling Water Reactor (ABWR) is a Gen-III BWR with a nominal power of 1350 MWe designed by General Electric (USA) and Hitachi (Japan). It is the first Gen-III reactor type to have been constructed, with the first unit in operation in Japan since 1996. The plant is designed for a 24-months fuel cycle (i.e. refueling once every 2 years) and a lifetime of at least 60 years. With the ABWR, the recirculation pumps and associated recirculation loops (such as the ones present at the NPP Leibstadt) are replaced by internal pumps, eliminating the occurrence of large break loss-of-coolant accidents (LBLOCA). In addition, the internal pumps allow power changes of up to 30% of rated power to be accomplished automatically by recirculation flow control alone, thus providing automatic electrical load following capability for the ABWR without the need to adjust control rod settings [2.2]. Besides enhanced protection of the safety system (e.g. bunkered emergency Diesel generators), several additional systems are included to cope with severe accidents, including controlled filtered containment venting through suppression pool scrubbing for fission product removal, and additional independent water injections for both RPV and containment drywell (the containment compartment in which the RPV is located). Explosions of hydrogen produced from fuel cladding oxidation are prevented through a combination of containment compartments inertization and the use of passive catalytic hydrogen recombiners. This reactor design has also a precursor of the EPR core-catcher, consisting of basalt concrete spreading area underneath the RPV that would allow dispersion and passive cooling of the corium in the event of a severe accident with fuel melt and RPV failure.

Core Damage Frequency (CDF) < 10^{-5}/year; Large Early Release Frequency () < 10^{-6}/year

AP1000: the Westinghouse AP1000 is a Gen-III+ two-loops PWR, which is characterized by the deployment of passive safety systems for heat removal from the core as well as for the cooling of the containment. In-vessel, passive core-retention strategy is deployed. Due to the high level of passivity, the plant design involves several simplifications, with increased reliability and cost saving. The AP1000 passive safety systems are designed to automatically establish and maintain core cooling and containment integrity for a significant period of time following design basis events assuming the most limiting single failure, no operator action, and with no onsite and offsite ac electrical power sources. The first units have been put in operation in China. Currently, AP1000 are in operation in China and USA, with an additional unit selected for Poland, with construction start scheduled for 2026.

CDF = 2.4 10^{-7}/year; LERF = 1.9 10^{-8}/year

APR1400: the Advanced Power Reactor (APR) 1400, designed by KEPCO (South Korea), is a Gen-III two-loops PWR. A legal dispute was filed in the US by Westinghouse in October 2021, based on the claim that the APR1400 incorporates design features of the System 80 PWR, originally designed by Combustion Engineering, a company which has been successively acquired by Westinghouse. The lawsuit could hamper the capability of KHNP/KEPCO to export the APR1400 and participate in bids for new NPP construction in Europe and other countries. However, the lawsuit was dismissed by the US federal court in September 2023. A final ruling by an arbitration panel is not expected until late 2025. Several APR1400 are in operation in South Korea and United Arab Emirates (UAE). In addition, at least 2 units are going to be built in Poland by 2035.

CDF = 2.25 10^{-6}/year; LERF = 7.19 10^{-7}/year

APWR: the Advanced Pressurized Water Reactor (APWR), developed via a joint venture between Mitsubishi Heavy Industries (MHI) and Westinghouse, is a 1534 MWe four-loop PWR. The APWR+ version features 1700 MWe instead. The first two APWR units are planned for construction in Japan at the Tsuruga plant. It features four trains of safety systems, inside-containment fuel storage, decay heat passive heat removal, hydrogen recombiners, and a system to provide cooling in case of ex-vessel core melt (core catcher concept)

CDF = ~ 10^{-7}/year; LERF = ~ 10^{-8}/year

EPR: the EPR is a Gen-III+ four-loop PWR designed jointly by Framatome and Siemens as evolution of the French N4 and German KONVOI designs. This is an evolutionary design with increased safety obtained through higher availability of safety systems (e.g. by increasing redundancy), additional protection systems aimed at limiting the possibility of radioactive releases (e.g. core catcher with passive cooling to contain and mitigate the effects of a potential core melt), additional protection against external events (e.g. double containment structure). Currently EPR units are in operation in China and Finland, while two additional units are in construction in UK, and one is supposed to start operation in France in 2024.

CDF = 2.9 10^{-7}/year; LERF = 2.7 10^{-8}/year

EPR-1200: the EPR-1200 is an adapted version of the EPR featuring lower power (1200 MWe instead of 1650 MWe), and developed to target calls for tenders for medium-sized reactors. It uses the same safety approach, overall architecture, materials and equipment as in the original EPR design. The main differences are given by the number of steam generators, which is reduced to three in the EPR-1200, the replacement of the EPR double containment with a single thick-walled containment, and a three-train architecture for the safety system instead of four.

VVER-428: also known as AES-91 or VVER-1000/428, is a Gen-III 4-loops PWR developed by JSC Atomenergoproekt (Saint-Petersburg), which includes a combination of active and passive safety systems and is certified to conform to European Utility Requirements (EUR). Active safety systems are used to manage design basis accidents, while an optimal combination of active and passive systems is used to manage severe accidents, including passive heat removal from the containment.

CDF = 2.7 10^{-6}/year; LERF 6.3 10^{-8}/year

VVER-412/466: also known as AES-92 or VVER-1000/412 e VVER-1000/466, is a Gen-III 4-loops PWR developed by JSC Atomenergoproekt (Moscow). As for the AES-91, it includes a combination of active and passive safety systems to cope with Beyond Design Basis Accidents (BDBAs) and is certified to conform to EUR.

CDF = 1.0 10^{-7}/year; LERF < 10^{-7}/year

VVER-491: also known as VVER1200/AES-2006, this is a Gen-III+ PWR designed by JSC "Atomenergoproekt" (St. Petersburg) to meet the Russian Regulatory Documents and considering the requirements of the IAEA and the European Utilities Requirements (EUR). It includes, passive emergency core cooling, a system for passive heat removal from the containment, and system of passive heat removal from the primary circuit through the steam generators, a double-envelope containment, and core catcher.

CDF = 2.5 10^{-7}/year; LERF = 2.0 10^{-8}/year

VVER-392M: also known as VVER1200/AES-2006, this is a Gen-III+ PWR designed by JSC "Atomenergoproekt" (Moscow) to meet the Russian Regulatory Documents and considering the requirements of the IAEA and the European Utilities Requirements (EUR). Compared to the V-491, it features an increased number of passive safety systems, including passive core cooling, passive core flooding system, passive containment cooling, passive long-term decay heat removal system, double-envelope containment and core catcher. These are complemented by a series of active safety systems.

CDF = 1.6 10^{-7}/year; LERF = 2.3 10^{-8}/year

Hualong-one: also known as **HPR1000**, is a Gen-III+ 3-loops PWR of Chinese development arising from the merge of the **ACP1000** design by China National Nuclear Corporation (CNNC) and the **ACPR1000** design by CGNPC, with the goal to obtain a single standardized design, even though CNNC and CGNPC have their own domestic supply chains. It incorporates a combination of active and passive safety system for the emergency core cooling, for residual heat removal, for the cooling of the containment, and for the cooling of the reactor cavity (in-vessel core retention strategy in case of core damage). Passive safety systems are introduced as back-up for active systems to cope with a potential loss of AC power. This reactor has been developed not only for inland use, but also for export. Three units are already in operation in China, and ten are under construction.

CDF = < 6.9 10^{-7}/year; LERF 3.0 10^{-8}/year

ACPR-1000: developed in China by CGNPC, is a Gen-III 3-loops PWR based on improvements of the predecessor CPR-1000 design. It incorporates typical features of Gen-III/III+ reactors such as double containment designed against large commercial aircraft crash, passive autocatalytic hydrogen recombiners, in-containment fuel storage tank, and a core catcher (ex-vessel cooling strategy in case of severe accidents involving core melt).

ACP-1000: developed in China by CNNC, is a Gen-III 3-loops PWR based on the CNP-1000 design. While the CNP-1000 design was developed by China with support by Westinghouse and Framatome, Chinese authorities claim full intellectual property rights on the ACP-1000 design. As for the ACPR-1000, also this design includes a double containment. It also features a combination of active and passive safety systems for the emergency core cooling, the removal of residual heat, and the cooling of the containment.

CDF = < 10^{-6}/year.

CAP-1000: this 1250 MWe Gen-III+ PWR design has been developed in China by the State Power Investment Corporation (SPIC) [formerly China Power Investment Corporation and SNPTC], based on the Westinghouse Gen-III+ AP1000 design. Four units are currently under construction. According to agreements with Westinghouse, China would own any derivatives of the AP1000 design with power over 1350 MWe. China has further developed the design, resulting in the higher power (1500 MWe) CAP1400, intended for export, and is currently working on the CAP1700 and CAP2100 designs. The first two CAP1400 units are currently under construction at Shidaowan site since 2019.

IPHWR-700: it is a Gen-III 700 MWe pressurized heavy water reactor designed by the Nuclear Power Corporation of India Limited (NPCIL), as an evolution of the Canadian CANDU design. It features double containment, a water-filed calandria vault, and a combination of active and passive safety systems for accident management. Decay heat removal is performed passively through natural circulation of the primary coolant. This is combined with passive cooling and recirculation of the secondary inventory of the steam generators. In case of severe accident conditions, the heavy water in the calandria and the surrounding water-filled vault act as passive heat sink.

ESBWR: this is a 1600 MWe Gen-III+ BWR developed by GE-Hitachi as a significant evolution from the ABWR design. It deploys fully passive systems both for the normal operation as well as for accident conditions. Natural circulation is used in the primary loop for the circulation of the coolant also in normal operation conditions, so that the need of pumps for the coolant circulation is eliminated. As for the AP1000, the significant deployment of passive systems allows for the simplification of the design and a significant reduction of the number of components (pumps, etc.). Removal of decay heat from the core, core flooding, as well as cooling of the containment are fully passive. In addition, passive flooding of the drywell is foreseen in case of a severe accident with core melt. All safety systems are designed such that in case of Design Basis Accidents (DBAs) no operator actions are needed to maintain safe, stable conditions for at least 72 hours. While this reactor is being offered on the market, there is currently no pending order and no unit in construction.

CDF = < 1.7 10^{-8}/year; LERF < 1.4 10^{-9}/year

EC 6: the Enhanced CANDU 6 (EC6) is a 740 MWe Gen-III+ PHWR developed by Atomic Energy of Canada Limited (AECL) in Canada as innovation of the CANDU-9 design, and meant for the Canadian market as well as for export. As the other PHWR designs, it uses natural uranium as fuel. Passive heat sinks for the core decay heat are provided by both the moderator in the calandria as well as by the water-filled vault surrounding the calandria. Other passive systems include the system to deliver make-up cooling water to the calandria vessel and the calandria vault, the reactor building spray system, and passive autocatalytic hydrogen recombiners. Compared to previous CANDU designs, the EC6 is characterized by increased passive safety features and enhanced provisions to prevent and mitigate severe accidents.

CDF = < 10^{-6}/year; LERF < 10^{-7}/year

The status on large Gen-III/III+ currently in operation or under construction is summarized in

Table 2.2, including individual units construction times.

2.2 Safety philosophy of Gen-III/III+ NPPs

Taking into account the advancements in technology and science and considering lessons learned from all past events, including the Fukushima Daiichi accident, new crucial requirements were introduced in the IAEA Specific Safety Requirements, SSR-2/1 [2.3] issued in 2012 to ensure an even higher level of safety for NPP operation.

Table 2.3 Modified plant design envelope for Gen-III/III+ and associated SSCs design basis [2.4]*

Plant design envelope

Operational states		Accident conditions		
NO	AOO	DBAs	Design Extension Conditions	
Normal Operation	Anticipated Operational Transients	Design Basis Accident	Without significant fuel degradation	With core melting (severe accidents)
Loads and conditions generated by External & Internal Hazards (for each plant state)				
Criteria for functionality, capability, margins, layout and reliability (for each plant state)				
Design basis of equipment for Operational states		Design Basis of Safety Systems including SSCs necessary to control DBAs and some AOOs	Design Basis of safety features for DECs including SSCs necessary to control DECs Features to prevent core melt	Features to mitigate core melt (Containment systems)

* SSC = single structure, system and component

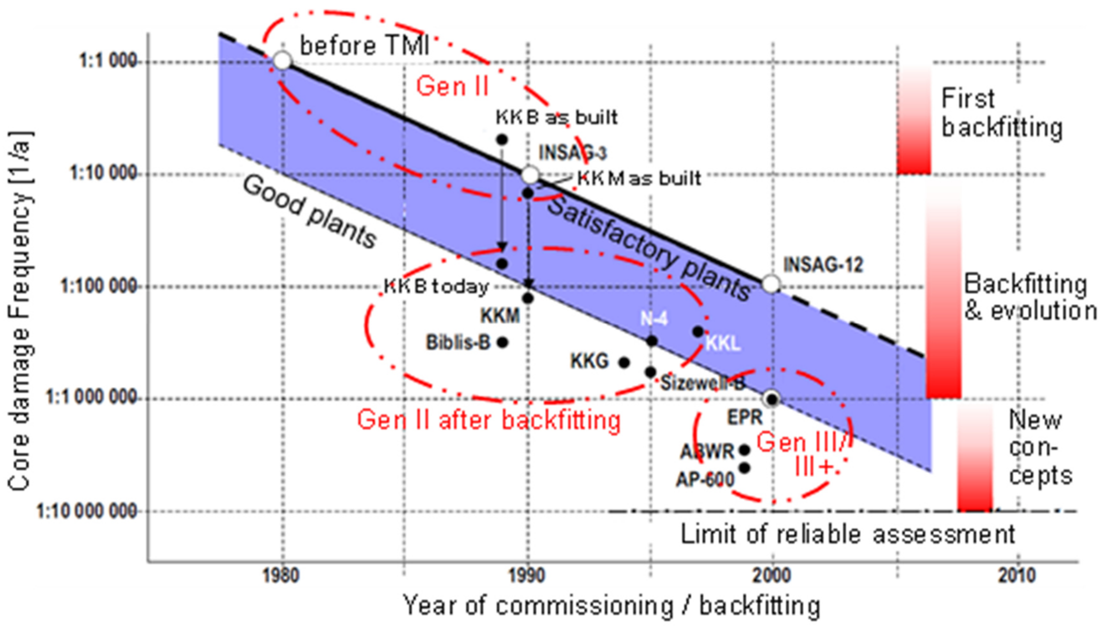

Figure 2.7 Continuous improvement of CDF over the past 50 years [2.6]

One of the most significant changes to past requirements is the inclusion of so-called Design Extension Conditions (DECs), highlighted in red in Table 2.3) in the plant design envelope, and the strengthened independence of different levels of defense in depth. In accordance with SSR-2/1 [2.3], the design is required to also address the necessary provisions for the mitigation of severe accidents and the practical elimination of event sequences that could lead to early or large releases [2.4]. In addition, the requirements on core damage frequency (CDF) and large early release frequency (LERF) have been strengthened, as shown in Table 2.4. In reality, values of CDF and LERF for Gen-III/III+ NPPs are well below these limits, as reported in section 2.1.

To note that the concept of DEC is not completely new, since it was already considered in the retrofitting of existing plants, including Swiss NPPs, taking into account events involving multiple failures of safety systems, as for example in the case of Station Blackout (SBO). However, the practical implementation of the new safety requirements means that, what were previously identified as Beyond Design basis accidents (BDBAs), and which were addressed with a mixture of emergency operator actions and accident management measures using equipment designed for other purposes, are now integral part of the plant design. As a consequence, a new series of engineered safety systems has been introduced in the design of Gen-III/III+ NPPs, to specifically address events falling in the DEC category, together with the provision of increased grace period from 30 minutes, typical of Gen-II designs, to a minimum of 3 to 7 or more days. The term "***grace period***" is used to describe the ability of a plant to remain in a safe condition for a substantial period of time after an incident or accident, without need for any human intervention [2.5].

According to the new SSR-2/1 requirements [2.3], "the plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that the possibility of plant states arising that could lead to an early radioactive release or a large radioactive release is ***'practically eliminated'***".

The resulting enhanced safety concept implemented in the design of Gen-III/III+ has allowed achieving a probability below 10^{-6}/year (less than once every million years of operation) for the occurrence of DEC with core melt, and a probability of a subsequent failure of the containment with releases to the environment below 10^{-7}/year (less than once every 10 million years). The continuous improvement in lowering the CDF is shown in Figure 2.7. In the figure, the improvements due to the continuous upgrades performed on the Swiss NPPs over the years is also evident. To note that KKG and KKL have similar CDF values (~ 2 10^{-7}/year) for internal events, however the CDF of both plants is dominated by external events (namely earthquakes), leading to a smaller CDF for KKG compared to KKL.

A summary of the new approach to defense in depth of the SSR-2/1 [1] is illustrated in Table 2.5, with the major changes highlighted in red, corresponding to level 3 and 4 of the defense-in-depth. Here, a clear distinction and separation is made between safety systems and provisions designed for accident scenarios not involving core melt, the ones predisposed for accident scenarios resulting in core melt, and the safety systems and measures designed to prevent off-site contamination (e.g. radioactive releases) following a severe accident with core melt. As a direct result of this new approach, the safety features designed to mitigate the consequences of core melt accidents must be entirely independent of the equipment designed to mitigate DBAs [2.4].

Table 2.4 Safety targets for new NPP designs

Target	Gen-II operating plants	New designs
CFD	10^{-4} /years	10^{-5} /years
LERF	10^{-5} /years	10^{-6} /years

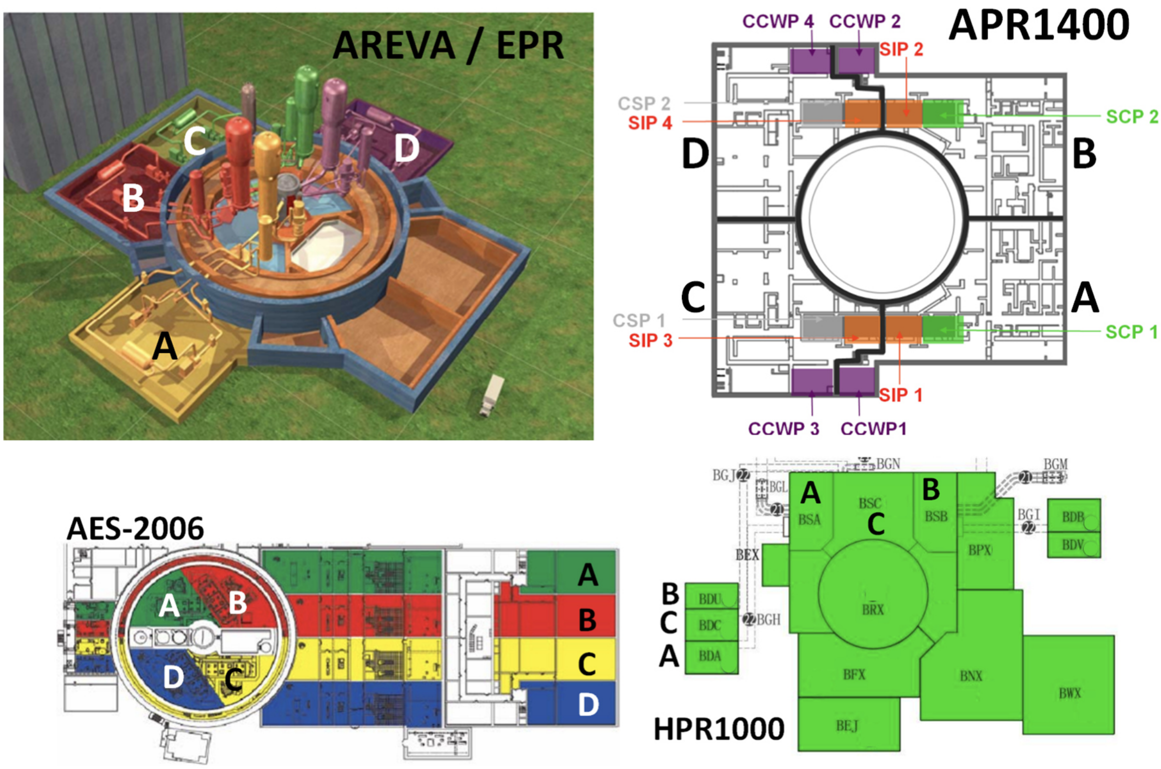

Figure 2.8 Physical separation of redundant safety trains in various NPP designs

The implementation of the new safety philosophy approach has resulted in several improvements:

- enhanced design basis safety through increased redundancy (e.g. EPR design with 4x100% instead of 4x50% redundancy), increased diversification of safety systems, complete physical separation of parallel redundant trains of safety systems to minimize common cause failure, enhanced inherent safety, and state-of-the-art list of accident scenarios;
- strengthen autarchy of safety system through introduction of passive safety and increased protection of the safety trains (e.g. bunkered emergency Diesel protected against floods, plane crash, earthquakes, fires);
- reduction of necessity for human intervention by automation of all safety functions and/or passive initiation of safety functions;
- increased autarchy (grace) time, through the introduction of large, autonomous heat sinks. The time after which operators' action is required (grace period) is increased from 30 minutes to a minimum of 72 hours;
- enhanced containment function through design of measures against severe accidents to prevent core melt or tackle the consequences of core melt, thus preventing radioactive releases to the environment. Examples include in-vessel core cooling, core catcher with ex-vessel cooling, hydrogen recombiners, filtered venting, etc.

- enhanced protection from external initiating events such as airplane crashes, earthquakes, floods, and fires, through the use of a stronger containment structure (e.g. double container, thicker containment walls, seismic decoupling from floor, etc) and by placing spent fuel pools within the containment.

The implementation of physical separation for parallel, redundant trains of safety systems is illustrated in Figure 2.8 for various designs. The double-containment structure is shown in Figure 2.9. Noteworthy is the location of the RPV with the reactor core, more than 40m below the top of the containment dome.

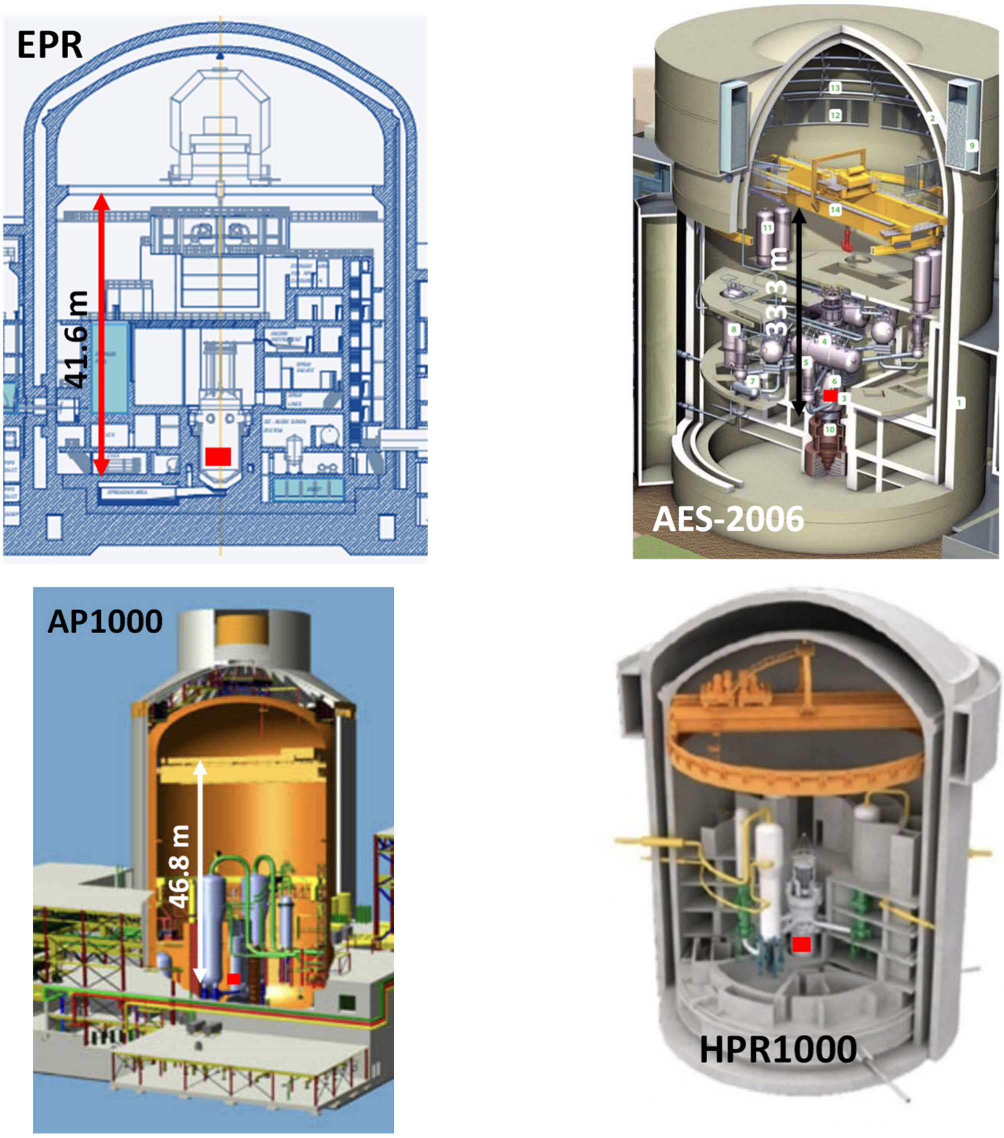

Figure 2.9 Double-containment structure for different Gen-III+ NPP designs. The approximate location of the core in the RPV is indicated in red.

Table 2.5 Level of the defense in depth for new NPPs [2.4]

Level of defence Approach 1	Objective	Essential design means	Essential operational means	Level of defence Approach 2
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures	Level 1
Level 2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features	Abnormal operating procedures/emergency operating procedures	Level 2
Level 3 — 3a	Control of design basis accidents	Engineered safety features (safety systems)	Emergency operating procedures	Level 3
Level 3 — 3b	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core melt	Emergency operating procedures	Level 4 — 4a
Level 4	Control of design extension conditions to mitigate the consequences of severe accidents	Safety features for design extension conditions with core melt. Technical Support Centre	Complementary emergency operating procedures/ severe accident management guidelines	Level 4 — 4b
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans	Level 5

2.2.1 Passive Safety Systems – Main concepts

Passive safety systems are systems whose working principle is based on natural forces such as pressure differences, gravity, temperature differences, free convection without requiring operator intervention, active controls, or an external energy source (e.g. electricity or Diesel engines) [2.5].

In Gen-II power plants, passive systems were limited to few systems such as the emergency reactor scram, accumulators for emergency core cooling during loss-of-coolant accidents (LOCAs) and hydrogen recombiners, used to limit the concentration of hydrogen in the containment and prevent the occurrence of hydrogen explosions. In Gen-II designs, safety systems mostly rely on active systems (e.g. pumps), which are powered by external energy sources (electricity, Diesel engines, etc).

With the new reactor designs of Gen-III/III+, there is an increased reliance on passive safety, with a group of these reactors relying on passive systems mostly for the safety function of the containment (needed during severe accident sequences), and another group relying on passive safety systems for the safety function of both containment and reactor itself (i.e.

passive safety systems are used to maintain the plant in safe conditions both in case of severe accidents, and in case of DBAs).

Passive safety systems have a higher degree of reliability compared to active systems. At the same time, they have the advantage of eliminating the costs associated with the installation, maintenance and operation of active safety systems that require multiple pumps with independent and redundant electric power supplies [2.7].

Table 2.6 Degree of passivity for passive safety systems

Degree of passivity:

	A	B	C	D
Structures (barriers, pressure proof)	X	X	X	X
Working fluids		X	X	X
Moving mechanical parts			X	X
Stored power				X
External activation signal				X

As summarized in Table 2.6, passive systems are classified according to four degrees of passivity (A to D) depending on whether their activation and functioning depends on moving fluids (e.g. natural circulation), moving parts (e.g. valves), stored energy (e.g. compressed fluid, batteries, fluids at higher elevation) and whether they need an activation signal [2.5].

Only stored energy sources such as batteries, compressed fluids or elevated fluids are acceptable as part of passive safety systems. Continuously generated power such as normal AC power from continuously rotating or reciprocating machinery is excluded (Diesel generators are therefore not accepted as part of passive safety systems). Acceptable active components of a passive safety system are limited to controls, instrumentation and valves. These though must be limited to single-action valves to activate the safety system operation and can only rely on stored energy. Manual initiation is excluded. A valve that has opened to allow the activation of a passive safety system cannot be reclosed.

The main concepts for passive safety systems include:

- depressurization of reactor pressure vessel (RPV), to also allow for subsequent core flooding through gravity;
- systems for removal of heat from the core/RPV;
- systems for core flooding through gravity;
- systems for the removal of heat from the containment atmosphere;
- systems for core retention.

Passive depressurization in case of over-pressurization transients or to decrease the pressure below ECCS pressure setpoint is achieved by releasing steam in water pools located within the containment of the NPP. The configurations typically used for PWR and BWR designs are illustrated in Figure 2.10. In case of PWRs, the steam is released from the pressurizer, while in case of BWRs, steam is released from the steam lines, upstream of the Main Steam Isolation Valves (MSIVs). Passive depressurization without loss of primary inventory and decay heat removal is achieved through natural circulation between the RPV and heat exchangers located in water pools placed at higher elevations inside or outside of the primary containment (see Figure 2.11). Additional passive systems are foreseen in advanced PWRs for the decay heat removal through passive cooling of the steam generators (see scheme in Figure 2.12).

The passive heat removal from the containment atmosphere is obtained through heat exchangers connected to outside-containment pools in open or closed natural circulation loop configuration, as illustrated in Figure 2.13 [2.7]. Configurations with the heat exchanger in direct contact with the containment atmosphere (for steam condensation) or submerged in the containment pools exist. The containment water pools are also directly connected to the RPV to allow for gravity-driven flooding of the reactor core at low pressures.

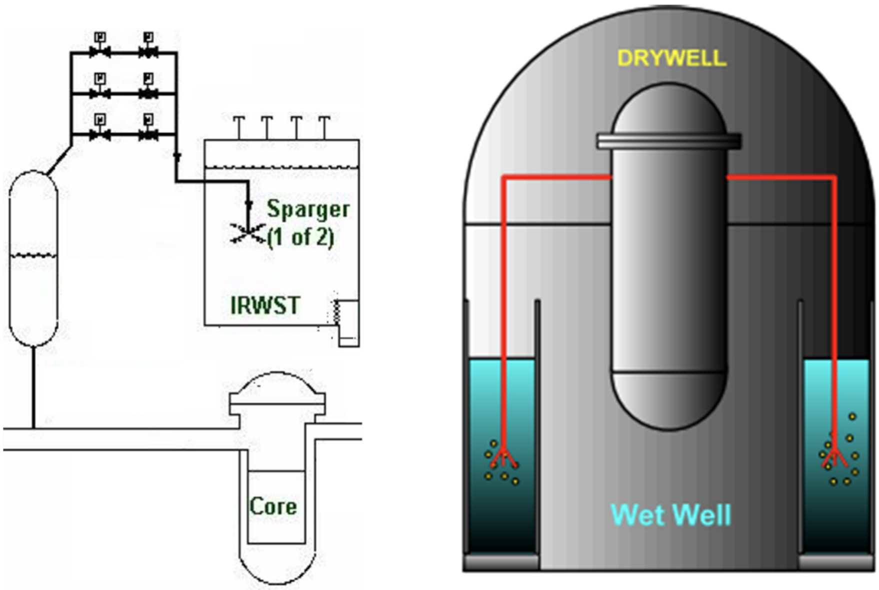

Figure 2.10 Passive depressurization through release of steam in containment pools. Left: PWR (AP1000), Right: BWR [2.7].

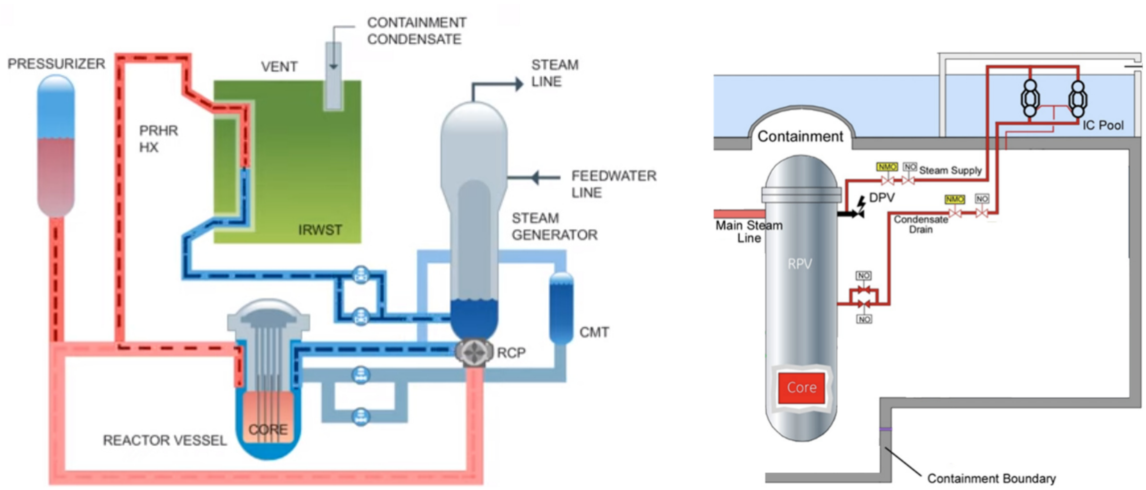

Figure 2.11 Passive decay heat removal. Left: PWR (AP1000), Right: BWR (ESBWR).

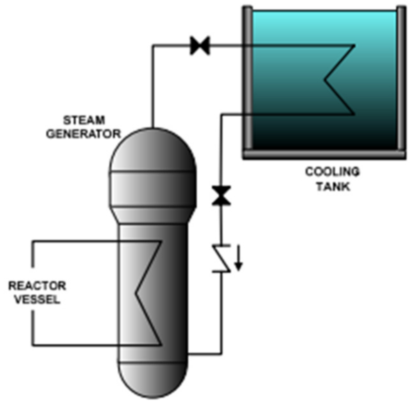

Figure 2.12 Passive decay heat removal through steam generator [2.7]

Finally, in the eventuality of severe core damage with melting of the fuel, passive cooling systems are in place to either guarantee that the core melt remains in the RPV (in-vessel core retention strategy) or to collect the molten core in dedicated structures (core catcher), where the molten core is then passively cooled (ex-vessel core retention strategy). The selection between the two strategy depends on the power density of the specific reactor design vs the RPV surface available for cooling.

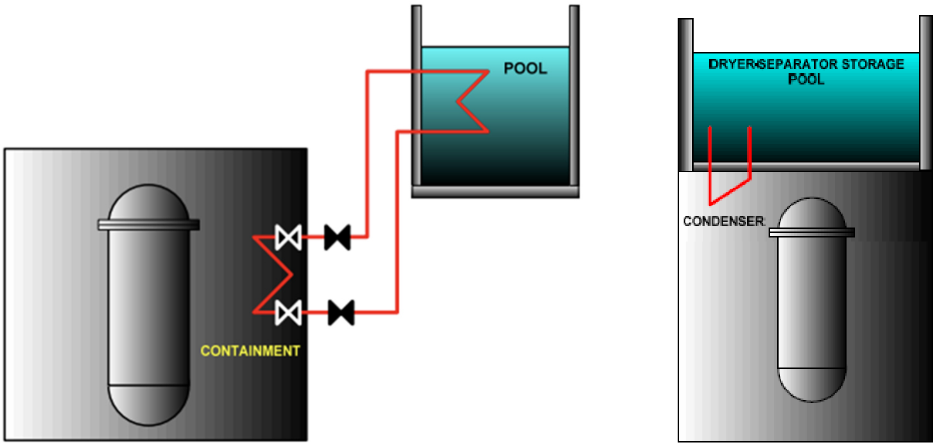

Figure 2.13 Passive containment heat removal through closed (left) or open (right) natural circulation loop configuration [2.7]

An example of how the ex-vessel core retention strategy with passive cooling is implemented by the EPR design (see Figure 2.14). In the EPR, the molten core is directed toward a large area (core catcher), where the corium can spread and be passively cooled (gravity-driven) with water from the in-containment water storage tank. With the in-vessel core-retention strategy instead, the reactor cavity is flooded with water (gravity-driven) to provide the necessary cooling and prevent failure of the RPV.

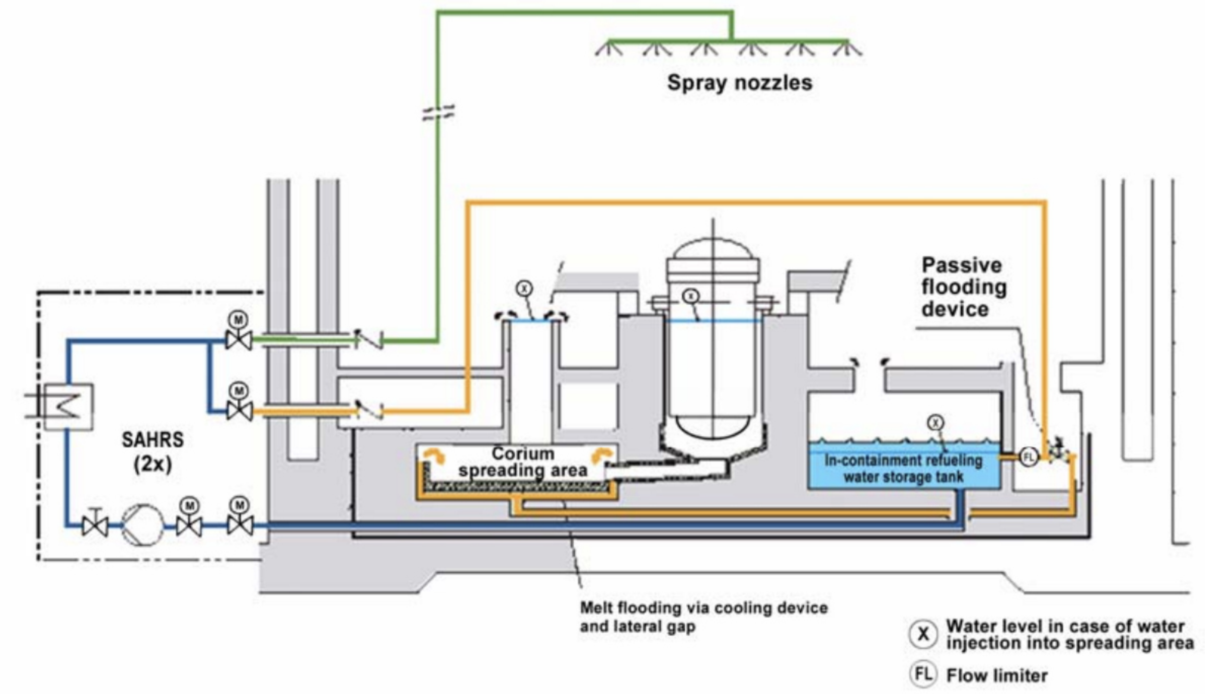

Figure 2.14 Ex-vessel core cooling strategy of EPR with core catcher and passive cooling

2.3 Conclusions

Large Gen-III/III+ NPPs represent a well-established, mature technology, featuring a wide spectrum of designs provided by numerous vendors across various countries. In total, there are 17 different designs currently available on the market, commercialized by vendors from France, Japan, USA, China, Canada, India, Russia and South Korea.

As of December 2023, 38 large Gen-III/III+ power plant units are in operation, and 51 are in construction. Construction durations vary significantly, ranging from as short as 4 years (all ABWRS built in Japan) to as long as 16 years, as for the first-of-a-kind EPR in Finland. To note however that the construction of the two EPR units currently underway in the UK at Hinkley Point (construction started in December 2018 for unit 1 and in December 2019 for unit 2) is anticipated to be finalized within 11 years. The EPR in Flamanville is expected to be connected to the grid in summer 2024. Overall, the average construction duration for large Gen-III/III+ units is approximately 7.7 years, with a median construction time of 8 years. A well establishment supply chain including a trained workforce, as well as the availability of a detailed design in advance stage of completion are essential factors for the plant construction duration.

Significant changes to the safety requirements have been adopted, with severe accidents now integral to the design of Gen-III/III+ NPPs. The implementation of the new safety philosophy has resulted in a new set of engineered safety systems and extended grace periods, with the goal to *"practically eliminate"* the occurrence of severe accidents with core meltdown and subsequent containment failure.

In particular, the new safety approach has led to an increase of the grace period (in which no human intervention is needed) from 30 minutes, typical of Gen-II designs, to a minimum of 3 days, core damage frequencies below 10^{-6}/year, and a probability of a subsequent failure of the containment with releases to the environment below 10^{-7}/year.

Noteworthy, while in past years the focus was on developing their own independent program, China has plans to exports their technology to the developing countries and beyond (see also Table 1.5). They have already exported two large Gen-III+ NPPs to Pakistan and they are participating to the bid for the construction of NPPs in Saudi Arabia.

2.4 References

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3 SMRs – state of technology and main actors

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This chapter is mostly focused on LWR SMRs. Details on the technology of non-LWR SMRs is presented in chapter 5. SMRs are expected to enter the market before 2030. NEA studies foresees that the SMRs share of the total new nuclear capacity by 2035 could be as high as 9% [3.1]. The interest in SMRs arises:

- from the need to power remote regions, and in general off-grid areas, currently relying on gas, oil or Diesel;
- for countries with small electricity grids where the deployment of large NPPs would not be possible or would not be economically efficient
- to replace coal-fired power plant in the range of 300-400MWe
- to supply energy-intensive industrial sites with electricity and heat
- for investors and operators not able or willing to invest large capitals

In addition, their increased flexibility for load-following operation make SMRs an attractive clean energy choice to integrate in a power grid with intermittent energy sources such wind and solar.

At a time of growing uncertainties in the security of energy supply, SMRs are becoming of interest also for energy-intensive companies which would like to build their own independent electricity/energy source, as recently seen for example in the United States. There, DOW chemical has shortlisted a SMR design for one of their industrial sites, Nucor Corporation as well as Amazon, Google and Microsoft have signed agreements for the potential use of SMRs to provide baseload electricity to their steel mills, and data centers respectively.

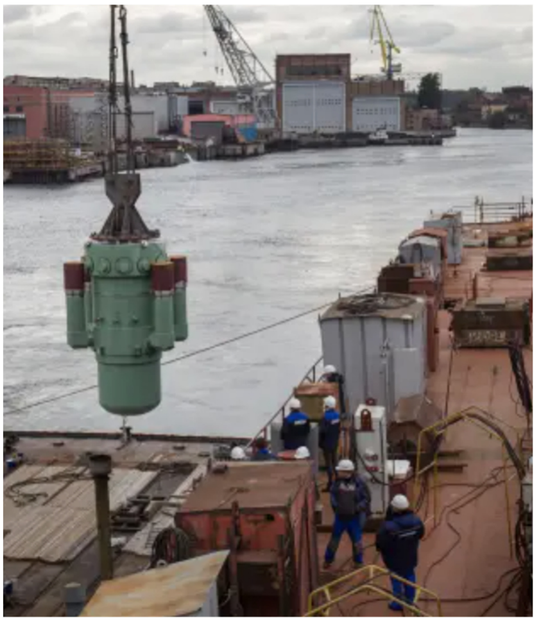


Figure 3.1 RITM-200 SMR

Small modular reactors (SMRs) are advanced reactors with a power capacity of typically up to 300 MWe per unit [3.2]. They are designed to be built in factories, shipped to the site of deployment and usually installed below ground level. Most of the designs are integral designs, in which all components (e.g. steam generators, pumps, pressurizer) are fully integrated in the RPV, making the design simpler and more compact. In case of integral SMRs, the factory-produced modules are complete reactor units, which are then transported to the construction site for installation (see example in Figure 3.1).

SMRs vendors pursue economy of series production, with SMR designs intended to be set up as independent components of a power plant with several units operating together, or as stand-alone units, with applications ranging from electricity production, heat generation to water desalination.

SMRs differ significantly from the early small power NPPs of the 60s, and they are not simply smaller version of large NPPs. Their design philosophy is significantly different. SMRs are designed with specific features in mind (see Refs.[3.2]-[3.8]):

- **Modular design:** SMRs are designed to be shop fabricated and then transported as modules to the construction site for installation. This simplifies the establishment of a supply chain, as well as reduces the complexity of the construction site. Both these factors can reduce construction risk, leading to cost savings.
- **Lower capital costs:** because of the reduced size, lower upfront capital investments are required for SMRs. This opens the market to a wider range of investors and operators, such as municipal utilities.
- **Shorter construction times**: because of the smaller size and the off-site factory production, construction times for SMR units can be strongly reduced compared to large NPPs. This can contribute to decrease of costs and lowering of financial risks.
- **Flexible power generation:** one of the ideas behind SMRs is to build multiple units at the same site, incrementally. While for a large NPP the average construction time is 7.7 years, a single SMR module can be built in 1.5 – 3 years. This means that electricity production can start sooner, and more units can be built while the previous ones have already started operation. In addition, the flexibility on the number of modules that can be installed at a given site allows to easily adapt to different customers' requirements.
- **Load-following operation:** an important feature of SMRs is that they are especially suited for load-following operation, meaning that their output can be adjusted based on the required load. Because of their smaller size and compact designs, they have much smaller inertia, allowing for faster response times for changing network loads. The SMR modularity, with several modules as part of the same power plant, allows for an increased flexibility in regulating power output. Because SMRs higher flexibility to adapt to changing load demands compared to large NPPs, they can be more easily integrated with intermittent energy sources such as wind and solar power.
- **Siting flexibility:** because of their smaller size, SMRs have a smaller footprint and can be placed at locations which would not be suited for large NPPs, including remote or off-grid areas, sites with limited water, or industrial clusters. SMRs are expected to be attractive options for the replacement of retiring fossil plants, or to provide an option for complementing existing industrial processes.
- **Enhanced safety:** SMRs adopt advanced safety features and passive systems to improve the safety performance and reduce the risk of severe accidents. Because of the lower power, core cooling during accident conditions is less demanding, so that a more effective deployment of passive safety systems can be realized, with even longer grace periods than large Gen-III/III+ NPPs. Moreover, the inventory of water in the RPV per unit of reactor core power is an order of magnitude larger for SMRs compared to large LWRs. Finally, the compact design opens up the possibility of construction below ground level, and "walk-away" safety concepts. As a result, the size of the Emergency Planning Zone (EPZ) could be significantly reduced, down to the plant site perimeter.

Noteworthy are also the recent developments pursued by Russia and China for shipyard-fabricated floating SMRs to power remote areas. China has a floating SMR currently under construction. Russia instead is already deploying SMRs in the Arctic, including a fleet of nuclear-powered ice-breakers and floating SMRs for heat and electricity generation to power remote communities as well as industry and mining operations. The aim is to replace coal and diesel energy sources and achieve large-scale savings of natural gas (more details are given in section 3.1 and Ref. [3.9]). Several other countries have also shown interest in shipyard-fabricated floating NPPs. In Figure 3.1 is a photo of the RITM-200 SMR Russia has developed for icebreakers and floating NPPs.

3.1 Main SMR designs available on the market

Currently, there are over 70 SMR designs with developers spread across many countries, as illustrated in Figure 3.2 [3.2] and Figure 3.3 (Ref. [3.5] and [3.6]), where the SMR concepts and associated developers headquarters are shown. The SMR designs under development include water-cooled technology, as well as designs based on alternative coolants as for Gen-IV reactors, such as liquid metal, gas-cooled and molten salt, with a large variation in size and electricity output, outlet temperatures in the range of 300°C to 900°C, with some advanced designs seeking to exceed 1000°C, and with various configurations from land-based to marine-based [3.5].

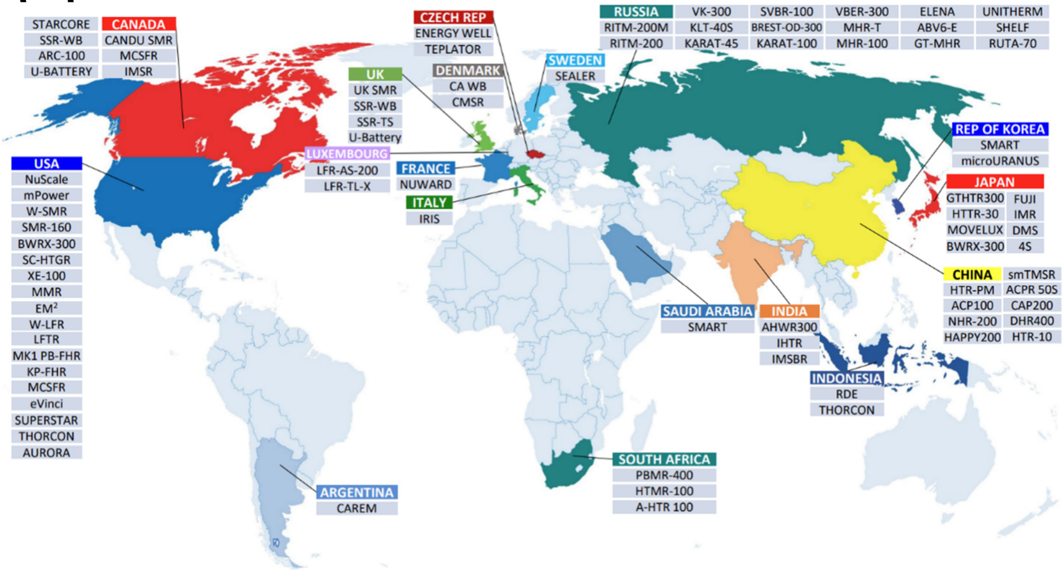


Figure 3.2 Global map of SMR technology development [3.2]

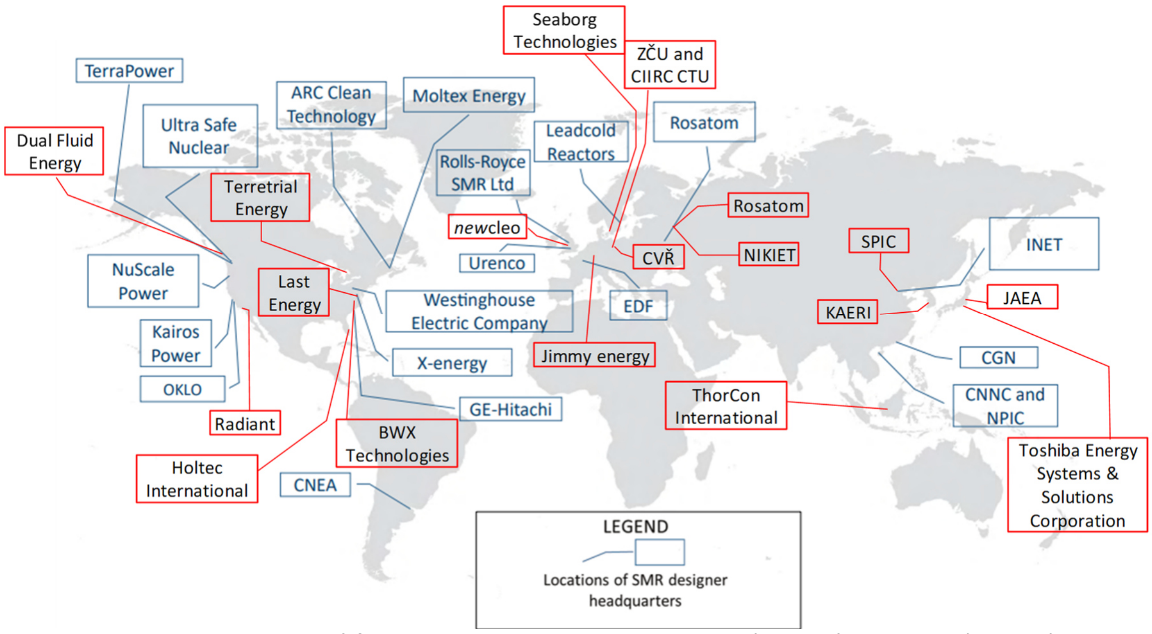


Figure 3.3 Location of SMR designers headquarter (blue: from ref. [3.6]; red from ref. [3.7])

Several of these SMR designs are still in the conceptual phase and will require significant research and development efforts. Only a few among all designs are at an advanced stage of development and are expected to be deployed by the early 2030s. The majority of SMR designs expected to be ready for deployment in the present decade are water-cooled. An attempt to assess the readiness level of the available designs has been recently carried out by OECD/NEA (see Ref. [3.5] and [3.6]). In Table 3.1 water-cooled SMRs in advanced stage of development are listed. In Table 3.2 and Table 3.3 non-water cooled SMRs are listed, using thermal and fast neutron spectrum, respectively. In these tables, the reactors which are already in operation are marked in green, and the ones that are currently under construction are marked in yellow.

Presently, ten SMRs are already in operation, two water-cooled KLT-40S units operating in Russia since December 2019, two HTR-PM units (gas-cooled) operating in China since December 2021, and six water-cooled RITM-200 operating in Russia, all commissioned between 2019 and 2022. In addition, three water-cooled SMR types are currently in construction, namely CAREM in Argentina, the ACP100 in China and several RITM-200N units in Russia, and one lead fast reactor (BREST-OD300) also in Russia. Four SMRs (US design) have been ordered in Canada for the Darlington site, where early site preparation has been already completed, as of February 2024. Romania, in collaboration with USA which provided a 3 billions USD funding, is expecting to finalize a decision on the first 6 SMR units by 2025 (NuScale design).

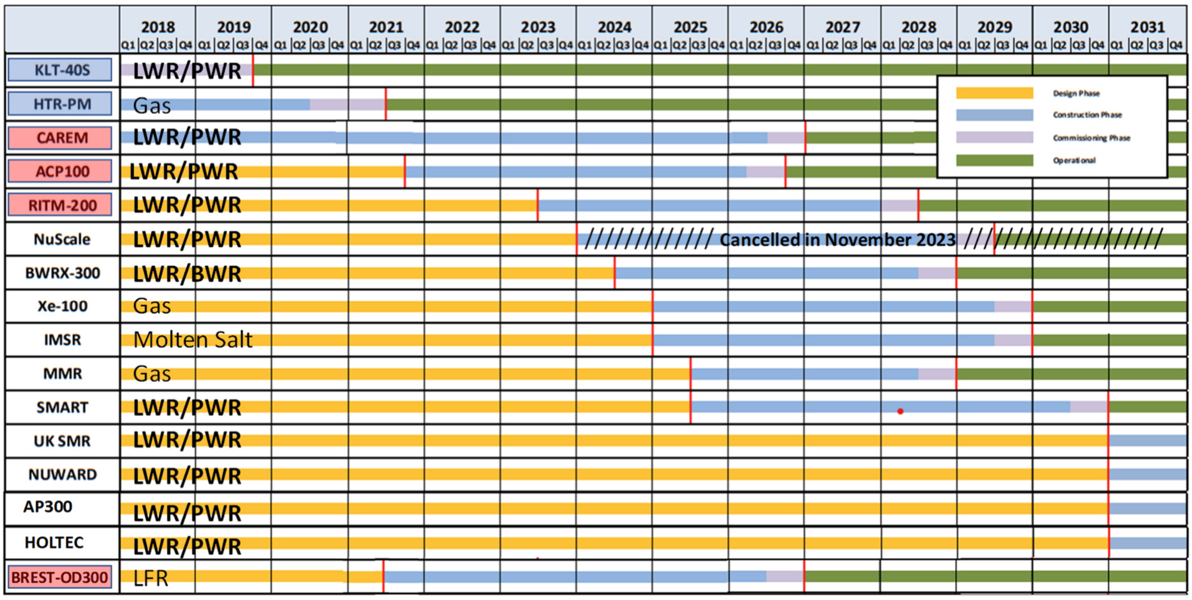


Figure 3.4 Timeline of deployment of SMR designs to 2030 (partially based on Ref. [3.5]). Names in blue boxes: SMRs already in operation. Names in red boxes: SMRs currently under construction.

A timeline for a selected SMRs expected to be deployed by early 2030 is reported in Figure 3.4 (partially based on Ref. [3.5]). Of the sixteen designs in the figure, eleven are LWRs (ten PWRs and one BWR), three are gas-cooled reactors (GCRs), one is a molten salt reactor (MSR), and one is a lead fast reactor (LFR). To note that NuScale was on track to submit a Combined License Application (COLA) to the US NRC in January 2024 for the construction of a six-modules SMR in Utah. The project was however cancelled in November 2023 because forecasted electricity costs raised to 8.9ct/kWh. This was caused by an increase of 150% of interest rates and a significant increase of material costs (e.g. 40% cost increase for steel) over the past 1.5 years. The project was deemed no longer competitive with the cheap gas and coal available in Utah. New gas plants are planned as replacement of the NuScale project.

Table 3.1 LWR SMRs in advanced stage of development (yellow: in construction, green: in operation)

Name	Thermal power [MWth (MWe)]	Type	Design organisation	Country	Status
CAREM	100 (30)	Integral PWR	CNEA	Argentina	Under construction in Lima. Expected to become operational by 2027. FOAK financed by government.
ACPR50S	200 (60)	Floating PWR	CGNCP	China	Under construction. Funded by CGNCP.
ACP100	385 (125)	Integral PWR	CNNC and NPIC	China	Construction started (Changjiang SMR 1) in 2021, with commercial operation targeted for 2026. Funded by CNNC.
KLT-40S	150 (35)	Floating PWR	"Afrikantov OKBM"	Russia	2 units in operation (Akademik Lomonosov 1&2). Funded by Russian gov + JSC Energoatom.
VOYGR	250 (77) 12 modules	Integral PWR	NuScale Power	USA	Up to 12 modules in one power plant. Licensed by US NRC. US gov + private funding. Shortlisted by UK for contract to be awarded in 2024. Shortlisted also in Romania and USA.
AP300	900 (300)	One-loop PWR	Westinghouse	USA	Scaled version of AP1000 already operating in USA and China. Pre-licensing stage with US NRC. Licensing expected to be simplified because based on AP1000 license. Shortlisted by UK for 2024 contract.
UK SMR	1,358 (470)	Integral PWR, 2 modules	Rolls-Royce SMR Ltd	UK	Phase 1 pre-licensing in UK. Four sites in UK pre-selected. Short-listed in Estonia. UK gov + private funding. Shortlisted by UK for 2024 contract.
NUWARD	540 (170) 2 modules	Integral PWR	EDF	France	Pre-licensing in France, Finland and Czech Republic. EUR 500 million by French gov. FOAK planned in France by 2030. Shortlisted by UK for contract to be awarded in 2024. Construction time ~40 months.
BWRX-300	870 (290)	Integral BWR	GE-Hitachi	USA	Phase 2 of pre-licensing in Canada completed. Financed by gov of Canada and USA, and Canadian Ontario province. Shortlisted by UK and Estonia. Ordered for construction in Ontario, Canada (4 units) and by TVA for site in Tennessee.
SMR-160	525 (160)	PWR	Holtec International	USA	Pre-licensing in USA and UK, and phase 2 pre-licensing in Canada. Fundings from US gov. MOUs with Entergy Corporation, Ukraine and Czech Republic. Shortlisted by UK for contract to be awarded in 2024. Short-listed by Ukraine with first unit to be operative by 2029 and plan to build 20 more units.
SMART	365 (107)	PWR	KAERI	Korea	Licensed by Korean Nuclear Safety Commission. MOU signed with Canada (2023) for deployment in Alberta, and with Saudi Arabia (2015). Funding by Korean gov + KEPCO.
RITM-200	175 (55)	Floating PWR	JSC "Afrikantov OKBM"	Russia	Six units are in operation on icebreakers, commissioned between 2019 and 2022. Additional units are currently under construction.
RITM-200N	190 (55)	On-shore PWR	JSC "Afrikantov OKBM"	Russia	To be built in Yakutia (Russia). Site license granted in April 2023. First concrete planned for 2024, operating license by 2027. Key electricity consumers of this first unit will be mining and processing facilities at the Kyuchus gold mine, rare-earth metal and tin deposits, and nearby towns.
RITM-200S	198	Floating PWR	JSC "Afrikantov OKBM"	Russia	Based on RITM-200 already in use in ice-breakers. Site at Baimskaya copper mine facility, to be deployed by 2027.
RITM-200M	175 (50)	Floating PWR	OKBM	Russia	Based on RITM-200 already in use in ice-breakers. MOU signed for deployment in Philippines and Myanmar. Funded by Russian gov.

A brief overview of the main characteristics of the LWR SMRs listed in Table 3.1 is presented in the following section. Most of the designs are integral PWRs, meaning that steam generators are integrated within the RPV and the main circulation pumps are welded to the RPV vessel, so that the entire primary loop is limited to the RPV itself (see Figure 3.5 for examples). The integration of the entire primary circuit within the RPV leads to a strong reduction of the containment size and to the elimination of Large Break LOCAs.

AP300 is a 300 MWe one-loop PWR developed by Westinghouse based on the large Gen-III+ AP1000 design (see chapter 2.1). AP1000 NPPs are already in operation in the US and in China. The AP300 is intended for electricity production as well as for district heating and water desalination. It is designed for an 80+ year life cycle. The design certification is anticipated by 2027, and the construction of the FOAK expected to start in the early 2030s. The AP300 has a compact, modular design with a fully passive safety concept. In October 2023, it was announced that the AP300 is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

ACP100 is an integral PWR developed by China National Nuclear Corporation (CNNC) with a power output of 125 MWe, and designed for electricity production, heating, steam production, or seawater desalination. The 16 once-through steam generators are all integrated in the RPV, the main circulation pumps are mounted directly on the RPV nozzles, while the pressurizer is mounted on top of the RPV. The RPV, the steam supply system and the spent fuel pool are all underground. The approach to safety is based on passive safety systems. Construction of the first unit was approved in July 2021. The unit is currently being built at the Changjiang nuclear power plant on China's southern island province of Hainan with operation scheduled to start by 2026, with a total construction duration of 58 months.

Table 3.2 Non-LWR SMRs in advanced stage of development (thermal spectrum)

Name	Thermal power (MWth)	Type	Design organisation	Country	Status
HTR-PM	500	HTGR	INET	China	2 units connected to the same turbine in operation since Dec 2021 (Huaneng Shidaowan). Additional 18 HTR-PM units are proposed for the Shidaowan site. A scaled-up version called HTR-PM600, with one large turbine rated at 650 MWe driven by some six HTR-PM reactor units is also planned. Feasibility studies for the HTR-PM600 deployment are under way for deployment in several provinces in China.
KP-FHR	311 (140 Mwe)	Molten salt cooled / solid fuel	Kairos Power	USA	Construction permit for demo unit received in December 2023. Site selected with TVA in Oak Ridge.
XE-100	200	HTGR	X-energy	USA	Pre-licensed completed in Canada, on-going the USA. Shortlisted for site in Richland (Washington, USA) and by Dow Chemical. Selected for construction in South Africa. Funded by US government and privates.
GTHTR300	600	HTGR	JAEA	Japan	No information. Development funded by Japan government.
IMSR	884	Integral MSR	Terrestrial Energy	Canada	Pre-licensing in USA, and phase 3 pre-licensing in Canada. Short-listed by AECL for site in Canada and MOU for siting at Idaho (USA). Funding from US and Canadian governments and privates.

ACPR50s is a 60 MWe offshore floating loop-type PWR developed by the China General Nuclear Power Corporation (CGNPC) with multiple target applications for both coastal and island sites, ranging from powering offshore oil drilling platform to offshore mining, nuclear power ships and distributed energy for islands, and drinking water production. The RPV is connected to two once-trough steam generators and associated main circulation pumps with very short piping. The pressurizer is connected to one of the two hot legs. The safety approached is based on passive safety systems. The system is mounted on a barge as a floating nuclear power plant. The first floating nuclear power plant in China is currently under construction at the Bohai shipyard with a single ACPR50s unit. Completion was planned for 2022, however CGNPC has not released further information.

Table 3.3 Non-LWR SMRs in advanced stage of development (fast spectrum)

Name	Thermal power (MWth)	Type	Design organisation	Country	Status
ARC-100	286	SFR	ARC Clean Technology	Canada	Pre-licensing phase 2 in Canada MOU for Point Lepreau site in Canada Funding from USA and Canada's gov + private.
SEALER-55	140	LFR	Leadcold Reactors	Sweden	Pre-licensing phase 1 in Canada (on hold)
Stable Salt Reactor - Wasteburner	750	MSR	Moltex Energy	Canada	Pre-licensing phase 2 in Canada MOU for Point Lepreau site in Canada Funding from USA and Canada's gov + private
Natrium	840	SFR	TerraPower	USA	Pre-lincensing in USA Selected to replace coal-fired Naughton Power Plant in Kemmerer, Wyoming (USA) Funded by US gov + private
BREST-OD-300	700	LFR	NIKIET	Russia	Under construction in Russia. Completion is planned for 2026. Funded by Russian gov.
CFR-600	1500 600 MWe	SFR	CNNC	China	2 units under construction in China (Xiapu 1&2). Expected to be connected to the grid between 2024 and 2025. A larger commercial-scale reactor, the CFR-1000, is also planned.

BWXR-300 is a SMR BWR developed by GE Hitachi Nuclear Energy, based on an evolution of the Gen-III+ ESBWR discussed in Chapter 2.1. It has an output of 300 MWe and its safety approach is fully passive. The circulation of the coolant in the reactor pressure vessel is achieved through natural circulation, therefore the need for RPV internal pumps, or for jet pumps and associated recirculation loops, is eliminated. The concept of primary coolant circulation through natural circulation has already been demonstrated by GE with the Dodewaard reactor, a 60 MWe small BWR which was successfully operated in The Netherlands from 1968 to 1997. The bottom elevation of the BWRX-300 reactor building foundation is approximately 36 m below grade, with the exterior dome extending to 30 m above ground. The RPV is housed in the primary containment vessel (RPV), which is almost entirely below grade and placed below a large water pool. The RPV can be passively cooled with three redundant trains of isolation condenser systems (ICS), which provide cooling for a minimum of seven days without power or operator action.

The BWRX-300 has entered phase-2 of the pre-licensing procedure with the Canadian Nuclear Safety Commission (CNSC). It has also entered pre-licensing with the UK Office of Nuclear Regulation (ONR) and with the US NRC. Recent developments:

- Ontario Power Generation (OPG) has signed a contract with GE Hitachi and other contractors in January 2023 and applied for a construction permit to the Canadian Nuclear Safety Commission for the Darlington site, where four units of the BWRX-300 are foreseen. Construction of the first unit is planned to start in 2024 (as soon as licensing is granted by the nuclear authority), with operation planned for end of 2028.
- Tennessee Valley Authority (TVA), one of the largest electric utilities in USA, is planning a first BWXR-300 unit at the Clinch River site in the US, for which it has already secured a site permit from the US Nuclear Regulatory Commission. TVA is preparing the submission for a construction license, while at the same time exploring additional sites in south east US for additional SMR units.
- In July 2022, Synthos Green Energy (SGE), a chemical company in Poland , together with GE Hitachi and BWXT Canada, and PKN Orlen, a Polish multinational oil refiner, has applied to the Polish nuclear authority for the evaluation of BWRX-300, and have announced a plan to deploy at least 10 units of the BWXR-300 in Poland in early 2030s. In March 2023, SGE has signed an agreement with Canadian OPG and TVA, to develop a BWRX-300 design consistent with European and Polish standards. In April 2023 seven potential locations have been announced for the first unit, all close to high energy intensive production plants. In May 2023, the BWXR-300 was found to be compliant with the Polish regulatory requirements.
- In June 2022, a MOU was signed by GE Hitachi with Saskatchewan Power Corporation for deployment of the SMR in mid-2030s.
- In September 2022 Estonia accepted tenders for SMRs by General Electric, NuScale and Rolls-Royce. In February 2023, the General Electric BWRX-300 was selected for a project development and a preliminary works contract.
- In October 2023, it was announced the BWRX-300 is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

CAREM is an integral PWR SMR with a power output of 30 MWe, designed by CNEA, Argentina. This SMR is currently under construction in Lima (Buenos Aires Province), and is expected to start operation by 2027. The twelve once-through steam generators and pressurizer are all integrated in the RPV, and the primary coolant is circulated through natural circulation. It therefore does not require main circulation pumps. The safety approach is based on a combination of passive and active safety systems, with the passive safety systems providing a grace period of 36 hours.

KLT-40S developed by JSC “Afrikantov OKBM”, with an output capacity of 35 MWe, is a floating PWR SMR that can be manufactured in shipyards and delivered via sea to the site of deployment fully assembled and ready to operate. It is based on the design of the KLT-40 marine propulsion nuclear plants which have been operating on several ice-breakers. The safety concept is based on a combination of active and passive safety systems, with the latter able to provide cooling without need of external power for 24 hours. Two units of KLT-40S are in operation since 2019 on the floating nuclear power station Akademik Lomonosov 1 in Pevek, a remote Arctic coastal city in Russia’s Far East, where they were transported after construction at the Baltic shipyard of St. Petersburg. They are deployed as non-self propelled power barge for electricity production, heat and water desalination. Refueling is every 3-4 years. The KLT-40 design was further modernized and improved to evolve into the RITM-200 SMR family.

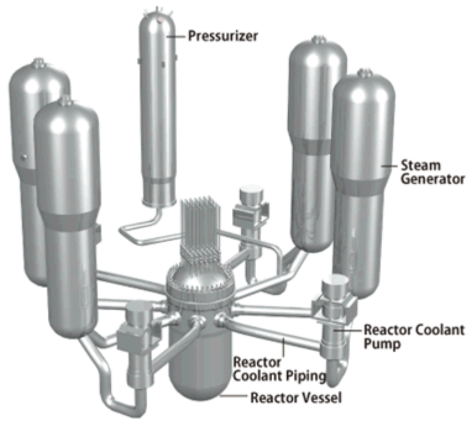
NUWARD is an integral PWR developed by a subsidiary of EDF (France), with contributions from CEA, Naval Group, TechnicAtome, Framatome and Tractebel (the latter are both private companies). It consists of two units, each with an output power of 170 MWe, for a total capacity of 340 MWe. The RPV of each unit is installed within a stainless steel containment vessel, completely submerged in the reactor water pool. The safety approach is fully passive, and the ultimate heat sink provided by the reactor pool allows for a grace period of 3 days, in which neither power or operator actions are required. Pre-licensing activities are on-going in France, Finland and Czech Republic. Lead by the French nuclear authority (ASN), a joint review of the NUWARD SMR design and safety concept was carried out with the Finnish (STUK) and Czech (SUJB) nuclear authorities and their technical support organizations (IRSN and SÚRO) with the goal of facilitating the harmonization and standardization of reactor designs and regulatory requirements across different countries, and to provide early feedback to EDF on its design and safety approach.

The project has strong support from the French government, which granted EUR 500 million to support NUWARD SMR development, including testing and licensing until 2030. The French government plans to build the reference NUWARD plant in France by 2030. In October 2023, it was announced that NUWARD is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

RITM-200 series is a family of Gen-III+ PWR SMRs developed by Afrikantov OKBM, a subsidiary of Rosatom, with the aim of providing reliable power generation for remote regions, industrial facilities, and naval vessels, based on an evolution of the KLT-40S design. The RIMT-200 series is used for icebreakers, as well as for floating or land-based platforms to supply electricity, heat and desalinated water to remote areas or industrial sites. There are four models in the series, namely: RITM-200, RITM-200M, RITM-200N, and RITM-200S, each model with different specifications and applications, but all sharing the same basic design principles and components. The reactor is an integral PWR, with the four steam generators and pressurizer integrated within the RPV. As for all integral design, this reduces the size and weight of the reactor, as well as the risk of loss of coolant accidents (large break LOCAs are eliminated by design). The safety concept is based on passive safety systems.

The **RITM-200** design is a 55 MWe integral PWR based on improvements of the KLT-40 design and developed for installation on icebreakers and floating NPPs. The RPV includes four integrated steam generators and four main circulation pumps. The integral, compact design improves the ability to operate in rolling and pitching seas. Six units of the RITM-200 are already in operation on three icebreakers, Arktika, Sibir and Ural commissioned between 2019 and 2022. Each of the icebreakers is powered by two reactor units. Refueling is every 10 years, with a 60 years lifespan. Two additional icebreakers Yakutia and Chukotka are under construction and scheduled to be completed in 2024.

The **RITM-200N** has a 190 MWe output is a land-based version of the RITM-200, with refueling every 5-6 years. A RITM-200N unit in planned for the isolated Ust-Kuyga town in Yakutia, to replace coal and oil-based electricity and heat generation. Construction is planned to start in 2024. Key electricity consumers will be mining and processing facilities at the large Kyuchus gold mine, rare-earth metal and tin deposits, and nearby towns. The RITM-200S is a variant designed for the Modernized Floating Power Units project. The project includes two of these reactors on a barge, with a nominal thermal power of 198 MW each and refueling needed about every five years. The first units are intended to deliver a power supply for the Baimskaya Mining and Processing Plant in Chukotka. The **RITM-200M** is a 50 MWe integral PWR similar to the RITM-200 design, much more compact than the KLT-40S units, developed for the Optimized Floating Power Unit (OPEB) and optimized for non-propulsion applications on floating nuclear power plants. The OPEB is an unpropelled, transportable barge designed to be moored at a protected pier. It will feature two RITM-200M units with refueling every 10 years, and a 60 years service life. Four OPEB units are expected to start operation between 2027 and 2028.



In integral PWR SMRs, all components of the primary circuit of a typical PWR (steam generators, pressurizer, main circulation pumps) are integrated in the RPV.

The compact design of SMRs is also illustrated in Figure 3.6, where the size of an EPR power plant is compared to a BWX-300 SMR.

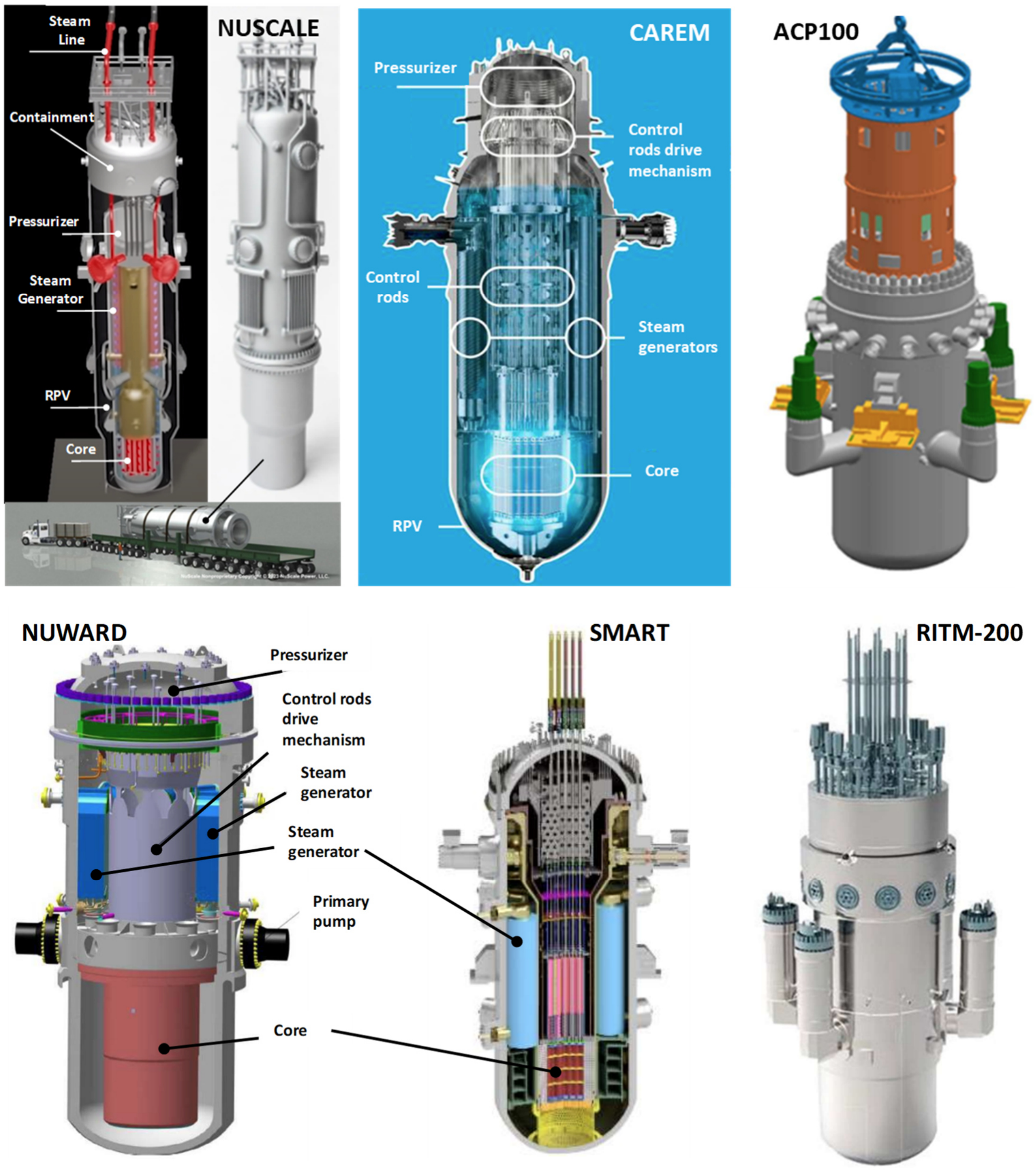


Figure 3.5 Examples of integral PWR SMRs. Top left: typical 4-loop PWR primary circuit with SGs

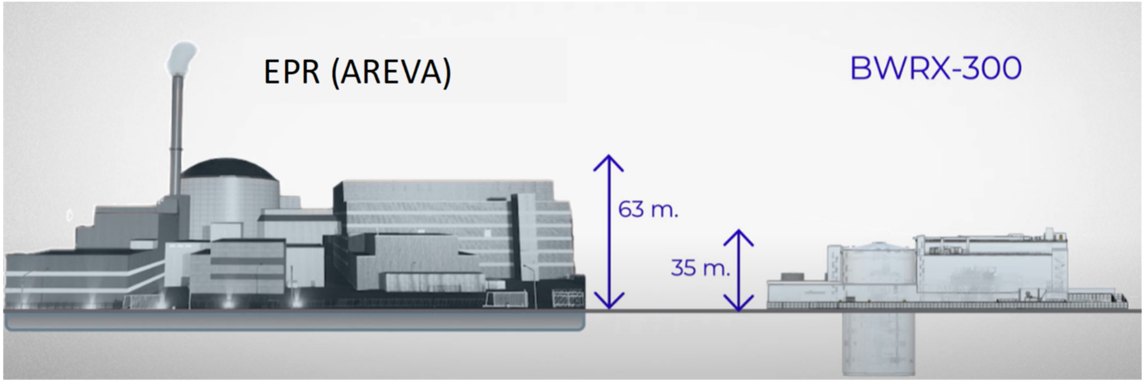


Figure 3.6 EPR (left) compared to BWRX-300 (right)

SMART, which stands for System-integrated modular advanced reactor, is an 107 MWe integral PWR developed by the Korea Atomic Energy Research Institute (KAERI). The eight once-through steam generators and the pressurizer are integrated within the RPV, with the main circulation pumps mounted horizontally directly on the RPV. The safety approach is based on passive safety systems, able to provide a grace period of 72 hours under DBAs. Standard design approval was received from the Korean Nuclear Safety and Security Commission (NSSC) in July 2012. In 2019 a pre-project engineering was completed for the deployment of SMART in Saudi Arabia. In the same year, Korea Hydro & Nuclear Power Co., (KHNP), KAERI and K.A.CARE (KSA's King Abdullah City for Atomic and Renewable Energy (KA-CARE) co-applied for standard design approval in Saudi Arabia.

SMR-160 is a PWR developed by Holtec International (US), with an output power of 160 MWe. The design is deployable in water-scarce locations using Holtec International's proprietary cooling systems with air as the ultimate heat sink. The safety concept is fully passive. The primary system consists of the RPV directly connected to a once through straight tube steam generator (SG), with an integral pressurizer present on top. The circulation of the coolant in the RPV is through natural circulation, eliminating the need for main recirculation pumps. The RPV and SG are situated inside a containment structure (CS), which is safeguarded by a containment enclosure structure (CES). This is fortified against missiles and shields the CS and safety systems from extreme environmental threats or sabotage. Almost half of the CS and CES are located underground, and all safety systems are accommodated within these structures. Holtec has entered pre-licensing in USA and UK, and is in phase 2 of pre-licensing in Canada. MOUs have been signed with Entergy Corporation, Ukraine and Czech Republic. It is one of six SMR designs shortlisted by UK for a contract to be awarded in 2024. It has also been short-listed by Ukraine with the first unit expected to become operative by 2029 and plan to build 20 more units. Recent developments:

- Holtec is investigating the potential to deploy the first SMR-160 unit at their Oyster Creek complex, where Holtec is decommissioning an older NPP.
- in April 2023, Holtec has signed a cooperation agreement with Energoatom for the deployment of the first SMR-160 unit, with connection to the grid planned by 2029. The plan involves expedited construction and commissioning of up to twenty (20) additional SMR-160 units in Ukraine and for establishing a manufacturing facility in the country for localizing the production of the variety of equipment required to build SMR-160 reactors [Ref].
- in April 2023 Holtec International and Hyundai Engineering & Construction have signed agreements with two Korean financial institutions, K-SURE and KEXIM, to support the deployment of SMR-160 nuclear reactors around the world.

- in September 2023 a long term power purchase has been signed by Wolverine Power Cooperative to restart the Palisades Power Plant in Michigan (USA). The agreement includes a provision to add up to two SMR at the Palisades site.
- in October 2023, it was announced that the SMR-160 is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

VOYGR developed by Nuscale (USA), it is an integral PWR SMR. The power plant can house up to 12 modules, each with a power output of 77 MWe, for a total of 924 MWe. Nuscale has received licensing by US NRC for its 50-MWe design, and is currently pursuing licensing for the 77-MWe uprated design. The Nuscale modular designs allow for smaller plants, with 4 (VOYGR-4) or 6 (VOYGR-6) modules. The approach to safety systems is fully passive. The circulation of the coolant in the RPV is achieved through natural circulation, so that the need for main circulation pumps is eliminated. The integral RPV is inside a stainless-steel containment, completely submerged in a water pool located in a below grade construction. The reactor is designed such that in case of a station blackout, sufficient cooling can be provided without the need for power or operator action and without the need for external water for an indefinite time. Containment, control room, and spent fuel pool are underground. Each reactor module is fully factory-built and shipped to the plant site by truck, rail, or barge.

Recent developments:

- As of November 2023, Nuscale was on track for submission of a Combined License Application (COLA) to the Nuclear Regulatory Commission in January 2024, for a six-modules SMR planned for construction in Utah starting 2024. This project was however cancelled in November 2023 due to a lack of sufficient subscribers among the 50 municipalities of the Utah Associated Municipal Power Systems (UAMPS). While 26 municipalities were still on-board with the project, a requirement was set by Nuscale and UAMPS to reach a minimum of 80% subscription level for the produced electricity. The cancelation was caused by increased interest rates (150% in 18 months) and increased material costs (steel, electrical equipment, and other construction commodities, with an increase of price index for steel by more than 50%). As a consequence of the costs increase, the Nuscale project was deemed to be no longer competitive with UAMPS gas and coal power plants. The project is planned to be replaced with a new gas plant.
- NuScale is currently still engaged in a project with Romania, in collaboration with Nuclearelectrica, for the construction of a VOYGR-6 plant at Doicesti. In August 2023 Romania’s National Commission for Nuclear Activities and Control (CNCAN) has approved VOYGR licensing basic document, confirming the design conforms with the national regulatory requirements. NuScale and RoPower Nuclear are currently conducting a Front-End Engineering and Design (FEED) study to analyse deployment at the Doicesti site, location of a former coal plant. A decision on the first 6 units is expected to be announced in 2025, while the NPP is planned to get commissioned in 2027. The project is co-financed by US with a contribution of 3 billion USD.
- In May 2023 NuScale has signed a MOU with Nucor Corporation (Nucor) to explore the construction of a VOYGR power plants to provide baseload electricity to Nucor’s scrap-based Electric Arc Furnace steel mills.
- In May 2023 the private power generation company GS Energy has signed a MOU with Uljin County in North Gyeongsang Province, South Korea, to consider the use of Nuscale VOYGR-6 plant for the Uljin Nuclear Hydrogen National Industrial Complex Project. , The SMR construction is planned to start in 2028.

- In October 2023 Nuscale was selected by Standard Power, a provider of infrastructure as a service to advanced data processing companies, to power their data centers in Ohio and Pennsylvania with a VOYGR-12 at each location, for a combined capacity of 1,848 MWe. Given the size of the Standard Power company, doubts have been raised on the realizability of the project.
- In October 2023, it was announced that the VOYGR is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

UK SMR is a PWR developed by Rolls-Royce (RR) with an output power of 470 MWe. Strictly speaking, it is a medium size PWR because its output power exceed 300 MWe, however its design is fully modular and can therefore be classified as a SMR. It is a 3-loop PWR, in which forced circulation is used for the primary coolant, with the main circulation pumps mounted directly to the outlet nozzle of the corresponding SG. In April 2023, Rolls-Royce has progressed to step 2 (out of three total) of the Generic Design Assessment (GDA). Memoranda of understanding are in place with Estonia, Turkey, Czech Republic, Ukraine and The Netherlands. They have also signed a MOU with Fortum in March 2023 to explore deployment opportunities in Finland and Sweden, and a memorandum of intent (MOI) with Industria (Poland) in February 2023 with plans to deploy the RR SMRs in central and southern Poland in the 2030s. A contract was signed with Westinghouse in October 2023 for the SMR fuel development. In October 2023, it was announced that the BWRX-300 is one of six designs shortlisted by the UK government for a contract to be awarded in 2024.

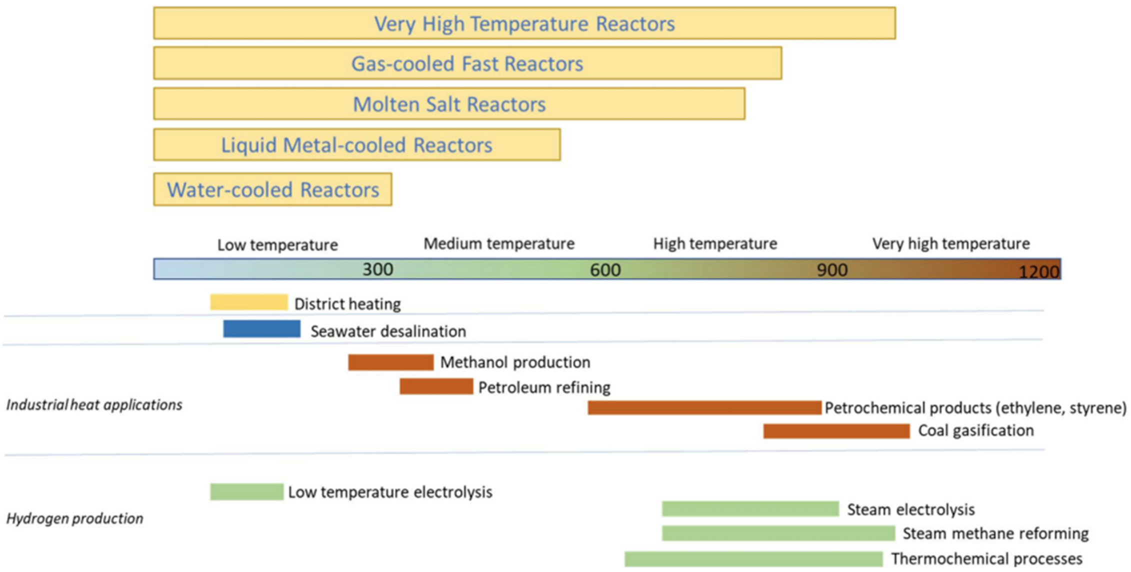


Figure 3.7 Output temperature of SMR technologies and corresponding non-electric applications [3.2]

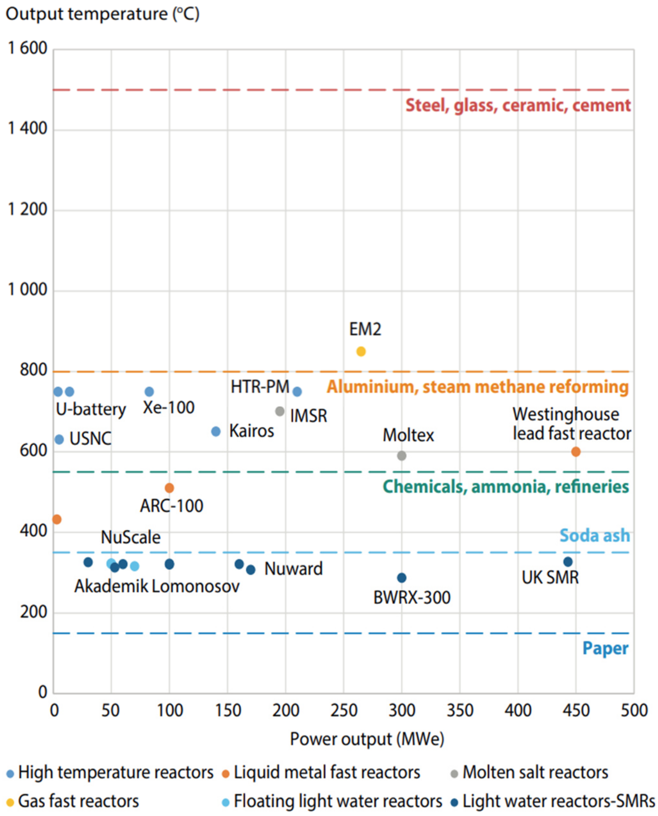
SMRs are intended for a variety of applications beyond just electricity production. In Figure 3.7 and Figure 3.8 the working temperature of SMR technologies and corresponding non electric applications are summarized.

LWR SMRs applications include, among others, district heating, heat for the paper industry, hydrogen production, and seawater desalination.

Use of NPPs for district heating, which requires steam or hot water in the range of 80 - 150°C with a distribution pipeline in the range of 10 - 15 km, has been already successfully deployed in Switzerland (with Refuna AG and the Beznau NPP) and several other countries, including Bulgaria, the Czech Republic, Hungary, Romania, the Russian Federation, Slovakia, Ukraine, and China.

LWRs have also already been deployed in several countries for water desalination, and recently two LWRs in the United States have been providing energy for a pilot hydrogen production plant, while since January 2023 Russia has started producing hydrogen through electrolysis at the Kola NPP.

The market for desalinated water is developing especially in arid regions with limited access to fresh water, such as the Middle East. The current global energy demand for desalination is estimated at 16 GWe of generating capacity and is therefore quite small Much larger is the market for hydrogen production, estimated to be already at a level of 574 GWe for electrolysis (or 803 GWth for thermochemical product processes) for the United States needs alone [3.12].



Examples of LWR SMRs non-electric applications:

- District heating
- Seawater desalination
- Hydrogen production
- Paper industry

Figure 3.8 Output temperature of SMR Technologies and corresponding non-electric applications [NEA].

3.2 Safety philosophy of LWR SMRs

As for the large Gen-III+ NPPs discussed in Chapter 2, the safety concept of LWR SMRs is based on passive systems, which do not require external power or operator action for their functioning. Some of the safety systems adopted for SMRs are similar to the one developed for the large Gen-III+ LWRs. An example is given by the BWXR-300 isolation condenser system (Figure 3.9), which passively removes heat from the RPV and transfers it to water pools, similarly to the design of the ESBWR plant (see Figure 2.11, right).

However, the reduced size of SMRs compared to large LWRs allow for further enhancement of the safety concept. In particular:

Elimination of certain accidents: because of the compact design, the risk of accidents scenario such as large brake LOCA is either eliminated by design, such as in integral SMRs, or strongly reduced. Many of the PWR designs (e.g. ACPR50S, CAREM, KLT-40S, NUWARD, UK SMR and RITM-200 series) do not foresee the use of soluble boron for reactivity control, eliminating the risk of boron dilution accidents. In addition, some of the SMR designs rely on natural circulation for the main coolant flow, eliminating the risk of loss of flow accidents (LOFAs).

Increased cooling efficiency: SMRs have a much larger RPV coolant inventory per unit power compared to large LWRs. This means that overheating transients will be slower in SMRs compared to large LWRs, providing for more time for the safety systems to intervene.

SMR have a higher RPV surface-to-power ratio. The larger heat transfer area per unit power allows for a more efficient deployment of passive heat removal.

In addition, the elongated design of the RPV leads to an enhancement of the natural circulation capabilities, yielding to higher coolant mass flux through the reactor core. This allows for efficient use of natural circulation during accident conditions (in some of the SMR designs, the natural circulation flow-rate is sufficient even for normal operation, so that the main circulation pumps are no longer needed).

As a consequence of all these features, the time passive safety systems can provide sufficient cooling without the need for integrating the reactor water pools with additional water can be significantly extended when compared to large Gen-III+ NPPs.

Use of containment vessel in large pools: because of the SMR compact size, a small containment volume is needed, which allows to use a steel vessel for the containment itself, instead of the large reinforced concrete structures needed for large NPPs. The RPV and steel containment vessel are then placed below or within a large pool of water, providing a reliable heat sink with a large heat capacity. Heat is removed through free convection in the water pool and latent heat of vaporization of the water mass. This arrangements allow for multiple days of independency for cooling the reactor, from a few days (e.g. 7 days for the BWXR-300) to unlimited time (e.g. NUSCALE).

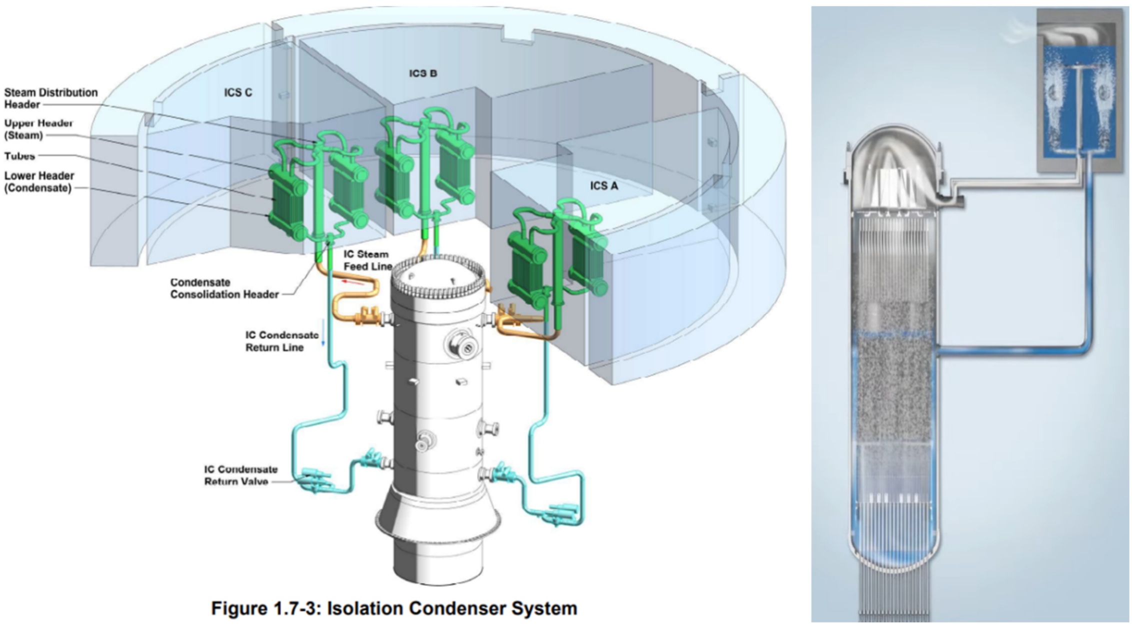


Figure 3.9 Passive heat removal system in BWRX-300

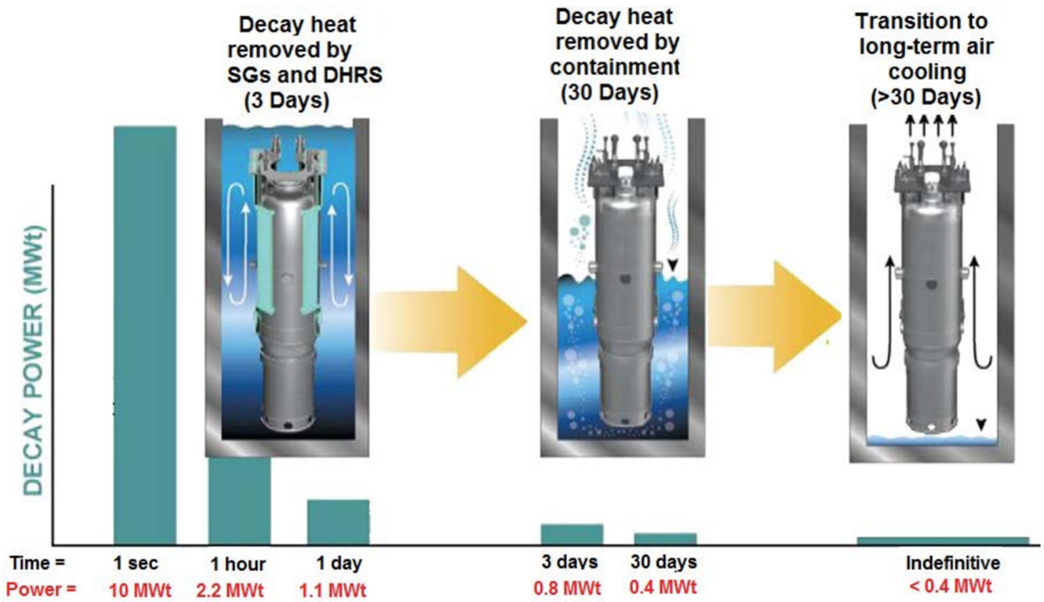


Figure 3.10 NUSCALE passive heat removal concept in case of a station black-out event [3.11]

In case of the NUSCALE design, for example, in the event of a complete station black-out (SBO) such as the one at Fukushima Daiichi, passive cooling is sufficient for an unlimited period of time without need for external action, as the water pool surrounding the containment containing the RPV is dimensioned such that by the time the pool water is evaporated, the decay heat has decreased to levels for which free convective heat transfer with the surrounding atmosphere is sufficient. In particular, the reactor pool is sufficiently large to passively cool the reactor for at least 30 days, without the need for any source of power, operator action, or make-up water. After that, water boil-off and passive air cooling later of the containment vessel are sufficient to provide adequate cooling for an unlimited period of time [3.11], as illustrated in Figure 3.10.

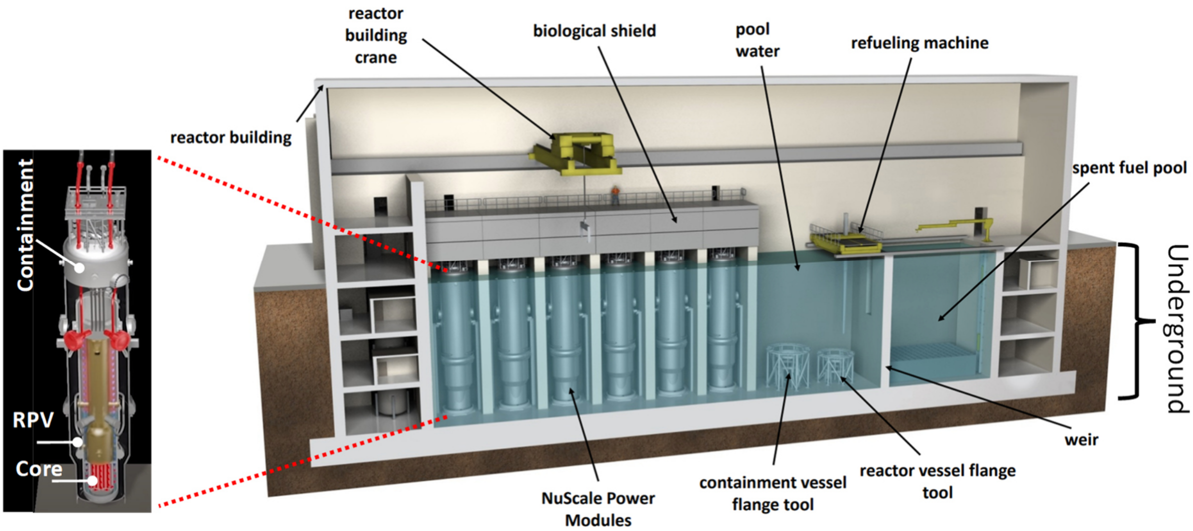


Figure 3.11 NUSCALE configuration for integral RPV, containment and reactor building

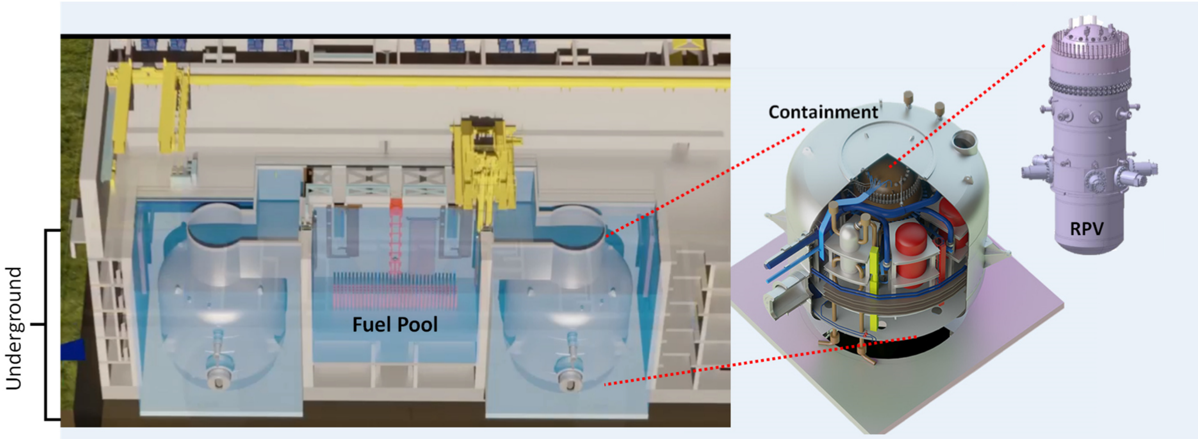


Figure 3.12 NUWARD configuration for integral RPV, containment and reactor building

Underground construction: because of the reduced size, it is economically feasible to construct the primary reactor system fully below ground level, which significantly increase the protection of the reactor against external events such as airplane crash or natural disasters, such as earthquakes or flood. The below-grade construction of both reactor and containment vessels allow therefore reducing the number of paths for fission-product release in the event of an accident.

The areas for fuel handling and the spent fuel pools are also typically underground within the containment building.

Increased reliability: the simpler design translates in a lower number of components and therefore a higher reliability of the plant and the safety systems. In the unlikely event of a severe accident with core melt, an in-vessel retention strategy can be pursued.

Reduced source term: SMRs have a smaller radioactivity inventory, since this is proportional to the reactor power level.

Examples of the implementation of such characteristics are shown in Figure 3.10 and Figure 3.11, representing the NUSCALE and NUWARD design respectively. In both cases, the RPV of the integral PWR is included in a steel containment vessel, which is placed underground in a water pool. In the figures, the placement of the spent fuel pool can also be seen, underground within the reactor building, which is designed to protect against external events, including plane crash.

3.3 Emergency planning zone for SMRs

All the features described in section 3.2 result in enhanced safety characteristics against the potential release of radioactivity to the environment. This is the reason why the United States, which are at the forefront of SMR licensing among the western countries, have adopted new regulations for the licensing of SMRs and for the sizing of the Emergency Planning Zone (EPZ).

The plume exposure pathways for the EPZ and ingesting pathway EPZ for large NPPs are set by US NRC at a 10-mile and 50-mile radius, respectively. Recently a new SMR emergency preparedness rule was approved by the US Nuclear Regulatory Commission (NRC). The rule would adopt a consequence-oriented approach, and therefore a scalable plume exposure pathway emergency planning zone, that is based on the following principles [3.15]:

- **risk-informed:** the rule considers the actual risks and consequences of potential accidents or incidents at SMRs and other new technologies, which may be lower or different than those at large LWRs (e.g. because of smaller radioactive inventory, new safety concepts, etc).

- **performance-based:** the rule sets performance objectives and criteria for emergency preparedness programmes, rather than prescribing specific requirements or methods. This allows flexibility and innovation for applicants and licensees of SMRs and other new technologies.
- **technology-inclusive:** the rule applies to a range of new technologies, such as non-LWRs, new research and test reactors, and medical radioisotope facilities. The rule does not apply to large LWRs, fuel cycle facilities, and currently operating research and test reactors, which have their own emergency preparedness requirements.
- **consequence-oriented:** the rule includes a scalable method to determine the size of the offsite emergency planning zone around a facility, based on the potential radiological impacts of an accident or incident.

According to the new approach, the area within which the public dose would exceed 10 millisieverts over 96 hours after a release is considered. That is, rather than setting a predetermined fixed distance for the EPZ, the distance would be determined by the potential consequence of an accident based on factors such as accident likelihood and source term, timing of the accident sequence, and meteorology.

As a result of the new regulations, US NRC has approved the size of the EPZ for the NUSCALE SMR to be limited to the plant site perimeter. It is expected that other SMRs would receive a similar ruling in the United States.

3.4 SMRs challenges

SMRs might face economical and regulatory hurdles (Refs. [3.8], [3.12]-[3.14]):

- **Economics:** SMRs still need to prove their economic competitiveness, especially in comparison with large NPPs and other low-carbon energy sources. SMRs do not have the same economy of scale as large NPPs. It would have to be seen whether enough units can be built in the future to replace the economy of scale of large NPPs with the economy of series production (resulting from the SMRs modularization and off-site factory production), and whether this in combination with the shorter construction times would lead to overall lower costs.

  Economy of series production would require a large and global market for a single design to achieve cost reductions through production of multiple units and learning effects. In China, where a well-established nuclear energy supply chain exists, the first water-based SMR, Linglong One (125 MWe), is currently being built for a total cost of 5 billion yuan (< 630 million CHF). The construction is running ahead of schedule, with the power plant completion to be expected in 53 months instead of the original 58.

- **Licensing:** the licensing process for a new design is lengthy and costly (e.g. the cost of licensing the NUSCALE design amounted to USD 500 million). To accommodate the innovative features and deployment modes of SMRs, there is a need to review and adapt the existing international and national legal and regulatory frameworks, which are mostly based on the experience with large LWRs. In addition, costs for construction and operation licenses are not expected to be less than for large NPPs. Obviously, the number of units built in any given country would have an important impact on licensing costs per unit. The cost and schedule for regulatory approval is therefore considered a major obstacle for FOAK units.

- **Fuel:** some advanced (non water-cooled) SMRs designs require the development of new fuel cycle facilities and capabilities, for the production and transport of high-assay low-enriched uranium (HALEU) fuel.

3.4.1 SMRs fuel

LWR SMRs have very similar fuel requirements as large LWR NPPs. Therefore, fuel does not pose a challenge for these types of designs.

Some advanced designs, such as the sodium-cooled Terrapower reactor, require instead HALEU fuel, which currently is produced at industrial scale only in Russia. To eliminated this geopolitical dependence, the US DOE has launched a new program to establish a commercial-scale, domestic production capabilities for HALEU fuel, and assure the availability of HALEU fuel for the demonstration and commercial deployment of advanced reactors in the United States. Consequently, a new fuel facility was built and is currently being operated in Ohio by Centrus, a nuclear fuel and services provider, and has already delivered the first batches of HALUE fuels. The company will supply the fuel for the Terrapower SMR units planned for construction in Wyoming.

3.4.2 SMRs licensing

In most industries, the bulk of the risks for new developments lie in the technology itself and the market. These risks are also present with nuclear technologies. However, an additional risk for nuclear is posed by the regulatory framework [3.16].

As a matter of fact, licensing is one of the main challenges for the deployment of SMRs because the current licensing frameworks existing in most countries are tailored on large NPPs and not well suited for SMRs, which have different characteristics than large NPPs. The licensing process is a multi-stage procedure involving regulatory bodies, industry stakeholders and the public consisting of the following steps:

<u>Pre-licensing:</u> in this stage, the SMR designers engages in informal discussions with the regulator to exchange information and receive guidance on the licensing process. It helps the designer identify potential issues before proceeding to the design certification.

<u>Design certification:</u> at this stage, the SMR designer submits a formal application to the regulator for the detailed design review. At the end of this stage, if successful, the regulator issues the design certification, indicating that the design is acceptable for use in that specific country

<u>Site permit:</u> at this stage, a site for construction is selected. The regulatory body will assess whether the site meets the required criteria.

<u>Construction and operating license:</u> in these two steps it must be demonstrated that the design, site and operator meet all safety, environmental and security requirements.

Regarding licensing in countries who do not have yet SMRs in operation or in construction, considerable advances have been made in the United States, where the US NRC has approved a new SMR emergency preparedness rule (see section 3.3 for details).

Even in the United States, though, under their existing framework, licensing of SMRs and other advanced designs is a severe hurdle for the financial viability of new nuclear projects. As of today, only the Nuscale VOYGR SMR has received a design certification from the US regulator NRC, with a lengthy and costly effort, as the licensing of the Nuscale VOYGER design took 10 years and $500 million. This because the current licensing framework is tailored for applications which have already a complete design and area backed by a commercial order.

This is why in 2016, based on more innovation-friendly licensing frameworks already existing in UK and Canada, recommendations were made by US industry to introduce a staged licensing process for new designs, which would help minimizing risks and attract investors [3.16], as illustrated in Figure 3.13. The recommendations called for the introduction of a licensing project plan (LPP), a living document providing a roadmap for the entire licensing process, including project schedules, testing requirements, deliverables, and NRC review budgets [3.16]. In 2019 the Nuclear Energy Innovation and Modernization Act (NEIMA) was signed into law that requires NRC, among other provisions, to follow a staged pre-licensing approach similar to the one in force in UK and Canada.

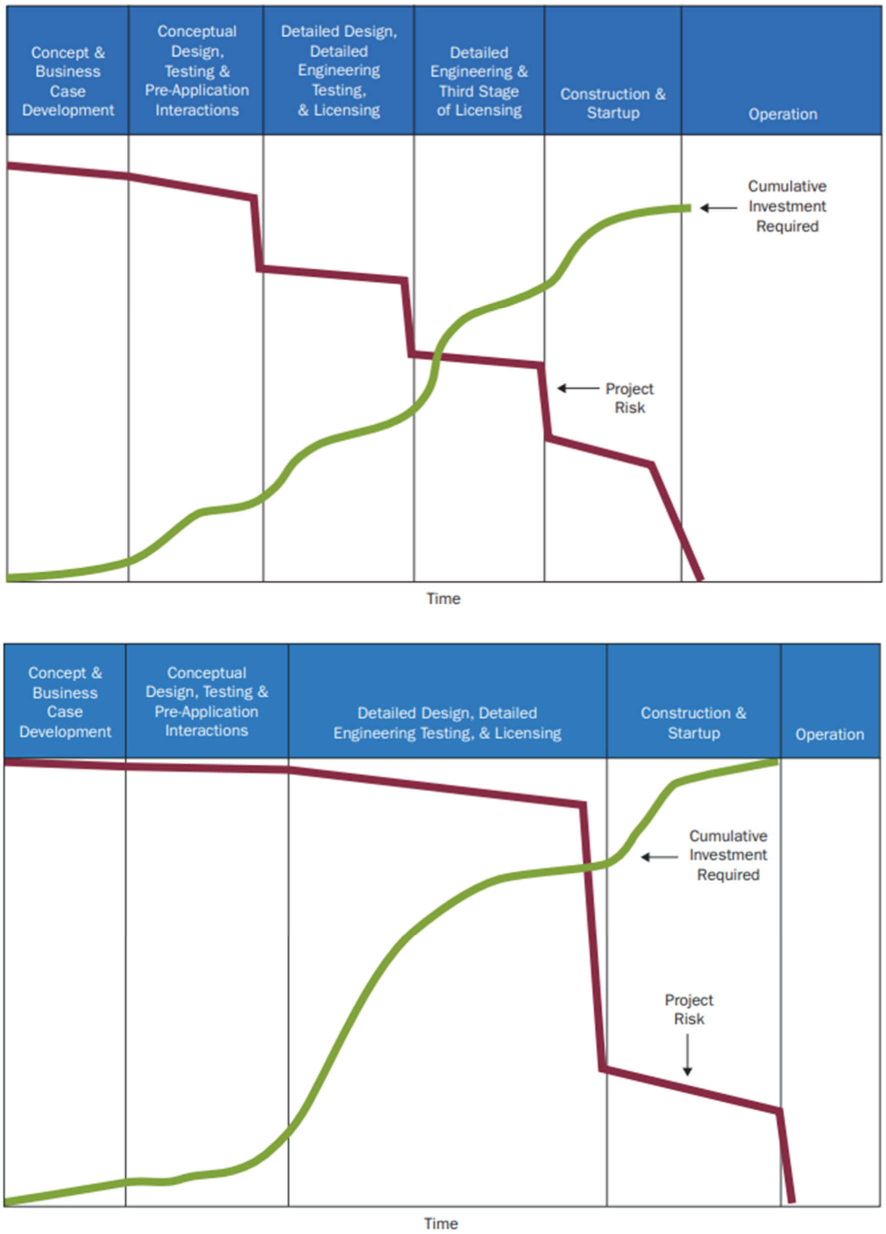


Figure 3.13 Desirable (top) vs current project risk in current US licensing frameworks for new nuclear technologies [3.16].

UK and Canada have a structured, step-wise pre-licensing design review process that provides earlier opportunities for reactor vendors to demonstrate to their investors and potential investors that the reactor design technology will be licensable [3.16], [3.17]. In Canada, the pre-licensing is organized into three steps:

1) Phase 1 (compliance with regulatory requirements) takes about 1 year long, in which the nuclear authority provides feedback on whether the design demonstrates an understanding of the licensing requirements. This phase can be started once the conceptual design is complete and the preliminary engineering is complete or at an advanced stage.
2) Phase 2 (pre-licensing assessment) takes about 1.5 – 2 years to complete, in which the authority will assess fundamental barriers to licensing. This phase can be started after the preliminary engineering is complete or at an advanced stage and phase-1 has been successfully completed.
3) Phase 3 (pre-construction follow-up), in which the vendor follows up with the nuclear authority on one or more areas covered under the previous two pre-licensing stages, with the goal of obtaining a throughout review of selected topics to avoid detailed reassessment of those areas during the construction license review.

In UK the pre-licensing process consists of four steps instead, at the end of which concerns or technical issues are highlighted. In UK a cost limitation agreement for pre-licensing exists that allow the vendor to agree on a ceiling for the costs that the nuclear authority can incur for pre-licensing review activities.

Several efforts are currently on-going world-wide to facilitate licensing of SMRs. Since 2022 the International Atomic Energy Agency (IAEA) has launched a SMR platform and the Nuclear Harmonization and Standardization Initiative (NHSI) to develop common regulatory and industrial approaches to SMRs, facilitating their safe and secure deployment [3.18], speeding up procurement, reducing production delays and costs, and ensure reliable supply chains compliant with safety requirements. The initiative consists of two separate but complementary tracks, the NHSI regulatory track and the NHSI industry track.

The **NHSI Regulatory Track** aims to increase regulatory collaboration among Member States, avoid duplication of efforts, increase efficiency, and facilitate the development of common regulatory positions without compromising nuclear safety and national sovereignty.

The **NHSI Industry Track** aims to develop more standardized industrial approaches for SMR development, manufacturing, construction, and operations. This track is focused on four main objectives:

- harmonization of high-level user requirements, to establish a common set of requirements that SMRs must meet, which can help streamline the design and licensing process;
- information sharing on national standards and codes used in different countries, to identify areas of commonality and divergence, which can inform efforts towards harmonization;
- experiments and validation of simulation computer codes to model SMRs;
- accelerating the implementation of a nuclear infrastructure for SMRs, to develop the necessary infrastructure to support the deployment of SMRs, including regulatory frameworks, supply chains, and workforce training programs.

Other harmonization efforts on-going for licensing of SMRs include the establishment of bilateral or multilateral cooperation agreements among regulators and vendors of SMRs. These agreements aim to facilitate the sharing of technical data, safety analysis, design evaluation and inspection results, as well as to harmonize the licensing requirements and criteria for SMRs. Some examples of these agreements are

- the Memorandum of Cooperation between the US NRC and the Canadian Nuclear Safety Commission (CNSC) on SMRs,
- the Memorandum of Understanding between the UK Office for Nuclear Regulation (ONR) and the US DOE on SMRs.
- The cooperation between the regulators of France (ASN), Finland (STUK) and Czech Republic (SÚJB) together with their technical support organizations (IRSN for France and SÚRO for Czech Republic), who between June 2022 and June 2023 together with their technical support organizations (IRSN for France and SÚRO for Czech Republic) have conducted a joint review of the EDF NUWARD design. This because energy companies in all these three countries have expressed interest in deploying this SMR design [3.19].

In Europe, WERNA, the Western European Nuclear Regulators Association has made several efforts to expedite the licensing process for SMRs, including harmonization of safety requirements, encouraging collaborations among regulatory bodies of different countries through joint safety assessments, and by developing processes to benefit from the regulatory assessments previously completed by other national regulatory bodies.

In US a government initiative was launched, the Foundational Infrastructure for Responsible Use of Small Modular Reactor Technology (FIRST) Program, with Japan and South Korea as contributing partners together with US. The goal of the FIRST program is to provide capacity-building support to help partner countries build SMRs or other advanced reactor technologies. Partnering countries include Estonia, Latvia, Ukraine, Romania, Serbia, Kazakhstan, Ghana, Kenya, Rwanda, Namibia, South Africa, Thailand, Indonesia, Philippines, and Malaysia.

These activities are complemented by bilateral or multilateral efforts, such as the pre-licensing activities for the SMR NUWARD design undertaken by the French nuclear authority ASN, with the Finnish authority STUK and the Czech authority SUJB.

Other regional and bilateral collaborations include also Argentina, China, Russia, and South Korea, among others.

3.5 Conclusions

SMRs offer smaller, simpler, and more flexible designs that can be deployed in remote locations, integrated with renewable energy sources, or used for non-electric applications. They have the potential to address some of the barriers to nuclear energy deployment, such as high upfront costs, long construction times, and grid limitations. However, they also pose technical, regulatory, and market uncertainties that need to be addressed before they can achieve commercial viability and widespread deployment.

3.6 References

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4 Microreactors

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Over the past seven years an unprecedented trend has emerged on so-called microreactors, designed to produce electrical power in the range of up to about 10 MWe [4.1] and defined by the US DOE as plug-and-play reactors. Specifically, microreactors are designed:

- to be fully factory-built and portable;
- to fit into a ISO container for easy transportation to the deployment site (see illustration in Figure 4.1);
- to be coupled to ultra-compact conversion systems (e.g. Stirling engines, supercritical CO_2 cycles, direct conversion devices);
- to have long fuel cycles, with refuelling every 5-10 years, depending on the design;
- to be capable of semi-autonomous or fully autonomous operation;
- to operate as part of the electric grid, independently from the electric grid, or as part of a microgrid. They are designed to power, among others, remote (e.g. rural) communities currently relying on diesel generators, mining sites, industrial complexes, military bases or expeditionary forces [4.2], off-shore platforms;
- to be deployable for restoring power quickly in communities affected by natural disasters and for restoration of critical infrastructure such as hospitals and drinking water supply;
- to provide electricity or industrial heat for desalination, hydrogen production and other industries.

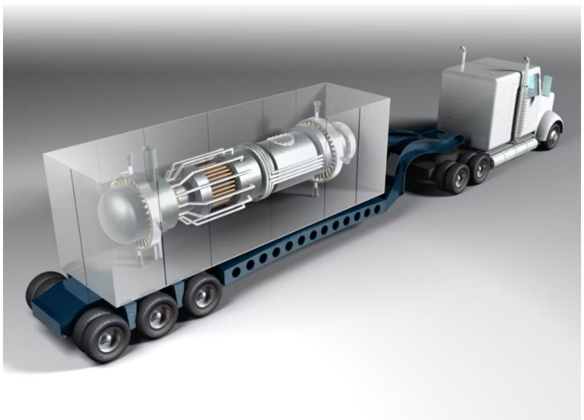


Figure 4.1 Microreactor in ISO container

In the latest IAEA summary [4.1] twelve microreactor designs are described, this list is however not exhaustive. Different types of coolants are being considered, including helium, molten salt, and liquid metal. Another cooling system option that has been proposed and considered to be very promising is the heat pipe.

A heat pipe consists of a fully sealed cylindrical pipe containing a fluid and a wick (porous) structure around the inner periphery of the tube (see scheme in Figure 4.2 and a reactor implementation in Figure 4.3).

One extremity of the heatpipe, called evaporator, is placed in the hot zone of the system (i.e. the reactor core). The fluid in the heatpipe evaporator section is vaporized and starts flowing toward the other extremity of the heat pipe, the condenser section. Here the heat is rejected to the heat sink (e.g. to produce electricity) and the vapor is condensed. Because of capillary forces in the wick structure the condensate flows back to the evaporator end. In this way a stationary circulation of the fluid inside the heatpipe is established and heat can be transferred from the evaporator region (reactor core) to the condenser region (outside the core). This fully passive cooling system is of particular interest for microreactors because it does not require gravity. Therefore, heatpipes can function in horizontal orientation and are also suitable for space applications. Different fluids can be used to design a heatpipe.

For nuclear applications, sodium heatpipes are most suitable in view of the high operating temperatures (700 °C and above).

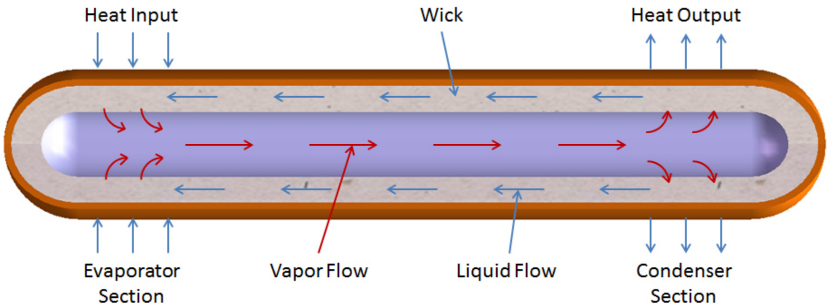
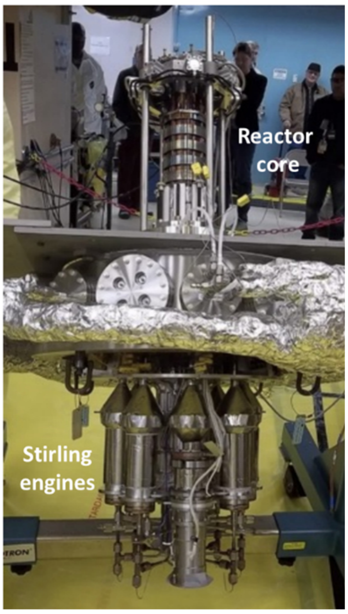


Figure 4.2 Working principle of a heatpipe

The first application of microreactors was originated by NASA. In 2015 NASA launched a collaboration with Los Alamos National Laboratory to work on the development of the Kilopower reactor, a very small-size (~ 1-10 kWe) reactor designed to power deep space missions (e.g. Mars mission) and intended to provide electric power in space, or on the surface of planets or moons. The project was completed within just three years, with a total cost of less than $20 million. Within the project, the prototype KRUSTY (Kilopower Reactor Using Stirling Technology) was successfully demonstrated in 2018, with an extensive experimental program aimed at testing the reactor concept operation, its stability and load follow characteristics [4.3]. KRUSTY operated at a nominal thermal power of 4 kWth and an outlet temperature of 800 °C. Highly enriched uranium (HEU) was used as fuel, while cooling of the core was achieved through sodium heatpipes.

A photo of the KRUSTY reactor is reported in Figure 4.3 (left) together with a representation of how the KRUSTY reactor design will be converted into the Kilopower design for space applications. An illustration of the Kilopower deployment in space is shown in Figure 4.4.



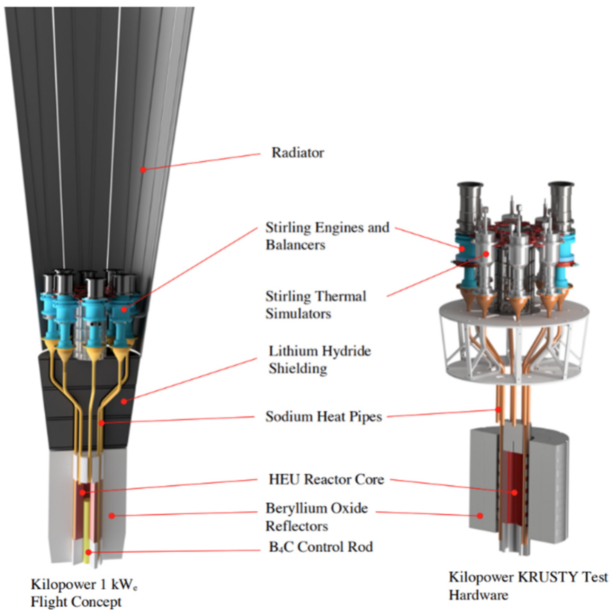


Figure 4.3 Photo of KRUSTY (left) and illustration of kilopower vs KRUSTY (right) [4.3].

Following the success with KRUSTY and the Kilopower project, NASA has partnered with DOE to issue a request of proposals from US companies to design a fission surface reactor that could be ready to launch within a decade for a demonstration on the Moon. In June 2022 three concepts were down-selected and are currently being developed:

- Lockheed Martin, partnering with BWXT and Creare;
- Westinghouse, partnering with Aerojet Rocketdyne;
- IX of Houston, a joint venture of Intuitive Machines and X-Energy, partnering with Maxar and Boeing.

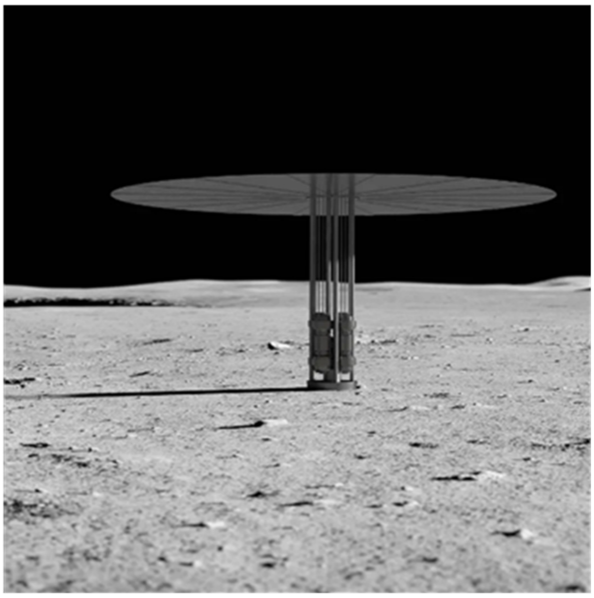


Figure 4.4 Impression of Kilopower reactor deployment in space [NASA].

Parallel to the development of nuclear fission for surface deployment, the Lockheed Martin/BWXT team together with the Space Nuclear Power Corp were also awarded a contract from the Defense Advanced Research Projects Agency (DARPA) to develop and demonstrate JETSON, a nuclear-powered spacecraft to benefit space exploration and national defense. The goal is to develop a reusable nuclear spacecraft for the transport of human and goods to a Lunar base. The reactor will power spacecraft payloads and electric thrusters for propulsion. The in-space flight demo of the nuclear thermal rocket engine vehicle is planned for launch no later than 2027. JETSON is expected to generate 6-20 kWe, four times the power of conventional solar arrays without the need to be in continuous sunlight.

The success of the Kilopower project sparked the interest of US DOE and the US department of defence (DOD), who subsequently launched new R&D programs leading to the development of several microreactor concepts in the US aimed at output powers in the MW power range. Development of micro-reactors is being supported in US through the DOE Advanced Reactor Technologies Program and by several companies, including General Atomics, Oklo, Westinghouse, X-Energy, HolosGen, and Ultra Safe Nuclear.

In Table 4.1 a summary of selected microreactor designs is presented. Some microreactors concept are being developed outside of the United States, but are still at an early stage of design. Currently, the only microreactor under construction is MARVEL, scheduled to start operation on the INL site by early 2025. Several designs are however undergoing licensing processes in Canada and in the USA for imminent deployment. The Hermes microreactor by Kairos has completed a construction license review in October 2023 and is currently awaiting US NRC license decision to start construction. The other designs in the list are at various stages of design pre-licensing. Because of the very small size, microreactors can be designed for complete factory-fabrication, have a much simpler design than SMRs or large NPPs and it is not a challenge to implement a fully passive safety strategy, with air as the ultimate heat sink. Therefore, it is expected that the development of microreactor designs would proceed quicker than for SMRs and large NPPs.

MARVEL: the Microreactor Applications Research Validation and EvaLuation (MARVEL) is a microreactor currently being built at Idaho National Laboratory with US DOE support. It is a liquid-metal (sodium-potassium) cooled microreactor with Stirling engines that will produce 100 kilowatts of energy using small amounts of high-assay, low-enriched uranium (HALEU). The fuel is in form of uranium zirconium hydride (U-ZrHx) with 19.75% U235 enrichment. The MARVEL project has been developed to test fundamental features, operation and behaviour of microreactors and to help researchers and end users understand how microreactors can integrate with other technologies (electrical and thermal applications). In particular, it will be used to test and demonstrate microreactors capability to manage grid demand to support a wide range of applications, including water purification, hydrogen production, industrial heat and integration with renewables. Operation is expected for early 2025. A photo of MARVEL is presented in Figure 4.5.

Figure 4.5 MARVEL microreactor. Left: photo of MARVEL; right: illustration of reactor placement in building cavity

BANR: the BWXT Advanced Nuclear Reactor (BANR) is a HTGR reactor part of the U.S. Department of Energy's Advanced Reactor Demonstration Program (ARDP). It has also been selected by the DOD for their mobile microreactor under the Pele project. Therefore, BWXT is pursuing two designs, a baseline design meant for stationary deployment, and a mobile design aimed at reducing the need for vulnerable fossil fuel deliveries relied on by the U.S. military, and to provide power for disaster response and recovery. The BANR microreactor uses helium as coolant and TRISO particles as fuel (19.75% enrichment), with refuelling once every 5 years. It is designed to provide as output electricity, steam for process heat, or both in cogeneration mode. As for other microreactors, cooling is passive and the design meets the standard shipping requirements for rail, trucks, and ships. An illustration is reported in Figure 4.6. In September 2023, BWXT was awarded a contract by the Wyoming Energy Authority to assess the deployment of microreactors to provide baseload power and heat for remote, off-grid applications in Wyoming (USA). In June 2024, a second-phase of the contract was awarded, focused on the completion of the microreactor conceptual design and demonstration of the supply chain.

EVinci: developed by Westinghouse, eVinci is a micro-reactor passively cooled with sodium heatpipes (same as for the KRUSTY demo), designed for semi-autonomous operation. and scalable energy generation ranging from 200 kW_e to 15 MW_e. As for other microreactors, also eVinci is meant to provide combined heat and power for military installations, remote communities and mining installations, and designed to be fully factory-assembled and transportable in shipping containers via rail, barge, and truck. HALEU (19.75% enrichment)

TRISO fuel is used with target of at least 10 years operation without refuelling and maintenance. The heat removal is completely passive through heatpipes, via conduction and radiative heat to the outer canister and via natural convection through air ducts. It requires no water for cooling or operation. Instrumentation and control is designed for remote monitoring. It is currently under Phase-2 of pre-licensing in Canada. In October 2023 Westinghouse also received an award by the US DOE for supporting a demonstration unit at the INL site. An illustration is reported in Figure 4.6.

MMR: the Micro Modular Reactor (MMR) Energy System developed by Ultra Safe Nuclear Corporation (USNC) is a HTGR microreactor using helium as coolant and TRISO particles as fuel. Heat from the reactor is stored in molten salt tanks similar to the ones used in concentrated solar power plants and in the Terrapower SMR design. This allows for additional flexibility in the supply of both electricity and process heat. As for other microreactors design, it uses no water and does not need an electrical rid or infrastructure support for operation. The University of Illinois at Urbana-Champaign (UIUC) plans to use a MMR unit as a test reactor for training and research, and to provide district heating and power to the university campus. UIUC is directly involved in MMR design and integration. At the same time, Ultra Safe Nuclear is pursuing licensing of the MMR design in Canada (currently under Phase-2 of pre-licensing), where the FOAK is planned for construction at the Canadian Nuclear Laboratories (Chalk River). In October 2023 Ultrasafe has also received an award by the US DOE for supporting the demonstration of a 1-5 MW unit (Pylon) at the INL site, with testing starting in 2026. An illustration is reported in Figure 4.6.

Hermes: it is a microreactor developed by US company Kairos Power LLC, cooled used a fluoride molten salt, Flibe, with excellent heat transfer properties and fission product solubility. HALEU TRISO particles are used as fuel. The high operating temperature allows for a superheated Rankine cycle. Passive safety is deployed. The first unit to be built at the Oak Ridge Site (Tennessee, USA) is meant to demonstrate the larger full-scale SMR developed by the same company. Two additional units instead will generate electricity through a shared Rankine cycle. The Hermes microreactor is part of a new approach to reactor development, which involves building a smaller version, perform testing, and then build a larger version, similar to the approach successfully deployed by high-tech enterprises like SpaceX. The construction license revision for the Oak Ridge unit has been completed by US NRC in October 2023 and Kairos is currently awaiting for US NRC to issue a decision on the construction permit. The operation of the first unit is targeted for 2026. A construction license for the additional units has been submitted in summer 2023 and the decision by US NRC is expected by fall 2024.

Xe-mobile: the Xe-mobile microreactor is developed by the US company X-energy, and consists of a HTGR pebble bed reactor cooled with helium and using HALEU graphite pebble fuel elements containing TRISO fuel particles. The reactor can be run autonomously, without any operators present on-site, with the FOAK demonstration plant requiring very few operators. As for the other microreactors, also Xe-mobile is designed for flexible operation supplying a combination of power and heat. The fuel cycle will allow a 10-years operation on a single core load. The design is being developed with financial support from the US DOE and DOD.

Kaleidos: this microreactor is being developed by Radiant Industries, and consists of a HTGR cooled with helium and using HALUE TRISO particles as fuel. Passive cooling with air as ultimate heat sink is deployed. A 5-years or longer fuel cycle is foreseen. Targeted commercialization is for 2028, with the first demonstration unit to be built at INL by 2026. Kaleidos was one of three microreactors which received funding by the US DOE in October 2023 to support a demonstration unit at the INL site. A centralized 24/7 fleet monitoring system would track the health of each microreactor. An illustration is reported in Figure 4.6.

All these designs make use of TRISO fuel, which is specifically design to withstand very high temperatures.

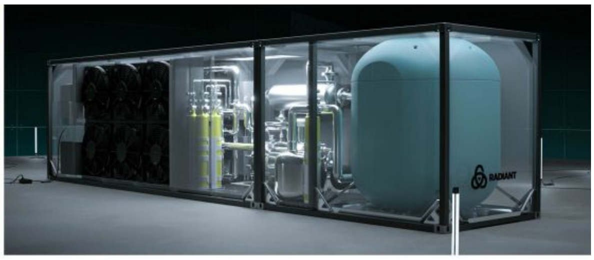
Table 4.1 Status of selected microreactors designs

Design	Developer & Reactor type	Power	Notes
MARVEL	INL (USA) NaK cooling	85 kWth 20 kWe	Under construction at INL. To start operation beginning of 2025.
Hermes	Kairos (USA) Fluoride cooled HTR	35 MWth	Construction permit received in December 2023. Operation targeted to start in 2026. A construction license application was submitted for two additional test units (with shared steam powered conversion system for electricity production) in July 2023. Decision expected by fall 2024.
eVinci	Westinghouse (USA) Na-heatpipes 15m^2 foot print < 2000m^2 site	13 MWth 5 MWe	Pre-licensing in Canada and USA. MOUs with Penn State Univ. (USA) and Canadian Saskatchewan Research Council. Funding from US and Canada governments. 8 years operation before refuelling. A manufacturing facility was launched in October 2023, scheduled to be completed before summer 2024. In November 2023 Canadian Saskatchewan Research Council received $80 million government funding to support licensing of eVinci in Canada.
MMR	Ultra Safe Nuclear (USA) HTGR (Helium)	10-45 MWth 3.5-15 MWe	Phase 2 pre-licensing in Canada. Submitted license application for site preparation. Letter of intention to US NRC for construction at Urbana-Champaign. Demo unit scheduled for 2026. In November 2023, the Manila Electric company Meralco has signed a contract to study the deployment of one or more MMR units in the Philippines.
Kaleidos	Radiant (USA) HTGR (Helium)	1.9 MWth 1.2 MWe	Pursuing a demo unit at Idaho National Laboratory by 2026.
BANR*	BWXT (USA) HTGR (Helium) 15m^2 foot print < 2000m^2 site	50 MWth 17 MWe	In September 2023 a contract signed with Wyoming Energy Authority to assess viability of deployment for baseload power and heat for remote, off-grid applications in Wyoming (USA). In June 2024 a second phase of the contract was signed focused on completion of conceptual design and demonstration of the supply chain. Goal is deployment of the microreactor Partially funded by US DOD (Pele project for a terrestrial application and DRACO project for nuclear thermal rocket application, in collaboration with NASA). In September 2023 shipping and energy supply chain firm Crowley has signed a MOU with BWX for a ship concept with a microreactor that will supply nuclear energy to shoreside locations for defense and disaster needs.
Xe-Mobile	X-energy (USA) HTGR (Helium)	2-7 MWe	Partially funded by US DOD (Pele project) and by US DOE (contract signed in October 2023)

MMR – Ultrasafe

eVinci - Westinghouse

BANR - BWXT



Kaleidos - Radiant

Figure 4.6 Example of microreactors currently under development

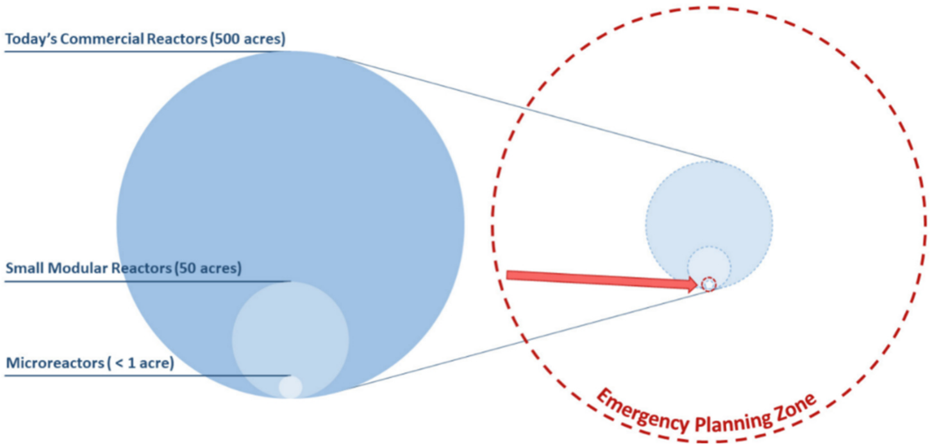


Figure 4.7 Comparison of estimated reactor footprints and emergency planning zones [4.4]

4.1 Microreactors challenges and opportunities

Several features of microreactors make them attractive for deployment, such as:

- the simplified design and passive safety features, including possibility of full underground construction (see examples in Figure 4.6).
- the factory construction which eliminates the uncertainty associated to large construction sites
- the transportability,
- the reduced infrastructure requirements (e.g. no need for cooling water or grid connections for off-site power),
- the reduced footprint (~15 m^2), small plant site (< 2000 m^2) and potentially reduced emergency planning zone (see Figure 4.7)
- the broad market opportunities because of the low capital costs and the high operating temperatures.

Among the main advantages of microreactors are manageable capital costs, a predictable construction schedule, and a reduced radiological risk [4.4].

Microreactors are considered a good competitor technology for applications (especially in remote areas) currently relying on Diesel generators, which have a very high cost per unit of power (see Figure 4.8, according to NEI estimates [4.5], based on 40 years operation and 10-years refuelling interval). Similar LCOEs for both microreactors and diesel generators are given in Ref. [4.7]. Here they have assumed Diesel generators capital costs of 200–2000 $/kWe and microreactors capital costs between 5000 – 20,000 $/kWe. The larger span for Diesel generators is caused by the fuel costs (product itself and transportation to remote areas). Microreactors are expected to be competitive with renewables in microgrids as well [4.7].

Microreactors are also of interest for industries requiring a certain level of independence from the electrical grid and a guarantee of security of energy supply. Because microreactors are entirely factory-built and therefore benefits from manufacturing productivity, a positive learning curve is to be expected, as for other industries [4.8] (e.g. Korea Hydro and Nuclear Power and the U.S. Navy are reported to have experienced learning rates of a 15% cost reduction per doubling of units [4.8]). also

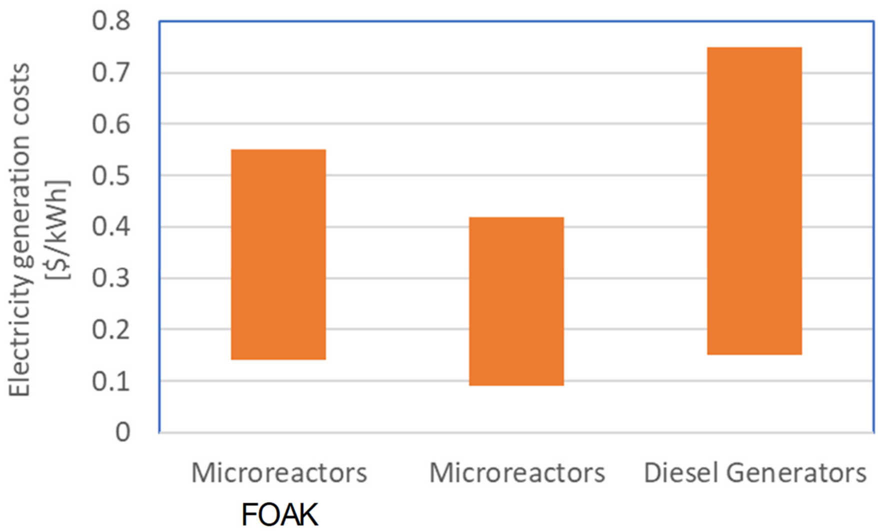


Figure 4.8 Estimated electricity generation costs for microreactors vs diesel generators [4.8]

Microreactors however have also a set of challenges:

- **Economics:** multiple orders are needed to benefit from the series production and justify investment in manufacturing factories.
- **Fuel availability:** while microreactors have the advantage of using low amount of fuel, as for non-LWR SMRs, microreactor designs require the development of new fuel cycle facilities and capabilities for the production and transport of high-assay low-enriched uranium (HALEU) fuel, and the production of TRISO fuel. Efforts to cope with this challenge are on-going. In January 2023, Ultra Safe Nuclear has formed a joint venture with Framatome for the fabrication of TRISO fuel, while in 2022 X-energy broke ground for a TRISO fuel fabrication facility in Tennessee. In addition, the operative experience on TRISO fuel is not as extensive as for LWR fuel.
- **Licensing:** recently, the Nuclear Energy Institute has issued a road-map [4.5] for a timely development of micro-reactors for deployment by the U.S Department of Defense (DoD). A critical point was found to be associated to the licensing process with U.S. NRC. This because, as for SMRs, licensing of microreactors require adaptations to the current licensing framework to address microreactors unique features (e.g. some microreactor designs foresee off-site staffing and semi-autonomous operation).
- **Security and proliferation risks:** the use of HALEU fuel poses a greater security and proliferation risk than the LEU grade fuel used in LWRs. However, all designs discussed in this chapter use the fuel in forms of TRISO particles, which are difficult to reprocess and therefore pose a significant barrier to the potential extraction of fissile material.
- **Public acceptance:** the use of nuclear energy as a distributed energy source, with many units spread across several different locations, might create additional barriers to public acceptance.

With regards to licensing, there are several innovative features in the microreactor designs that pose regulatory challenges. For example, an interesting feature of microreactor designs is that high-pressure conditions are not expected to be generated during accident conditions, meaning that a pressure-retaining containment would not be needed. The US NRC has approved the concept of “functional containment”, which would not necessarily retain pressure at all phases of an accident, but would still adequately retain radionuclides to protect the public and the environment [4.6]. On the other hand, because of the lower power and therefore smaller source terms compared to large NPPs, microreactors are similar to research reactors installations. Because of this, a more expeditive licensing process is expected compared to standard NPPs. The deployment of mobile microreactors, for example for military use, pose additional licensing challenges with regards to physical security and emergency response capabilities. On the other hand, because of the simpler operation, the time for operator training and licensing is expected to be significantly reduced [4.5].

4.2 Conclusions

Starting from the developments by NASA and Los Alamos National Laboratory focused on space applications of very small reactors, the concept of MW output capacity microreactors raised the interest of both the US department of energy and the US department of defense. While there are some developments outside of the United States, the great majority of the efforts on microreactors are taking place in USA, given the availability of funding from the government through NASA, DOE and DOE. All concepts are still in early phase of development with first demonstration units either under construction or undergoing licensing approval.

Because all concepts have high operational temperature, microreactors are well suited for cogeneration and for providing energy to several industrial processes which require heat at high temperatures. They are expected to cater to niche electricity and heat markets, such as powering micro-grids and remote off-grid areas, restoring power in communities affected by natural disasters, aiding in the rapid restoration of critical services (e.g., hospitals, water supply), and for industrial applications such as seawater desalination and hydrogen production. Commercialization is expected by 2030.

4.3 References

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5 State of GEN IV technologies including SMRs

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5.1 Generation-IV International Forum (GIF)

Around the year 2000 there was an international effort to identify reactor technology for the future. More than 2000 concepts have been evaluated and grouped according to the applied technology. Finally, six systems were selected as so called Generation-IV reactors and Generation-IV International Forum (GIF) was founded. It represents a collaborative international initiative aimed at conducting research and development to assess the viability and performance potential of the next generation of nuclear energy systems. The GIF Charter was initially signed by Argentina, Brazil, Canada, France, Japan, Korea, South Africa, the UK, and the US in July 2001. Switzerland joined in 2002 (and still is a GIF member), followed by Euratom in 2003, and China and Russia in 2006. Australia became a member in 2016. The primary objectives guiding the development of GIF systems are outlined in Table 5.1 and briefly presented below.

Table 5.1 GIF Goals [5.1]

Top-level goal	**Details**
G1. Sustainability	– Long term fuel supply – Minimization of waste and long term stewardship burden
G2. Safety & Reliability	– Very low likelihood and degree of core damage – Practical elimination of offsite emergency response
G3. Economics	– Life cycle cost advantage over other energy sources – Financial risk comparable to other energy projects
G4. Proliferation Resistance & Physical Protection	– Unattractive materials diversion pathway – Enhanced physical protection against terrorism

G1. Sustainability

The sustainability is the ability to meet present needs without compromising the ability of future generations to meet their own needs. It relies on two interconnected pillars: 1) "long-term fuel supply" means that Gen-IV system with the associated fuel cycle should use natural resources efficiently (ideally, only natural uranium or thorium should be used as a feed fuel for the system of the reactor and fuel cycle); and 2) "minimization of waste and long term stewardship burden" means that the waste has minimum amount of material that potentially can be burned in the reactor (i.e. actinides) and ideally, the waste should consists of fission products only. For both these pillars, the best results can be obtained by recycling of all actinides (in other words by the fuel cycle closure on all actinides) in a fast neutron spectrum reactor.

G2. Safety & Reliability

This goal is similar to the safety and reliability goal of the currently operated reactors. On one hand, the Gen-IV system should be designed so that the combination of active and passive safety measures makes the core damage probability at least not higher than in the operating reactors. On the other hand, the radioactivity release outside the Gen-IV plant should be practically eliminated. This means that the Gen-IV systems should be designed so that the chain reaction can be reliably shut down in any emergency situation and the residual heat generated in the reactor after the reactor shutdown is removed passively due to gravity and circulation of atmospheric air.

G3. Economics

Competitive economic performance is crucial in the marketplace and remains indispensable for Gen-IV nuclear energy systems. The future of nuclear energy should support various ownership models for plants and foresee a diverse range of potential roles in energy supply. Gen-IV nuclear energy systems have the potential to serve beyond electricity production, offering a wider spectrum of energy products, e.g. hydrogen production.

G4. Proliferation Resistance & Physical Protection

Enhanced proliferation resistance and physical protection stand as critical priorities for the increased implementation of nuclear energy systems. Alongside the immediate utilization of international safeguards, advanced Gen-IV reactor technologies advocate for incorporating safety, security, and safeguard necessities directly into the design of new fuel cycles and reactors.

5.2 Overview of GIF reactor systems

GIF initially chose six reactor systems for in-depth analysis and development at the outset of the initiative [5.1]:

1. Supercritical Water Reactor (SCWR)
2. Very High Temperature Reactor (VHTR)
3. Gas Fast Reactor (GFR)
4. Sodium Fast Reactor (SFR)
5. Lead Fast Reactor (LFR)
6. Molten Salt Reactor (MSR)

Table 5.2 **presents overall characteristics of advanced nuclear fission reactors, including light-water SMR and six GIF systems.**

Table 5.3 summarizes some information about six existing fast-spectrum nuclear fission reactors (all of them are of SFR type). Table 5.4 presents the list of GIF members and the GIF systems these countries contribute to.

Table 5.2 Characteristics of advanced nuclear fission reactors (Light Water SMR and Gen-IV) [5.2]

Reactor	Neutron spectrum	Coolant	Coolant pressure, bar	Outlet temperature (°C)	Fuel cycle
Light Water SMR	Thermal	Water	~160	300-330	Open
SCWR	Thermal/Fast	Water	~250	510-625	Open/Closed
HTGR/VHTR	Thermal	Helium	~70	700-1000	Open
GFR	Fast	Helium	~70	850	Closed
SFR	Fast	Sodium	~1	500-550	Closed
LFR	Fast	Lead	~1	480-570	Closed
MSR	Thermal/Fast	Molten salts	~1	700-800	Open/Closed

Table 5.3 Existing global fast reactors (all of SFR type) [5.3]

Country	Reactor name	Operation years	Current status
China	CEFR	2010-present	Active
India	FBTR	1985-present	Active
Russia	BOR-60	1969-present	Active
India	PFBR	Scheduled for 2024	Under construction
Russia	BN-600	1980-present	Active
Russia	BN-800	2014-present	Active

Table 5.4 Parties to GIF system arrangements [5.4]

	SCWR	VHTR	GFR	SFR	LFR	MSR
Argentina						
Australia		×				×
Brazil						
Canada	×	×				×
Euratom	×	×	×	×	×	×
France		×	×	×		×
Japan	×	×	×	×	×	
China	×	×		×	×	
Korea		×		×	×	
South Africa						
Russia	×			×	×	×
Switzerland		×				×
United Kingdom		×		×		
United States		×		×	×	×

The six Gen-IV system are discussed in details in the following chapters. A summary about the readiness of non-LWR technologies is presented in Table 5.5 (Ref. [5.5]).

Table 5.5 Readiness of non-LWR technologies [5.5]

Coolant	Technology and Prior Operating Experience	Technology Readiness Level	Outstanding Technology Development Issues
Helium	Small HTGR; half dozen around the world	Moderate to high for 750°C outlet design; Low to moderate for 900°C outlet temperature design	For designs with 750°C outlet temperature, complete fuel and graphite qualification by 2022. For 900°C outlet temperature, additional work is required for intermediate heat exchanger and other system components that can operate at that high temperature.
	GFR; none	Lowest. A gas-cooled fast reactor has never been built anywhere in the world	Fuel requires qualification. Internal structures must be able to sustain high levels of radiation damage for the long-lived cores being proposed. Safety systems require demonstration.
Liquid Metal	Small SFR like PRISM or ARC; several dozens around the world	High for traditional small SFR. Medium for longer lived, higher burnup breed-and-burn cores or for SFRs using advanced steel alloys that have yet to be qualified	Source term experiments for metallic fuel to reduce conservatism in safety analysis. Major fuel qualification effort if reactor is to be used with transmutation fuel. Development of a fuel vendor for commercialization. For more advanced SFRs, longer lived, higher burnup breed-and-burn cores need to be demonstrated and concepts using advanced structural alloys need to be qualified
	LFR; Russian subs (none elsewhere)	Lowest	Corrosion and erosion of coated cladding by lead at higher temperatures (≈700°C-750°C outlet) and higher flow velocities. Testing of passive safety behavior of the plant/key systems. Major qualification effort is needed for transmutation fuel (if reactor is used in that manner). Some concepts favor the use of nitride fuel—qualifying this fuel system in an LFR would require a significant effort and increase time to reach commercial readiness.
Molten Salt	FHR; none	Low to moderate	Prove REDOX corrosion/control in non-uranium-based salt in the presence of a neutron field. Need a demonstrated material solution (strength, corrosion resistance, irradiation stability) for long-term operation. Demonstrate tritium mitigation solution. Test passive safety aspects of the plant/design.
	MSR thermal; two thermal experiment systems (MSRE and ARE). No power conversion in these experiments	Low to moderate for FLiBe-based thermal reactor systems. Low for other salts in both thermal and fast spectrums. Most systems outside the reactor dealing with removal of fission products, actinides and fissile/fertile material are low maturity. Such systems are needed to maintain reactivity at high burnup. Many of these proposed reactor systems have not yet been built anywhere in the world.	Depending on the design, long-term corrosion is an issue unless major components are replaced frequently as is proposed in some designs. Demonstration of REDOX control in a neutron field in the salt will be required if the salt is not FLiBe. For lithium-containing salts, tritium mitigation solution must be demonstrated. Behavior of noble fission products plating out in IHX and corrosion and embrittlement by fission product tellurium (leading to cracking) over longer operation are also uncertain. High nickel alloy (Hastelloy X) developed at Oak Ridge National Laboratory in the United States is not code qualified for use by the American Society of Mechanical Engineers (ASME). This alloy also has poor irradiation stability and insufficient strength above 700°C. A demonstrated material solution is needed for long-term operation. (Some designers propose replacing structural materials every 4 to 6 years for this reason with an associated increase in cost and reduction in availability.) Unclear how inspection requirements per ASME Section XI will be implemented. Instrumentation in this system will require some development. Proliferation and materials accounting issues remain in this system since fission products and actinides are removed from the system and fissile material can in principle be diverted.
	MSR fast; no fast systems have ever been built	Lowest. Similar concerns as thermal MSRs	Chloride salts, strategies for corrosion control must be demonstrated. A materials solution must be demonstrated. Similar issues on inspection as MSR thermal systems. High power densities in some systems require high flow rates that can lead to erosion/corrosion concerns. Also, high flow rates can lead to very low delayed neutron fraction in the core, which makes reactivity control problematic.

5.3 Supercritical Water Reactor (SCWR)

5.3.1 Overview of SCWR

The Supercritical Water Reactor (SCWR) is a nuclear fission reactor cooled and moderated by light water above the thermodynamic critical point (374°C, 22.1 MPa).

This reactor can be considered as an evolution from Gen-III Light Water Reactor aiming at the increase in the thermodynamic cycle (power conversion) efficiency.

There are two groups of conceptual designs of SCWRs: pressure-vessel concepts proposed first by Japan and more recently by a Euratom partnership (see Section 5.3.2) and China, and a pressure-tube concept proposed by Canada.

Supercritical water temperature at the reactor outlet range from 510°C to 625°C at pressure of about 25 MPa (250 bar). The reference fuel is a classical fuel rod with enriched uranium dioxide fuel pellets and stainless-steel cladding.

The **main advantages** of SCWR are as follows:

- Improved economics is provided by higher thermodynamic efficiency of power conversion thank to the higher temperatures compared to Gen-III LWRs.
- Both thermal and fast neutron spectra are possible and therefore closure of fuel cycle is potentially achievable for the fast neutron spectrum option.
- Transparent coolant simplifies in-service inspection and repair as well as fuel handling.

The **main challenges** of SCWR include:

- In case of the thermal neutron spectrum the fuel cycle closure is impossible.
- Materials, including fuel, should be selected and qualified for the conditions of high-temperature and high-pressure water environment.
- Reference water chemistry is to be selected to minimize degradation of materials.
- Water radiolysis under irradiation conditions requires to be investigated.
- Additional thermal hydraulic experiments are needed to fill gaps in SCW heat transfer and critical flow databases.
- Overall safety demonstration is required.
- No operational experience exists.

5.3.2 Examples of SCWR projects including SMRs

All Gen-IV SCWR concepts presented below uses solid fuel, are cooled and moderated by light water at supercritical pressure, resulting in a thermal neutron spectrum. They are designed to operate in an open once-through fuel cycle. Below we elaborate on one concept (HPLWR) more extensively while providing a concise overview of the others.

HPLWR (EU) [5.6]

The High Performance Light Water Reactor (HPLWR) is a European Supercritical Water Cooled Reactor designed and analyzed in the European HPLWR project. The reactor is of a pressure-vessel type and operates at 25 MPa and 500ºC average core outlet temperature, resulting in envisaged increase in thermodynamic efficiency of power conversion to about 44%. However, the improving efficiency comes at the expense of a more intricate design. Compared to PWR, the HPLWR experiences a significant increase in coolant heating within the core, rising from 31ºC to 220ºC and wider range of water densities. This poses a challenge concerning the peak cladding temperature, which is targeted at 630ºC to avoid an extensive creep deformation. The chosen approach to minimizing the peak cladding temperature is to heat up the coolant in three stages with intermediate coolant mixing as shown in Figure 5.1 left. The HPLWR fuel assemblies consist of 40 fuel pins each, incorporating a central water box to enhance neutron moderation under conditions of the low coolant density. The elevated pressure requires also an increase of the reactor vessel thickness up to 45 cm. Overall the resulting reactor design (Figure 5.1 right) becomes very complicated. Another "trade-off" for the enhanced efficiency is the concern about corrosion. The cladding alloys being considered for this purpose encompass ferritic–martensitic steels, stainless steels, nickel-base alloys, and ODS alloys.

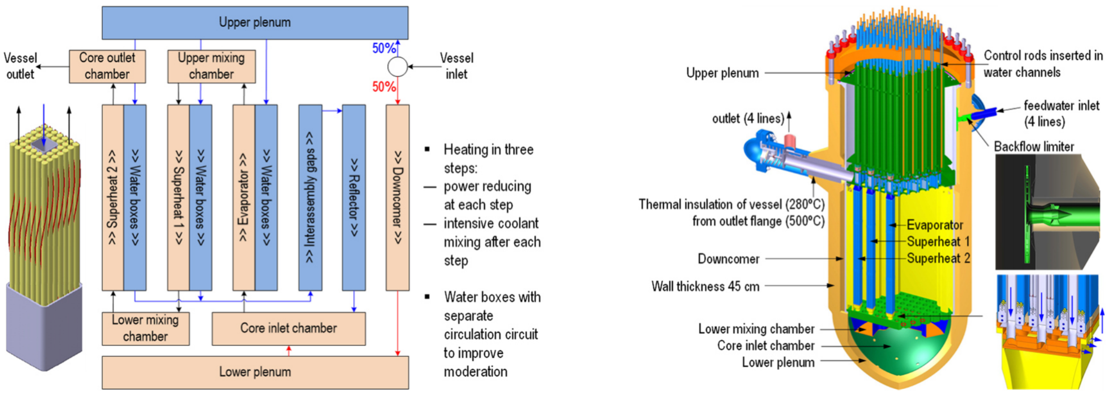


Figure 5.1 HPLWR fuel assembly and three-step heating diagram (left) as well as reactor vessel internals (right) [5.6]

CSR1000 (China) [5.7]

China designs two kinds of large-power SCWR concepts, and one SCW-SMR: CSR1000, SCWR-M, and the CSR-150, respectively.

SCW-SMR (EU, Canada, China) [5.8]

The supercritical water cooled SMR of 200 to 300 MWe is developed in frame of the Euratom Joint European Canadian Chinese project (ECC-SMART).

VVER-SCP (Russia) [5.9]

Two variants of SCWR concepts are developed in Russia: with direct and indirect power conversions (VVER-SKD and VVER-SCP, respectively).

5.4 Very High Temperature Reactor (VHTR)

5.4.1 Overview of VHTR

The High-Temperature Gas-cooled Reactor (HTR or HTGR) is a helium-cooled graphite-moderated thermal-spectrum nuclear fission reactor.

This reactor can be considered as an evolution from Gen-IV SCWR aiming at overcoming the SCWR challenges related to the material compatibility with high-temperature and high-pressure water, water chemistry, radiolysis, etc.

HTR uses ceramic tri-structural isotropic (TRISO) coated particle-based fuels. TRISO is a fuel particle with UO_2 kernel coated with several layers of pyrolytic carbon and silicon carbide (see Figure 5.2 left). The coatings perform as barriers. Excellent fission product retention and safety characteristics of the fuel make it possible to operate it at high temperature. The particles are dispersed in graphite pebbles or prisms. The graphite serves as a moderator.

Helium coolant temperature at the reactor outlet range from 700°C to 950°C at pressure of about 7 MPa (70 bar), thus enabling power conversion efficiencies of up to 48% and delivery of industrial process heat, which can be used, in particular, for hydrogen production.

VHTR is an evolutionary step targeting even higher thermodynamic efficiency and various applications of the process heat by further increasing the helium outlet temperature to 1000°C or even higher. VHTRs can be built with power outputs that are typical of SMRs, primarily dedicated to the electricity and process heat cogeneration, e.g. for hydrogen production.

The **main advantages** of (V)HTR are as follows:

- High coolant temperature at the reactor outlet enables non-electric applications.
- High thermodynamics efficiency is due to the high coolant temperature.
- Very high passive safety is a result of low power density and large mass of the moderator (graphite). Even in case of a depressurization event the reactor shuts down itself without operator intervention (due to reactivity feedbacks and heat removal by radiation).
- Inert gas coolant eliminates corrosion issues.
- Transparent coolant simplifies in-service inspection and repair as well as fuel handling.
- There exists some operational experience: HTTR (Japan) is currently in operation.

The **main challenges** of (V)HTR include:

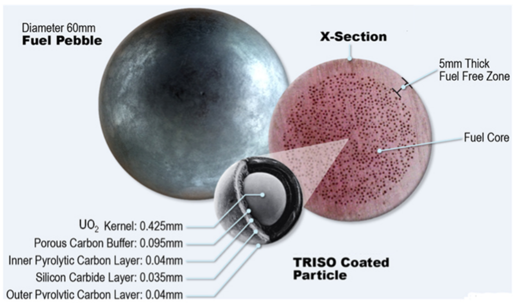
- Because of the thermal neutron spectrum the fuel cycle closure on all actinides is impossible.
- Temperature of ~1000ºC (for hydrogen production) cannot fully achieved with the currently available materials.
- Coupling with process heat applications is a technological challenge under research and development.
- Irradiated graphite is a waste of significant volume, which should be processed and safely stored.
- Operational experience is quite limited.

5.4.2 Examples of HTR projects including SMRs

All Gen-IV VHTR concepts listed below uses solid fuel (TRISO), are cooled by pressurized helium and moderated by carbon, resulting in a thermal neutron spectrum. They are designed to operate in an open once-through fuel cycle. Below we elaborate on one concept (HTR-PM) more extensively while providing a concise overview of the others.

HTR-PM (China) [5.10]

High-Temperature gas-cooled Reactor Pebble-bed Module (HTR-PM) is a demonstration power plant in China commissioned at the end of 2021. This is a small modular nuclear reactor. Graphite fuel pebbles (see Figure 5.2 left) are continuously added to the core from the top, slowly floating through the core and discharged from the bottom. Every pebble passes the core six times. A single HTR-PM reactor module (Figure 5.2 right) produces 250 MW of thermal power, with helium temperatures at the reactor core inlet and outlet at 250°C and 750°C, respectively. It generates steam at a pressure of 13.25 MPa and a temperature of 567°C at the steam generator outlet. Two such reactor modules are connected to a steam turbine, creating a 210 MWe. The power density is approximately 3.3 MW per cubic meter, which is about 30 times lower than in a PWR. Due to this low power density and high thermal inertia of graphite, there's no need for a core emergency cooling system, as decay heat is naturally removed in case of accidents by radiation. Showing very high safety level, HTR's main disadvantage is very low sustainability from viewpoint of the fuel resources and waste management (see G1 in Section 5.1).



Fuel	UO_2
Number of pebbles in the core	420'000
Heavy metal per pebble	~7 g
Number of coated particles in each pebble	~ 11'660
Fuel loading scheme	Multi-pass (six times)
Average discharge burnup	90 MWd/kgU

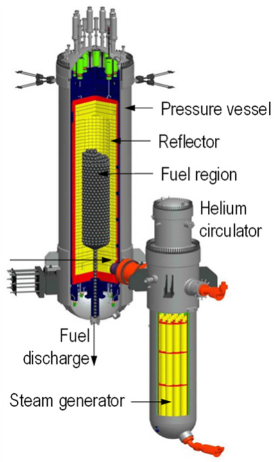


Figure 5.2 HTR TRISO fuel concept (left) and single module of HTR-PM (right) [5.10]

BANR SMR (USA) [5.11]

BWXT Advanced Nuclear Reactor (BANR) is a high-temperature gas microreactor of a thermal power of 50 MW that utilizes TRISO fuel. In December 2020, the DOE chose BANR as one of five projects to receive a portion of the initial $30 million in funding for risk reduction projects as part of its Advanced Reactor Demonstration Program (ARDP).

Kaleidos (USA) [5.12]

The concept of this microreactor is developed by a private company Radiant (USA). The Kaleidos reactor is a 3.5 MWth high-temperature reactor cooled by helium and fueled by TRISO particles dispersed in cylindrical compacts with a zirconium hydride moderator. The US National Reactor Innovation Center is currently preparing the EBR-II dome facility for hosting reactor demonstration projects.

Jimmy (France) [5.13]

Jimmy is a 10 MWth gas-cooled microreactor designed for industrial process heat production (no electricity generation). In May 2022, the Autorité de Sûreté Nucléaire (ASN) and the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) initiated a preliminary pre-licensing assessment of Jimmy's Dossier d'Options de Sûreté (DOS), also known as the safety options file. The IRSN, which serves as the primary technical support entity for ASN, concluded its analysis and published the results in September 2022.

GTHTR300 (Japan) [5.14]

The Gas Turbine High Temperature Reactor (GTHTR300) was designed by the Japan Atomic Energy Agency (JAEA), drawing upon their prior experience with the High Temperature

Engineering Test Reactor (HTTR). As of the assessment, no publicly accessible and verifiable information was available regarding licensing or pre-licensing activities for the GTHTR300.

5.5 Gas Fast Reactor (GFR)

5.5.1 Overview of GFR

The Gas Fast Reactor is a helium-cooled non-moderated fast-spectrum nuclear fission reactor.

This reactor can be considered as an evolution from Gen-IV (V)HTR aiming at overcoming the SCWR challenges related to thermal spectrum, i.e. impossibility to close a fuel cycle.

The reference GFR fuel is a classical fuel rod with mixed uranium-plutonium fuel pellets and ceramic (in high-temperature concepts) or stainless-steel (in low-temperature concepts) cladding. The coolant temperature at the reactor outlet is around 850°C.

The **main advantages** of GFR are as follows:

- Fast neutron spectrum provides a potential for new fissile breeding and therefore a potential to close the fuel cycle with recycling of all actinides.
- High operating temperature allows increased thermal efficiency and industrial applications of heat similar to the (V)HTR.
- Transparent coolant simplifies in-service inspection and repair as well as fuel handling.
- Inert gas coolant eliminates corrosion issues.

The **main challenges** of GFR include:

- Safety demonstration is required and in particular decay heat removal in case of loss of flow and depressurization accidents. The absence of moderator significantly reduces the core thermal inertia compared to the (V)HTR. Expensive measures such as a compact spherical containment and injection of heavy gas are required to manage depressurization accidents.
- High-temperature materials and fuel qualification are required similarly to the (V)HTR.
- No operational experience exists for this reactor.

5.5.2 Examples of GFR projects including SMRs

Gas Fast Reactor technology is developed in Europe along the two routes: a large GFR and an ALLEGRO demonstrator. Below we elaborate on one concept (GCFR) more extensively while providing a concise overview of ALLEGRO.

GCFR (EU) [5.15]

The large 2400 MWth GFR is developed by consortium of European countries in frame of the Euratom projects. Figure 5.3 shows the internals of the pressurized reactor vessel (left) and a spherical compact containment as a necessary barrier for providing a backup pressure in case of the primary system depressurization (right). The safety issue of decay heat removal in a depressurization event dictates many safety measures and technical solutions making the design and procedures complicated, e.g. necessity to reconfigure the circulation circuit while switching from normal operation to the decay heat removal regime, injection of heavy gas from the pressurized containers to provide additional cooling power, use of Bryton cycle machines for passive decay heat removal, etc.

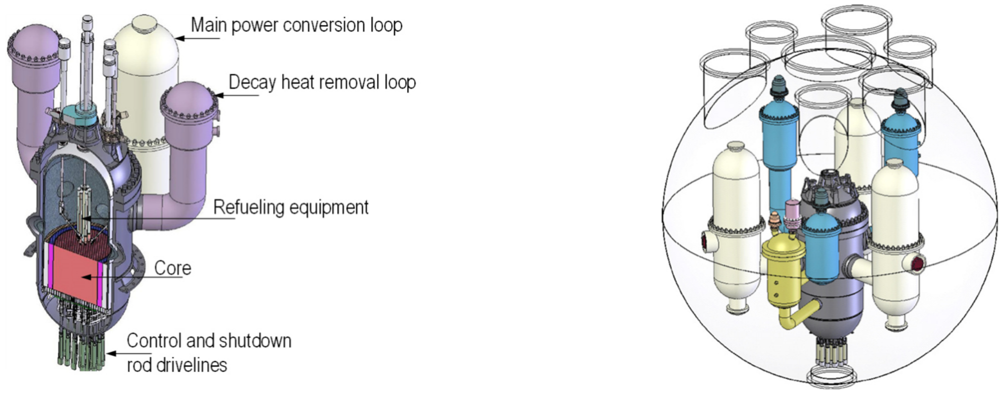


Figure 5.3. GCFR primary system diagram (left), compact spherical containment (right) [5.15]

ALLEGRO (V4, EU) [5.16]

The GFR technology demonstration relies on testing essential aspects in the 75 MWth European Gas Fast Reactor Demonstrator Project (ALLEGRO). ALLEGRO can be considered an SMR and aims to validate GFR technologies, including fuel, fuel elements, helium-related systems, and safety measures, particularly decay heat removal. The project seeks to confirm the successful integration of these features into a representative system. The development of ALLEGRO is overseen by the V4G4 Centre of Excellence consortium, comprising four founding members (EK from Hungary, NCBJ from Poland, UJV Rez from the Czech Republic, VUJE from Slovakia) and two associated members (CEA from France and CVR from the Czech Republic).

5.6 Sodium Fast Reactor (SFR)

5.6.1 Overview of SFR

The Sodium Fast Reactor is a sodium-cooled non-moderated fast-spectrum nuclear fission reactor.

This reactor can be considered as an evolution from Gen-IV GFR aiming at overcoming the GFR challenges related to low thermal inertia and safety issues in case of depressurization events.

Thank to the exceptional thermophysical properties of sodium (high boiling point, heat of vaporization, heat capacity and thermal conductivity), the liquid sodium coolant allows high-power density with low-coolant volume fraction for improvement of fuel performance in the closed fuel cycle. Sodium coolant provides also high thermal inertia and significant level of natural convection important for the reactor safety. Existing technologies of sodium cleaning allow for a complete prevention of corrosion of structural materials. On the negative side, sodium reacts chemically with air and water and requires a sealed coolant system with intermediate sodium circuit to practically eliminate a contact of activated primary sodium with the steam-water in steam generator.

The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C. Under these conditions, austenitic and ferritic steel structural materials can be used, and a large margin to coolant boiling at low pressure can be maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. SFR sizes can range from small modular systems to large monolithic reactors.

There are two reference fuels: the first design (developed in Europe) is a cylindrical fuel rod with mixed uranium-plutonium dioxide fuel pellets in stainless-steel cladding with helium inside the fuel rod, while the second design (developed in USA) is a U-Pu-Zr metallic alloy slug in stainless steel cladding filled with a sodium (sodium bond).

The **main advantages** of SFR are as follows:

- Fast neutron spectrum provides a potential for new fissile breeding and therefore a potential to close the fuel cycle with recycling of all actinides.
- Excellent thermal conductivity of sodium provides very efficient cooling with minimal coolant volume fraction (beneficial for efficient closed fuel cycle performance).
- Large margin to boiling (~350ºC between outlet temperature and sodium boiling point) means that no coolant pressurization is required. This is a very important advantage for reactor safety, in particular, excluding depressurization events.
- Significant operational experience exists (300+ reactor-years). Significant number of SFRs are currently in operation: BOR-60, BN-600, BN-800 (all Russia), CEFR (China).

The **main challenges** of SFR include:

- Sodium is chemically active in contact with water or air and therefore an intermediate sodium circuit is needed between primary sodium and steam-water circuit.
- Sodium has a significant scattering cross section and, therefore, spectrum hardening when sodium is removed from the core the neutron spectrum becomes harder. The spectrum hardening in fast reactor results in a positive reactivity effect (so-called positive void effect). In particular, to mitigate the consequences of the sodium boiling special safety measures are needed.

5.6.2 Examples of SFR projects including SMRs

All Gen-IV SFR concepts listed below uses solid fuel, are cooled by non-pressurized sodium and not moderated, resulting in a fast neutron spectrum. They are designed to operate in a closed fuel cycle. Below we elaborate on one concept (ESFR) more extensively while providing a concise overview of the others.

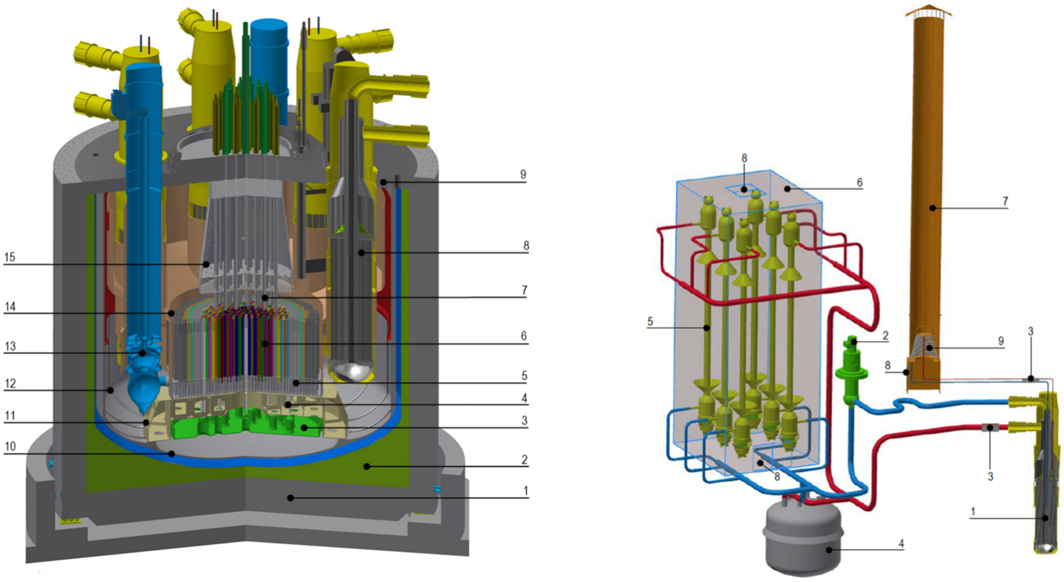
ESFR (EU) [5.17]

Following up the previous European projects CP ESFR project), the ESFR-SMART project (European sodium fast reactor—safety measures assessment and research tools) has made a next step in development of a large European Sodium Fast Reactor of 3600 MWt and 1500 MWe (Figure 5.4 left). The main new features include:

- New core concept with
  - reduced sodium void effect,
  - passive control rods and
  - corium discharge tubes.
- In-vessel core catcher.
- Reactor pit with functions of the safety vessel.
- Massive metallic roof with leak tightness of roof penetrations.
- New decay heat removal (DHR) systems:
  - connected in parallel to the Intermediate Heat Exchanger (Figure 5.4 right),
  - based on air cooling of steam generator surface (Figure 5.4 right).
  - Passive electromagnetic pumps in the secondary system for enhancing natural convection.

BOR-60 (Russia) [5.18]

Fast 60MW sodium-cooled reactor BOR-60 is a low-power SFR prototype and research reactor. It transfers generated heat to the local heat network and produce electricity. This research reactor is used to test fuel cycle, sodium coolant technologies and a wide range of design concepts for fast reactors. Being a powerful source of fast neutrons, this reactor is used to study the effect of neutron irradiation on various structural, fuel and absorbing materials. The reactor was commissioned in 1969 and currently in operation.



1 – Reactor pit

2 -- Insulation

3 – Core catcher

4 – Corium discharge tubes

5 – Diagrid

6 – Core

7 – Control rod driveline

8 – Intermediate heat exchanger

9 – Reactor roof

10 – Reactor vessel

11 – Strongback

12 – Vessel cooling pipes

13 – Primary pump

14 – Inner vessel (redan)

15 – Above core structure

1 – Intermediate heat exchanger

2 – Secondary pump

3 – Thermal pumps

4 – Sodium storage tank

5 – Steam generator

6 – Casing of Decay Heat Removal System (DHRS-2)

7 – Air stack of DHRS-1

8 – Openings for air circulation

9 – Sodium-air heat exchanger of DHRS-1

Figure 5.4. ESFR primary (Left) and secondary (Right) systems [5.17]

BN-600 (Russia) [5.19]

The BN-600 reactor is a sodium-cooled fast breeder reactor, built at the Beloyarsk Nuclear Power Station, in Sverdlovsk Region of Russia. It has a 600 MWe gross capacity and a 560 MWe net capacity, dispatched to the Middle Urals power grid. It has been in operation since 1980 and currently in operation. It represents an evolution on the preceding BN-350 reactor. The plant is of a pool type, fuelled by the enriched uranium dioxide fuel in an open fuel cycle. As of 2022, the cumulative energy availability factor recorded by the IAEA was 76.3%.

BN-800 (Russia) [5.20]

The BN-800 reactor is a 880 MWe sodium-cooled fast breeder reactor, built at the Beloyarsk Nuclear Power Station, in Sverdlovsk Region of Russia. The reactor is fuelled by the mixed uranium-plutonium fuel and considered part of the weapons-grade Plutonium Management and Disposition Agreement signed between the United States and Russia, with the reactor being part of the final step for a plutonium-burner core (a core designed to burn and, in the process, destroy, and recover energy from, plutonium). The plant reached its full power production in August 2016. It currently works in an open fuel cycle.

CFR (China) [5.21]

The China Institute of Atomic Energy (CIEA) expanded on the CEFR project to develop the CFR-600, a 600 MWe reactor. Construction of a demonstration unit at the Xiapu site in China started in December 2017. Initially, it will use mixed uranium-plutonium oxide fuel (MOX), followed by metallic fuel. Notably, construction of a second CFR-600 unit at the Xiapu site began in December 2020.

FBTR (India) [5.22]

The Fast Breeder Test Reactor (FBTR) is a sodium-cooled reactor located at Kalpakkam, Tamil Nadu, India. It first reached criticality in October 1985 and currently is in operation. The reactor uses a mixed uranium-plutonium carbide fuel consisting of 70 percent PuC and 30 percent UC. Plutonium for the fuel is extracted from fuel irradiated in the Madras power reactors and reprocessed in Tarapur. Some of uranium was created by irradiation of natural thorium.

PFBR (India) [5.23]

The Prototype Fast Breeder Reactor (PFBR) is a 500 MWe fast breeder nuclear reactor currently under construction at the Madras Atomic Power Station (MAPS) in Kalpakkam, India. Originally scheduled for commissioning in 2010, the construction of the reactor faced numerous delays. As of December 2022, the Prototype Fast Breeder Reactor is anticipated to be completed in 2024, marking a 20-year timeline from the project's inception.

ASTRID (France) [5.24]

ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) was a 600 MW sodium-cooled fast breeder reactor project proposed by the Commissariat à l'énergie atomique (CEA) to be constructed at the Marcoule Nuclear Site in France. In August 2019, France officially canceled the ASTRID project, citing that “In the current energy market situation, the perspective of industrial development of fourth-generation reactors is not planned before the second half of this century.”

BN-1200 (Russia) [5.25]

The BN-1200 reactor is a sodium-cooled fast breeder reactor project of about 1220 MWe currently in development by Russia. It builds upon the earlier BN-600 and BN-800 reactors, sharing several key features with them.

MBIR (Russia) [5.26]

MBIR is a sodium-cooled fast reactor designed for a lifespan of up to 50 years. This versatile technology is a multi-loop research reactor capable of testing various coolants, including lead, lead-bismuth, and gas. It operates using mixed uranium-plutonium oxide fuel (MOX), as specified by Rosatom. The RIAR plans to establish on-site closed fuel cycle facilities for the MBIR, employing the pyrochemical reprocessing method that it has developed at a pilot scale.

Natrium (US) [5.27]

The Natrium system comprises a 345 MWe sodium fast reactor and can be customized for specific markets. Its innovative molten salt based thermal storage capability has the potential to increase the system's power output to 500 MWe for over 5.5 hours when required.

5.7 Lead Fast Reactor (LFR)

5.7.1 Overview of LFR

The Lead Fast Reactor is a pure lead-cooled non-moderated fast-spectrum nuclear fission reactor.

This reactor can be considered as an evolution from Gen-IV Sodium Fast Reactor aiming at the practical elimination of the risk of sodium fires and improvement of economy due to elimination of the intermediate circuit.

The reference fuel is a classical fuel rod with mixed uranium-plutonium fuel pellets and stainless-steel cladding. Operational temperature of the lead coolant at the reactor outlet is in the range of 480-570°C and limited by the material corrosion issues.

The **main advantages** of LFR are as follows:

- Fast neutron spectrum provides a potential for new fissile breeding and therefore a potential to close the fuel cycle with recycling of all actinides.
- High density of the heavy liquid metal coolant provides a very high thermal inertia.
- High thermal conductivity and expansion coefficient of the coolant results in an efficient heat removal at low velocities and high natural circulation level.
- The coolant is chemically passive with water and air and therefore there is no need in an intermediate circuit (the advantage compared to SFR).
- Large margin to boiling (~1260ºC between outlet temperature and lead boiling point) means that no coolant pressurization is required. This is a very important advantage for reactor safety, in particular, excluding depressurization events.
- There is some operational experience: small fast reactors cooled by lead-bismuth eutectics were used at the Soviet military submarines.

The **main challenges** of LFR include:

- High density of the lead coolant is the reason of erosion of the structural materials as well as seismic and refueling issues.
- At high temperature structural materials (such as iron or nickel) are slowly dissolving in lead flow and therefore a complicated technology for material protection is needed.
- The positive void reactivity effect is high (e.g. due to the gas entry).
- The margin between operational temperature and lead freezing point (327ºC) is low. Therefore, special safety measures needed to avoid lead freezing.
- No reactors are currently in operation, available operational experience is low.

Gen-IV LFR concepts include three reference systems [5.1]:

1) European Lead Fast reactor (ELFR) with power of 600 MWe;
2) Russian BREST-OD-300 demonstrator with power of 300 MWe; and
3) US Small Secure Transportable Autonomous Reactor (SSTAR) with power range of 10-100 Mwe, featuring a very long core life.

The main GIF related activities in Europe are:

1) Multi-purpose hybrid research reactor for high-tech applications (MYRRHA), carried out by SCK-CEN (Belgium) with the goal of demonstrating an accelerator-driven system technology as well as supporting the development of fast-neutron spectrum Generation-IV systems [5.28];
2) Advanced Lead Fast Reactor European Demonstrator (ALFRED) project developed by consortium of European countries [5.29]; and
3) Advanced Modular Reactor (AMR) Program [5.30] conducted by UK in collaboration with several EU organizations in support of the development of the Westinghouse LFR concept [5.31].

5.7.2 Examples of LFR projects including SMRs

All Gen-IV LFR concepts listed below uses solid fuel, are cooled by non-pressurized pure lead and not moderated, resulting in a fast neutron spectrum. They are designed to operate in a closed fuel cycle. Below we elaborate on one concept (ALFRED) more extensively while providing a concise overview of the others.

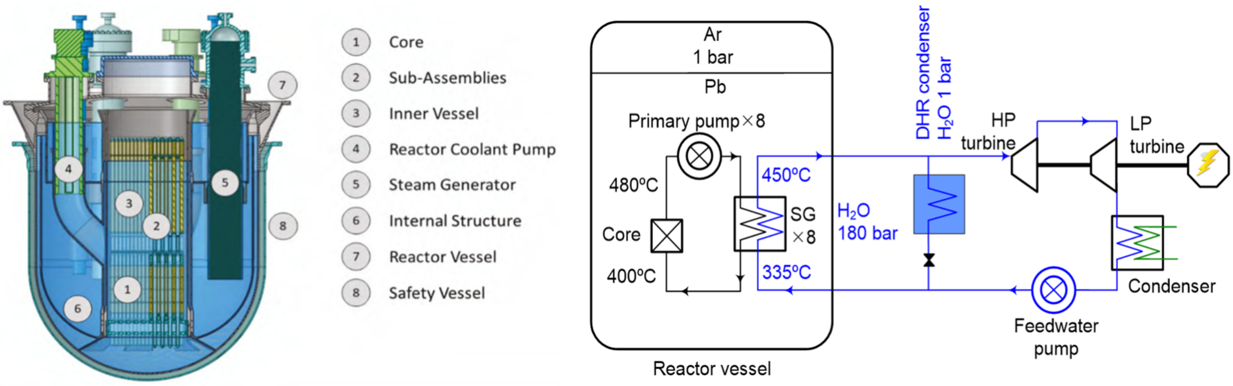


Figure 5.5. ALFRED LFR primary system configuration (Left) and schematics and main parameters of primary and secondary systems (Right) [5.29]

ALFRED (EU) [5.29]

ALFRED is envisioned as the Lead Fast Reactor (LFR) demonstrator, featuring a core power in the range of a few hundred MWth. This design is an SMR and allows for straightforward scalability to commercial sizes. Concurrently, ALFRED strives to integrate prototype technological solutions and components for a Lead Fast Reactor First of a Kind (FOAK) with a target power output of 250–300 MWe, aligning with the GIF sustainability objective (G1). At European level, an international consortium, FALCON (Fostering ALfred CONstruction), was established among Ansaldo Nucleare[6] (Italy), ENEA[7] (Italy) and RATEN-ICN[8] (Romania) in 2013 and joined by six supporting organizations in the following years.

The ALFRED primary system (Figure 5.5 left) is of pool type with mechanical pumps and steam generators located in the hot pool. Decay Heat Removal (DHR) systems are isolation condensers located outside the reactor vessel (see Figure 5.5 right). Both coatings or oxygen-driven self-passivation are considered as material protection measures. It is crucial to maintain a narrow temperature range between 400°C as the lower limit, which prevents lead from freezing at 327°C, and 550°C as the upper limit, which is related to corrosion issues concerning cladding temperature.

BREST-OD-300 (Russia) [5.32]

BREST is an innovative lead-cooled fast reactor of 300 MWe/700 MWth with mixed nitride fuel and secondary system with supercritical water developed in Russia. BREST-OD-300 received the construction license in February 2021 and in June 2021 the construction began in Seversk (near Tomsk).

CLEAR-M (China) [5.33]

The China develops a concept of lead-based mini reactor CLEAR-M10 of 10 MWe as a representative concept for a small modular energy supply system. The main purpose of the project is to provide electricity as a flexible power system for islands, remote districts or industrial parks. The project is supported by a lead-bismuth experiment loop platform KYLIN-II operated for more than 30 000 h for corrosion tests, thermal-hydraulic experiments and prototype component proof tests.

Newcleo (Italy) [5.34]

Private company aiming at series of LFRs: 1) 10 MW electrically heated/nonnuclear facility with turbogenerator by 2026; 2) 30 MW demonstrator and test reactor with core outlet at 430/440° (later 530°), using mixed uranium-plutonium fuel (MOX) as fuel by 2030; 3) 200 MW nuclear waste-to-energy SMR, for stand-alone or fleet type configuration, using MOX as fuel by 2032; 4) 30 MW mini nuclear reactor for industrial and maritime applications working as a nuclear battery, with infrequent refuelling (>10 years) by 2032. All systems feature spiral-tube steam generator, extended stem fuel assemblies and amphora shaped inner vessel.

Westinghouse-LFR (USA) [5.31]

The Westinghouse LFR is a ~450 MWe, highly simplified, passively safe, compact and scalable reactor plant. The Westinghouse LFR will achieve the following important objectives for its customers: walk-away safety; reduced capital/overnight costs; competitive Levelized Cost Of Electricity (LCOE) even in the most challenging global markets; Variable electricity output to complement renewables; capability for non-electricity applications such as cogeneration and seawater desalination; reduced nuclear fuel waste volume per unit of electricity generated.

MYRRHA (Belgium) [5.28]

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[6] https://www.ansaldoenergia.com/companies/ansaldo-nucleare

[7] https://www.enea.it/

[8] https://www.raten.ro/?lang=en

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is the world's first large scale Accelerator Driven System (ADS) that consists of a subcritical fast-spectrum nuclear fission reactor cooled by lead-bismuth eutectics and driven by a high power linear accelerator. With the subcritical concentration of fission material, the nuclear reaction is sustained by the particle accelerator only. Turning off the proton beam results in an immediate and safe halt of the nuclear reactions. MYRRHA will have a maximum output of 100 MWth.

TRANSMUTEX (Switzerland) [5.35]

Transmutex is a private company developing an Accelerator-Driven, subcritical system cooled by lead, with a specific focus on waste-burning and Thorium deployment. Unlike the MYRRHA reactor, the system will be driven by a high-power cyclotron.

5.8 Molten Salt Reactor (MSR)

5.8.1 Overview of MSR

Based on the most general definition of MSR from IAEA TRS-489 document [5.36] "an MSR is any reactor where a molten salt has a prominent role in the reactor core (i.e., fuel, coolant, and/or moderator)". This definition is broad and includes also HTRs cooled by salt, which were historically not considered as MSRs. Already from the variety of the functions mentioned in the definition, it is clear that the MSR is rather a whole category of reactors than a single concept.

This reactor can be considered as an evolution from Gen-IV Lead Fast Reactor aiming at the practical elimination of the core meltdown event.

The fuel ranges from liquid molten salt containing fuel components to graphite pebbles with TRISO particles. The temperature at the reactor outlet ranges from 700 to 800°C.

The associated fuel cycles vary from one concept to another, but generally an online separation of gaseous fission products are considered in almost all projects, while a closed fuel cycle with recycling of selected actinides is assumed in fast-spectrum systems, and thermal-spectrum systems mainly rely on open fuel cycles. In some cases thorium-uranium closed fuel cycle is considered in thermal-spectrum MSRs.

At increased temperatures, these molten salts become more corrosive, influenced by thermodynamics, impurity effects, and fluctuating activity and temperature gradients. Proficiency in comprehending and controlling material corrosion in these molten salt environments necessitates careful material selection and exacting control over salt chemistry. These measures stand as crucial steps in seamlessly integrating these salts into nuclear energy systems.

The **main advantages** of liquid-fuel MSR are as follows:

- Fast neutron spectrum MSRs provides a potential for new fissile breeding and therefore a potential to close the fuel cycle with recycling of all actinides.
- Large margin to boiling (few hundred degrees between outlet temperature and molten salt boiling point) means that no coolant pressurization is required. This is a very important advantage for reactor safety, in particular, excluding depressurization events).
- Strongly negative fuel salt density (void) reactivity effect is the basis for the reactor safety.
- High thermodynamics efficiency is due to the high coolant temperature.

- The absence of structural materials in the case of a homogeneous core[9] results in no radiation damages of the structural materials.
- There is a flexible possibility to add or remove fuel salt. The fuel reprocessing can be significantly simplified.
- Continuous removal of insoluble fission products is possible. This is beneficial for neutron balance and reactor safety.
- There is some operational experience: Molten Salt Reactor Experiment (MSRE) was conducted in ORNL (USA) in 1960s [5.37].

The **main challenges** of liquid-fuel MSR include:

- Molten salt fuels are strongly corrosive.
- Lack of the usual barriers on the way of the radioactivity (e.g. fuel cladding) requires development of a new safety approach.
- There is a high fluence on vessel.
- Part of fuel is always located outside the core. This results in the need of larger fuel inventory to sustain the chain reactor; moreover, this reduces the effective fraction of delayed neutrons β with an impact on safety.
- The margin between operational temperature and molten salt freezing point is low. Therefore, special safety measures needed to avoid freezing.
- Solubility of compounds formed during operation are low and often unknown. The corresponding research is needed.
- There are no reactors in operation, available operational experience is low.

The IAEA TRS-489 document [5.36] includes an **MSR taxonomy**, which was adopted by GIF (see Figure 5.6). This taxonomy has several levels; on the highest level reactors are divided into classes based on technological aspects related to kind of solid materials, which are present in the core. The graphite moderator, as the only moderator directly compatible with the salt, has a prominent position and form the first class: I. Graphite based MSRs. In these MSRs, the graphite moderator shapes the core filled by the liquid salt. Since there are only these two materials in the core, the actinides can be embedded either in the graphite matrix or in the salt. The neutron spectrum of Graphite based MSRs is obviously thermal. The second class, II. Homogenous MSRs, is defined by general absence of solid materials in the active core, which is thus solely filled by the actinides carrying salt. These reactors are considered as fast; however, presence of F, Li and Be in the carrier salt can soften the fast spectrum.. Since it is not straightforward to dissolve the lightest moderating elements in the salt, homogenous MSRs are usually not foreseen as thermal reactors. The third class, III. Heterogeneous MSRs, is represented by reactors with at least three materials in the core. Typically, there are two molten salts that need to be separated by a wall. These reactors can be either thermal, where the wall separates fuel salt and the moderator, or fast, where the wall separates fuel salt and the dedicated coolant.

The three basic MSR classes cover the majority of recent concepts. However, there are some historical or less populated concepts, which form the last class: IV. Other MSRs. Even with this additional class, it is sometime difficult to apply this taxonomy; for instance, there are reactors

[9] The homogeneous core is a cavity filled with the liquid fuel without heterogeneous structures like fuel rod lattice.

that use in the core simultaneously graphite as moderator and structural materials to separate two different salts. In this particular case, the presence of graphite is considered as the dominant feature. Each of the three major classes can be divided into two families based on the class specific features. Hence, there are six major MSRs families, illustrated in Figure 5.6. These six MSRs families have many common features; at the same time, there are many dissimilarities.

MSR shares many features with other reactors. However, there are also some difference, which are listed in [5.36] and in shorter form adopted in this chapter.

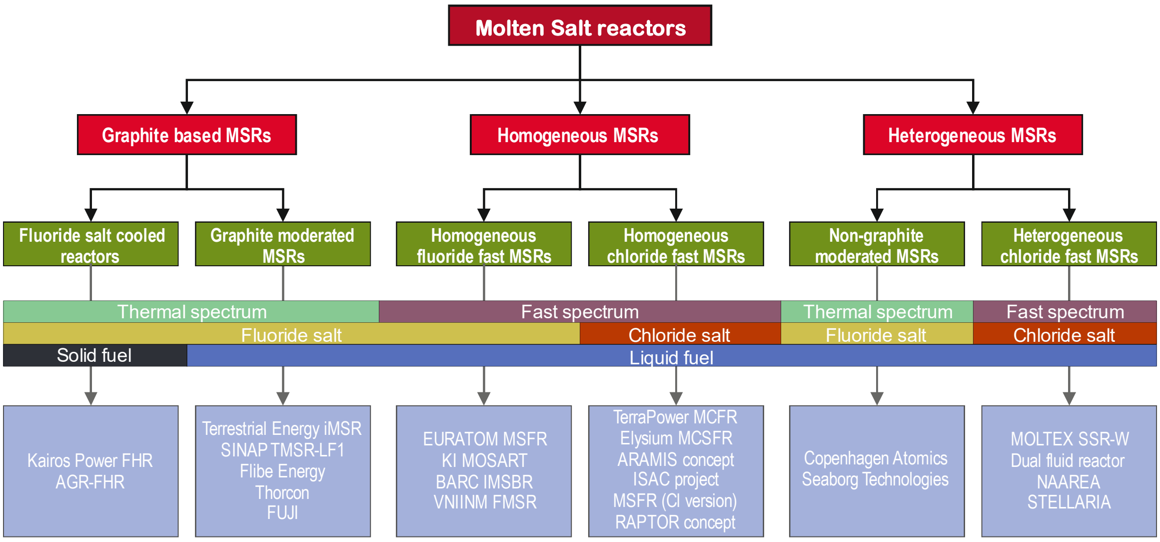

Figure 5.6 MSR taxonomy as defined in IAEA TRS-489 [5.36]

Double heterogeneity (family: I.1.)

In I.1 family, fuel is based on TRISO particles embedded in graphite moderator and cooled by dedicated molten salt coolant. There is thus double heterogeneity in the core.

Graphite limited lifespan and positive temperature effect (family: I.2)

In the I.2 family, the graphite does not contain fuel and its replacement is thus driven solely be its irradiation lifespan. Accordingly, these concepts needs to replace the graphite moderator regularly. Furthermore, in some particular cases of Th-U breeding cycle the temperature feedback from graphite may be positive.

Positive coolant and blanket density effect (families: I.1, III.5, III.6)

In some concepts, dedicated coolant is used. Since there is no fuel dissolved in this coolant, it may have positive temperature feedback coefficient. This is valid also for blanket salt with dissolved fertile fuel. Any conclusion about MSR safety should be thus related to given MSR family.

Large migration area (families: I.1, I.2, II.4, partly III.6)

Large migration area for neutrons is a synonym for core transparency for neutrons. Some MSR concepts are quite transparent for neutrons and as a results they are large to minimize the neutron leakage.

Volumetric heat up and homogenization (families: I.2, II.3, II.4, partly III.5, III.6)

Whenever the fuel is dissolved in the molten salt, the fission energy and the fission products is released directly in the liquid. As a consequence, the fuel acts as volumetrically heated liquid, what should be accounted for in the safety systems layout and their simulation. Furthermore, the burnup is being smeared by homogenization for whole fuel volume.

Power level and peaking (families: I.1, partly III.5, III.6)

Solid fuel embedded in graphite or non-circulating liquid fuel enclosed in pins does not profit from the homogenization. It behaves as a standard reactor with solid fuel and the power peaking should be optimized to flatten the burnup distribution.

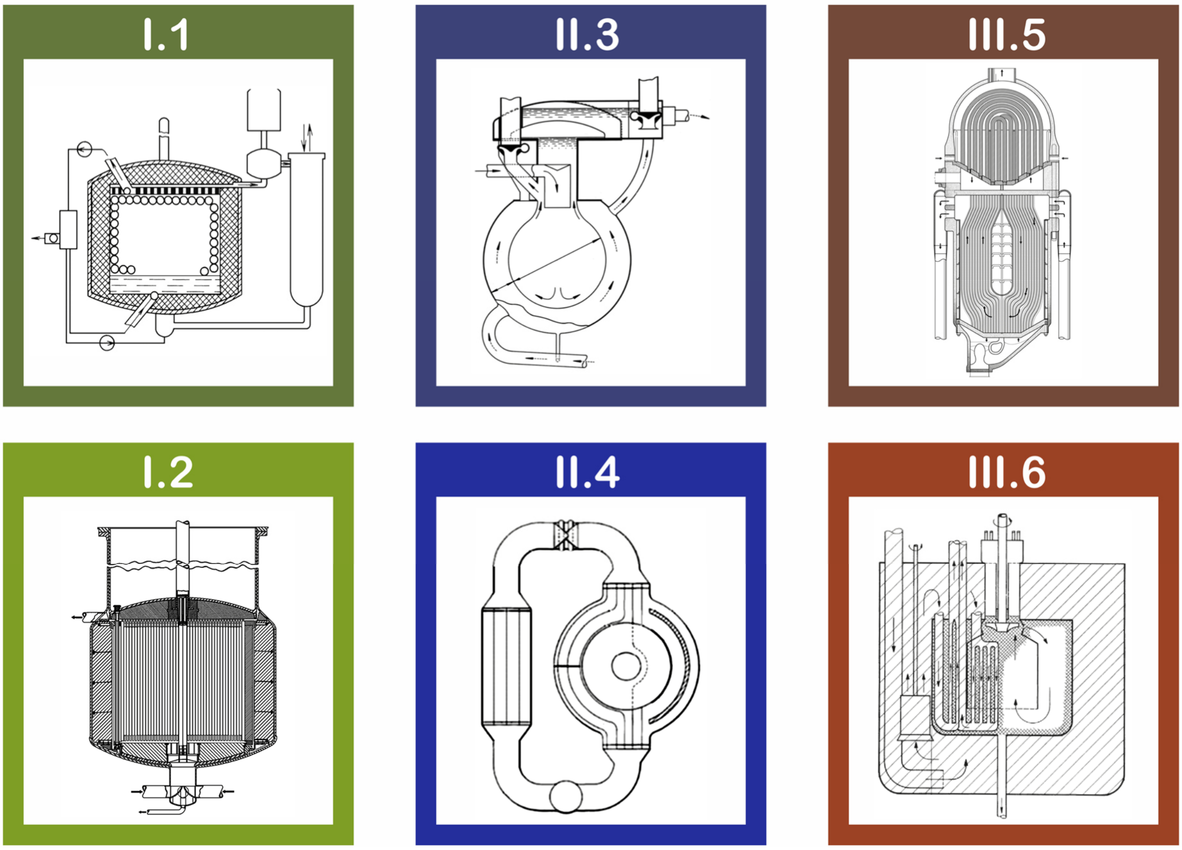

Figure 5.7 Illustration of the six major MSRs classes [5.36].

Local overheating or excessive burnup (families: I.2, II.3, II.4, partly III.5, III.6)

Even if the fuel is dissolved in the molten salt and instantly homogenized, there can be stagnation zones, with risk of local overheating and excessive burnup.

Gaseous and non-soluble fission products (families: I.2, II.3, II.4, III.5, III.6)

In all MSR concepts with liquid fuel some fission products have limited solubility in the salt. Hence, noble gases (Ex, Kr) or semi noble metallic fission products (Mo, Te, etc.) may leave the salt. Gases may cumulate in the cover gas and pressurize it. Semi noble metals can plate our on the primary circuit wall. Their treatment may be thus necessary.

Fission products circulation (families: I.2, II.3, II.4, optionally III.5, III.6)

In many concepts fuel dissolved in liquid salt acts simultaneously also as a heat transport medium. It is thus extensively circulation through the primary circuit. As a result all actinides and fission products circulate too. In case of fission products, which are emitting delayed neutrons, it results in a new safety phenomena and stronger coupling between thermal-hydraulics and neutron-kinetics.

^{233}Pa longer half-life (all families when operated in Th-U cycle)

The ^{233}Pa nuclide is an intermediate product of ^{232}Th transmutation into ^{233}U. Since it has relatively long half-live of 27 days, it can be chemically separated from the salt. On one hand, this separation reduces the parasitic neutron captures on ^{233}Pa. On the other, hand it provides a method how a weapon grade material could be separated.

Limited structural material lifespan (all families)

Lifespan of all structural material directly exposed to the neutron flux is limited. In case of heterogeneous concepts, such material is located directly in the core; in case of homogeneous concepts, it represents the core wall. Accordingly, there will be need of regular component replacements.

5.8.2 Examples of MSR projects including SMRs

Majority of Gen-IV MSR concepts listed below uses liquid fuel in the form of various molten salts, are cooled by the fuel salt itself and either moderated by various moderators or non-moderated, resulting in either thermal or fast neutron spectra, respectively. Thermal systems are designed to operate in an open fuel cycle, while fast reactors are intended for operation in a closed fuel cycle.

For classical solid fuel Gen-IV reactors the designing freedom is not so large as for MSR based on liquid fuel. The fact that the number of presented MSR concept may outnumber these others does not mean that they are more advanced or more perspective. Especially for MSR concepts it is symptomatic that they are based on a single idea, typically related to fuel cycle or core size and location. Technology readiness levels of these single ideas differs and some of them are close to 8 and 9.

Below we elaborate on one concept (MSFR) more extensively while providing a concise overview of the others.

MSFR (EU) [5.38]

The Molten Salt Fast Reactor (MSFR) is an academic design from CNRS Grenoble and was selected here as a reference design. Its advantage is that all related studies are published and not kept secret as in the case of other developers. It can be thus used to illustrate typical MSR properties. MSFR rely on fluoride salts and homogeneous active core. Accordingly, the innermost wall of the reactor vessel is forming the space for fuel salt and gives thus the shape to the active core (see Figure 5.8 left). In the past US research at ORNL, eutectic salt ^{7}LiF-BeF_2 was identified as a candidate with low neutron capture and low melting temperature (<460°C). Therefore, all graphite moderated Th-U breeders were relying on this carrier salt, which was mixed with fluorides of actinides. The addition of ThF_4 increases the melting temperature by 50°C to ~510°C, therefore increasing the operational temperature, which is a disadvantage from viewpoint of operational procedures. Since MSFR is a fast spectrum system, the Be, as a moderating material, needs to be removed and the eutectic consist of UF_4-ThF_4-^{7}LiF mixture. As a consequence, the melting temperature raised again by further 50°C to 560°C.

The structural materials, which can withstand neutron irradiation and potentially corrosive salt environment are usually based on nickel alloys. Their structural integrity is assured by the producers typically up to 700-750°C. The salt melting temperature on one side and the temperature limit of the structural material on the other side create a narrow window for MSFR operation. This window should also include a margin between melting and the minimal operation temperature and a margin for accidents due to a temperature increase. From several perspectives, the very high salt melting temperature and the narrow operating window complicate operational procedures and therefore represent a hurdle, which is slowing down the MSR deployment. There are methods how to deal with corrosion or radiation. However, when they are combined with high temperature, many established tools and methods are failing.

The salt has relatively low heat conductivity similar to water and it is not an optimal heat-exchange medium. Furthermore, the salt temperature at the reactor outlet is limited to 700-750°C. The margin to melting/solidification temperature is not large; however, the margin to boiling temperature is more than sufficient (~1500°C). One should understand here, that already before the boiling onset, some compound with high partial vapor pressure might rapidly evaporate from the salt. The salt retention capability is a function of temperature. Moreover, gaseous fission products tend to leave the salt and cumulate in cover gas. Alternatively, they are removed by a so-called off-gas system. Similarly, metallic fission products form small particles, which tend to plate out on the walls of the primary circuit or enter the He bubbles in the off-gas system to promote the separation of the gaseous fission products. The primary circuit of MSR thus resembles a vented or unvented fuel pin. Since the gaseous and metallic fission products tend to leave the salt and since the liquid state of the fuel allows for novel reprocessing methods, many designers integrate the reprocessing unit with the reactor system. As can be seen from Figure 5.8 right MSFR also relies on such integrated reprocessing unit. The removal of gaseous and metallic fission products and the possible reprocessing strongly changes the distribution of radiotoxicity in the system and has strong impact on the risk evaluation.

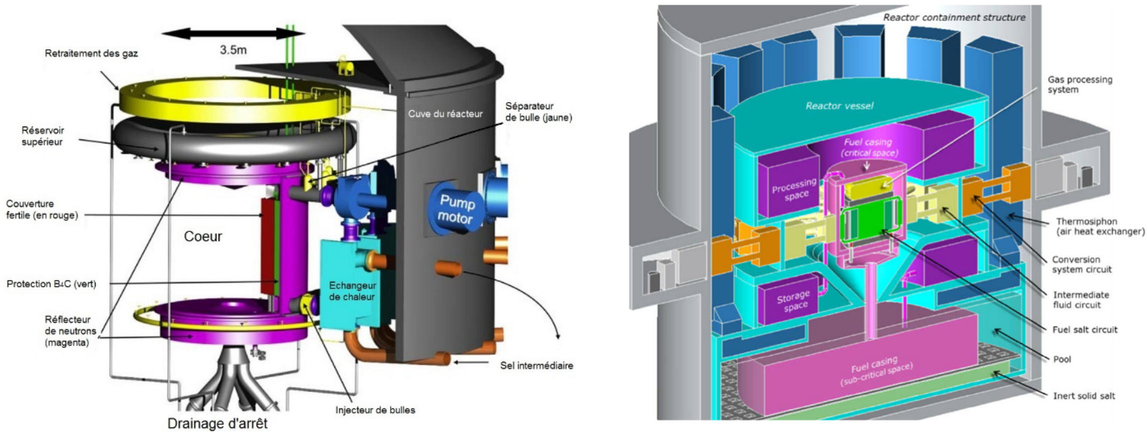

Figure 5.8 Molten Salt Fast Reactor – academic design from CNRS Grenoble; reactor core (left), containment layout (right).

Transatomic Power (USA) [5.39]

Few years ago, there were articles about “Green” reactor designed by the Transatomic Power. The idea of this company was to used moderator based on zirconium hydride (ZrH). All the superior claims have been later shown as false. The performance is somewhat similar to water moderated reactors. Furthermore, the difficulty to operate the ZrH at temperatures required for molten salt fuel resulted in termination of the project.

Seaborg (Denmark) [5.40]

The Seaborg had two ideas. One related to fuel cycle was based on utilizing liquid moderator, sodium hydroxide (NaOH). The second idea related to core location, was its placement on a ship to create so called Power Barge. Some experts have been pointing out the difficulty to operate the reactor with such an aggressive liquid. After some attempts the development of the NaOH concept was terminated and the company recently swapped to graphite as the moderator. Since they utilize salt with low tritium production based on $NaF-KF-UF_4$, the concept thus shares the same materials as the Terrestrial Energy iMSR.

Terrestrial Energy (Canada) [5.41]

The single idea behind the Terrestrial Energy was slightly different: to use materials which have been already tested in the past, so that the utilization of the reactor can be accelerated. Therefore, it is based on graphite moderator, as all concept designed by ORNL in the past. The fuel salt avoids ^{7}Li and Be to minimize the tritium production and rely on $NaF-KF-UF_4$. This concept recently completed the Canadian pre-licensing review; however, with several reservations.

Copenhagen Atomics (Denmark) [5.42]

The idea of Copenhagen Atomics was to use heavy water as a moderator and place the reactor into the standardized transport container. Since the details of the design are not public, it acts as a black box. To preserve the breeding capability and to keep the core small, it relies on fissile central zone being surrounded by thorium containing fertile blanket. The fissile fuel and the heavy water moderator should be separated by a composite material. Based on the used materials, the neutronics performance could be great. Nonetheless, the initial fuel should rely on reactor grade Pu or on minor actinides (refer to the chapter 6 for more details about fuel cycle). Should these materials not be available, a transition design fueled by High-Assay Low-Enriched Uranium (HALEU) may be needed to cumulate ^{233}U for starting the normal operation in Th-U closed cycle. It is interesting to point out here that the company has already successfully brought products on the market. They are manufacturing pumps for molten salts and experimental loops for research institutions (e.g. INL or MIT).

TerraPower (USA) [5.43]

TerraPower, a company founded by B. Gates, in long term follows the breed-and-burn concept (viz. alternative fuel cycles in chapter 6) in two major designs: solid fuel SFR and liquid fuel MCFR (Molten Chloride Fast Reactor). Breed-and-burn fuel cycle can be started by HALEU or mixed uranium-plutonium oxide fuel (MOX). Without the need of fuel reprocessing it can utilize more than 20% of the natural uranium resources. Recent/Today's LWR do not utilize more than 1% of natural uranium resources. The weakness of the concept is limited knowledge of chloride salts. The past research was almost exclusively focusing on fluorides. As for any other MSR concept, the search for optimal salt composition does not provide satisfactory results. The operation temperatures are high and the temperature window narrow.

SINAP TMSR-LF1 (China) [5.44]

The SINAP (Shanghai Institute of Applied Physics) is commissioning an experimental graphite moderated reactor which is similar to the past Molten Salt Reactor Experiment (MSRE) from ORNL. It is foreseen as a first stage of larger reactor construction. The possible position of MSR in Chinese reactor fleet is, however, not clear.

Kairos Power (USA) [5.45]

Kairos Power relies on solid TRISO fuel particles based fuel and molten salt as a coolant. It is thus rather HTR than MSR. Since it fulfils the bordered IAEA definition of MSR, it is listed here. HTRs have usually the advantage that accidents can be avoided by heat dissipation through radiation. It is enabled by temperature resistance of the fuel and by lower specific power of the core. The lower specific power is an inevitable consequence of the very low specific density of the fuel and He coolant properties. By replacing the usual He coolant by molten salt, the specific fuel density and the specific reactor power can be increased. Nonetheless, it results in loss of the passive decay heat removal feature. This concept is somewhere between HTR and MSR and its advantages are not necessary obvious.

MOSART (Kurchatov Institute, Russia) [5.46]

In Russia nuclear industry is practically state controlled and aims at a three-component structure, including Light Water Reactors, Fast Reactors and dedicated Minor Actinides Burners:

1) LEU is burned in LWR;
2) the recycled mixed uranium-plutonium oxide fuel (MOX) is used in SFR and
3) the remaining minor actinides should be burned in dedicated transmutors.

MOSART is a Molten Salt Reactor considered as one of the possible minor actinides transmuter options. A core loaded solely with minor actinides can have low or even positive Doppler effect. To oversome this safety challenge, the MOSART transmutor uses liquid fuel salt. The reactivity effect of liquid fuel salt thermal expansion is negative, because the expansion is reducing the fuel amount in the core. Another limiting characteristic of the fuel salt is a solubility of some compounds, for instance PuF_3, which may be limited and depends on temperature.

MOLTEX (Canada) [5.47]

The idea behind all MOLTEX concepts is the separation of fuel and coolant salts in the core. The structural material needed for this separation, however, deteriorates the neutronics performance of the core. It can provide some advantages when used as a minor actinides burner. Nonetheless, as any other heterogeneous system, pin break can result in fluids mixing and very long outage of the unit.

Other MSR concepts

In Figure 5.6 many additional concepts are listed: AGR-FHR, Flibe Energy, Thorcon, FUJI, BARCS IMSBR, VNINM FMSR, Elysium MCSFR, Aramis concept, ISAC project, MSFR (CI version), Raptor concept, Dual Fluid Reactor, NAAREA, STELLARIA. It is a miscellaneous mixture of projects, which are at different stages of development and at different TRS level. Some of them may vanish soon, other may stay. There are some additional concepts presented in the media. The presented information is, however, not even sufficient to categorize these concept in the taxonomy.

5.9 Questions and Concluding remarks

Question: What is main difference of Gen-IV from Gen-III? Short answer: Improved sustainability, higher operating temperature and other risk structure.

One of the GIF objectives is sustainability, including waste minimization and long term fuel supply. The major strength of Gen-IV rectors is the capability to operate in so called closed fuel cycle and consume ^{232}Th and ^{238}U. Not all 6 selected concepts have this capability. VHTR, SCWR and some MSRs concepts cannot support the closed cycle in a sustainable way because of the thermal neutron spectrum. SFR, LFR and GFR can be operated in both closed Th-U and U-Pu cycles, where the technology is established for U-Pu cycle. MSR as a category of reactors cannot be evaluated at once. There are many concept, which can breed in one or in both fuel cycles and there are concepts with mediocre neutronic performance, which cannot breed.

All Gen-IV reactor coolants have boiling temperatures higher than boiling temperature of water at pressures typical for PWR. Higher operating temperature obviously increases the thermal efficiency of the thermodynamic cycle. The more important consequence is that salt and liquid metal coolants can be operated at high temperature and low pressure at once. From safety perspective, the low pressure operation opens many new options and reduces the potential driving force for radiotoxicity release. The higher temperature and low pressure provide potential to design robust air-cooling passive safety system. The overall risk thus may be similar to LWR, although its internal structure could differ.

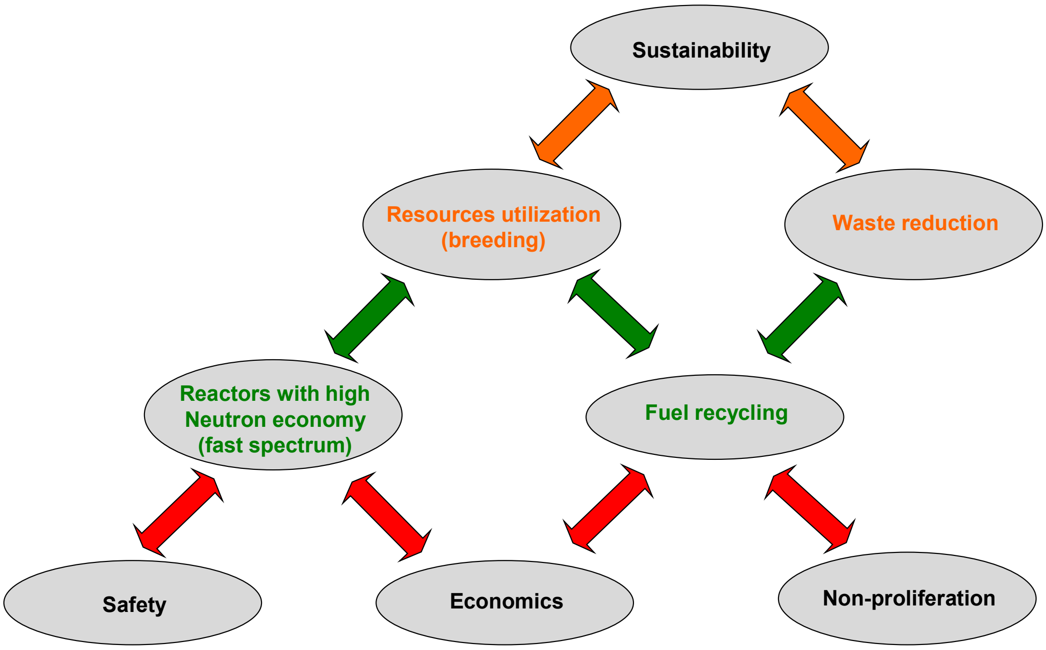

Figure 5.9 Conflict between sustainability on one hand and safety, economics and non-proliferation on the other hand.

Question: Does improved sustainability influence safety, economics and proliferation resistance?
Short answer: Yes.

Current LWRs are safe, economic and their fuel cycle is proliferation resistant. Gen-IV systems should at best equalize in these three features and add sustainability on top of it. Nonetheless, it seems that sustainability collides with these three features (safety, economics and non-proliferation) (see Figure 5.9). The sustainability requires minimal waste production, which is not possible without reprocessing. Similarly, high resources utilization based on closed Th-U or U-Pu cycle requires fast spectrum reactors and reprocessing. Reprocessing is increasing the costs and challenging the economics. Simultaneously, it increases the proliferation risk. Moreover, fast spectrum reactors may have positive temperature feedback coefficient of the coolant and thus an issue with the safety. On top of it, lead as a coolant is heavy and problematic be earthquake, sodium has low boiling temperature and exothermic reaction with water and air and gas coolant needs to be pressurized to stay effective and depressurization is a challenging problem with potentially strong consequences. Adding sustainability on top of existing LWR safety, economics and proliferation resistance is thus not trivial. Since the economics and sustainability are competing each other, either technological breakthrough should happen, which will reduce the costs, or the lack of other resources should increase the acceptable cost level, or a political decision should prioritize sustainability over economics, with a hope that in long term these system will be more economic. Especially MSR has potential to profit from simplicity of the advanced designs and to combine sustainability, safety and economics.

Question: Could Gen-IV reactors be designed as SMRs?
Short answer: Yes, especially MSR, but small core size is not good for sustainability.

Recently concepts of so called Small Modular Reactors (SMR) are discussed. Since breeding process has very tight neutron economy, downsizing of the Gen-IV reactors goes against the breeding capability. Figure 5.10 illustrates several potential advantages of fast SMRs. In general, the modularity could be more important than the small core size. It may be that in the future BMR (Big Modular Reactors) will overtake the lead from SMRs. The best potential and safety scalability from SMR to BMR may have some MSR concepts. They have potential to combine breeding capability with negative temperature feedbacks and passive safety.

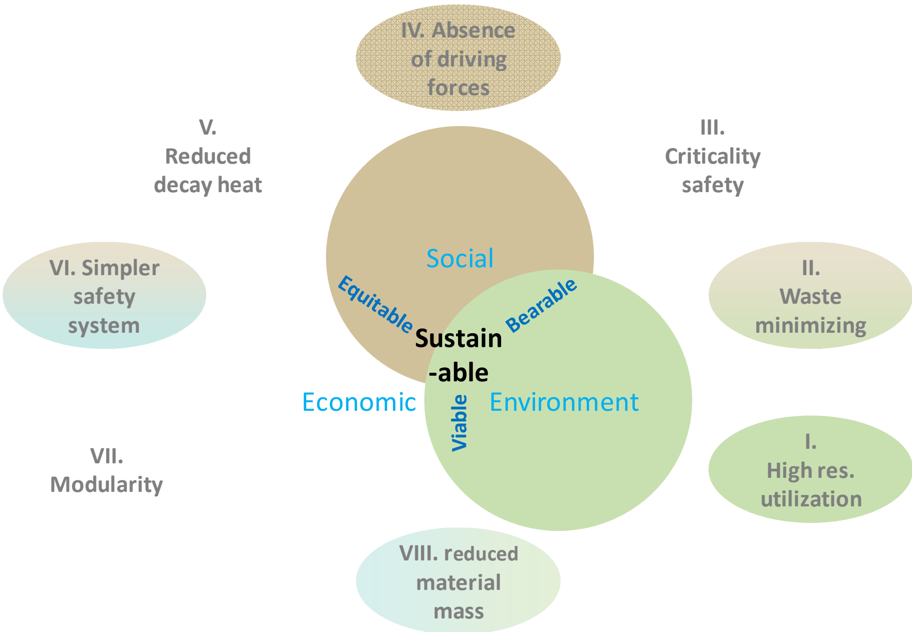

Figure 5.10 Potential advantages/features of fast SMRs for the three general pillars of sustainability.

Question: By when can the commercialization be expected? At what cost?

The costs are always very hard to estimate for first-of-a-kind construction. The closed fuel cycle would require recycling capability and its establishing needs rather long term strategic planning and sufficient budget, which allows for failures. Good example here is the Rokkasho Reprocessing Plant in Japan, where the construction started in 1993 and the MOX fuel manufacturing plant is expected to be in operation in 2024. Such a delay is common for complex technological platforms, which are built as first of a kind in given country. Another example of first-of-a-kind construction is EPR, where even four first of a kind units have been / are being constructed in China, Finland, France and UK. The cost and construction times strongly differs between these cases. The original supplier AREVA even bankrupted and was overtaken by EDF. Still it may be a successful story, if France were to build additional EPRs, as recently announced. In that case, EDF would profit from the collected know how and have a chance to average the high initial costs.

Another first-of-a-kind reactor is China's HTR-PM reactor. It is in operation and the construction cost are known for the respective construction company. There was a project between INET and PSI and as a part of this cooperation the costs of HTR-PM construction in China should have been transposed to Switzerland to estimate the reactor cost in different countries. The obvious conclusion was that cost from one country cannot be easily superimposed to another country.

To certain extend, the nuclear industry can be compared with aircraft industry. In both cases the development cost are enormous, prototypes are costly and regulatory environment tough. The Airbus A380 could be mentioned here. There were 254 aircraft constructed; however, to cover the initial development costs, more aircrafts should have been constructed (presumably 280). It is convoluted case, where state subventions played a role. Compared to aircrafts production, energy supply is even more strategic and governments around the globe support domestic vendors in development of new reactors. It is not obvious that the result will differ from Airbus A380.

For large and strategic projects, which last for many years the capital cost could be crucial. The construction of nuclear plant should be as short as possible so the capital cost are reduced and so that the operating plant generate an income. The highest extra cost from EPR delay were not related to additional components of repetitive construction of some segments. The major part was capital cost. Some countries realize that and tried to decrease capital cost of private companies by governmental guaranties or even by nationalization of the utility. It was the case of French EDF and it might be the case of Czech CEZ utility. Part of the SMRs success story is also the reduction of capital cost, where first module can already generate income, whereas others are in construction.

Question: Are Gen-IV reactors the only one capable to operate in closed cycle? Short answer: no.

There is a difference between closed fuel cycle operation and sustainability. The closure of a fuel cycle means that some fraction of the fuel after irradiation in the reactor will be used to produce a new fuel. If no irradiated fuel is used for manufacturing a new fuel, the fuel cycle is considered open. The sustainability means two interconnected pillars: 1) the fuel cycle uses natural resources efficiently (ideally the reactor is fuelled by natural uranium or thorium only); and 2) the waste output is minimized, i.e. the amount of material in the waste that potentially can be burned in the reactor is minimal (ideally, the waste should consists of fission products only). For both these pillars, the best results can be obtained by fuel recycling in a fast neutron spectrum reactor. In an ideal situation the fuel recycling means that after irradiation 1) all remaining heavy metals are separated from fission products; 2) the new fuel is produced by mixing these heavy metals with natural U or Th (the mass of the natural U or Th should equal the mass of the separated fission products) and 3) the fission products go to the final repository.

As for many other materials and industries (steel, concrete, glass, plastic, etc.), circular economy should be introduced also for actinides in nuclear technology. Nonetheless, recycling of actinides for any fuel cycle requires a reprocessing plant. To establish reprocessing capacity is thus initial and most important step to introduce actinides circular economy. The consideration to construct a Gen-IV reactor should thus always include also the consideration about a reprocessing plant. The added value of Gen-IV reactors operated in open cycle is lower, unless it is seen as temporary phase in closed cycle utilization.

The actinides circular economy could be introduced already for current or future thermal-spectrum LWRs. They can recycle U and Pu in the form of mixed uranium-plutonium oxide fuel (MOX) or U only in the form of reprocessed uranium (RepU) (see Chapter 6). Such recycling is technologically the most available method of reducing actinides amount in the waste stream, because in this case only fission products and minor actinides (Np, Am, Cm, etc.) end up as vitrified waste. Nonetheless, even if the fuel cycle would be closed, LWRs still rely on fissile ^{235}U. This is because the composition of fissile and fertile isotopes generated from the main fertile isotope (^{238}U or ^{232}Th) in the thermal spectrum does not allow for a sustainable chain reaction without adding external fissile material (e.g. ^{235}U). Therefore, even in a closed fuel cycle LWRs cannot fully profit from ^{238}U (or ^{232}Th) reserves. The utilization of natural uranium resources will be still less than 1%.

The major added value of Gen-IV reactors is capability to utilize ^{232}Th and ^{238}U in closed fuel cycle. Since the reserves of these nuclides are by more than two order of magnitude higher than for ^{235}U, nuclear reactors fuelled by ^{232}Th and ^{238}U could provide energy for the entire Earth for centuries or even thousands of years. As such it is a technology, which should be compared or considered equally important as a nuclear fusion.

Question: What opportunities will this technology open up for the reuse of partially used fuel? Short answer: In can be seen as initial resource for Gen-IV.

The Gen-IV reactors are designed to be operated in a closed fuel cycle. Once circular economy for actinides is introduced, the only actinides in the waste stream will originate from the reprocessing losses. The efficiency of the reprocessing technique would thus determine the final radiotoxicity of the waste. The stewardship burden of the irradiated fuel is caused by long-term radiotoxicity of spent fuel driven by Pu and minor actinides. In actinides circular economy Pu in MOX form may serve as an initial fuel for the Gen-IV reactors or be recycled in LWR. The Pu amount cumulated during presumed 60 years of Swiss nuclear power plants operation will be sufficient to fabricate 2.5 full core loadings of the ESFR reactor (see Section 5.6.2). From this perspective, the amount is not large.

Question: Are Gen-IV the only reactors, which can reuse the partially used fuel? Short answer: No.

The used fuel has three major heavy-metal components: Pu, minor actinides and irradiated uranium, called reprocessed uranium (RepU) after separation. LWR can utilize Pu as MOX fuel and RepU after re-enrichment as a fuel. There exist also dedicated burners, which can burn Pu and minor actinides or even solely minor actinides. The performance of any burner in an open cycle is, however, low and the best results for waste minimization can be obtained by multi-recycling in a closed fuel cycle. Even with the repetitive recycling, at the end there will be amount of spent fuel, which will correspond to at least one if not two core loading of the burner. The mass reduction will thus be at best 90%. Since there are also reprocessing losses, the final spent fuel repository could never be avoided. For countries, which are phasing out nuclear energy, spent fuel deposition in a final repository is thus the least risky option. Nonetheless, the repository should allow for actinides retrieving in the future, because it is strategic material.

Question: Are there going to be also Western vendors offering this technology? Short answer: Yes.

SFR is the most mature technology among the six GIF systems. It has 300+ reactor-years of operational experience, six reactors are under operation (see Table 5.3) and one of them (BN-800) is very close to actually being the Gen-IV system. From this simple comparison we can conclude that SFR is a GIF concept which is most close to commercialization. This is in a way confirmed by the Terrapower's Natrium™project [5.27], which is a small sodium fast reactor. In 2020 the DOE awarded TerraPower $80 million to demonstrate the Natrium™ reactor and integrated energy system with its technology co-developer GE Hitachi Nuclear Energy and engineering and construction partner Bechtel. In 2022 TerraPower purchased land in Kemmerer, Wyoming for Natrium Reactor Demonstration Project. According to Terrapower the first Natrium plant will begin commercial operations in 2030. Unfortunately, as of November 2023 there is no such a commercial project of small SFR in Europe.

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6 Review of fuel availability

Tony Williams (AXPO)

6.1 Introduction

In this chapter, the long-term, global availability of nuclear fuel is addressed, with a particular focus placed on Swiss requirements.

The analysis is based upon a handful of reputable standard reference works, the contents of which are, where appropriate, compared and challenged to try and present as balanced a picture as possible. Since the reference works typically extend to 2040, this is also the timeline chosen for most of the analyses. However, since the operation of Swiss plants beyond 2040 cannot be excluded, the supply situation beyond this date is also briefly discussed.

6.2 Some Fundamentals of the Nuclear Fuel Supply Chain

We define nuclear fuels as materials that are used to generate energy in the form of heat in nuclear reactors. Nuclear fuels inevitably contain “fissile” isotopes. These include the isotopes uranium-235, uranium-233, plutonium-239 and plutonium-241. Of these, only the isotope uranium-235 occurs in nature, the other isotopes can be produced from thorium-232 and uranium-238, which are therefore known as “fertile” isotopes. The conversion of thorium-232 and uranium-238 into fissile isotopes is typically carried out in so-called "breeder" reactors.

Most of the over 400 nuclear power plants (NPP) in operation today are so-called Light Water Reactors (LWR) whose nuclear fuel cycle is based upon uranium-235 (U-235)[10]. Consequently, and because it would go beyond the scope of this analysis to include all types of nuclear fuel, the main body of the analysis is devoted to the U-235 based fuel supply chain.

The isotopic composition of natural uranium (^{nat}U) is as follows[11]:

U-235:	0.72 %
U-238:	99.27 %
U-234:	0.0055 %

LWRs require a fraction of U-235 increased from 0.72% to approximately 4-5%. In fact, the production process for nuclear fuel consists of 4 consecutive steps (see Figure 6.1).

Each of the four physical processes have their own specific supply chains and can be procured separately. In practice however, the separate components are often “bundled”. For example, it is possible to purchase $^{nat}UF_6$ directly from the converter without providing U_3O_8, or $^{enr}UF_6$ from the enricher without providing $^{nat}UF_6$. It is even possible to buy complete fuel assemblies without having to worry about sourcing the fissile material at all.

[10] https://pris.iaea.org/PRIS/WorldStatistics/OperationalReactorsByType.aspx

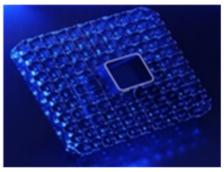
[11] https://www.chemie.de/lexikon/Uran.html

Exactly how to procure is a strategic decision and depends on the commercial environment but also on the internal resources available. In the Swiss plants, a variety of strategies have been followed, for instance Beznau has typically purchased complete fuel assemblies, whereas Leibstadt has procured fabrication services and $^{enr}UF_6$ separately. As this aspect has little influence on the overall security of supply of fuel, here we will look at the four components separately. Current and future demand is addressed in section 6.4 and primary supply in section 6.5.

1. **Uranium mining**: Natural uranium is mined and refined, resulting in $^{nat}U_3O_8$ ("yellowcake") as a starting material for conversion.

2. **Conversion**[12]: $^{nat}U_3O_8$ is converted into $^{nat}UF_6$ (uranium hexafluoride) as a starting material ("feed") for enrichment.

3. **Enrichment:** $^{nat}UF_6$, which sublimes at 56°C, is heated to above this temperature and enriched in centrifuges. This results in $^{enr}UF_6$ ("EUP")[13] and serves as a starting material to produce fuel pellets for fuel assemblies.



4. **Fuel assembly fabrication**: The $^{enr}UF_6$ is converted into $^{enr}UO_2$ and formed into pellets, which are filled into rods and incorporated into the fuel assembly structure.

Figure 6.1 Steps of fuel production

In addition to the primary supply dealt with in detail in section 6.5, there also exists a complementary secondary market. This secondary market is significant and needs to be considered when analyzing supply and demand scenarios. To this end a discussion is included in section 6.6.

A fifth, somewhat neglected step is transport/logistics. Typically, the individual fuel components are produced in geographically diverse locations which necessitates their physical movement

[12] Confusingly, "conversion" also refers to the conversion of $^{enr}UF_6$ to UO_2 for pellet fabrication

[13] Enriched Uranium Product

from one location to the other. A brief discussion of transport/logistics is included in section 6.7.

The availability of a commodity such as nuclear fuel is of course related to its price and whether buyers are able/willing to pay that price. A brief discourse on the financial aspects of nuclear fuel supply is found in section 6.8.

The global nature of the nuclear supply chain makes it susceptible to geopolitical influences, a recent example being the Russian invasion of Ukraine in 2022. A discussion of current and potential future geopolitical trends and their possible consequences is therefore provided in section 6.9.

In section 6.10 the outlook for fuel supply beyond 2040 is addressed, also considering unconventional as well as alternative sources. The chapter concludes with section 6.11 in which a summary and conclusions are provided.

6.3 Sources of Information and Assumptions

The analyses are based primarily on the following standard reference publications [6.1] to [6.5]:

- Uranium Resources, Production, and Demand 2022, OECD/NEA (NEA/IAEA, 2023)
- World Energy Outlook 2020, IEA (IEA, 2020)
- Global Scenarios for Demand and Supply Availability 2021-2040, WNA (WNA, 2022)
- Nuclear Energy Data 2021, OECD (OECD/NEA, 2022)
- Nuclear Fuel Cycle Supply and Price Report 2022, ERI (ERI, 2022)

For the future scenarios, the reference year 2040 is chosen, mainly because the available reliable sources are limited to this period. In some cases, the reference years of the various sources do not exactly coincide. Due to the general long-term nature of the topic, this plays only a subsidiary role for the statements and conclusions.

6.4 Current and Future Demand for Nuclear Fuel

In this section we discuss the current and future demand for nuclear fuel, in terms of the four components, both globally and more specifically in Switzerland.

As of January 1, 2021, a total of 442 commercial nuclear reactors with a total capacity of 390 GWe were in operation worldwide and in 2020 a total of 2’523 TWh of electricity was generated from nuclear energy [6.1].

According to the federal government's electricity statistics, in the hydrological year 2022, Swiss nuclear power plants generated 21.16 TWh[14] of electricity.

[14] https://www.bfe.admin.ch/bfe/de/home/versorgung/statistik-und geodaten/ energiestatistiken/ elektrizitaetsstatistik.html

6.4.1 Uranium

Current requirements

According to (NEA/IAEA, 2023) [6.1], global annual demand for ^{nat}U on 1 January 2021 amounted to around 60’100 tonnes (t) of ^{nat}U. In Switzerland, the current uranium demand is around 500 t[15] or less than 1% of global annual demand.

Future requirements

The future demand for nuclear fuel is, of course, directly linked to the future development of nuclear energy which can only be based on estimates or forecasts.

In (NEA/IAEA, 2023) [6.1]*high growth* and *low growth* scenarios are used to estimate a range of future uranium demands. In the low growth scenario, nuclear power generation is assumed to remain at approximately the current level until 2040. In the high growth scenario, an increase to 677 GWe is postulated.

The World Nuclear Association (WNA), seemingly more optimistic about the future development of nuclear power, proposes a range of 449 – 839 GWe (WNA, 2022) [6.3]. In contrast, the International Energy Agency (IEA) [6.2], postulates a more conservative range of 455 – 569 GWe for 2040[16], as does the ERI [6.5] with a range of 332 – 693 GWe. The WNA generation forecast clearly represents an outlier compared with the other three references, see for instance *Figure 6.2* in which the reference generation forecasts from a number of sources, including [6.1] and [6.5] are compared with the WNA.

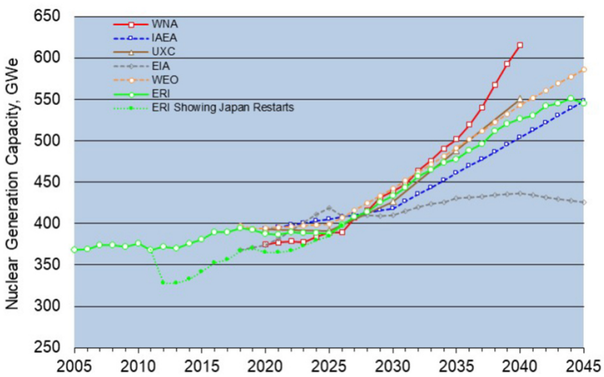


Figure 6.2 A Comparison of nuclear generating capacity forecasts to 2045, from (ERI, 2022).

The implications of the scenarios in [6.1] for uranium demand are shown in *Figure 6.3*. While the low growth scenario predicts a future uranium demand that will remain at around today's level, the high growth scenario shows an increase of around 80% to approximately 108’000 t by 2040, This value has been estimated using an assumed tails assay of 0.25%, the relevance of which is addressed in 6.4.3. A similar high case scenario is presented in [6.5].

[15] This value corresponds to actual requirements with an assumed tails assay of 0.25%

[16] 479 – 599 GWe (gross) reduced by 5% to yield GWe (net) – to be compatible with (1), (3) and (5)

If we take the high scenario from the WNA report [6.3], the demand for uranium increases to 156’000 t[17], almost three times current demand.

During this period, i.e., until 2040, and assuming that KKB will be permanently shut down in the period 2030 – 2040, Swiss demand will fall to around 380 t U per year.

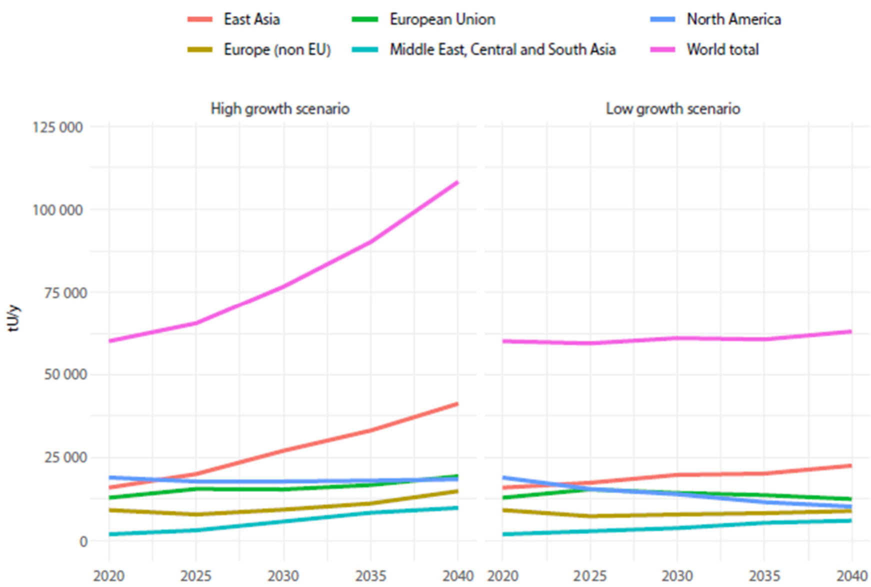


Figure 6.3 Estimation of global demand for uranium for high and low growth scenarios according to OECD/NEA [6.1]

6.4.2 Conversion

Current requirements

In principle, the demand for conversion goes together with the demand for ^{nat}U. If uranium is required, then so is conversion, because most NPPs require $^{nat}UF_6$ to facilitate enrichment. As a result, the scenarios from 6.4.1 can also be applied to conversion.

In 2020, approximately 61’500 t of conversion were needed globally [6.3]. For Switzerland, requirements were around 500 t of conversion.

Future requirements

Here too the need for conversion is analogous to ^{nat}U. In 2040, referring to section 6.4.1, the range of necessary conversion capacity will correspond to 60,000 t – 110,000 t.

By 2040, by which time KKB will probably have been decommissioned, Swiss annual conversion requirements will have reduced to approximately 380 t.

[17] WNA assumes a tails assay of 0.22%

6.4.3 Enrichment

As mentioned above, enrichment represents the increase of the fraction of the isotope U-235 from 0.7% in $^{nat}UF_6$ to around 4-5% in $^{enr}UF_6$, the final product being known as Enriched Uranium Product (EUP). Now, although enrichment requirements are clearly related to the requirements for EUP, the relationship is not unique due to the influence of the choice of the so-called “Tails Assay”[18] on the relative amounts of ^{nat}U and enrichment required to produce a given enrichment level (see *Figure 6.4* and following text).

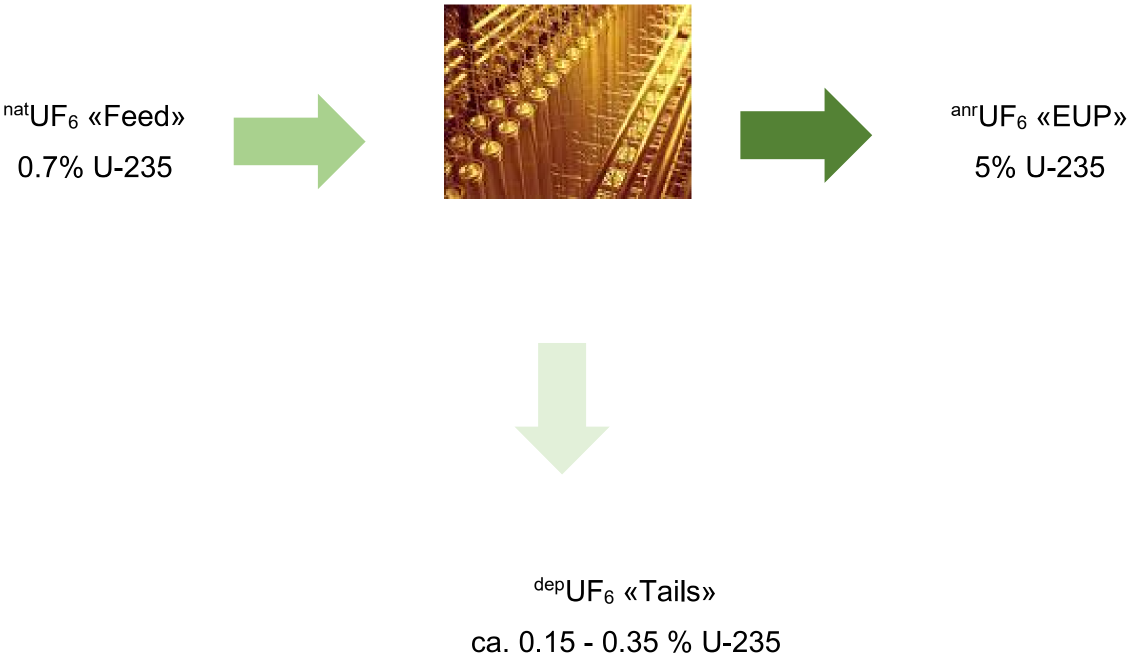


Figure 6.4 Schematic representation of the enrichment process

This influence of tails assay always needs to be considered when assessing uranium supply and enrichment capacity. Here an example:

Assuming the operator of a NPP requires 20 t of EUP enriched to 5%, requirements would amount to 144’000 SWU[19] and 229 t of $^{nat}UF_{6,}$ assuming a tails assay of 0.3%.

If the required tails assay were to be changed from 0.3% to 0.2% however, then the respective requirements would change significantly, to namely 177’000 SWU and 187 t of $^{nat}UF_6$.

The tails assay is always defined in enrichment purchases and often it can be specified by the customer over a limited range typically 0.25 – 0.35%. However, this information is only used to define the price of the enrichment services and the amount of $^{nat}UF_6$ which the customer

---

[18] Tails Assay : the enrichment of the depleted stream ($^{dep}UF_6$) in the enrichment process, see Figure 6.4.

[19] Separative Work Unit : the term used to define a unit of enrichment.

needs to deliver. The tails assay actually applied in the centrifuge plant is chosen by the enricher as a result of optimizing their own internal costs and processes.

For instance, in a market in which $^{nat}UF_6$ prices are low and enrichment prices high, a customer would choose a high tails assay e.g., 0.35%, thus minimizing the amount of enrichment they need but consequently increasing the amount of $_{nat}UF^6$ they need to deliver. Depending on the enricher's internal costs and surplus capacity however, they may choose to "underfeed", meaning that they would physically enrich to a lower tails-assay e.g., 0.25%, thus expending more enrichment than the customer purchased, but using less $^{nat}UF_6$ than the customer delivered. The surplus uranium belongs to the enricher and can be sold into the market.

From the example above, a decrease of 0.1% in tails assay reduces the feed ($^{nat}UF_6$) requirements by 20% thereby increasing the SWU requirements by 20%. This interchangeability between uranium and enrichment requirements causes a considerable uncertainty in the assessment of future supply / demand scenarios and is addressed in section 6.6.

Current requirements

According to [6.3], the global enrichment requirements in 2020, assuming a tails assay of 0.22%, were some 50'200'000 SWU. As mentioned on several occasions in this chapter, this figure does not necessarily represent the amount of enrichment physically needed to be carried out, since we do not know which tails assay the enrichers apply in practice. What we do know, is that for several years now, mainly due to the mothballing of a large fraction of the Japanese fleet following Fukushima, global enrichment capacities have significantly exceeded requirements. This has incentivized enrichers to resort to underfeeding (i.e. utilizing less uranium and more enrichment than is contractually specified), which justifies the relatively low tails assay assumed in [6.3]. This also means that significant additional enrichment capacity is available if the demand situation should change, see section 6.5.3.

Enrichment requirements in Switzerland in 2020 amounted to roughly 430'000 SWU.

Future requirements

Figure 6.5 shows the estimated enrichment requirements until 2040 according to [6.3]. In the high case, the requirements will more than double, reaching 120'000'000 SWU in 2040. Even in the low case, an increase of around 30% is predicted. Again, a tails assay of 0.22% was assumed; if this were to be increased, for example as a result of a shortage of enrichment, then the requirements would significantly reduce. In fact, this is already happening; since the Russian invasion of Ukraine, and with it the possibility that Russia's share of the global enrichment market will no longer be available to western utilities, western enrichers have begun to increase tails assays again.

Again, compared with other studies, the WNA scenarios are rather bullish when it comes to the future development of nuclear energy. In [6.5], for instance, the high case enrichment requirements represent only 95'000'000 SWU, which seems to us to be a more reasonable assumption.

Enrichment requirements in Switzerland in 2040 will amount to roughly 320'000 SWU.

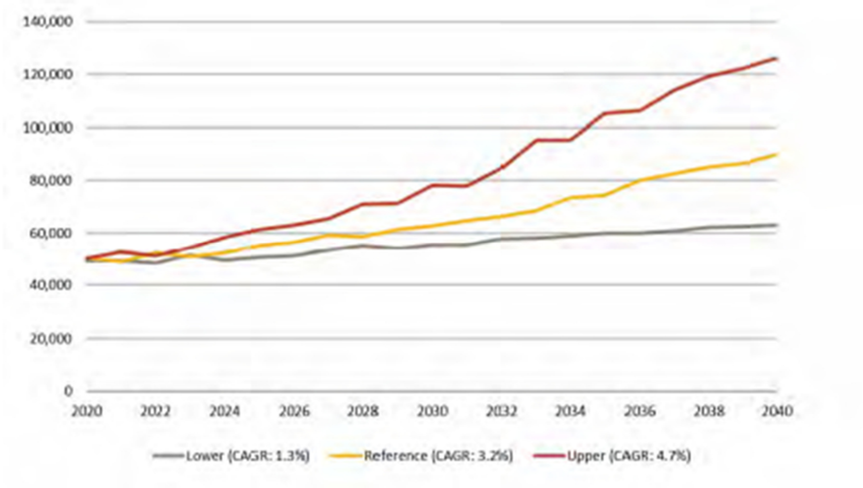


Figure 6.5 Enrichment requirements for various scenarios (‘000 SWU), WNA [6.3]

6.4.4 Fabrication

Fuel fabrication capacities are quantified in terms of the mass of nuclear fuel, expressed as “*tonnes of heavy metal (tHM)*”, contained within the fuel assembly, excluding the mass of the metal fuel assembly structure itself.

Fuel fabrication is the only one of the four components of nuclear fuel which is not fungible. In other words, fabrication is specific to a reactor type and sometimes even to an individual reactor. In this analysis we will focus on the reactor designs relevant for Switzerland, i.e., Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR), collectively known as Light Water Reactors (LWR).

Furthermore, requirements for fabrication services are typically split between reloads and first-core requirements. Every operating reactor requires the fabrication of a reload of fuel every year or two (depending on the cycle length). First cores on the other hand are only required once at the beginning of each reactor lifetime. However, since first cores require typically five times more fabrication services than annual reloads, these can have a significant effect on market dynamics. In principle, this is also the case for uranium, conversion, and enrichment, but because these components are fungible, the effect is less directly relevant for a particular plant or vendor.

Current requirements

Current global [6.3] and Swiss requirements for fuel fabrication services are summarized below in Table 6.1. Although the most common PWR fuel type today is the 17x17 design, the Swiss PWRs KKB and KKG utilize 14x14 and 15x15 designs respectively.

Table 6.1 Current LWR Fuel Fabrication Requirements (tHM)

	PWR (non 17x17)	PWR 17x17	BWR	Total LWR
Global [6.3]	*1’164*	*3’471*	*982*	*~6’000*
Switzerland	*30*	*0*	*21*	

Future requirements

According to[6.3], the reference case global BWR fabrication requirements will remain roughly constant at 1000 tHM until 2040. In the high demand scenario, they are predicted to rise to 1'300 tHM until 2040.

Whereas the "*non 17x17 PWR*" market (relevant for KKG & KKB) is predicted to remain roughly constant at around 1000 tHM (no further reactors which utilise this type of fuel will be built), the 17x17 market is expected to increase in both the high and low scenarios, from 3500 tHM in 2020 to between 4900 – 9700 tHM in 2040.

In Switzerland, again assuming the decommissioning of KKB before 2040, the PWR fabrication requirements will reduce to about 18 tHM by 2040, whereas the BWR requirements will remain constant at 21 tHM.

6.5 Current and Future Fuel Supply

6.5.1 Uranium

Historical and current production

Uranium has been mined since the mid-1940s, initially for military purposes, later predominantly for energy production in civilian reactors.Figure 6.6 shows the development of uranium production compared to actual requirements in the period 1945 – 2020.

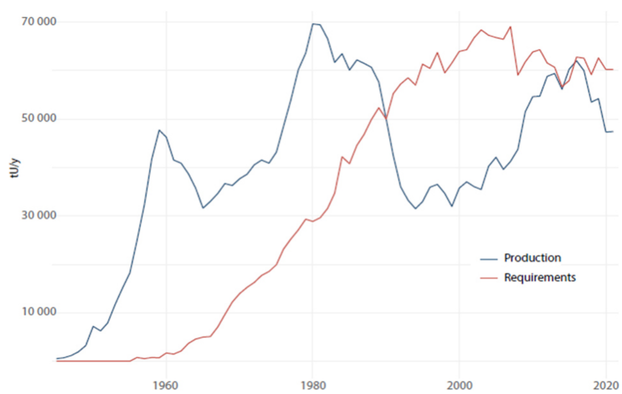


Figure 6.6 Historical development of global production and demand for natural uranium

Several phases can be identified:

1945-65: Uranium mining driven mainly by military needs, large quantities of uranium were discovered, mined and stored.

1965-85: Sharp increase in production, driven by the anticipated rapid development of civil nuclear energy. Hundreds of plants were built, and hundreds more were planned.

1985-05: After the events in Three Mile Island and Chernobyl, many new construction projects were abandoned. The demand for uranium decreased, and so did its price. In response, production and exploration fell sharply. Large quantities of stockpiled uranium flowed into the market and kept prices below average production costs for more than 20 years, see section 6.8.

2005-present: Since around 2005, the anticipated renaissance of nuclear energy has driven up prices and thus exploration activities. Despite the effects of Fukushima, production volumes have risen steadily since around 2005. In 2017/18, for the first time since 1990, as much uranium was mined as was required annually. From 2018, however, due to falling prices, production was again partially curtailed.

It may seem surprising that the nuclear energy industry has survived a significant production deficit for over two decades. This points to the fact that demand has not been fully stilled by primary production, but that a number of so-called „secondary sources ", as mentioned in section 6.2, have been required. More about these in section 6.6.

Figure 6.7 shows the development of the uranium production for the most important countries of origin since 2009.

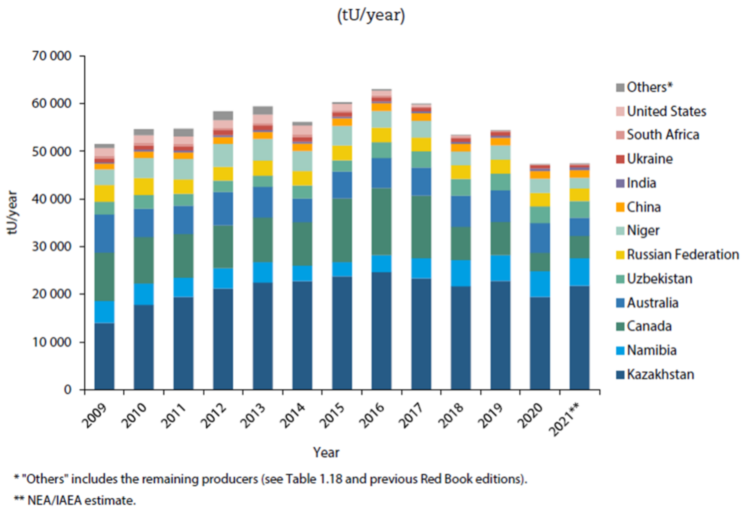


Figure 6.7 Evolution of uranium production in the most important countries [6.1]

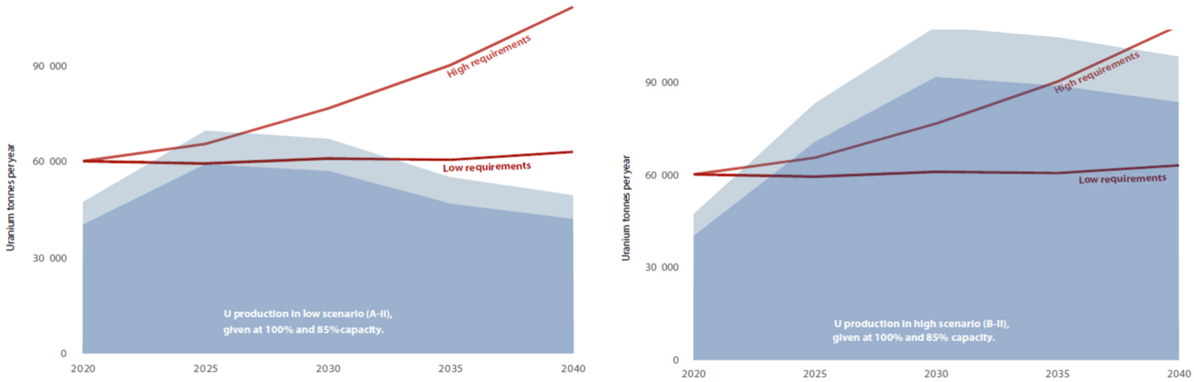
Noteworthy is the increase in production from Kazakhstan, as well as the reduction of Canadian production since 2017, partially due to oversupply and partially due to the Covid-19 pandemic. Furthermore, it can be seen that Russia plays only minor role in uranium production. In 2021, around 47,000 t of U were produced, with the following geographical distribution:

Kasachstan	41%
Australia	13%
Namibia	12%
Canada	8%
Usbekistan	8%
Niger	6%
Russia	6%
Others	6%

Comparing this to a demand side of more than 60‘000 t points to a supply shortfall of over 13‘000 t or 20% in 2021. As mentioned above, this shortfall was compensated by secondary supply, predominantly the drawdown of inventories.

Future supply

Figure 6.8 shows the projected uranium requirements from section 6.4.1 compared with the estimated production volumes, also in a low and high production scenario, according to the NEA/OECD [6.1].



a) Low production scenario *b) High production scenario*

Figure 6.8 Projected world uranium production to 2040 (supported by identified resources at a cost of <USD 130/kgU) compared with reactor requirements [6.1]

The production volumes relate to identified resources which could be retrieved at or below a price of 130USD/kg$^{nat}UF_6$ (prices for $^{nat}UF_6$ in June 2023 were around 190 USD/kgUF_6). The defintion of Identified Resources is provided in Figure 6.9, whereby:

Reasonably assured resources (RAR) refers to uranium that occurs in known mineral deposits of delineated size, grade and configuration such that the quantities, which could be recovered within the given production cost ranges with currently proven mining and processing technology,can be specified. Estimates of tonnage and grade are based on specific sample data and measurements of the deposits and on knowledge of deposit characteristics. Reasonably assured resources have a high assurance of existence. Unless otherwise noted, RAR are expressed in terms of quantities of uranium recoverable from mineable ore.

Inferred resources (IR) refers to uranium, in addition to RAR, that is inferred to occur based on direct geological evidence, in extensions of well-explored deposits, or in deposits in which geological continuity has been established but where specific data, including measurements of the deposits, and knowledge of the deposit's characteristics, are considered to be inadequate to classify the resource as RAR. Estimates of tonnage, grade and cost of further delineation and recovery are based on such sampling as is available and on knowledge of the deposit characteristics as determined in the best known parts of the deposit or in similar deposits. Less reliance can be placed on the estimates in this category than on those for RAR. Unless otherwise noted, inferred resources are expressed in terms of quantities of uranium recoverable from mineable ore.

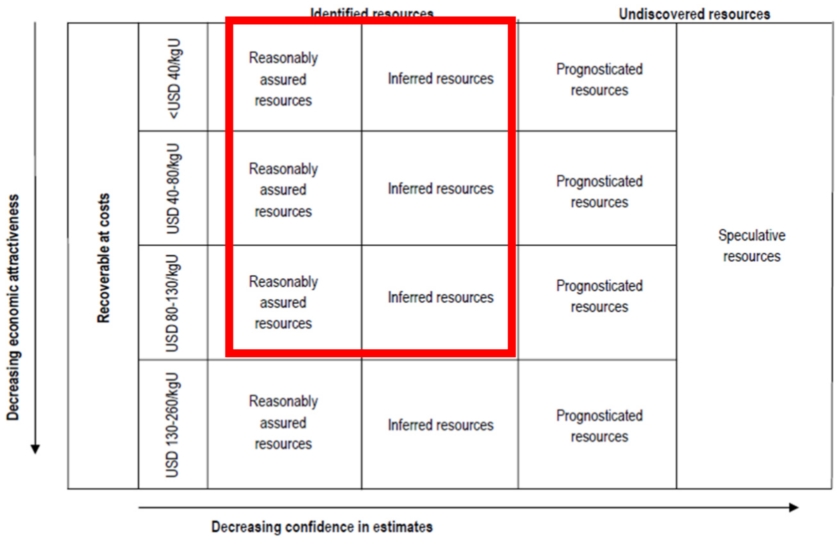
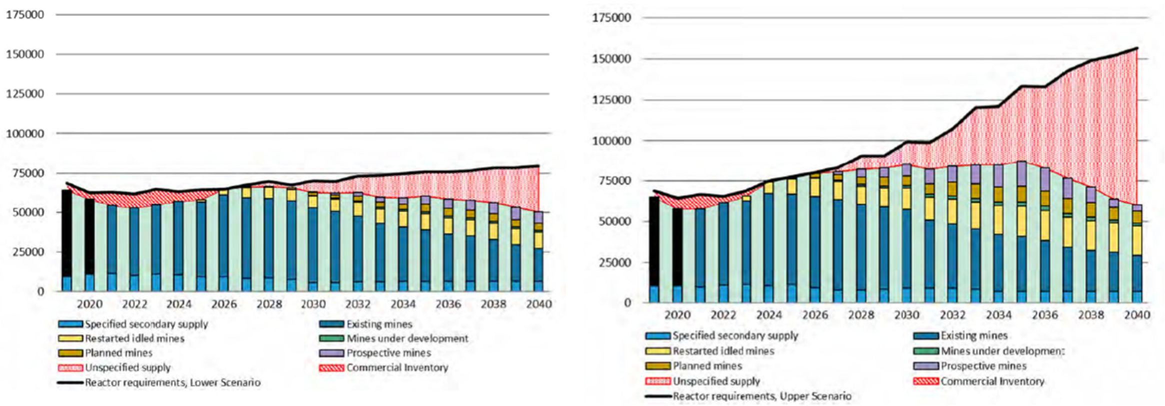


Figure 6.9 NEA/IAEA classification scheme for uranium resources, the red border marks the categories which are assumed in Figure 6.8 [6.1].

Referring to Figure 6.8 , we observe that, whereas the low production scenario hardly covers the low requirements scenario, the high production scenario covers even the high requirements scenario until practically 2040. Clearly, the supply and demand curves are not independent. As exploration, development and production are strongly linked to demand and therefore prices, it is very unlikely that a low production scenario will be coupled with a high demand. With this logic, the WNA, in assessing uranium supply and demand, have only combined low-low and high-high scenarios [6.3]. Nevertheless, since the WNA high demand scenario is significantly higher than the equivalent OECD/NEA scenario, the WNA uncovered demand (see *Figure 6.10* «unspecified supply») is significantly larger than the undersupply shown in *Figure 6.8* b).



a) low production, low demand **b) high production, high demand**

Figure 6.10 Uranium supply and demand scenarios (tU) according to WNA (WNA, 2022)

In their conclusions, the WNA rationalise the shortfall as follows [6.3]:

«Regardless of the scenario, in the long term, the industry needs to at least double its development pipeline of current, under development, planned and «potential supply» projects by 2040. Undoubtedly there are more than adequate project extensions, uranium resources and other projects in the potential supply category to accomplish this, assuming economic conditions incentivize the necessary investment in the timeframe required. »

OECD/NEA [6.1] arrive at similar conclusions to WNA:

«The uranium resource base described in this report is more than adequate to meet currently projected growth requirements to 2040. As far as the availability of physical resources is concerned, there is no reason to assume major changes in this picture even beyond 2040.

However, consumers and producers need to ensure that adequate framework conditions for the exploration, mining, transformation, and transport of uranium are in place. This includes pricing mechanisms that allow for sufficient visibility in order to allow for the considerable long-term investments required. »

...as do ERI [6.5]:

«While the high forecast requirements are considered to be low probability, this scenario provides a bounding assessment of supply adequacy. The figure indicates that the current planned and prospective uranium production capacity … is sufficient to meet the high requirements through the year 2040. However, in order to do so, potential new supply must be brought into production at a high rate. Exploration and development would be needed to find additional properties beyond the prospective mines … in order to meet post-2040 demand.»

6.5.2 Conversion

Historical and Current Capacities

Conversion plants are industrial chemical plants in which U_3O_8 is converted into UF_4 in a first process step and into UF_6 in a second. Since such plants only process ^{nat}U, the nuclear criticality aspects are less onerous than in enrichment plants, the main concern being the relatively large amounts of fluorine and HF present. It is observed however, that the conversion process can be unpredictable, such that nameplate capacities are often not achieved or at least not sustainably.

Today, five countries operate conversion plants, see Table 6.2. Although conversion is an indispensable step in the nuclear fuel supply chain, it has been somewhat neglected in the past. Referring to Figure 6.11

Figure 6.11 , between 1995 and 2018, the price for conversion lay mostly between 5 - 10 USD/kg, which was not a sustainable situation for the producers. As a consequence, plants were severely throttled back or put into cold shutdown. This explains why, despite having a global nominal capacity of 62'000 t, only 31'600 t of uranium were converted in 2020. As clearly seen in Figure 6.11 the throttling back of production has had a significant effect on spot prices since 2018 which today are at around 40 USD/kg UF_6, as well as on the more relevant long-term price indicator (which has doubled since 2020). As a result, the Converdyn plant is currently being prepared for a resumption of operations, the Orano plant is ramping up its output and Westinghouse are planning to refurbish and restart their existing plant in Springfields, UK.

Table 6.2 Conversion plants worldwide and their availability in 2020 [6.3]

Converter	Country	Location	Nameplate Capacity (tU)	Capacity utilization (%)	Capacity utilization (tU)
Cameco	Canada	Port Hope	12’500	72%	9’000
CNNC*	China	Lanzhou & Hengyang	15’000	53%	8’000
ConverDyn**	USA	Metropolis	7’000	0%	0
Orano***	France	Pierrelatte, Malvési	15’000	17%	2’600
Rosatom	Russia	Seversk	12’500	96%	12’500
Total			62’000	51%	31’600

* Assumption that China will develop its conversion capacity sufficient for domestic needs

** In January 2021 a restart plan was announced, targeting to resume production in 2023

*** Orano’s new conversion facility is still in the process of production ramp-up, which is expected to be finalised by 2023

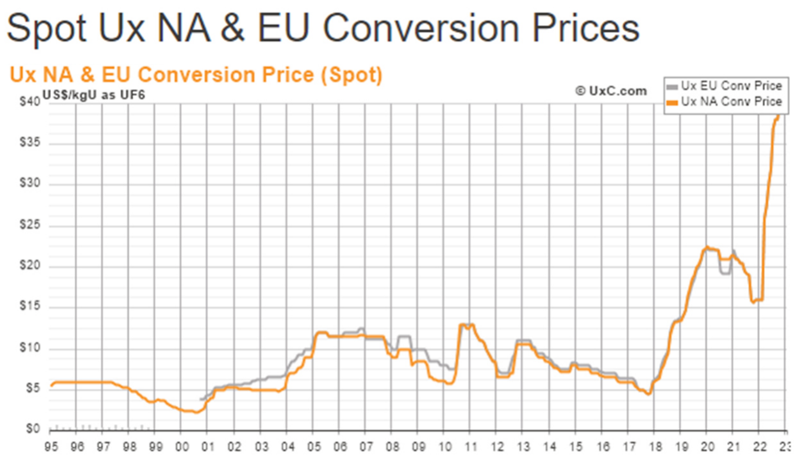


Figure 6.11 Historical spot conversion prices in Europe and North America (Source UxC, LLC, http://www.uxc.com/)

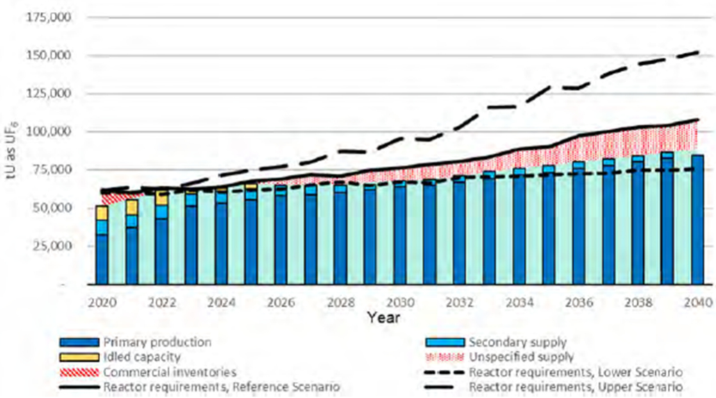


Figure 6.12 Supply and demand curves for conversion until 2040 according to WNA [6.3]

An alternative view of the future conversion market is provided in *Figure 6.13* .

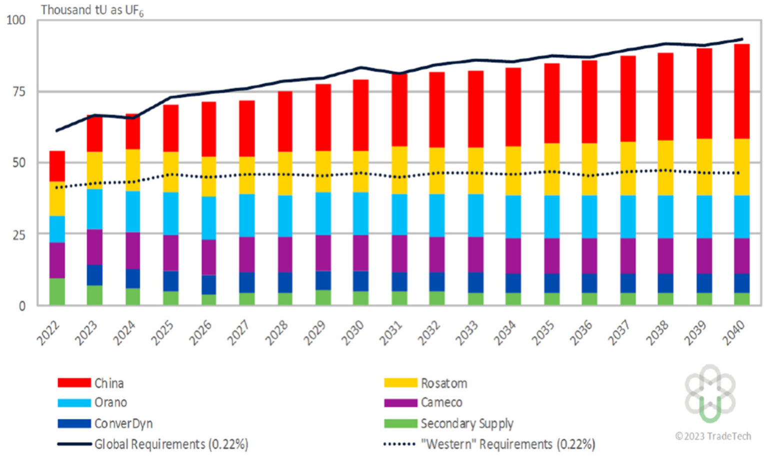


Figure 6.13 **Supply and demand curves for conversion services (WNFM 2023)**

Future Conversion Capacities

Figure 6.12 shows the Supply and Demand situation according to the WNA [6.3] until 2040. The supply curve assumes that all existing facilities can be operated at 90% of their capacity from 2026 onwards. Whereas the WNA lower scenario is comfortably covered, for reference and high scenarios, between 20'000 and 60'000 t of conversion capacity need to be brought on-line by 2040. Some of this demand can be provided by secondary supply and further capacity utilization and optimization, but nevertheless some new capacity will be required.

This perspective, based on a market view by TradeTech[20], and presented at the *49th Meeting of the World Nuclear Fuel Market*[21] by Clark Beyer of Global Fuel Solutions, shows a more balanced situation, with demand lying between the WNA low and reference scenarios. Conversion plants are not particularly complex constructions and additional capacity could be added if required. Orano's Comhurex II Conversion Project[22] for instance, having a nameplate capacity of 15'000 t, was launched in 2007 and was commissioned in 2018. According to planning, full production will be achieved in 2023. Another example is the recently announced refurbishment of the plant at Springfields, UK[23], which plans to be operational in 2028.

In summary, if a renaissance of nuclear energy leads to a significant increase in the demand for nuclear fuel, as predicted in the WNA high scenario, conversion capacity can be built on a timescale comparable to the construction of new nuclear plants. In the meantime, and as discussed in Section 6.6, any gaps will be filled by secondary supply. WNA describes the situation as follows:

[20] https://www.uranium.info/

[21] https://www.wnfm.com/

[22] Orano commissions new conversion facility : Uranium & Fuel - World Nuclear News (world-nuclear-news.org)

[23] https://world-nuclear-news.org/Articles/Westinghouse-targets-new-UK-based-uranium-conversi

«Overall, the change in the conversion market … is characterised by a significant curtailment in primary production levels resulting in a heavy reliance on and corresponding reduction in inventories. After these inventories are exhausted over the near to medium term, the market will incentivize increased production output via increased capacity utilization factors, the expansion of existing conversion plants, or even the construction of a new conversion facility».

6.5.3 Enrichment

Current and Future Capacities

Table 6.3 shows the current and future predicted global enrichment capacities according to [6.3]. As centrifuge facilities do not lend themselves to load following, installed capacity is usually operated at 100% load factor irrespective of demand. Instead, when demand for enrichment is low, enrichers either underfeed (i.e., use less uranium than the customer has provided whilst reducing the tails assay) or produce more EUP than required and store this for later use. Therefore, in contrast to conversion, the concept of capacity utilization for enrichment is not applicable.

Table 6.3 Current and predicted future global enrichment capacities until 2030 ('000 SWU)

Operator	2020	2025	2030
CNNC	6'300	11'000	17'000
Orano	7'500	7'500	7'500
Rosatom	27'700	26'200	24'800
Urenco	18'300	17300	16'300
Other (INB, JNFL)	66	375	525
Total	**59'866**	**62'375**	**66'125**

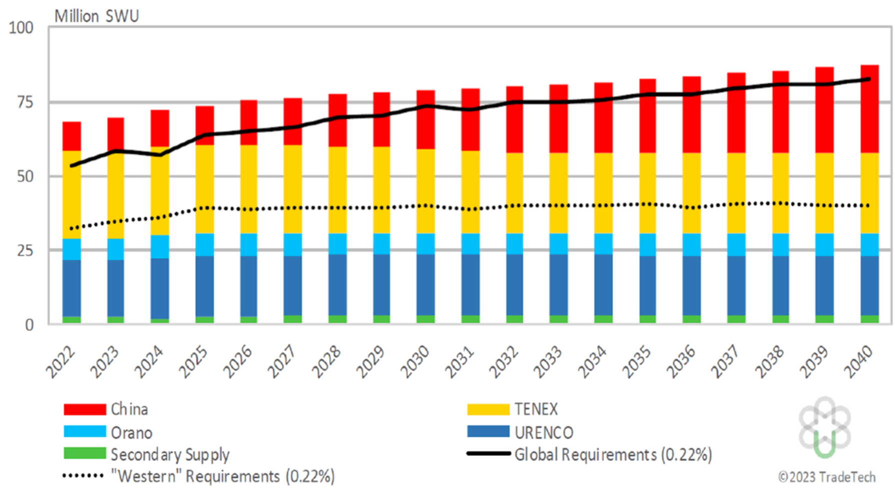


Figure 6.14 **Supply and Demand Curves for Enrichment (WNFM 2023)**

A comparison of Table 6.3 with Figure 6.5 indicates that the WNA reference case requirements can be covered by the predicted available enrichment capacities until 2030. Beyond this and for the upper demand scenario, significant amounts of additional capacity will be required.

An independent view is provided in Figure 6.14 This perspective, based on a market view by TradeTech[24] , and presented at the 49th Meeting of the World Nuclear Fuel Market[25] by Clark Beyer of Global Fuel Solutions, assumes a more balanced view of enrichment supply and demand, such that the reference demand is comfortably covered until 2040.

Also ERI [6.5] are confident that their reference demand case can be covered by existing capacities out to 2025, see Figure 6.15.

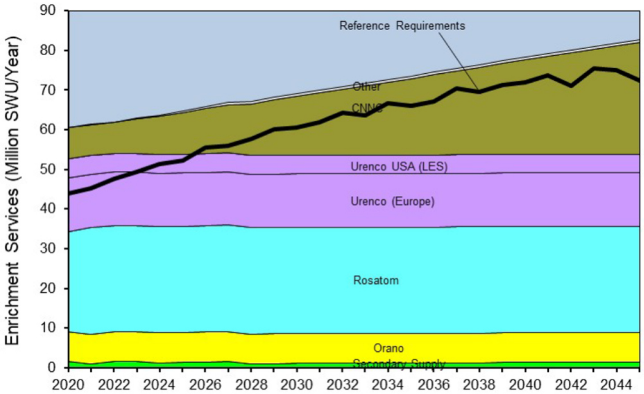


Figure 6.15 Enrichment supply adequacy assuming no retirement losses

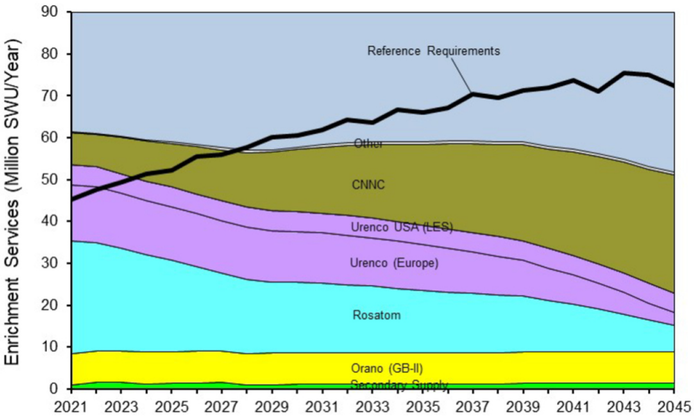


Figure 6.16 Enrichment supply adequacy assuming retirement losses

However, due to the high technological demands placed on them, centrifuges are subject to breakdown rates in the range of a few percent per year. Since individual centrifuges within a cascade cannot be realistically replaced of repaired, a minimum level of annual replacement is required just to maintain nominal capacities. Assuming a nominal lifetime of 35 years, *Figure 6.16* shows the adjusted picture in which an additional 20 Mio. SWU are required by 2045.

For the ERI high demand case, an additional 30 - 40 Mio. SWU capacity will be required by 2040, see Figure 6.17.

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[24] https://www.uranium.info/

[25] https://www.wnfm.com/

It is worth mentioning that, although Orano and Urenco are two fully independent western enrichers, both companies depend exclusively on a single entity ETC[26] for their supply of centrifuges. This clearly represents a single-source supply risk, although the fact that ETC is jointly owned by Urenco and Orano does help to mitigate this risk.

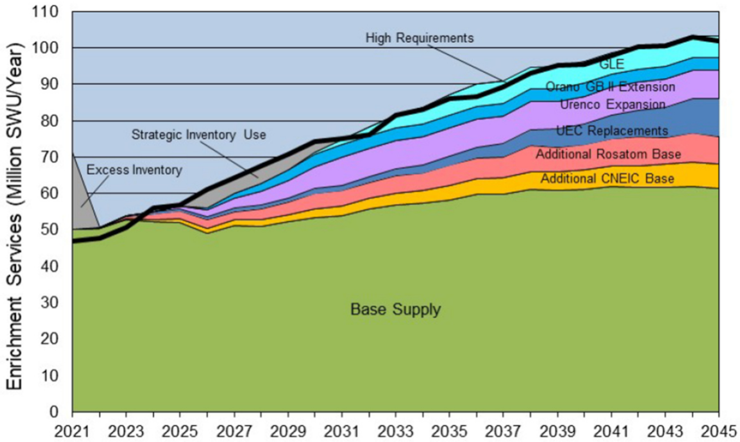

Figure 6.17 Additional SWU requirements for the ERI high demand case.

6.5.4 Fabrication

Current and Future LWR Capacities

As mentioned in section 6.4.4, fuel assembly fabrication is dependent on reactor type and the market is strongly segmented. Although many different fuel types exist, PWR and BWR fuels are clearly the most relevant for the western market.

Regarding current fabrication capacities, different values exist for the three main fabrication steps, namely conversion[27], pelletizing and assembly. Referring to *Table 6.4* , currently installed global LWR fabrication capacities amount to 13’517, 14’313 and 15'326 tHM for conversion, pelletizing and assembly respectively [6.5], which is to be compared with a current demand of around 6’000 tHM (see section 6.4.4). It should be mentioned however, that the capacities noted are licensed volumes which are rarely, if ever, achieved. Furthermore, the demand figure of 6'000 tHM does not include the fabrication of new cores, which currently amounts to an additional 1'000 tHM. Nevertheless, even taking this into account, there is still a significant oversupply of global LWR fabrication capacity available at the current time. WNA (WNA, 2022) is of the opinion that the current overcapacity is not sustainable, thus leading to more consolidation in the market in the near- to mid-term.

Regarding future capacities, Figure 6.18 shows the predicted development of global LWR fabrication supply[28] and demand until 2045, according to ERI [6.5]. Even for the high demand case, sufficient capacity is available.

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[26] https://enritec.com/

[27] In this case conversion refers to the conversion from $^{enr}UF_6$ to $^{enr}UO_2$

[28] Based upon $^{enr}UF_6$ to $^{enr}UO_2$ conversion capacity, the most limiting step in the fabrication process

Table 6.4 Summary of all LWR Fuel Fabricator Production facilities and Capacities

Country	Fuel Fabricator	Location of Fuel Fabrication Facility	Type of Fuel (LWR Only)	Production Capacity, MTU per Year		
				Conversion	Pellet	Assembly
East Asia						
Japan	Global Nuclear Fuel - Japan	Yokosuka	BWR	0	620	630
	Mitsubishi Nuclear Fuel	Tokai Mura	PWR	450	440	440
	Nuclear Fuel Industries	Kumatori	PWR	0	383	284
	Nuclear Fuel Industries	Tokai Mura	BWR	0	250	250
Korea	KEPCO Nuclear Fuel Company	Taejon	PWR	700	700	550
China	CNNC-China South Nuclear Fuel Co.	Yibin, Sichuan	PWR	800	800	800
	CNNC-China Nuclear Northern	Baotou, Inner Mongolia	PWR	200	200	200
	CNNC-Baotou Nuclear Fuel Co. (AP1000)	Baotou, Inner Mongolia	PWR	400	400	400
			Asia Subtotal	**2,550**	**3,793**	**3,554**
Western Europe						
France	Framatome	Romans	PWR	1,800	1,400	1,400
Germany	Framatome	Lingen	BWR / PWR	800	650	650
Spain	ENUSA	Juzbado	BWR / PWR	0	500	500
Sweden	Westinghouse Atom	Vasteras	BWR / PWR	787	600	600
U.K.	Westinghouse (long term lease)	Springfields (a)	PWR	950	600	860
			Europe Subtotal	**4,337**	**3,750**	**4,010**
U.S.						
U.S.	Framatome	Richland, WA	BWR / PWR	1,200	1,200	1,200
U.S.	Global Nuclear Fuel - Americas	Wilmington, NC	BWR	1,200	1,000	1,000
U.S.	Westinghouse Electric Company	Columbia, SC	PWR/BWR	1,600	1,594	2,154
			U.S. Subtotal	**4,000**	**3,794**	**4,354**
Total Installed LWR Fabrication Capacity for East Asia, W. Europe and U.S.				**10,887**	**11,337**	**11,918**
Other						
Brazil	Industrias Nucleares do Brasil	Resende	PWR	160	120	400
India	Nuclear Fuel Complex (DAE)	Hyderabad	BWR	48	48	48
Russia	TVEL, JSC Machine Building Plant	Elecktrostal (c)	PWR	1,500	1,500	1,560
	TVEL, Novosibirsk Chem. Conc. Plant	Novosibirsk	PWR	450	1,200	1,200
Kazakhstan	Ulba Metallurgical Plant	Ust-Kamenogorsk (d)	PWR	472	108	0
	Ulba-FA LLP			0	0	200
			Other Subtotal	**2,630**	**2,976**	**3,408**
Total Installed LWR Fuel Fabrication Capacity for World				**13,517**	**14,313**	**15,326**

(a) Includes an estimated 200 MTU for fabrication of AGR fuel.
(b) TVEL capacity as JSC MBP includes production of up to 220 MTU of fuel for RBMK reactors.
(c) Kazatomprom comissioned a 200 MTU fuel assembly fabrication facility in cooperatio with CGNPC. Production began in 2021 with deliveries to Chinese reactors in early 2022.

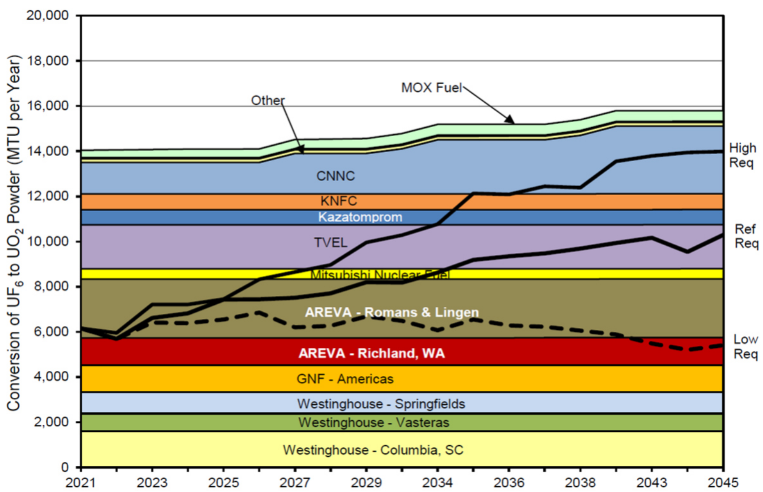

Figure 6.18 World LWR Fuel Fabrication Capacity Adequacy [6.5]

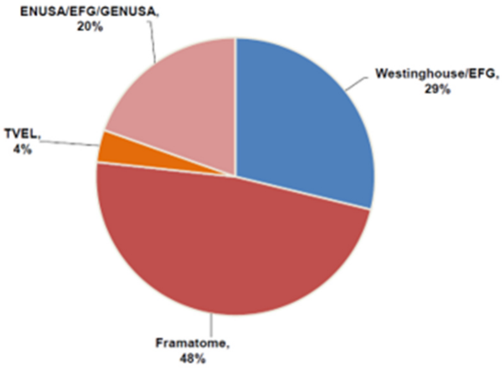

Figure 6.19 Current Distribution of the Western European Fabrication Market

Figure 6.19 on the other hand shows the current market shares for the western LWR fabrication market, also taken from Ref. [6.5]. Immediately apparent is the very modest market share which TVEL holds (supply to a VVER in Finland and the fabrication under contract with Framatome, e.g., for KKB). The large market share of Framatome is due to the significant demand for PWR in France.

Table 6.5 Summary of Alternative Fuel Fabrication Capacities

Country	Fuel Fabricator	Location of Fuel Fabrication Facility	Type of Fuel	Production Capacity Assembly
Western Europe				
U.K.	Westinghouse	Springfields	GCR	200
			Europe Subtotal	**200**
Other				
China	CNNC	Baotou, Inner Mongolia	HTR	300,000 pebbles
India	Advanced Fuel Fabrication Facility (DAE)	Tarapur	LMR	Prototype
Japan	Nuclear Fuel Industries	Tokai	HTR	0.40
Russia	TVEL, JSC Machine Building Plant (MSZ)	Elecktrostal	LMR (BN-600)	50
	TVEL, JSC Machine Building Plant (MSZ)	Elecktrostal	GMR (RBMK)	460
	TVEL, Mining & Chemical Complex (MCC)	Zheleznogorsk	MOX (LMR)	60
U.S.	BWX Technologies	Lynchburg, VA	TRISO	Engineering Scale
	X-energy	Oak Ridge National Laboratory	TRISO	Pilot Scale
			Other Subtotal	**570**
Total Installed Other Fuel Fabrication Capacity for World				**770**

Typically, when a reactor is newly built, the reactor vendor will also supply "*original equipment*" fuel to that reactor for a number of years. With time however, it is common that the operator diversifies his supply chain and changes his fuel supplier on a semi-regular basis. Due to regulatory requirements, achieving diversity in fuel fabrication is a time consuming and costly endeavour and is an important strategic decision to be taken by the operator.

In Switzerland, a variety of strategies have been adopted. Whereas KKM remained true to the reactor vendor throughout its operational lifetime, other plants have tried to diversify their fuel fabrication supply. For KKL, for instance, several fuel types are licensed:

- Westinghouse: *Optima2, Optima 3* (Fabrication in Sweden, USA, Spain)
- Framatome: *Atrium 10XM* (Fabrication in Germany, USA)

For KKB there is unfortunately only one fuel design available worldwide namely *Agora* from Framatome. However, this fuel type can be fabricated in Germany, USA and Russia and KKB along with KKG is currently qualifying the Romans facility in France.

Whether this level of diversity can be maintained in the future depends on whether the consolidation postulated by WNA actually takes place.

Alternative Fuel Types

Whereas the majority of this section has been devoted to LWR fuel, there are of course reactor technologies in operation and in planning which require other types of fuel. Some of these technologies are addressed in 6.10. Suffice it say that fabrication capacity for these alternative fuel types already exists on a limited scale, see for instance Table 6.5.

6.6 Secondary Supply

It has already been mentioned that so-called secondary sources have played and continue to play a significant role in the various nuclear fuel supply chains. Thanks to these secondary sources, the supply-demand imbalances, especially in the area of uranium and conversion, have been fully compensated, thus avoiding disruptions in the supply chain. *Figure 6.20* shows the significant role that secondary supply has played for natural uranium in the past.

Secondary supply does not however only relate to uranium. The use of stockpiled $^{nat}UF_6$ represents a source of secondary conversion and EUP can be regarded as stored enrichment. The most relevant sources of secondary supply and an estimate of some of their respective quantities are given below, mainly from (3):

Secondary Source	**Estimated Quantity**
Commercial inventories	> 280'000 tU (globally, excluding Russia)
Government stockpiles	~ 100'000 tU (USA)
Down-blending weapons-grade uranium	152’000 tU / 133 Mio. SWU[29]
Re-enrichment of tails	1.2 Mio. tU (eq. to 200'000 t ^{nat}U) : assuming feed assay of 0.25% and tails assay of 0.15%
Underfeeding by enrichers	unknown
Reprocessing and recycling	unknown
- Reprocessed Uranium	
- MOX (Plutonium)	

WNA has attempted to quantify the availability of secondary sources of uranium, conversion an enrichment and assign these to high and low scenarios. Figure 6.21 , Figure 6.22 and Figure 6.23 show their estimates of the annual contributions to uranium, conversion and enrichment of secondary supply.

[29] Only taking into account 500 t of highly enriched uranium (HEU) from Russia, no account is taken of US down-blending.

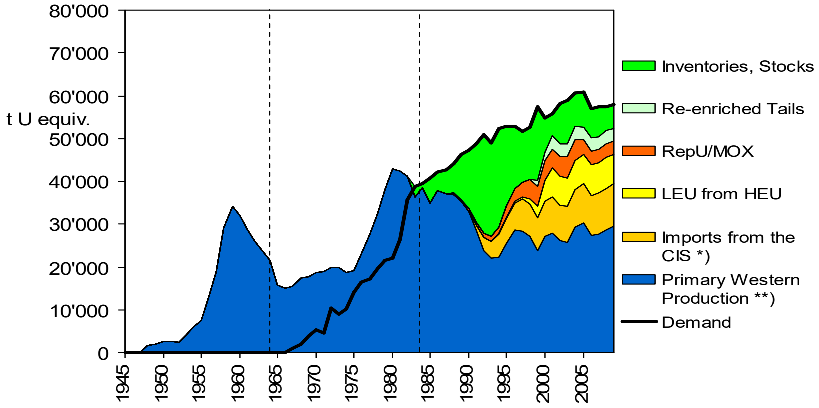

Figure 6.20 Supply and demand curves showing the role that secondary supply has played in the past (source unknown, but probably WNA)

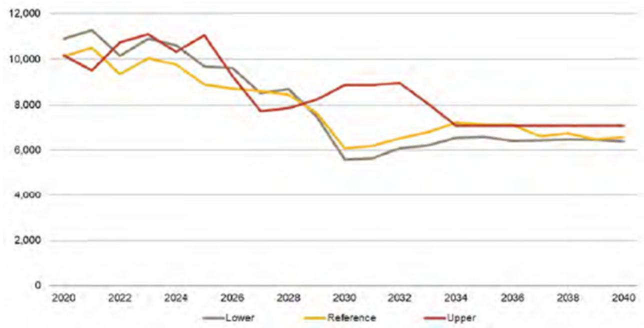

Figure 6.21 Estimates of secondary uranium supply [6.3]

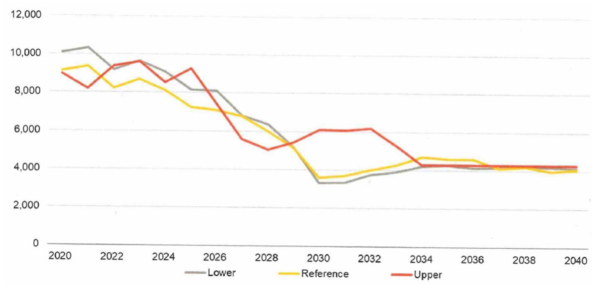

Figure 6.22 Estimates of secondary conversion supply [6.3]

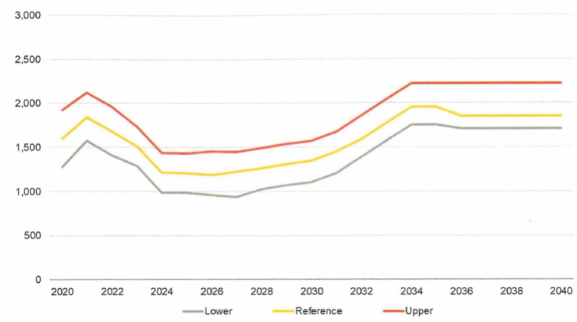

Figure 6.23 Estimates of secondary enrichment supply [6.3]

The findings are summarised below in which it can be seen that secondary supplies can contribute more than 10% of annual uranium and conversion requirements until 2040, for enrichment the contribution is significantly less, around 2%.

	Today	2040
Uranium	*10’000 t U*	*6’000 t U*
Conversion	*9’0000 t U*	*4’000 t U*
Enrichment	*1’500-2’000 t SWU*	*1'500 – 2'000 t SWU*

6.7 Transport and Logistics

As mentioned above, each part of the nuclear fuel supply chain is connected by physical transports. Each component requires a different type of transport package, each package needs to be licensed according to international regulations and has finally to be validated by national authorities. In addition, the export and import of nuclear materials needs to be approved by the appropriate authorities.

In 2020 the Euratom Supply Agency identified «*Lack of transport hubs open to nuclear shipments*» [6.6], as the number one risk to the nuclear fuel supply chain. They justify this as follows:

«a number of ports in Europe have taken the decision not to accept shipments of nuclear material any more. This was the case recently in Hamburg, where local political parties agreed on this in the coalition agreement and the port followed this agreement by a voluntary statement. An increasing number of shipping companies are deciding to refuse nuclear fissile materials on their vessels (for instance Grimaldi, Hapag Lloyd, Stena Line)»

Germany, not only in connection with port availability but also concerning the approval of transports from and through Germany, represents a particular risk. Fuel deliveries to Switzerland were already endangered in 2020/1 due to unfounded claims that the transports were illegal.

Since Germany is not only an important source of supply for Swiss plants but also an important transit country for nuclear products from other countries, this represents a significant risk which can only partly be addressed by the industry itself. For example, since Westinghouse currently has to transport fuel to Switzerland via Germany, they are currently trying to qualify an alternative transport route through France.

Nevertheless, since the hinderances are often politically motivated, governmental / diplomatic support may become necessary in the future.

6.8 Nuclear Fuel Markets – The Cost of Nuclear Fuel

The uranium, conversion and enrichment markets involve transactions between:

- Producers or suppliers (uranium miners, converters, enrichers, or fuel fabricators),
- Public and private electrical nuclear utilities or fuel consumers,
- Other uranium market participants that buy and sell uranium (agents, traders, intermediaries).

A fully mature exchange for the trading of these materials, as for other commodities, does not exist and price definitions are therefore not fully transparent. Instead, a small number of independent market observers[30] publish prices on a regular basis, according to their market insights. Despite the significant potential to misuse this system, these prices are often used in term-contract pricing mechanisms.

The demand for nuclear fuel is inelastic, meaning that even significant changes in the price of one or more components do not fundamentally affect the purchaser's behaviour. This is because the fixed (non-fuel) costs of nuclear power generation dominate such that the electricity price is relatively insensitive to the price of nuclear fuel, as demonstrated in the example shown in Table 6.6. Comparing the total fuel costs of 53 Mio. CHF from Table 6.6 with the total annual costs of KKL in 2022[31] of roughly 450 Mio. CHF, we arrive at a fractional cost of around 12% (equivalent to 0.53 Rp/kWh). Even if we artificially double the price of uranium, which is an extreme assumption (we will see in section 6.10 that there are proven reserves of Uranium for the next 100 years even at prices below current prices), this figure only increases to 18%.

Table 6.6 Costs of nuclear p.a., based upon long-term prices and KKL requirements.

	Annual Requirements	**Long Term Prices 26.6.23**[a]	**Costs Mio. CHF**
Uranium	225'000 kg	131 CHF / kg	29.5
Conversion	225'000 kg	26 CHF / kg	5.9
Enrichment	135'000 SWU	131 CHF / SWU	17.7
			53.1

[a] www.uxc.com, USD = 0.9CHF

On the supply side however, the situation is somewhat different. When prices fall, suppliers tend to react either by cutting back production, mothballing plants, reducing investments in capacity replacement, or adjusting their processes (underfeeding in the case of enrichment). When prices rise however, suppliers put reserve capacities into operation, invest more in exploration (in the case of Uranium) and adjust their processes (overfeeding in the case of Enrichment). These actions serve to keep the supply / demand balance roughly in check in the mid-term. The mismatches which nevertheless inevitably arise have in the past been filled by secondary supply and there is no reason to believe that this won't continue in the future. The historical development of uranium, conversion and enrichment prices, not adjusted for the value of money is shown in *Figure 6.24*

[30] In particular UxC (https://www.uxc.com/) and Trade Tech (https://www.uranium.info/)

[31] geschaeftsbericht_2022_V5.indd (kkl.ch)

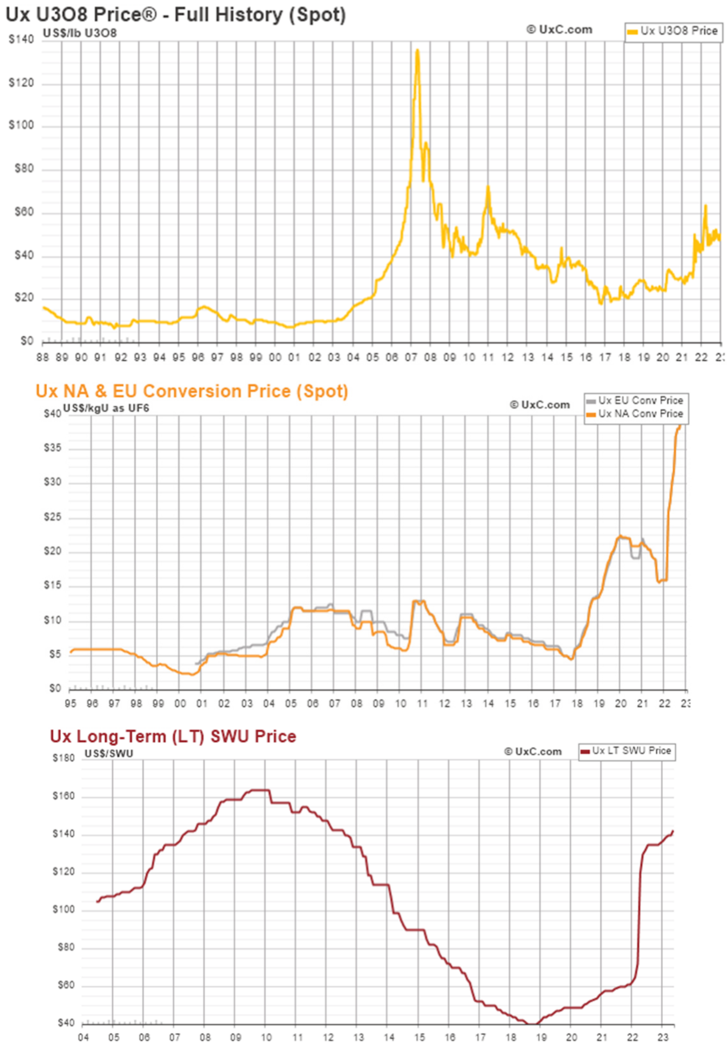

Figure 6.24 Historical Spot prices for Uranium, Conversion and Enrichmen (Source UxC, LLC, http://www.uxc.com/)

6.9 Geopolitical Trends and Consequences

The analyses presented in sections 6.4 to 6.7 are based upon objective assessments of current and predicted supply and demand scenarios without speculating on potential geopolitical events or trends or on how these could influence the balance.

Given the current political situation, this analysis would not be complete without some words on the consequences of the war on Ukraine on future nuclear fuel supply.

Since the dissolution of the Soviet Union, Russia has played an increasingly important role in global nuclear supply chains. For instance:

– The U.S. and Russia entered into an agreement in 1993, providing for the

U.S. to buy 500 t [32] of highly enriched uranium resulting from the dismantling of former Soviet Union nuclear weapons over a period of 20 years (U.S.-Russia HEU Agreement). The enrichment contained in these deliveries served to offset the demise of the US enrichment industry when the US closed their diffusion plants in Portsmouth and Paducah in 2001 and 2013.

- From 2014 until 2020, under the so-called Amendment to the Russian Suspension Agreement (RSA) US utilities imported the equivalent of 20% of their enrichment requirements. In a further amendment, the agreement was extended to 2040, although the amounts will be reduced to 15% from 2028.
- Many European operators, including Switzerland, have sourced considerable amounts of fuel from Russia, although EU members are party to the Corfu Declaration of 1994, whereby the fraction of Russian enrichment supply is limited to 20% (this does not apply to Switzerland).
- Rosatom has been successful in marketing its VVER 1200 reactor outside Russia. According to the WNA[33] currently 18 VVER plants are under construction in India, China, Turkey, Egypt, and Bangladesh.

Taking this into account, it is hardly surprising that no direct sanctions have yet been imposed on the import of nuclear fuel from Russia; western dependence is still too strong. This does not mean however, that the procurement of nuclear fuel from Russia has remained unaffected. Numerous measures have been imposed, which have introduced significant uncertainties into the Russian supply chain. These include the following[34]:

US: Although no direct trade restrictions are in place, there have over the last year been sustained efforts to limit trade with Russia. The proposals under consideration all assume Russia will remain a U.S. supplier for the next few years while providing for a full ban on Russian imports before the end of this decade.

In April 2023, the U.S. government sanctioned five Rosatom subsidiaries.

Financial sanctions have disrupted the ease of usual global transactions in foreign currencies for nearly all parties doing business with Rosatom and its subsidiaries.

Canada: Has introduced a policy that prohibited any person in Canada and any Canadian outside Canada to provide to Russia or any person in Russia any service falling under the category of “*Water transport services – Freight transportation*.”

EU: In early 2023, the European Parliament adopted a resolution urging sanctions against Rosatom and for an immediate and full embargo on EU imports of uranium from Russia along with all fossil fuels. The resolution has predictably drawn objections from Hungary and Bulgaria.

[32] Equivalent to 152’000 tU and 133 Mio. SWU or more than twice global requirements

[33] https://world-nuclear.org/information-library/current-and-future-generation/plans-for-new-reactors-worldwide.aspx

[34] Ux Weekly, 14.8.2023, www.uxc.com

ESA[35] is likely to tighten its position on Russian fuel supplies going forward. After all, the agency is the final signatory on every nuclear fuel contract in the EU market, giving it a strong and effective tool for shaping the region's fuel procurement.

UK: The United Kingdom has instituted multiple packages of internationally coordinated sanctions and trade measures against Russia. In the latest sanctions package, the UK government linked Rosatom with Russia's defence complex, adding senior Rosatom executives to the designated list.

The UK has notably banned all Russian-affiliated vessels from entering its ports, thus disrupting scheduled nuclear fuel deliveries to Westinghouse's Springfields Fuels Ltd. fuel fabrication facility in England. While efforts to overcome the transportation issues have since been largely successful, the new issue that has arisen due to an inadvertent duty (~30%) imposed on the imports of the Russian nuclear material entering the UK. It is understood that the duty has effectively halted all imports of EUP from Russia to Springfields.

These measures, together with the expectation that direct sanctions on nuclear fuel imports from Russia will sooner or later be imposed have provoked many western operators, including Switzerland, to seek alternative suppliers in the short term and to rethink their fuel supply strategy in the long term. Also, the western suppliers are reacting to the situation by utilizing idle capacities and engaging with operators to secure financing for the construction of additional capacity. Currently, and with respect to the individual components we see the future supply situation as follows.

Uranium: As mentioned in section 6.5.1 the Russian share of Uranium production is modest and is mainly needed to supply its own fleet. As far as Kazakhstan is concerned, they currently represent a significant share of uranium production capacity which until recently has been dependent on the transports via St Petersburg. In the meantime, an alternative transport route via the Caspian Sea has been established. It does have to be said however, that the position of Kazakhstan with regards to Russia is not very transparent and it may be that Kazakhstan will sooner or later also have to be excluded from western supply chains.

Russia (possibly in collaboration with Kazakhstan) is changing its focus eastwards, predominantly towards China and India. This on the one hand will remove Russian supply from the western markets, but at the same time reduce eastern demand from China and India.

All the analysts agree that the mid-term expansion of existing and development of new mines in the west in necessary, but also feasible, and is already beginning to happen. This will almost certainly be accompanied by higher prices in the short to mid term. Whereas supply risks in enrichment have already translated into higher SWU prices, the full effect has not yet been felt for uranium, which, as already discussed, is a substitute for enrichment.

Conversion: In contrast to uranium, the Russian (and Chinese) conversion capacities are significant. ERI (ERI, 2022) have analyzed a situation in which the considerable capacities in Russia and China are no longer available to the West.

[35] Euratom Supply Agency

Figure 6.25 shows the results for the ERI reference case in which, despite the additional restrictions, western supply and western demand remain roughly in balance, out to 2045. It cannot be excluded that additional capacity will need to be built but this does not seem to present a fundamental problem.

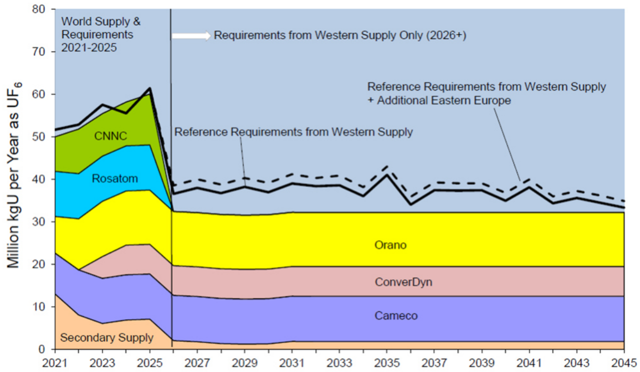

Figure 6.25 Projected western conversion supply adequacy, 85% capacity factor [6.5]

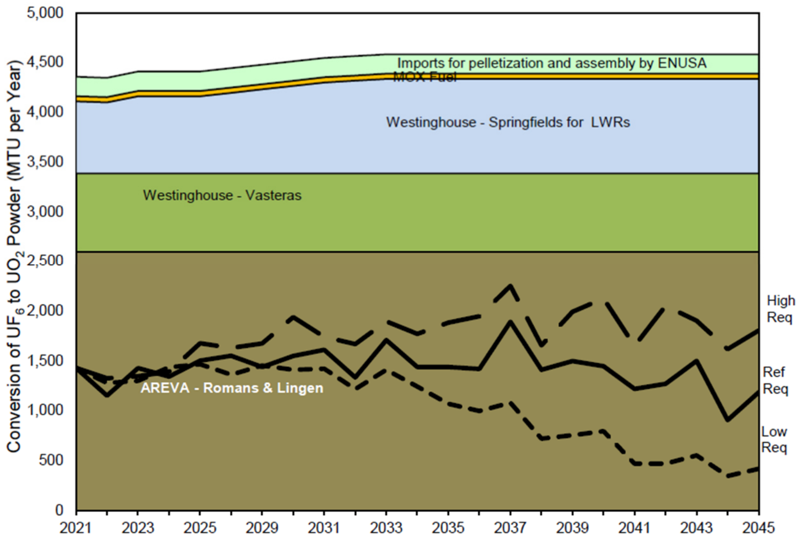

Figure 6.26 Adequacy of Western European LWR Fuel Fabrication Capacity

Enrichment: Also here Russia holds a significant fraction of the global enrichment capacity, nevertheless, ERI [6.5] predicts that their reference case demand of around 75 Mio. SWU in 2040 can be fulfilled with a modest 10 Mio. SWU of additional capacity, even assuming that enrichment deliveries from Russia to the US and Western Europe will be curtailed from 2026 onwards. For the ERI high demand case, an additional 30 - 40 Mio. SWU capacity will be required by 2040, see Figure 6.17 . Again, this does not represent a fundamental problem, if customers are available and in need of enrichment services, these services can be built up and provided. Indeed, the full effect on enrichment prices likely has also not yet been registered in SWU prices, as the extent and duration of Russia's separation and the industry reactions to it are not yet fully known.

Fabrication. This probably the least onerous issue in connection with Russia. Basically, Russia only produces fuel for their own VVER reactors in Russia and for several ex-Soviet countries. There are some exceptions, in which Rosatom fabricate western fuel under license (e.g. KKB) but this represents an insignificant fraction of the market.

Most ex-Soviet countries are now turning to the west for fuel supply, but until now, only Westinghouse has been able to provide VVER fuel for these reactors. Framatome however are also working on licensing a VVER product. This business has little or no impact on western supply, except perhaps for short term resource issues at the fabricators. Figure 6.26 compares western European demand and capacity and clearly indicates a **significant over-capacity out to 2045**. Figure 6.26 [6.5] clearly shows that there is now and in the future sufficient fabrication capacity in the west to satisfy western demand.

6.10 Fuel Supply Beyond 2040

The focus of this report has been the supply of nuclear fuel up until 2040. However, the reactors being built today and, in the future, will require nuclear fuel on a timescale much longer than this and Swiss plants could also still be operating post 2040. Whereas conversion and enrichment are industrial processes whose capacities can, if necessary, be increased, uranium is a naturally occurring raw material and by definition a finite resource. In this section we will therefore focus on the various conventional and unconventional sources of uranium and its alternatives.

6.10.1 Conventional Uranium Supply

Table 6.7 , extracted from [6.1], shows the NEA/OECD estimates of uranium resources for the categories «*reasonably assured*» and «*inferred*» and for various cost categories, see also Figure 6.9.

Table 6.7 Identified resources (recoverable) as of 1 January 2021, t U

Resource category	2019	2021	Change[a]	% change
Identified (total)				
<USD 260/kgU	8 070 400	7 917 500	-152 900	-1.9
<USD 130/kgU	6 147 800	6 078 500	-69 300	-1.1
<USD 80/kgU	2 007 600	1 990 800	-16 800	-0.8
<USD 40/kgU[b]	1 080 500	775 900	-304 600	-28.2
RAR				
<USD 260/kgU	4 723 700	4 688 300	-35 400	-0.7
<USD 130/kgU	3 791 700	3 814 500	22 800	0.6
<USD 80/kgU	1 243 900	1 211 300	-32 600	-2.6
<USD 40/kgU[b]	744 500	457 200	-287 300	-38.6
Inferred resources				
<USD 260/kgU	3 346 400	3 229 200	-117 200	-3.5
<USD 130/kgU	2 355 700	2 263 900	-91 800	-3.9
<USD 80/kgU	763 600	779 600	16 000	2.1
<USD 40/kgU[b]	335 900	318 700	-17 200	-5.1

At a price level of 130 USD/kgU, which is well below todays level of 190USD/kgU, there exist over 6 Mio. t of recoverable Uranium, which would satisfy current demand for roughly 100 years.

6.10.2 Unconventional Sources of Uranium

Apart from the conventional extraction of Uranium from ore bodies in dedicated mines, there are several other significant so-called unconventional sources, the two most important of which are:

Uranium as a by-product of Phosphate mining

According to the WNA[36], the main unconventional source of uranium is rock phosphate, or phosphorite, and some 20,000 t of Uranium has already been recovered as a by-product of agricultural phosphate production to the 1990s, until this became uneconomic.

Phosphorite contains 18-40% P_2O_5, as well as some uranium, often 70 to 200 ppm, and sometimes up to 800 ppm (the average concentration of Uranium in the earth's crust is 3 ppm).

Estimates of the amount of uranium available from this source range from 9 to 22 million t, although the recent discovery of a significant deposit in Norway[37] which practically doubles the proven world resources will certainly increase this estimate.

Uranium from seawater

The world's oceans have long been regarded as a possible source of uranium because of the

large amount of contained uranium (over 4 billion tU) and its practically inexhaustible nature. However, because seawater contains such low concentrations of uranium (3-4 parts per billion), developing a cost-effective method of extraction remains a challenge [6.1]. There have been some recent developments[38] in China however which promise to retrieve Uranium from seawater more efficiently.

Even if only 1% of the available resources could be extracted from the oceans, this would represent some 40 Mio. t of uranium or more than 600 years of supply at current usage rates.

6.10.3 Alternative Sources of Nuclear Fuel

As mentioned in Section 6.2, the majority of NPPs today utilize nuclear fuel which is based upon the fission of U-235, and this will be the case for decades to come. This is however not the only source of nuclear energy; there are alternatives, some of which require the use of different reactor technologies. Of particular relevance are the following:

Breeding of fissile Pu-239/241 from fertile U-238:

Referring to Figure 6.27, roughly 3 Mio. t of Uranium has been mined since the discovery of nuclear fission. Although most of the U-235 from this uranium has already been used in NPPs, the U-238, either in the form of depleted Uranium following enrichment (so called tails) or residing in spent fuel following discharge, is still available.

Taking only the tails material into account, this represents significantly more than 2 Mio. t of U-238, which is currently stored, mainly at enrichment plants. If this material would be utilized in fast breeder reactors to generate Plutonium *in situ* as a fissionable material, this would represent many hundreds of years of additional fuel supply, assuming today's requirements.

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[36] https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/uranium-resources/uranium-from-phosphates.aspx

[37] https://www.euractiv.com/section/energy-environment/news/great-news-eu-hails-discovery-of-massive-phosphate-rock-deposit-in-norway/

[38] https://www.world-nuclear-news.org/Articles/CNNC-launches-test-platform-to-extract-uranium-fro

Reprocessing spent fuel would provide additional significant amounts both of U-238, but also Plutonium which could be used for use in breeder reactors or the manufacture of mixed U-Pu (MOX) fuel for light water reactors.

The use of MOX and Reprocessed Uranium in Switzerland

Between 1970 and the early 2000's the Swiss NPPs transported spent fuel to the reprocessing plants in La Hague (F) and Sellafield (UK). In total, some 1'200 tons of fuel (around 4'000 fuel assemblies) were reprocessed, the ensuing waste was transported back and stored in ZWILAG, and the plutonium and uranium were used to produce new MOX and Reprocessed Uranium (RepU) fuel assemblies.

At the outset, in the 1960's, it was taken for granted that spent fuel should be reprocessed. It was assumed that fast breeder reactors would quickly take over from light water reactors and the plutonium needed to "start" such reactors was in short supply. At that time, plutonium had a significant market value, which provided an added incentive to reprocess. In 1978 however, when the then president of the US, Jimmy Carter, banned the reprocessing of spent nuclear fuel and cancelled the US fast reactor programme, the plutonium economy collapsed. In the meantime, many of the NPPS built during this period had been equipped with relatively small fuel ponds, causing them to be dependent upon the reprocessing option, not for economic but rather for spent fuel management reasons. In addition, because plutonium no longer had a market value, these utilities were obliged to manufacture MOX and RepU fuel for use in their own reactors.

Alongside the fact that reprocessing clearly possesses advantages with respect to resource use, there are further aspects, some positive, some negative, which need to be addressed when adopting this option.

+ The plutonium and uranium arisings reduce the requirements for natural uranium. Typically, for every 8 spent fuel assemblies one fresh RepU and one fresh MOX assembly can be manufactured, thereby reducing the requirements for natural uranium by roughly 25%.
+ The waste returned to the customer represents a fraction of the volume of the initial spent fuel and is already conditioned for deep geological disposal.
+ The radiotoxicity of the high-level waste returned is significantly shorter lived than is that of spent nuclear fuel.
- The manufacture and transport of fresh MOX fuel is more complex and expensive than normal uranium fuel.
- Spent MOX fuel generates significantly more decay heat and a larger neutron dose than does uranium fuel with the same burn-up. This can lead to challenges and additional costs in the storage and handling of spent MOX fuel.
- The "useability" of the plutonium and uranium arising decreases with increasing burnup the spent fuel delivered, since the number and amounts of "unwanted" isotopes.
- Most reactors must be specially licensed to use MOX fuel, which behaves slightly differently during operation.

In summary, reprocessing and recycling were successfully carried out in Switzerland for many years and could be adopted again if the need/opportunity arose. The costs were generally higher than those associated with direct geological disposal, but this situation can reverse in a high-cost nuclear fuel scenario. As mentioned above, fourth generation reactors, such as molten salt, lead cooled, thorium, will in any case require some form of reprocessing as an integral part of their fuel cycle.

Currently however, the reprocessing of spent fuel is forbidden in Switzerland, and the remaining spent nuclear fuel will instead be directly disposed of in a deep geological repository.

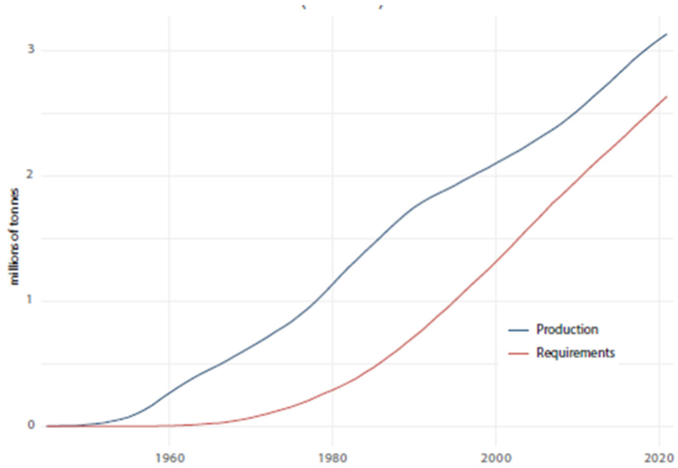

Figure 6.27 Cumulative Uranium Production from 1949 until today [6.1]

The use of Thorium:

Thorium-232 is a fertile isotope analogous to U-238, which occurs naturally and is more abundant in the earth's crust than is uranium. Thorium is used to breed U-233 as U-238 is used to breed Pu-239. The use of thorium also requires some adaptation of todays technology. However, if this would be implemented, it would provide hundreds to thousands of years of supply, at the current requirements.

6.10.4 Summary

In summary, the exploitation of alternative sources of uranium such as phosphates and sea water will significantly stretch global uranium supply in the long term.

Moving to advanced reactors and to the reprocessing and recycling of spent fuel would further extend the long-term potential of uranium fuel from hundreds to potentially thousands of years. This would require however the commercialisation of reactor concepts such as sodium or lead cooled fast reactors or molten salt reactors.

And finally, if alternative fuel cycles were developed and successfully deployed, thorium could also be a potential contributor to the nuclear fuel cycle provided existing initial fissile inventories to start such thorium fuel cycles are readily available.

Although a serious quantitative estimate of the actual potential is difficult to make, we can confidently say that the various future options will be able to provide for thousands of years of energy production, irrespective of how the use of nuclear power develops.

6.11 Summary and Conclusions

In this chapter we have tried, based on the most up-to-date, reliable references available to us, to provide a quantification of the long-term supply perspectives for nuclear fuel. Clearly, any long-term projections extending as far into the future as 2040 are subject to considerable uncertainties, and the nuclear industry is no exception in this respect. With nuclear capacity forecasts for 2040 ranging from 393 GWe to 839 GWe, caution must be exercised in interpreting any conclusions arising therefrom.

However, as the authors regard it as very unlikely that today's global nuclear capacity will be more than doubled in the next 17 years, all feasible future scenarios are covered here.

The information provided in sections 6.4, 6.5 and 6.6 can be summarised as follows. In Figure 6.28, in which the current global supply and demand situation is presented, a strong dependence on secondary supply is observed in the uranium and conversion sectors. Although sufficient primary capacity would be available in these sectors, production has been throttled back due to sustained depressed prices. Recently, as a consequence of rising prices, these capacities are gradually being brought back on-line. In the enrichment sector on the other hand, there is currently sufficient installed capacity to meet demand.

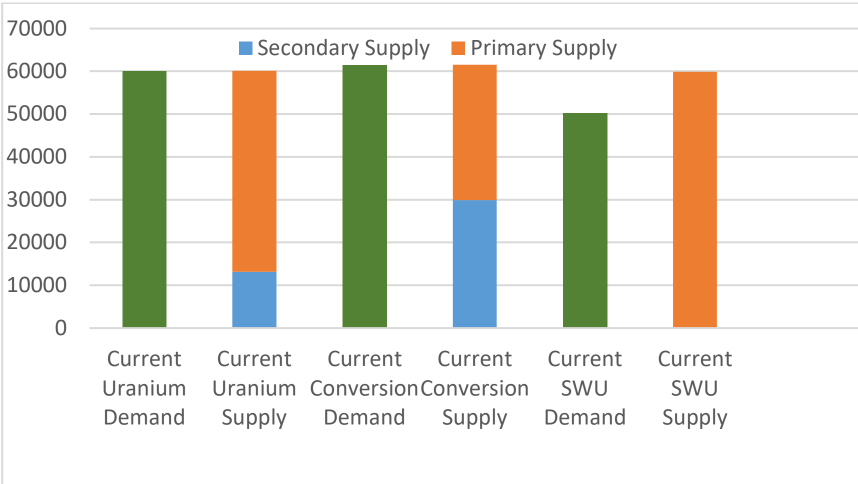

Figure 6.28 Current Supply and Demand Situation for Uranium, Conversion and Enrichment

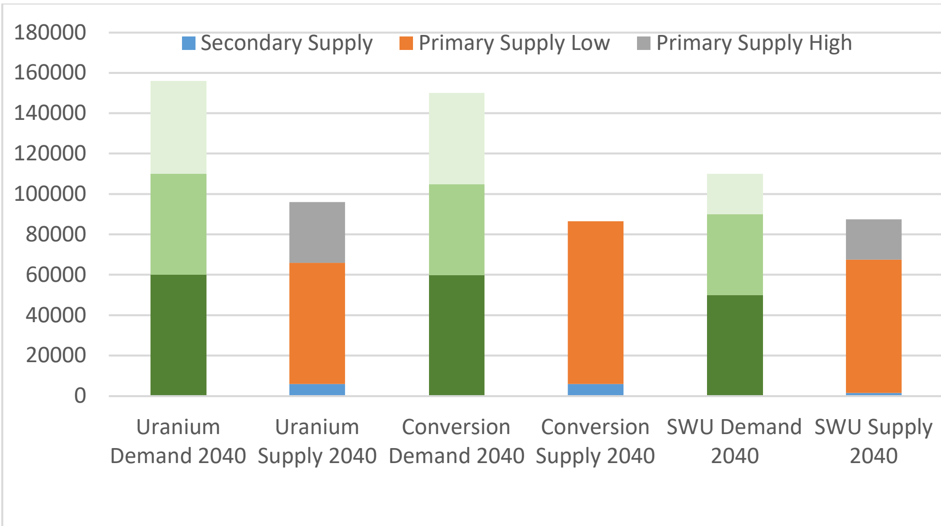

Figure 6.29 Projected Supply and Demand Situation for Uranium, Conversion and Enrichment in 2040

Looking now at Figure 6.29 in which the supply and demand situation for various scenarios in 2040 is presented, it is seen that, very broadly, the projected high supply scenarios can satisfy a significant amount of the reference demand, without assuming any significant amounts of secondary supply. In other words, it seems that a reference demand scenario could just be satisfied with existing planning. The high demand scenarios however would require a significant expansion of capacity, probably involving the construction of new plants.

The WNA [6.3] puts it like this:

«*There is no doubt that sufficient uranium resources exist to meet future needs; however, the producers are waiting for the market to rebalance in order to start reinvesting in new capacity and bringing idled and shutdown projects back to production. Additional conversion capacity is also likely to be needed, while enrichment and fuel fabrication capacities appear to be sufficient to cope with demand*».

At this point, it is once again important to point out that the WNA study [6.3], is significantly more optimistic regarding the expansion of the nuclear fleet until 2040. Correspondingly, the WNA high demand scenarios are significantly higher than in other analyses. This is a known phenomenon resulting from a certain bias arising from the fact that the WNA report is compiled predominantly by suppliers and that the WNA is ostensibly a pro nuclear lobby organisation. We are by no means suggesting that the results have been unfairly biased to benefit the suppliers, only that the future requirements proposed by the WNA probably lie at the upper bounds of plausibility. In other words, the authors of this report do not expect the WNA upper scenarios to materialise. For this reason, we are confident that future requirements can be comfortably accommodated by the available and planned infrastructure.

With this in mind, and taking into account that, even in the event that KKL, KKB and KKG would all continue operating into the 2050s, the Swiss requirements for nuclear fuel will represent significantly less than 1% of the global requirements, we see no long-term risks for the security of supply of nuclear fuel to Switzerland.

Where we do see some risks, are the short-term upheavals to the market caused by the Russian invasion of Ukraine and the associated disruption and potential disruptions to the nuclear supply chain. What we are observing is a separation into two markets, east and west. The western markets are rapidly reducing their dependence on Russian enrichment by means of significant efforts by the enrichers to increase tails assays, to make use of redundant capacity and to expand centrifuge cascades to licensed limits. Russia on its part is moving its focus eastwards to China and India. Ironically, the disruptions to the nuclear supply chain by the war in Ukraine, will in the next 5-10 years serve to strengthen and enlarge western supply capabilities and resilience.

Apart from this, hinderance of nuclear logistics by antinuclear groups remains a very real risk. Although operators can take some measures to mitigate these risks e.g. setting up alternative transport routes not involving Germany and qualifying production outside Germany, there remain some risks which can only be addressed on a diplomatic level.

Looking further ahead into the second half of this century, we believe that an increased need for nuclear power will lead to increased exploration activities and therefore to increased uranium reserves. In addition, on this timescale, reactor technology will develop to such an extent that other fuels with much greater energy potential than U-235 can be used, thus extending the availability of nuclear fuel from hundreds to many thousands of years.

6.12 References

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[6.4] OECD/NEA. (2022). Nuclear Energy data 2021. OECD.

[6.5] ERI. (2022). Nuclear fuel Cycle Supply and Price Report.

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7 Alternative fuel cycles

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7.1 General background for alternative fuel cycles

In the first part of this chapter the fuel availability was revived and many terms and relevant data have been discussed. In the second part of this chapter, alternative fuel cycle would be discussed. For the needs of this part, some terms will be re-introduced from different perspective. As it was already mentioned, nuclear fuel cycle relies on actinides as a natural resource. It has so called front end, where all preparatory steps belong: exploration, mining, milling, conversion, enrichment and fabrication. The middle part of the fuel cycle is the actual fuel irradiation and energy production in a nuclear reactor. The back end of the fuel cycle is represented by interim storage, transportation, reprocessing, partition, transmutation and/or waste disposal. The foremost objective of the nuclear fuel cycle is energy production in several forms (electricity, district heating, water desalination, hydrogen production, etc.). The major side products of this production are Fission Products – FPs and synthetic actinides. Especially the amount of synthetic actinides in the waste stream determines the long term stewardship burden related to the final spent fuel repository. On the highest level, the measure for fuel cycle performance evaluation can be defined as maximal natural resources utilization and minimal waste production. Nonetheless, it has also safety and economic implications.

7.1.1 Characterization of natural resources

The three actinides nuclides present in the nature: ^{235}U, ^{238}U and ^{232}Th are called primordial. They are unstable and decay through the decay chains. Accordingly, already the front end of the nuclear fuel cycle represent a source of radiotoxicity, because the respective radionuclides from the decay chains are extracted and separated from the host rock, which was isolating them from the biosphere. The half-lives for radioactive decay are: 0.7 bn. years for ^{235}U, 4.5 bn. years for ^{238}U, and 14 bn. years for ^{232}Th. The estimated reserves are somewhat related to these half-lives. Accordingly, there is 140 times more ^{238}U than ^{235}U and 3-4 times more ^{232}Th than ^{238}U.

Primordial actinides have been created presumably during the supernova explosion by rapid neutron capture process and by the consecutive decays. As such, they represent a forepast energy conserve. For the discussion of alternative fuel cycle options it is advantageous to imagine their nucleus as a droplet, pressurized by strong repulsive force of more than 90 protons, which is kept together by the surface tension. Fission reaction represent braking of this fragile forces balance and the actual energy release originates from proton repulsion between the two FPs. ^{235}U has high probability of fission for all possible energies (velocities) of the interacting neutron. Hence, it can be fissioned by any neutron present in the reactor. Actinides nuclides with such characteristic are called fissile. In contrary, only neutrons with high energy can cause fission of ^{232}Th and ^{238}U. Their fission is thus a threshold reaction and below certain energy level, the interaction with neutron results in its capture. In general, neutron capture causes nuclear transmutation (change of number of protons or neutrons in nucleus). Since the final products of ^{232}Th and ^{238}U transmutations (the ^{233}U and ^{239}Pu nuclides) are fissile, this capture reaction is called breeding and ^{232}Th and ^{238}U are called fertile. The conserved energy can be released from any actinide nuclide. The only difference between fissile and fertile nuclides is the number of neutrons necessary for the release. In case of ^{232}Th or ^{238}U roughly two neutrons are needed, one for the transmutation and second for the actual fission; in case of ^{235}U only one neutron is necessary, solely for the fission. Hence, it is more demanding to utilize ^{232}Th and ^{238}U resources for which the reserves are higher.

7.1.2 ^{235}U as a part of natural uranium

^{235}U is the only primordial nuclide capable to self-sustain fission chain reaction. At the same time, in occurs in the nature as natural uranium, a mixture of ^{234}U, ^{235}U and ^{238}U nuclides. ^{234}U is a decay product and member of the ^{238}U decay chain and its share in natural uranium is thus very small. The share of ^{235}U is approximately 0.72%. Hence, more than 99% of natural uranium consist of ^{238}U, a nuclide which predominantly captures neutrons. To self-sustain fission chain reaction with natural uranium is demanding and can be achieved only in thermal (moderated) reactors with excellent neutron economy; for instance in graphite or heavy water moderated systems. The fission of ^{235}U produces in average 2.5 neutrons. One of these neutrons is necessary for maintaining the chain reaction. However, in case of natural uranium fueled reactor, the ^{238}U isotope could in average capture one additional neutron. As a result, only about 0.5 neutrons would be left to cover other neutron losses caused by neutron leakage or capture on structural materials and graphite or heavy water.

Since it is much more convenient to use light (normal) water as a moderating coolant, the share of ^{238}U is typically reduced so that higher neutron losses caused by the hydrogen parasitic captures can be allowed. Combination of enriched uranium and light water as a moderating coolant is used in vast majority of current power reactors, producing more than 90% of the energy from fission. Heavy water moderated reactors produce around 6% of the energy. There are several Russian RBMK type reactors still in operation producing less than 1.5% of the energy. This Chernobyl-like reactors are moderated by graphite and cooled by light water. Overall, 98% of the energy from fission is produced in water moderated and/or cooled reactors [7.1].

The uranium enrichment process is costly and demanding; however, it not only allows operation with light water, but also increases the amount of energy which can be produced from one fuel assembly. For moderated thermal reactors the typical uranium enrichment is up to 5%. In unmoderated, so called fast reactors, the enrichment could be more than 10%. In the past, uranium was enriched to levels above 90% of ^{235}U for military purposes and for experimental reactors. Nowadays, the enrichment process is limited to 20% from non-proliferation reasons. During the enrichment process also the concentration of lighter ^{234}U nuclide is increased. Based on internationally accepted product specification mentioned in [7.2] the concentration of ^{234}U in enriched uranium should be kept below 1% of the ^{235}U concentration.

7.1.3 Fuel burnup

In all power reactors relying on enriched uranium, the ^{235}U and ^{238}U nuclides are inevitably irradiated simultaneously. The measure of fuel irradiation is expressed either as an energy produced per mass of loaded actinides (MWd/kg or GWd/t), or as a share of already fissioned atoms in percent (FIssile MAterial – FIMA %). Since there is some parallel between fission and fire, both being kind of chain reaction, the consumption of actinides by fission is called burning and the measure of irradiation as burnup. The maximal achievable burnup is limited by many factors; nonetheless, the major one is the reduction of fission probability due to the changing actinides composition and the increase of neutron parasitic capture caused by FPs accumulation. Fuel relying on 5% enriched uranium typically reaches burnup of 5% FIMA (50 GWd/t). Hence, 95% of actinides are still in the discharged fuel.

7.1.4 Synthetic actinides

The interaction of all three primordial actinide nuclides ^{235}U, ^{238}U and ^{232}Th with a neutron results either in fission, which releases the energy and originates two FPs, or in a transmutation, which is generally a source of synthetic actinide nuclides. Here it should be noted that fission is also a transmutation; however, the term is not used for it.

To illustrate the evolution of fuel composition the irradiation of 5% enriched uranium in LWR was simulated for the purposes of this report (see Figure 7.1) [7.3]. The maximal presented burnup of 10 % FIMA is not realistic. Yet, it pro provide certain insight for the evolution trends. The detailed composition of the irradiated fuel is presented in the right part of the Figure 7.1 for the realistic burnup of 5 % FIMA.

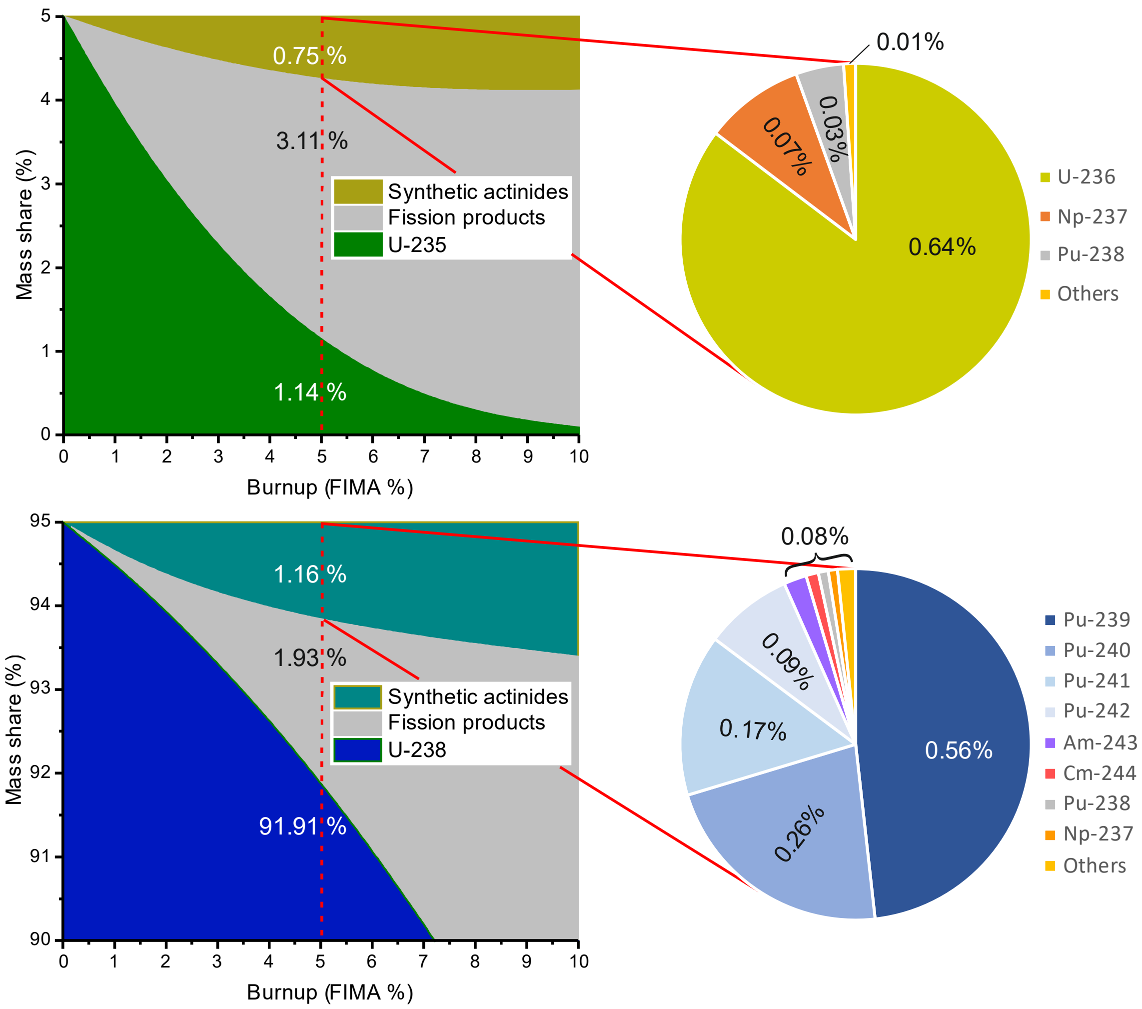

Figure 7.1 Evolution of primordial and synthetic actinides during 5% enriched uranium irradiation in light water reactor. The evolution (left) and detailed composition (right) at 5% FIMA burnup of ^{235}U (top) and ^{238}U (bottom) is separated to underline the origin of each synthetic nuclide [7.3].

From Figure 7.1 it can be seen that from the originally loaded 5% of ^{235}U, 1.14% is still present in the discharged fuel, 3.11% were fissioned and 0.75% transmuted into the synthetic actinides. Synthetic actinides originated from ^{235}U are dominated by ^{236}U (0.64%). It is a synthetic actinide with one of the longest half-life for decay (see Figure 7.2). There is also small amount of ^{237}Np and ^{238}Pu. None of these synthetic actinides is fissile. Furthermore, since their utilization for energy production may require more than 3 neutrons, they are not even fertile. Hence, they represent a burden from both neutronics and waste stream perspective. Moreover, synthetic isotopes ^{236}U and ^{234}U are mixed with primordial uranium isotopes. Thus, the irradiated uranium, which represents more than 93% of the discharged fuel mass, has higher radiotoxicity than the natural uranium.

Irradiation of ^{238}U results in its direct fission. Nonetheless, it is a threshold reaction, which represents not more than 10% of all fissions. From originally loaded 95% of ^{238}U approximately 0.5% are directly fissioned, whereas roughly 2.6% are transmuted into synthetic actinides by the more dominating neutron capture reaction. Major component of these synthetic actinides are plutonium isotopes ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. Since ^{239}Pu and ^{241}Pu are fissile, more than the half of the generated synthetic actinides are fissioned. The ^{238}U thus contributes with its direct and indirect fission by 1.9 % and ^{235}U by 3.1 % to the 5% FPs generation.

All synthetic actinides have shorter half-life for radioactive decay than primordial actinides (see Figure 7.2). Highly probably they have been also created by the supernova explosion; however, due to their faster decay they are not anymore present in the nature. The only exception is probably ^{244}Pu with 0.08 bn. years half-life, which can be found in trace amounts. From geological perspective, the transmutation of primordial actinides results in shortening of the half-lives and accelerates the disappearance of actinides from the earth. From human time perspective, transmutation leads to an increase of radiotoxicity. Hence, synthetic actinides as a side-product of the energy generation are unwanted component of the waste stream. They dramatically prolongs the long-term stewardship burden related to the final repository. Simultaneously, synthetic actinides ^{233}U and ^{239}Pu can be considered as an asset and act as a catalyzer or actually intermediate product of fertile primordial actinides utilization.

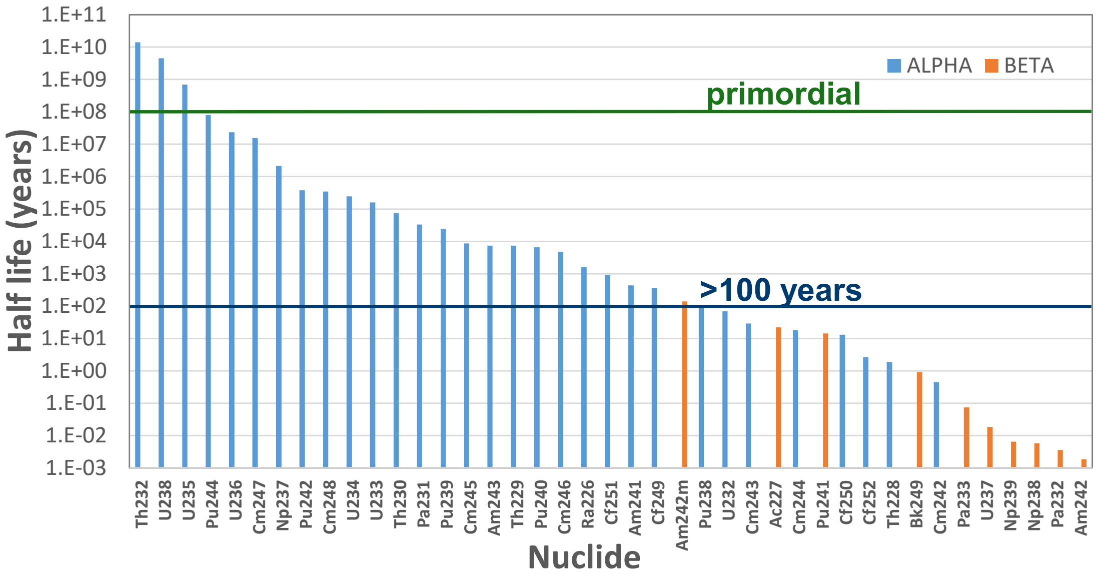

Figure 7.2 Half-life and type (alpha or beta) of radioactive decay for primordial and synthetic actinides [7.2].

7.1.5 Induced radiotoxicity

Synthetic actinides, but also FPs, have shorter half-live for radioactive decay than primordial actinides. The irradiation of enriched uranium thus increases the radiotoxicity. To illustrate this increase, radiotoxicity of irradiated LWR fuel is shown in Figure 7.3. The data are normalized to the initial radiotoxicity of 5% enriched uranium. Here it should be stressed that such uranium includes not only 5% of ^{235}U, but also almost 0.5% of ^{234}U. This uranium isotope is member of

^{238}U decay chain and it has by 4 orders of magnitude shorter half-live than ^{238}U. Consequently, in higher amounts ^{234}U is a source of large radiotoxicity. Its presence in enriched uranium increases the radiotoxicity of the fresh fuel almost by one order. Furthermore, as Figure 7.3 shows, when stored without any irradiation (0% FIMA) the radiotoxicity of fresh fuel would due to ^{234}U presence increase and culminate around 300'000 years being up to 30 times higher than the initial value. The share of ^{234}U in enriched uranium is over-proportional when compared with the natural or depleted uranium. It is transmuted in the reactor and source of additional ^{235}U. This transmutation can be considered as a radiotoxicity reduction. This is not a commonly presented fact and it helps to balance in longer perspective the amount of overall induced radiotoxicity.

From radiotoxicity perspective, there are two major preparatory steps in the front end of the fuel cycle. After the mining of uranium ore, natural uranium in a form of yellowcake is separated from the decay chain nuclides. These nuclides are more radiotoxic than the actual natural uranium. In the next step, natural uranium is enriched. The enrichment process produces depleted uranium as a side product. To reflect this process Figure 7.3 also include relative ingestion radiotoxicity of uranium ore and of natural uranium, which was utilized for 5% enriched fuel preparation.

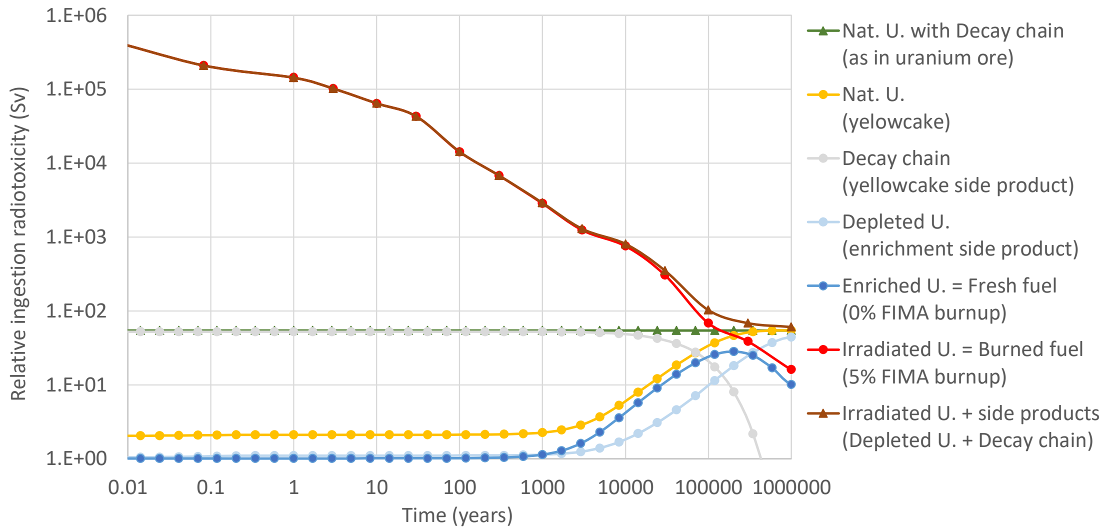

Figure 7.3 Overview of ingestion radiotoxicity related to uranium fuel production and utilization. Data relates to 5% enriched uranium and are normalized to its initial radiotoxicity before irradiation.

There are two major ways how to evaluate or actually define the man-made radiotoxicity as a side product of energy production:

1) The first option is to compare the fresh and irradiated fuel radiotoxicities (dark blue and red curves in Figure 7.3). This method relies on radiotoxicity induction in the power plant during irradiation, where assemblies loaded and unloaded from the reactor are compared. It can serve as a definition of the radiotoxicity induced by irradiation.
2) The second option is a broader comparison of the original radiotoxicity of the utilized natural resources (green curve in Figure 7.3) with the remaining radiotoxicity after energy production. The remaining radiotoxicity has 3 components: a) the irradiated fuel itself (red curve in Figure 7.3), b) the separated decay chain nuclides as a side product of the yellowcake preparation (grey curve in Figure 7.3) and c) the depleted uranium as a side product of the enrichment process (light blue curve in Figure 7.3). This option represents a global radiotoxicity balance.

For both these methods, relative induced radiotoxicity can be defined as a ratio between the two respective radiotoxicities. Using the first definition, it can be seen from Figure 7.4 that immediately after the end of irradiation the radiotoxicity is increased 1’000’000 times. After 100 years it is still 10’000 times higher. However, after 30’000 years it is only 30 times higher. Around 100’000 years the induced radiotoxicity is only by factor of 3 higher than radiotoxicity of fresh fuel (0% FIMA) when stored for the same time. The second definition provides even lower values, because the radiotoxicity of natural resources provides higher comparative basis. Hence, immediately after the end of irradiation the radiotoxicity is increased 10’000 times and after 100 year 300 times. In 10’000 years the ratio drops to 15. No matter the definition, the induced radiotoxicity ratio is below 3 after 100’000 year. After that time point, it stays elevated for additional several hundred years; nonetheless, it is of the same order as the original radiotoxicity of the fresh fuel in one case and of the natural resources in the other case.

On one hand, the induced radiotoxicity practically disappears after circa 100’000 years. On the other hand, the original resource in form of uranium ore has been stabilized in the host rock and isolated from the biosphere, before we have extracted it. We are thus responsible to create equally good and long lasting protection similar to the original host rock, to isolate all related waste streams from the biosphere. Even if the best fuel breeding or waste burning reactor and closed fuel cycle with fuel recycling would be applied, synthetic actinides will be always present in the waste stream. The final spent fuel repository is thus unavoidable. Only its size may differ according of the actinides recycling policy.

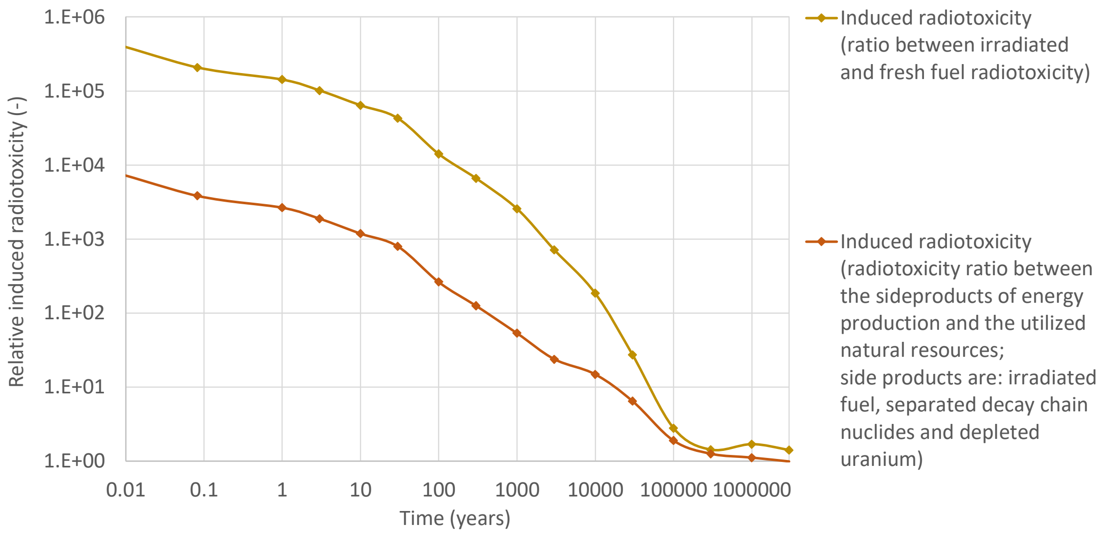

Figure 7.4 Relative induced radiotoxicity as a function of time for 5% enriched uranium after 5% FIMA burnup.

7.2 Alternative fuel cycles

Since the enriched uranium burning in LWR is world-wide used as a major nuclear technology for energy production, it serves as the reference fuel cycle. In this sub-chapter the alternative fuel cycle will be discussed. Firstly, the list of available fuels will be introduced; secondly, their combination in different fuel and fuel cycle types will be evaluated. Since the reprocessing is an inevitable part of practically all alterative cycles, it is also shortly discussed here.

7.2.1 List of possible fuels

The fuel for fission chain reaction consists always from actinides. There are several fissile and fertile actinides and actinides groups listed hereafter, which can serve as a fuel or its part. These are either primordial or synthetic.

Natural uranium

Natural uranium could be utilized only in thermal reactors with excellent neutron economy (graphite or heavy water moderated). Its reserves were discussed in the first part of this chapter and are sufficient for several centuries. The major reactor type for natural uranium utilization is Canadian CANDU reactor. It relies on heavy water as moderator and light water as coolant. In the past, there were also prototypes of heavy water moderated gas cooled reactors (e.g. Lucens in Switzerland or KS-150 in Czechoslovakia).

Low Enriched Uranium (LEU)

Enriched uranium with up to 5% of ^{235}U can be labelled as Low Enriched Uranium (LEU). It is produced from natural uranium by the enrichment process. LEU is the most common fuel for LWR, with well established supply chain and marked, which was extensively described in the first part of this chapter. During the 5% enriched LEU production, natural uranium is separated into 90% of depleted uranium and 10% LEU. Accordingly, the reserves of 5% LEU corresponds to one tenth of the natural uranium reserves.

High Assay Low Enriched Uranium (HALEU)

The uranium enrichment is limited by the non-proliferation agreement to 20%. The uranium enriched in the range from 5% to 20% of ^{235}U is called High Assay Low Enriched Uranium (HALEU). The marked and so the supply chain for HALEU was limited in the past it is not well developed. Nonetheless, practically all non-water cooled reactors being operated, constructed or designed nowadays rely on HALEU; with exception of graphite-moderated gas-cooled MAGNOX reactors and some MSR concepts, which can be fueled by LEU. Especially in USA, where fuel reprocessing is not allowed, HALEU is the only available fuel for advanced systems. There are ongoing efforts to assure HALEU availability sponsored by DOE [7.4].

Highly Enriched Uranium (HEU)

Highly Enriched Uranium (HEU) is enriched to more than 20% of ^{235}U. The enrichment could reach more than 90%. HEU was produced as a weapon material and as a fuel for small typically experimental or military reactors. It is relevant to mention here that in the past it was also mixed with thorium and considered as an initial fuel for thorium breeders. Based on the non-proliferation treaty, HEU is not any more allowed material outside of the nuclear powers.

Depleted Uranium

The side product of uranium enrichment is called depleted uranium. It contents only 0.2%-0.25% ^{235}U. It cannot be utilized as a fissile fuel in a reactor. However, it can serve as fertile fuel, where the energy stored in the ^{238}U nucleus can be released by means of two neutrons.

Thorium

The reserves of thorium are estimated to be 3-4 times higher than for natural uranium. It is probably related to its longer half-live for radioactive decay. The major and practically only isotope in the nature is ^{232}Th. However, in the nature there exist also ^{230}Th as a member of ^{238}U decay chain and ^{228}Th as a member of ^{232}Th decay chain. The half-live of ^{228}Th is less than 2 years. Accordingly, it decays fast and thorium separated from its decay chain will include practically only ^{232}Th. Similarly as depleted uranium, thorium cannot be utilized in a reactor as a fissile fuel. However, it can serve as fertile fuel, where the energy stored in the ^{232}Th nucleus can be released by means of two neutrons.

Minor actinides (MAs)

During LEU irradiation in LWR or HALEU irradiation on other reactors so called minor actinides are generated. This sub-group of synthetic actinides includes mainly Np, Am and Cm isotopes (see Figure 7.1). As all other actinides, they can be utilized for energy production; however, the number of neutrons needed for energy release strongly differs between them. In the most common PUREX reprocessing method they stay all together with FPs in the waste stream. Since their share is low, they are often vitrified together with FPs and so prepared for final disposal.

Reprocessed uranium (RepU) from LEU irradiation

The irradiated LEU consists by more than 93% from uranium. It includes rest of ^{235}U (~1%), its transmutation product ^{236}U (~0.6%) and the bulk mass of ^{238}U. There are also other uranium isotopes; however, with much lower concentrations. It can be separated by the PUREX process from plutonium and from the mixture of FPs and MA. RepU can be optionally re-enriched and reused in LWR. RepU re-enriching and recycling in LWR is one of the simplest option how to reduce the actinides bulk amount in the waste stream.

Depleted Uranium from RepU re-enrichment

The side product of RepU re-enrichment is similar to depleted uranium. However, since RepU has higher share of ^{234}U and ^{236}U isotopes, their concentration will be higher also in the produced depleted uranium.

Plutonium from ^{238}U irradiation

The irradiation on ^{238}U results in plutonium creation. The isotopic composition of the plutonium depends on the reactor type and burnup level. Should the plutonium be separated from irradiated fuel with low burnup, its predominant isotope would be ^{239}Pu. In such case, we speak about weapon grade Pu. During the usual irradiation of LEU in LWR the reactor grade Pu is created with higher share of ^{240}Pu and other higher Pu isotopes.

Uranium from ^{232}Th irradiation

The irradiation on ^{232}Th results in uranium creation. The isotopic composition of the uranium strongly differs from the natural uranium and the major isotope is ^{233}U. The isotopic composition depends on the reactor type and burnup. As for plutonium, uranium separated from irradiated fuel with low burnup, consists predominantly from ^{233}U. There is however one big difference between ^{232}Th and ^{238}U irradiation. The intermediate product of ^{232}Th transmutation into ^{233}U is ^{233}Pa, which has quite long half-life for radioactive decay (27 days). It is sufficient time so that it can be chemically separated to obtain practically pure ^{233}U. This may be seen as fuel cycle advantage. At the same time, it is concern from proliferation perspective. Longer irradiation of thorium leads to higher uranium isotopes creation: ^{234}U, ^{235}U, and ^{236}U. The stockpile of ^{233}U in the world is negligible and mainly in USA.

7.2.2 Natural uranium burning in heavy water moderated reactors

The obvious advantage of natural uranium fuel is the absence of enrichment process. This fuel cycle option was preferred be countries with uranium reserves and related to national security, because the entire front end of the fuel cycle was covered by domestic means. At the same time, the neutron economy of this fuel cycle is tight and the low ^{235}U concentration does not allow for higher burnups. Accordingly, the fuel with slight enrichment 0.9% - 1.2% can provide better performance and reduction of fuel cycle costs by 20% to 30% [7.5].

7.2.3 LEU burning in water moderated reactors in once through cycle

The reference fuel cycle utilized today is enriched uranium burning in water moderated and/or cooled reactor. It is an open fuel cycle, where the enriched uranium is only once irradiated in the reactor. The resulting irradiated fuel is declared as a waste and stored in temporary or final repositories. In this fuel cycle the natural resources utilization is low. Less than 1% of the originally mined uranium is used for energy production. The utilization can be increased by higher burnup or actually higher energy production per fuel assembly. However, higher burnup may require higher uranium enrichment, which will result in higher amount of depleted uranium in the waste stream. The overall impact on utilization is thus neutral. In any case, higher burnup results in lower spent fuel assembly count or actually lower waste mass per unit of produced energy. The irradiated fuel from LWR is declared as a waste on in some countries; in others it is understood as a resource of Pu and RepU and foreseen for recycling.

Recently there are several companies proposing LWR technology in SMR format (see Chapters 3 and 4). The modularity and core down-sizing have implication predominantly on economy and safety of the reactor. The fuel cycle performance is influenced by higher neutron leakage from the smaller core and by the possible absence of boron burnable absorber. Nonetheless, it does not have substantial impact on the fuel cycle performance. The resources utilization is still small and fuel burnup comparable or even slightly deteriorated when compared to large LWR with the same LEU fuel enrichment.

7.2.4 HALEU burning in non-water moderated reactors in once through cycle

There exists many innovative reactor concepts in different stage of development. Based on Figure 7.5 [7.6] 80 % of these concepts are moderated, from which only one third utilizes water. The non-water moderated reactors relies on graphite. There are two major core layouts: 1) fuel embedded in graphite as a Triso particles, 2) fuel dissolved in molten salt. In the first case, the prismatic graphite-based fuel can be cooled by gas or dedicated molten salt coolant. In the second case, fuel is dissolved in a liquid and circulates through the reactor in the channels formed from graphite. The two respective groups of reactors are HTRs and MSRs.

In case of HTRs, the higher uranium enrichment is necessary to compensate the graphite moderating properties. For instance, the Chinese HTR-PM reactor relies on HALEU with circa 8% enriched uranium. The natural resources utilization in HTRs is similar to LWRs. However, the fuel specific density is much lower and the HTR spent fuel is thus much more volumetric. Furthermore, the supply chain and market is not yet well established for graphite based Triso particles fuel. The technology is known and world-wide; however, there are only few companies having or establishing the capability.

In case on MSRs, HALEU may be used it some cases, but it is usually not necessary. Since the fuel is dissolved in the molten salt, the specific fuel density in MSRs is higher than in HTRs. Hence, they can be operated also with LEU. Moreover, the liquid fuel state opens new options for increasing the burnup by continuous FPs removal, for burning of other fuels than LEU and HALEU and even for closing of the fuel cycle. Since the last two options are demanding from regulatory and technologic perspective, open fuel cycle and LEU or HALEU burning is the first choice option for the near term deployment. It is frequent that the concept designers claim capability to breed in Th-U cycle and simultaneously prepare design for LEU or HALEU burning.

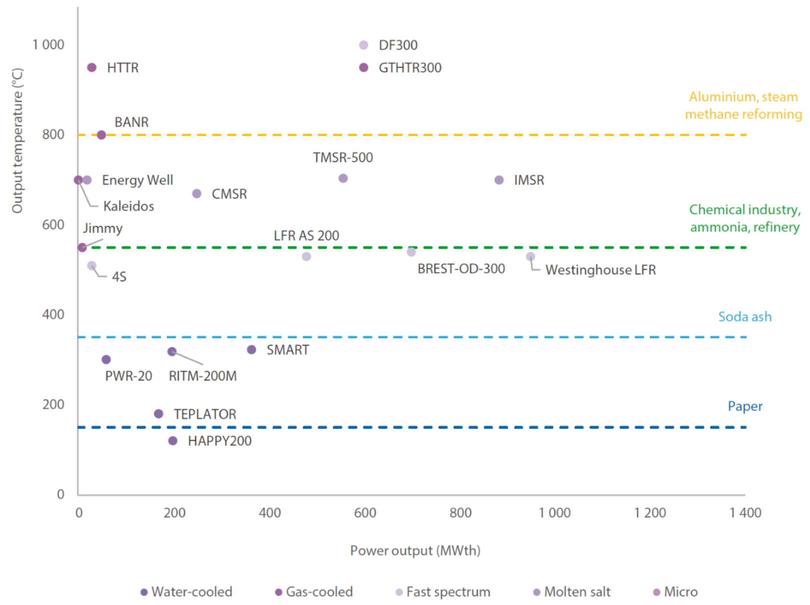

Figure 7.5 Range of sizes and temperatures for heat application [6.6].

7.2.5 HALEU burning with thorium in once through cycle

One additional option how to combine primordial actinides into a fuel is HALEU burning with thorium. In the past, rather HEU was combined with thorium, to suppress the ^{238}U share in the fuel. Assuming that a fast reactor needs circa 10% of the actinides to be fissile and assuming HALEU with 20% ^{235}U, the fresh fuel composition would be 50% thorium, 40% ^{238}U and 10% ^{235}U. Obviously, irradiation of such fuel will generate synthetic actinides from all three primordial actinides, which will be mixed together. Hence, the new fissile material ^{233}U generated from ^{232}Th irradiation will be mixed with the remaining ^{238}U. It is unpractical for possible further use. This issue can be avoided by a seed-blanket configuration, where the fertile thorium pins are separated from the fissile uranium pins. There are studies for LWR showing that the fissile material balance in such a seed-blanket configuration stays constant, what would indicate that there is chance to operated LWR as a thorium breeder. This conclusion is however based only on the first fuel cycle. Should the fuel be irradiated for longer time after repetitive recycling, ^{234}U concentration will grow and deteriorate the neutron economy.

7.2.6 Fuel reprocessing

Neglecting heavy water moderated systems capable to operate with natural uranium and small reactors fueled with HEU, the LEU and HALEU are the only primordial fissile materials, which can be reasonably used as a nuclear fuel. Their burning in once through cycle is at current uranium market price the most economic option. At the same time, any open cycle is waste intensive and provides very low resources utilization. All other fuel cycles require reprocessing. Closed cycle with repetitive fuel irradiation is the major option to increase natural resources utilization and minimize the waste production.

Since the spent LEU fuel is initially strongly radiotoxic (see Figure 7.3) and a source of decay heat, the reprocessing takes place usually several years after the fuel discharge. The cooling period depends also on the reprocessing method. It can be shorter for pyro-reprocessing, which is generally more resistant to radiation, and longer for aqueous processes, which are more sensitive. The reprocessing typically separates actinides from FPs and actinide elements from each other. Sometimes, only one or two actinide elements are separated from the remaining mixture of actinides and fission products. The individual actinide separation in the partitioning process is often limited by the selected method. For instance, the industrially established PUREX process extracts separately uranium and plutonium from the spent fuel with a certain efficiency. Other synthetic actinides and uranium and plutonium residue are left in the so called PUREX raffinate. This raffinate is usually vitrified and foreseen for later final disposal. There exist a modified PUREX method and alternative aqueous processes, which allow also for americium and curium separation. However, on a large industrial scale nowadays only the uranium and plutonium are extracted for further use.

Table 7.1 World-wide capacity of reprocessing facilities in tons of heavy metals [6.7].

Country	Company Name	Location	Plant	Capacity (tU*/year)	Commercial Operation
France	Orano R La Hague	La Hague	La Hague Plant	1,700tHM	1966
U.K.	Sellafield Ltd.	Cumbria Seascale	Sellafield (Magnox Reprocessing Plant)	1,000	1964
Russia	PA Mayak	Ozersk	Joint Mayak Reprocessing Plant RT-1 Plant	400tHM	1977
	Mining and Chemical Complex (MCC)	Zheleznogorsk	Pilot Demonstration Center (PDC)	4.4tHM (PhaseⅠ)	2016 (PhaseⅠ)
				226tHM (PhaseⅡ)	Scheduled for 2024 (PhaseⅡ)
			RT-2 Plant	800tHM	Scheduled for 2035
Japan	Japan Atomic Energy Agency (JAEA)	Tokai, Ibaraki	Tokai Reprocessing Plant	120tHM	1981 (Decommissioning plan was Approved at June 2018)
	Japan Nuclear Fuel Ltd. (JNFL)	Rokkasho, Aomori	Rokkasho Nuclear Fuel Cycle Facility	800	First half of 2022 (completion)
China	Lanzhou Nuclear Fuel Complex	Lanzhou, Gansu	Lanzhou Pilot Reprocessing Plant	0.1tHM	Construction started in 2006

* U: The weight of uranium in its metal state HM : The mass of metal component of plutonium and uranium in MOX fuel

The existing aqueous reprocessing facilities are listed in Table 7.1 [7.7]. The prominent leading position takes France with the La Hague site and its world-wide larges reprocessing capacity in tons of Heavy Metal (t_{HM}/year). It is also interesting to mention here that Japan is the only non-nuclear-weapon country with reprocessing capacity. In the past, there was also a smaller reprocessing plant in operation at Tokai. In 1993 the construction of the Rokkasho reprocessing plant started, with planned capacity of 800 tons per year. The plant is in active testing since 2006 and the MOX fuel manufacturing plant is expected to be operational in 2024.

Since 2015, China has been constructing a civil demonstration reprocessing plant for spent LWR fuel with a capacity of 200 t_{HM}/year. The construction activities and equipment purchased suggest that this first plant (Project I) could complete its civil engineering stage and begin equipment installation in late 2020. Project 1 is expected to be operational in 2025. The scheduled start of work on the second reprocessing plant (Project II) in late 2020 or early 2021 suggests that it could be commissioned before 2030 [7.8]. Obviously, installation of a reprocessing plant is a strategic decision and very long lasting process, which needs governmental support. It may change in far future; however, any near term deployment must rely on existing reprocessing capacity or on existing stockpile of reprocessed materials.

The 3000 t stockpile of Swiss spent fuel seems to be large amount. Compared to the reprocessing throughput of the existing reprocessing plants, it is small and therefore it may not be sufficient to justify the construction of reprocessing plant in Switzerland. Moreover, the existing reprocessing plants are often a non-negligible source of radioactive emissions and therefore often face a strong public opposition. The siting of a reprocessing plant in Switzerland could be thus highly problematic. The pyro-reprocessing methods are more robust from radiation perspective and can process fuel with higher activity. The overall reprocessing time could be thus shorter. However, their TRL level is low and they are not yet operated at industrial scale.

7.2.7 Pu and RepU single- or multi- recycling in LWR

The Swiss utilities rely on LEU burning in LWR open cycle as on the reference fuel cycle option. Assuming that it would be the only fuel cycle option and assuming 60 years operation time of the remaining NPPs, the mass of spent fuel in Switzerland would be roughly 4000 t. However, in the past, the spent fuel from all Swiss power plants was reprocessed, during limited time period. The Pu and RepU was separated and reused. In case of Pu in was mixed with uranium to Mixed OXides fuel (MOX). The RepU was utilized in its re-enriched form. This recycling helped to reduce the mass of spent fuel. Accordingly, after 60 years of operation, there should be only 3000 t of spent fuel in assemblies form and some amount of vitrified MAs and FPs.

Generally, Pu of both grades when mixed with natural or depleted uranium can be used as a MOX. Based on the Pu share, it can serve as initial fuel for fast breeder operated in the U-Pu cycle or as a recycled fuel in LWR. There exist relatively large Pu stockpile of both grades. The utilization of weapon grade Pu outside of nuclear powers is, however, problematic. The recycling of Pu and RepU involves reprocessing and manufacturing plants. Hence, it was/is usually applied in countries where such plants exist. For the respective countries, refer to the **Fehler! Verweisquelle konnte nicht gefunden werden.**. Recycled MOX is extensively used in French LWR. Japan also uses MOX in its thermal reactors. However, it was historically reprocessed either in France or in UK, because the indigenous production was limited. MOX fuel assemblies were to a smaller extent also applied in UK, Belgium, Germany and as already discussed in Switzerland.

The plutonium isotopic composition evolves during irradiation and has generally a higher share of ^{240}Pu and higher Pu isotopes with increasing burnup. Due to this evolution the plutonium vector quality is deteriorating. It is one of the reasons why MOX is typically recycled only once and why the spent MOX assemblies from LWR are usually not reprocessed. Nonetheless, in France there is an ongoing program, which should evaluate the potential benefits of MOX multi-recycling in LWR. Since the mass of Pu is relatively low, this multi-recycling will not substantially change the utilization of natural resources. Nonetheless, it can help to reduce the long-term stewardship burden related to the final repository.

The reduction of spent fuel mass in Switzerland by 1000 t during the past fuel recycling was not achieved only by MOX itself. Plutonium share in the reprocessed fuel presents only around 1 %. The bulk saved mass resulted from RepU recycling, which was re-enriched and/or blended with HALEU. It strongly reduced the mass of irradiated uranium from final repository perspective. The impact on natural resources utilization was only mild.

Since the irradiation, cooling and transportation of a spent fuel assembly covers time periods of many years and since some of the European nuclear power plants are at the end of their expected lifetime, Orano, formerly Areva, offered a precycling strategy (see Figure 7.6) [7.9]. The simple idea behind is that firstly the MOX assemblies are provided from existing Orano plutonium stockpile and the actual spent fuel assemblies are sent for reprocessing later. From the process description it is not obvious if this encompasses only spent uranium fuel or also spent MOX fuel and how the RepU is utilized.

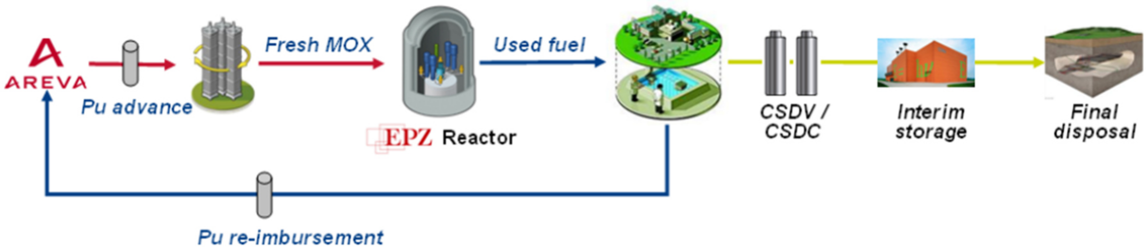

Figure 7.6 MOX precycling strategy, where firstly the plutonium from Orano (formerly Areva) stockpiles is used and only later the respective spent fuel assemblies are sent for reprocessing [7.9].

In general, MOX and RepU recycling in LWR reduces strongly the actinides mass in the waste stream. The recycling in existing Swiss NPP would be, because of the limited time, possible probably only with the Orano precycling option or similar arrangement. Should there be a new LWR build in Switzerland, the existing spent fuel could be recycled so that its mass would stay close to the already existing 3000 t even after the end of the new reactor operation. Only the mass of vitrified FPs and MAs would linearly grow as a function of energy production. Nonetheless, single- or multi- recycling of MOX and RepU is not the most economic fuel cycle option and it will not be applied by utilities unless it would provide some other advantage.

7.2.8 MOX as initial fuel for fast breeder reactor operated in closed U-Pu cycle

MOX fuel can also serve as an initial fuel for fast reactors operated in closed U-Pu cycle. There are three major fast reactor designs based each on their respective coolant type. These are: SFR, LFR and GFR. The LFR family also include reactors cooled by lead-bismuth eutectic. These reactors have been proposed in the past for closed U-Pu breeding cycle; however, to design a burner based on the same technology, it is sufficient to deteriorate the neutron balance by core downsizing or by any other options. There are no big differences in the neutronics performance of all fast reactors. Technologically most advanced is SFR. There are two big units in operation in the Russian Federation BN600 and BN800. There exist also plans for the BN1200 reactor. The USA was considering to build a versatile test reactor based on SFR technology. It was however dropped. Together with the Terrapower company an SFR demonstration reactor is planned for construction. China is building two units of the CFR-600 reactor and planning the CFR-1000. In Europa, the SFR technology is followed mainly by France. However, the project was recently put on hold and France is considering MOX multi-recycling in LWR.

The Pu from irradiated LWR fuel can be used to fabricate initial MOX fuel for a fast breeder reactor. Based on the detailed knowledge of ESFR-SMART, a reference concept of European project with the same name, some conclusions can be done from the Swiss spent fuel perspective. The core loading of the ESFR-SMART reactor corresponds to circa 80 t of fuel with 15 % plutonium share. Swiss power plants are expected, assuming 60 years operation, to generate 30 t of plutonium. Hence, it could be used to fabricate 2.5 full core loadings of the ESFR-SMART. The reactor acts as a breeder, and the plutonium could be understood as a catalyzer of uranium burning. Hence, the loaded 30 t of plutonium will rather be maintained than burned. Since the fuel reprocessing takes a certain time, 2.5 core loadings are minimal appropriate amount to operate the reactor with a small reserve. Should there be more than one fast breeder reactor in operation, the other units should be started by HALEU fuel. From this perspective, LWR plutonium can be seen as resource and not as a burden.

7.2.9 Thorium utilization in closed Th-U fuel cycle

The closed U-Pu cycle can rely on initial fuel in form of MOX or HALEU. The LWR MOX consists from Pu and bulk mass of ^{238}U. Hence, its composition will not evolve strongly during the repetitive irradiation and recycling in the closed U-Pu cycle. Similarly, the HALEU, as primordial material, is a mixture of ^{235}U and ^{238}U. During its irradiation, ^{235}U is swiftly replaced by Pu. Hence, both MOX and HALEU provide convenient and natural option for transition from fresh fuel to the equilibrium fuel, with stabilized isotopic.

In case of closed Th-U cycle, the situation is more complex. ^{232}Th, as the only primordial isotope, is fertile and it does not have fissile neighbor similar to ^{235}U and ^{238}U. In the past HEU mixed with ^{232}Th was used as the initial fuel. From non-proliferation reasons, it is not anymore possible. For similar reason the weapon grade Pu can be excluded. Hence, reactor grade Pu or HALEU or their combination are the only three options to start the closed Th-U cycle. In case of Pu-Th mixture, the transition is relatively short, but reactor grade Pu is not a natural part of the Th-U cycle equilibrium fuel. It may cause material and chemical incompatibilities and difficulties. In case of HALEU, the transition depends on recycling strategy. The mixture of HALEU and Th includes substantial amount of uranium. The generated fissile ^{233}U in the spent fuel will be thus mixed with left over ^{238}U. A solution, which bypasses these issues is to produce ^{233}U in dedicated fertile blankets, which would be surrounding the core or to use the seed-blanket lattice layout.

Starting of the Th-U cycle relies in one case on recycled Pu, which is not so common material as LEU; in other case, it relies on intermediate reactor with dedicated breeding zones and the actual closed Th-U cycle breeder can be started firstly when sufficient ^{233}U amount is generated. The breeding in closed U-Pu and Th-U cycles in fast reactor is from neutronics, waste and radiotoxicity performance very comparable. There are small differences. The U-Pu cycle is technologically more mastered and its startup is more convenient. Thorium is valuable asset, but it is rather a fuel for the future. Nonetheless, both Th-U and U-Pu closed cycles will rely on fuel recycling and the respective reprocessing capacity should be assured by national project with long governmental support or by long-term industrial contract. Operating fast breeder reactor in open cycle with HALEU fuel provides only part of all possible advantages.

7.2.10 Breeding in open fuel cycle without reprocessing

There exist an option, how to run U-Pu breeding in open cycle. It is based on the fact, that certain reactors can provide high excess of fissile material during the irradiation. Should this bred excess be higher than the mass of fissile fuel in discharged fuel, its recycling is not necessary. The process is called breed-and-burn and there are two major reactor concepts capable of it: SFR and MCFR (Molten Chloride Fast Reactor). Both these concepts are developed by B. Gates company Terrapower. The breed-and-burn capability of MCFR was in open literature firstly pointed by PSI, which belongs to the pioneers in that field. The open breed-and-burn cycle can be started by HALEU or MOX and later it will operate only with natural or depleted uranium refill. Resources utilization in such open cycle is circa 20%. At the same time, it is an open fuel cycle, thus waste intensive. In general, it provides the highest natural resources utilization without reprocessing. The major issue in SFR case is that the fuel cladding should withstand very high burnups. In case of MCFR the major drawback is the enormous core size, which is necessary to minimize neutron leakage.

7.2.11 Dedicated MAs burning

Irradiation of primordial actinides generate synthetic actinides. MAs represent small subgroup of synthetic actinides, where for instance Pu or ^{234}U and ^{236}U isotopes are not included.

Whenever fertile primordial nuclides, especially ^{232}Th and ^{238}U, are avoided in the fuel, the reactor acts as a strong burner. Excluding HEU fueled experimental reactors, solely synthetic actinides are present in such burner core. As already explained, all actinides, when irradiated by neutrons for sufficiently long time, will be fissioned or transmuted to a fissile nuclide and then fissioned. Since all actinides decay relatively slowly through alpha decay series and since FPs decay much faster, fission is the ultimate transmutation process to replace the long-term radiotoxicity with short-term radiotoxicity. Fission is thus the actual “waste burning” process.

In a dedicated transmutor or actually MAs burner, fertile primordial nuclides ^{232}Th and ^{238}U should be avoided. Their absence simplifies the expression for transmutation speed, because it is equal to the fission rate. A dedicated transmutor is usually a fast spectrum system. It helps to minimize the neutron costs of transmutation. However, thermal spectrum transmutors are also possible. The neutron balance of a burner is relaxed, because the neutron captures of ^{232}Th or ^{238}U are absent. Compared to a breeder, burner can be constructed as a fast system with lower fuel density or smaller core size or even as a thermal system. Thermal spectrum is not so often proposed, because both the neutron costs for transmutation and maximal achievable burnup would be deteriorated.

In a fast transmutor a burnup of about 10 % FIMA can be achieved. Since the burnup is limited, as in any other reactor, fuel recycling is necessary for substantial transmutation efficiency. Closing of the fuel cycle requires fuel reprocessing, partitioning and fabrication. Based on the burnup of the reprocessed fuel and on the reprocessing losses, the overall utilization of fuel can be calculated as it was shown in the hypothetical transmutation chapter. Assuming the burnup of 10 % FIMA and the reprocessing losses of 0.1 %, the overall utilization would be 99 % of recycled material. In case of 1 % reprocessing losses, it would be only 90 %.

The absence of ^{232}Th and ^{238}U in the dedicated transmutor can result in unacceptable temperature feedback coefficients. The Doppler effect of the fuel can be close to zero or even positive. The coolant density effect may be also positive, because it is not used for neutron moderation and its neutron capture rate is reduced with its decreasing density. Accordingly, dedicated transmutors can have a criticality safety flaw. Some of the concepts are compensating it by external neutron source. There exist also dedicate burners based on liquid molten salt fuel, which profit from the liquid state of the fuel and its expansion coefficient. The density of the liquid fuel is reduced when it heats up and lower fuel amount in the core introduces negative reactivity. Hence, the density effect can compensate the week or positive Doppler effect. In case of solid fuel this expansion effect is less pronounced.

Dedicated transmutors are designed for waste burning and the energy production can be seen as a side products. Transmutation of slowly decaying synthetic actinides into faster decaying FPs decrease the long-term hazard, but increase the short-term hazard. It may represent a shift in the geological storing times needed for the final disposal. Here the Figure 7.3 should be recalled and it should be stressed that the side products of nuclear fuel production (nuclides from the decay chain and depleted uranium) carry substantial radiotoxicity, which cannot be reduced in a transmutor. Furthermore, since there are always reprocessing losses, small part of MAs will end up in the waste stream. Finally yet importantly, in case of transmutation based on small national project and limited spent fuel stockpile, there is an issue of the last core loading of the transmutor. There will be several tons of MAs left after the reactor shut down. Assuming typical transmutor, the core loading could be 4 t. The expected mass of Pu and MAs after 60 years of Swiss power plants operation is 30 t and 7 t, respectively. Hence, the last core loading of the transmutor represents 10% of the waste to be burned. The reduction by 90% is impressive. Nonetheless, the effort and increased risk to achieve it is not necessary worth it and the related space savings in the final repository may not balance the costs for the transmutor operation. Furthermore, should the Swiss spent fuel stockpile be reprocessed and

Pu and MAs burned, there will be still circa 3000 t of RepU, which will need the long lasting isolation from the biosphere.

Transmutation, could be convenient in a country, which is recycling MOX and RepU for energy production purpose and the dedicated MAs transmutor is only a part of the waste minimization plan. In such case, the reprocessing is driven by energy production and not by the transmutation itself. For instance, Orano as the world-wide major fuel reprocessing company has a project to design dedicated Pu and MAs burner in so called Mimosa project (MultI-recycling strategies of LWR SNF focusing on MOlten SAlt technology) [7.10]. The technology selected in this project relies on molten salt fuel, to address the safety of MAs burning system and to avoid the costly external neutron source.

7.3 Conclusions

- Natural uranium reserves are sufficient for next few centuries and depend also on the marked price.
- LEU is produced in several enrichment plants and there is sufficiency variety to assure its supply.
- HALEU production is being increased, because it is major fuel for advanced reactors.
- For majority of advanced fuel cycles reprocessing is necessary.
- MOX and RepU can be conveniently recycled in LWR.
- Closed Th-U and U-Pu cycles should both rely on reprocessing.
- Starting of U-Pu cycle is more convenient.
- Reprocessing of MOX from U-Pu cycle is well established.
- Starting of Th-U cycle is demanding and may require dedicated transition reactors.
- Breeding in U-Pu cycle is possible also in open cycle in so called breed-and-burn reactors.
- Dedicated MAs burner can be applied to minimize the radiotoxicity of existing spent fuel.
- The reduction is however limited and the related reprocessing and transmutor operation increases the risks.
- Moreover, the bulk reprocessed mass would be RepU, whose further handling is not addressed by the transmutor.
- In nuclear phasing out country, spent fuel deposition in final repository is convenient option with lower risk than its reprocessing and transmutation.
- Should the reprocessing by operated for energy production reasons, dedicated transmutor could support other reactors and minimize the waste stream.
- As in any other industry, introducing circular economy for actinides, or actually recycling the nuclear fuel is the best option to increase resources utilization and waste minimization.
- At the same time, it is more costly than utilization of fresh resources. Hence, there should be an additional motivation for the companies to utilize it.

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8 Nuclear energy life cycle analysis

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The purpose of is chapter is to provide a concise overview of the state-of-the art regarding environmental Life Cycle Assessment (LCA) of nuclear power generation based on available literature. This includes a discussion of LCA results with a focus on impacts on climate change (i.e., life-cycle greenhouse gas (GHG) emissions), key driving factors and parameters in the LCA of nuclear power, and a discussion of current data gaps and main uncertainties in the LCA literature addressing nuclear power, which potentially represent focus areas of future research.

8.1 Methodology – Environmental Life Cycle Assessment (LCA)

Environmental Life Cycle Assessment (LCA) is the method of choice to quantify a broad range of environmental burdens of goods and services over their entire life cycle including their production, use, and end-of-life. Usually, LCA is performed according to established standards [8.1]-[8.3]. Environmental burdens quantified by LCA include for example impacts on climate change, emissions of air pollutants and toxic substances as well as land, water and other resource consumption. Such LCA results can be used to compare the environmental performance of different power generation technologies.

8.1.1 System boundaries and functional unit (FU)

An LCA of nuclear power includes all processes from uranium mining and milling to enrichment to electricity generation at the power plant to final geological storage of radioactive waste in its foreground process system, as visualized in Figure 8.1.

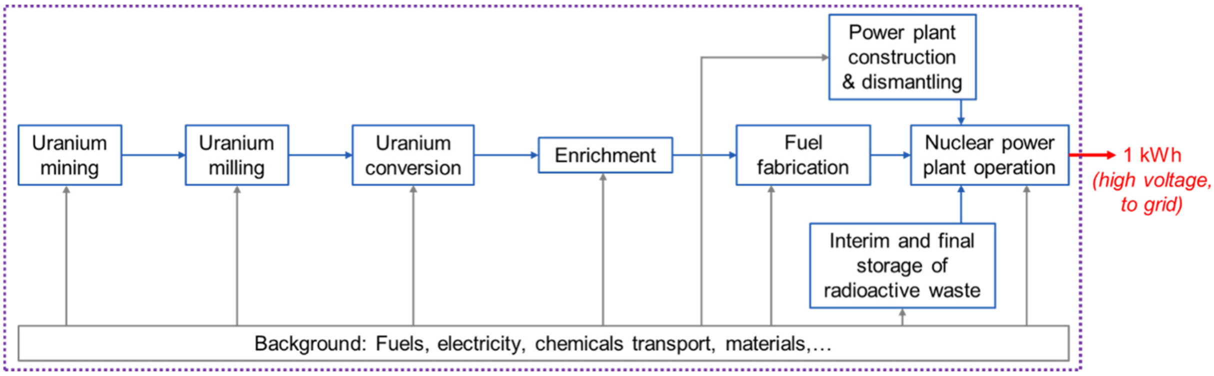

Figure 8.1 System boundaries of nuclear power generation LCA.

This means that a) all direct environmental burdens of these processes are quantified, ideally representing specific nuclear power chains, and b) all indirect environmental burdens (i.e. all those from e.g. material and energy supply chains as well as transport services) are quantified using generic so-called "background" processes based on background LCA databases such as ecoinvent [8.4]. Most often, one unit of electricity, generated by the power plant at its busbar to be fed into the electricity grid, is chosen as functional unit, to which all environmental burdens are scaled.

8.1.2 Life cycle inventory data

Challenges in collecting inventory data (i.e., material and energy flows as well as process emission data) for nuclear power generation process chains are most often related to uranium mining and milling, as information related to these activities is scarce, often not publicly available, and also associated with large uncertainties. Depending on the source of uranium used for fuel element production, this can also be the case for subsequent steps in the fuel chain, especially if military uranium sources are involved [8.5].

8.2 Literature review – life-cycle GHG emissions

The most recent review of nuclear power generation LCA studies (academic and grey literature) has been performed by Gibon and Hahn [8.6]. They list all the 59 reviewed studies, which provide a total of 275 estimates of life cycle GHG emissions, in the supporting information of their open-access article. Studies which do not provide characteristics of their LCA in a systematic manner were excluded from this review. The scope of the review is comprehensive, i.e., no exclusion criteria regarding geographies, technologies, or other parameters have been applied. Figure 8.2 shows the range of LCA results in terms of impacts on climate change from all the studies included in the review, classified according to the time individual studies have been performed (left), according to reactor technology (middle) and according to fuel enrichment technology (right) [8.6]. Several observations can be made:

1) Average GHG emissions are – for all sorts of classifications of studies – in general in the order of about 5-20 g CO_2eq/kWh of electricity. Few outliers report substantially higher GHG emissions.
2) Life-cycle GHG emissions tend to decrease over time, i.e. older studies report higher emissions than more recent studies.
3) The vast majority of LCA has been performed for light water reactors (BWR and PWR) of generations II and III and only very few studies are available for advanced reactor concepts.

The fact that older studies report higher emissions is mostly due to a) the global trend in enrichment technology away from diffusion towards centrifugation, which is more energy efficient and b) the fact that the background inventories used in the LCA studies tend to be less carbon-intensive, as for example the share of low-carbon technologies in the power generation mix increased during the last decades.

Light water reactors have been in focus of LCA studies so far, as they represent the best known technologies with – as opposed to more advanced concepts – data available for performing LCA.

The results regarding life-cycle GHG emissions show a consistent picture. The few outliers with substantially higher GHG emissions represent LCA studies with very unfavorable assumptions in terms of for example uranium ore grade, power source used for enrichment, reactor lifetime, etc., which can, however, not be considered as representative for current nuclear power. As demonstrated by Gibon and Hahn in a very transparent way, it is indeed possible to set parameters in the LCA in a way which results in GHG emissions slightly above 100 g CO_2eq/kWh [8.6].

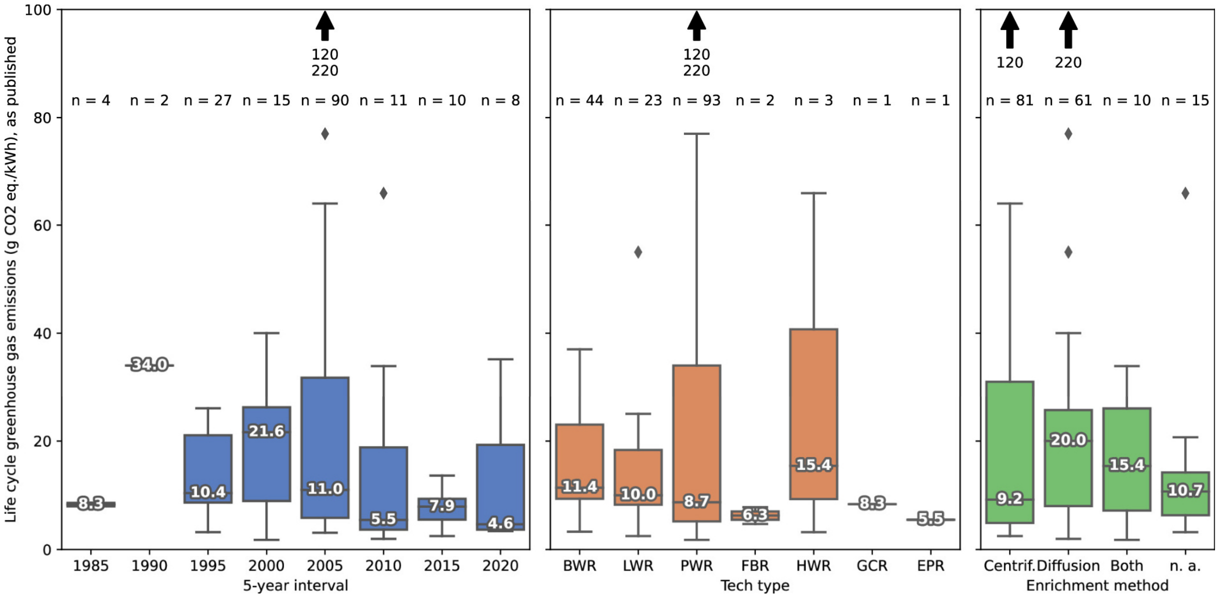

Figure 8.2 Overview of life-cycle GHG emissions of nuclear power according to a broad range of studies [8.6]. Median, sample size, and potential outliers are provided for each series of each category.

8.3 Parametrization of a generic nuclear LCA model

Based on the available literature, a generic, parameterized LCA model for nuclear power, representing light water reactors and associated fuel chains, has been compiled [8.6]. This LCA model can be used to analyze sensitivities, determine driving factors and main contributions to LCA results and to quantify LCA results for specific parameter settings and modeling choices.[39] This LCA model for nuclear power represents an important step forward towards a transparent and context-specific evaluation of the environmental impacts of nuclear power, also in comparison to previous assessments in the context of SFOE's technology monitoring [8.7]-[8.9] as these previous assessments relied exclusively on life cycle inventories of the ecoinvent database [8.4], which did not allow for adjustment of important parameters without some expertise in Life Cycle Assessment.

Parameterization covers all foreground processes in the nuclear LCA (Figure 8.1); main parameters include the uranium ore grade of the uranium deposit, the shares of mining techniques (open-pit, underground, in-situ-leaching (ISL)), mining and milling energy requirements and the type of energy source used, energy demand for conversion and enrichment as well as enrichment technology (gaseous diffusion, centrifugation), enrichment rate, power plant lifetime and electric efficiency, and type of cooling (river, coastal). For each of those parameters, ranges, default values, and uncertainty distributions are specified. The parameter with the largest variation is the ore grade: currently economically viable mines exploit ore bodies with ore grades between 300 ppm (0.03%) and up to 20%. On average, the ore grade of uranium mines operated today is about 0.15% [8.10].

[39] The model can be downloaded here: https://pubs.acs.org/doi/suppl/10.1021/acs.est.3c03190/suppl_file/es3c03190_si_003.xlsx

Figure 8.3 shows a contribution analysis for different environmental burdens based on the generic LCA model for nuclear power, for fuel enrichment per centrifuge and the default parameter setting as specified in [8.6].

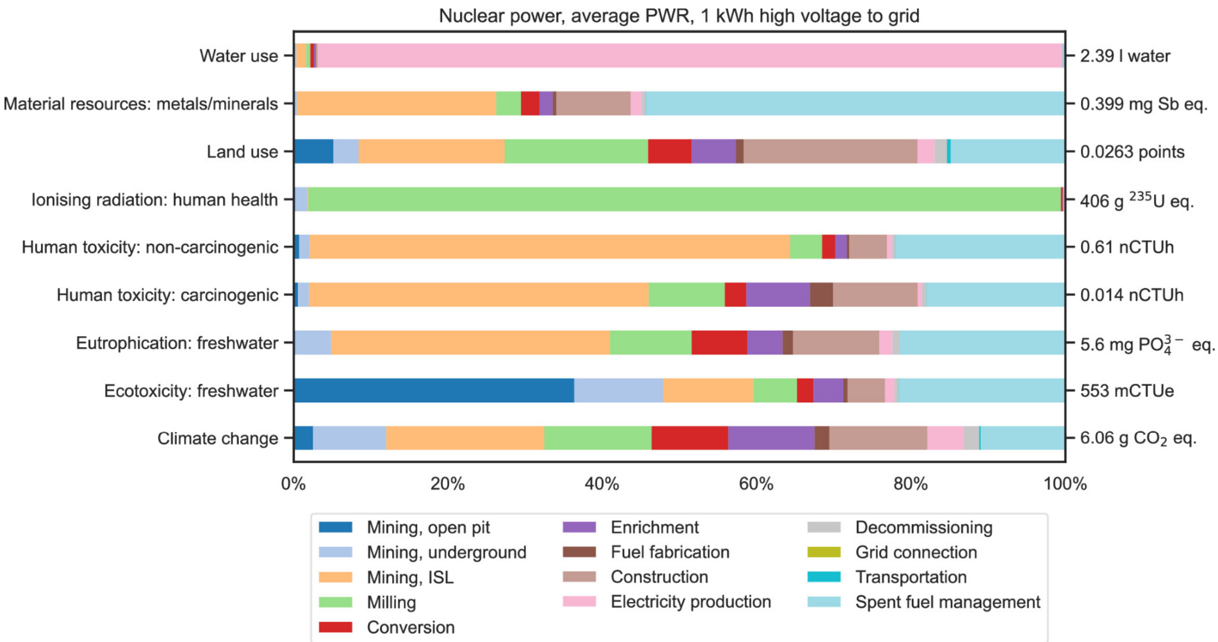

Figure 8.3 Overall LA results and contribution analysis for a range of environmental burdens of nuclear power according to the generic LCA model established in [8.6].

In terms of GHG emissions, the contributions to the total score are distributed among most of the processes from uranium mining to waste management and disposal. Mining and milling together contribute almost 50% of total GHG emissions. If enrichment via diffusions were chosen, the contribution from this process would substantially increase, unless electricity with a very low climate impact were used for this diffusion process. Regarding other environmental burdens, mining and milling of uranium often represent the main contributors due to associated releases of pollutants to air, water, and soil. An exception is the use of water, which is dominated by the power plant operation due to cooling water use (in case river-fed cooling towers are used). However, power plant operation only shows very minor contributions to other burdens and the same holds true for fuel fabrication and power plant decommissioning as well as transport processes in general. Finally, land use is rather evenly distributed among all steps in the nuclear power generation chain.

Based on a global sensitivity analysis, Gibon and Hahn identified the most influential processes and parameters regarding their impact on LCA results [6]. Figure 8.4 shows Sobol indices quantifying the contribution of each model input parameter to the total variance of each impact category for the nuclear LCA model [6]. The higher these indices for specific parameters, the higher their contributions to the overall variances regarding each impact (i.e., environmental burden) category.

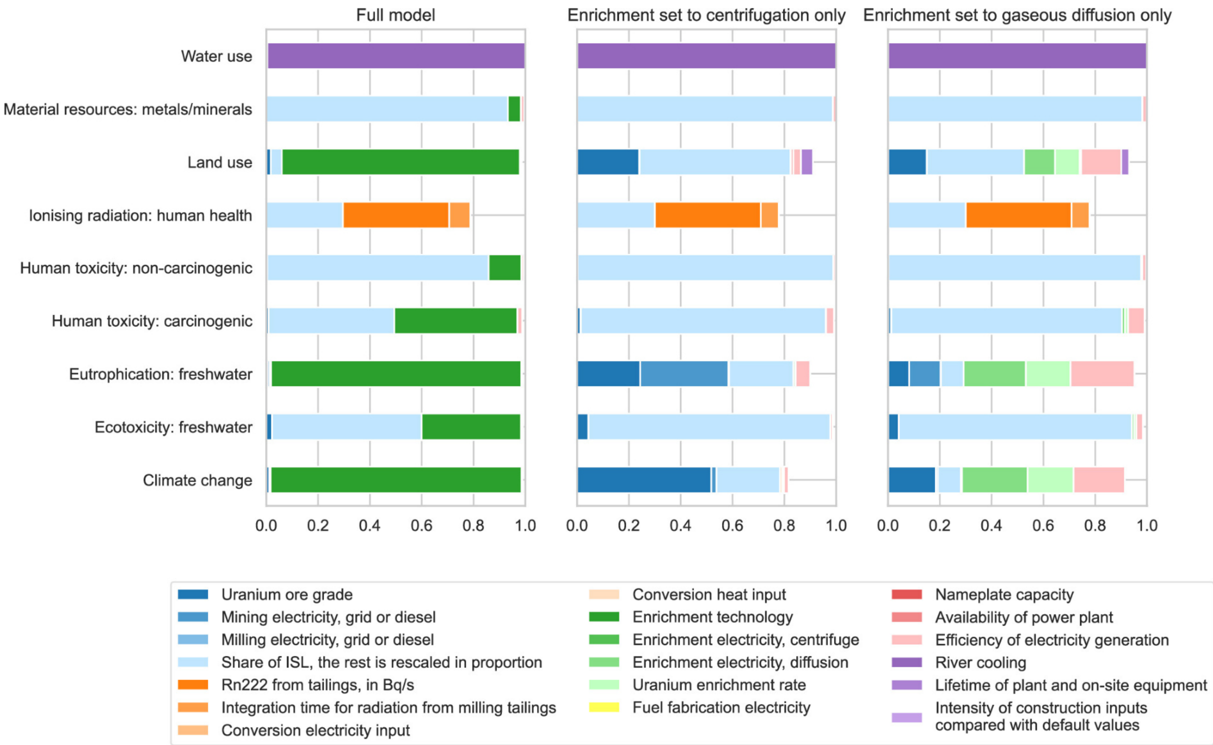

Figure 8.4 First-order Sobol indices quantifying the contribution of each input parameter to the total variance of each impact category for the generic LCA model (left), enrichment set to centrifugation (middle) and to gaseous diffusion (right) [8.6].

For example, the variance of results in terms of impacts on climate change is almost exclusively determined by the choice of enrichment technology (diffusion vs. centrifugation). Similarly, the variance of results in terms of water use is determined by the choice of river-based cooling per default. The share of ISL for uranium extraction from the deposits is the most important factor explaining the variability regarding pollutant emission related impact categories and mineral resource use. If enrichment is performed via centrifuges, the uranium ore grade plays an important role in terms of GHG emissions: at very low ore grades (10 ppm used as minimum in the model), climate impacts per kWh nuclear power increase substantially up to levels around 100 g CO_2eq/kWh, which reflects the outliers identified in the literature review. In general, comparing LCA results based on optimistic versus pessimistic parameter settings, LCA results show variations of up to one to two orders of magnitude.

8.4 Swiss-specific nuclear LCA

The latest Swiss-specific LCA of nuclear power has been carried out in 2017/18 by PSI on behalf of swissnuclear and in collaboration with the utilities operating the power plants in Gösgen (KKG, pressurized water reactor) and Leibstadt (KKL, boiling water reactor) [8.5]. As part of this work and in close collaboration with the plant operators and the responsible parties for fuel supply, inventory data of all processes of the nuclear chains have been revised and updated for the reference year 2017, also taking into account the recent, very detailed LCA concerning management of radioactive waste from Swiss sources [8.11]; in addition, few new processes were integrated, for example the decommissioning of the power plants at the end of their operation periods. Therefore, this study represents the state-of-the-art LCA for nuclear power in Switzerland based on accessible data. For most processes, data quality can be considered as good. The exception is uranium extraction and processing for fuel element fabrication in Russia, which was a relevant part of the fuel supply for KKL at the time the analysis has been carried out. Information regarding these processes is partially insufficient and as a consequence, these processes could not be properly modeled. An (unknown) fraction of this uranium originates from disarmed nuclear weapons – complete data for the associated processes were not available. Instead, own assumptions and approximations have been used.

8.4.1 Swiss-specific LCA: modeling and results

Table 8.1 lists key parameters in the LCA of Swiss nuclear power referring to the reference year 2017 according to [8.5].

In general, the LCA results of the Swiss-specific analysis are in line with those of the generic LCA performed by Gibon and Hahn [8.6] and the literature they reviewed. The Swiss-specific LCA results and an analysis of the main contributors along the power generation chain show that the environmental performance of Swiss nuclear power is to a large extent determined by the origin of uranium. Most important factors in this context are uranium ore concentrations as well as technologies and energy carriers used to mine and process uranium resources. The tailings of uranium mining and milling cause a large fraction of environmental burdens and decreasing ore grades will lead to an increase in the quantity of these tailings. The burdens due to uranium enrichment are comparatively small, as fuel is enriched with centrifuge technology only, which is much more energy efficient than enrichment by diffusion, which has been used in the Swiss fuel supply chain years ago.

Table 8.1 Key parameters in the LCA of Swiss nuclear power [8.5]; reference year for the analysis: 2017.

Data	Unit	PWR (KKG)	BWR (KKL)
Plant thermal capacity	MW_{th}	3002	3600
Plant lifetime[7]	years	50	50
Annual net electricity generation	GWh/year	8022	9458
Fuel type	-	UO_2	UO_2
Fuel assembly	kg of UO_2/assembly	502	200
Enrichment grade	-	4.95%	4.50%
Efficiency	-	33.6%	33.3%
Discharge fuel burnup	MWd_{th}/kgU	62.4	53.9
Fuel Consumption per kWh of Net Electricity Generation	kg of U/kWh net electricity	1.98E-06	2.32E-06
Intermediate level radioactive waste generation	m^3/kWh net electricity	1.57E-9	3.67E-9
Low level radioactive waste generation	m^3/kWh net electricity	2.24E-9	4.89E-9
Spent fuel generation	kg/kWh net electricity	2.96E-6	3.36E-6

Figure 8.5 shows life-cycle GHG emissions of Swiss nuclear power according to [8.5]. Per kWh of electricity generated, GHG emissions are around 6 g CO_2eq and 9 g CO_2eq for PWR and BWR, respectively. Contributions from uranium mining and milling represent around 50% of the totals in both cases.

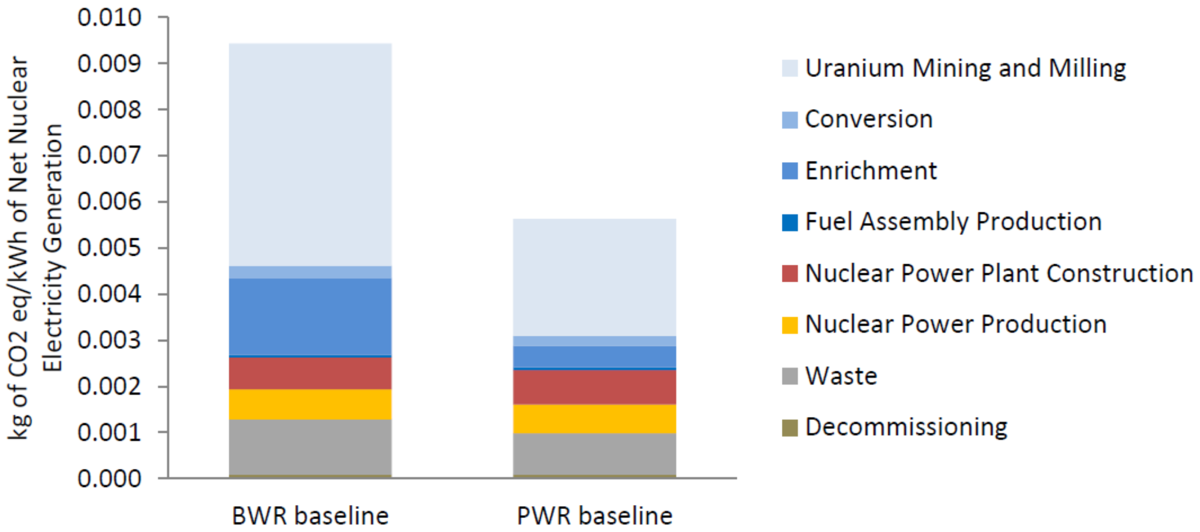

Figure 8.5 Life-cycle impacts on climate change for Swiss nuclear power [8.5]; "baseline" refers to the default parameter setting in the analysis.

Zhang and Bauer [8.5] also addressed other environmental burdens than impacts on climate change. Figure 8.6 and Figure 8.7 show a contribution analysis for the Swiss PWR and BWR chain, respectively.

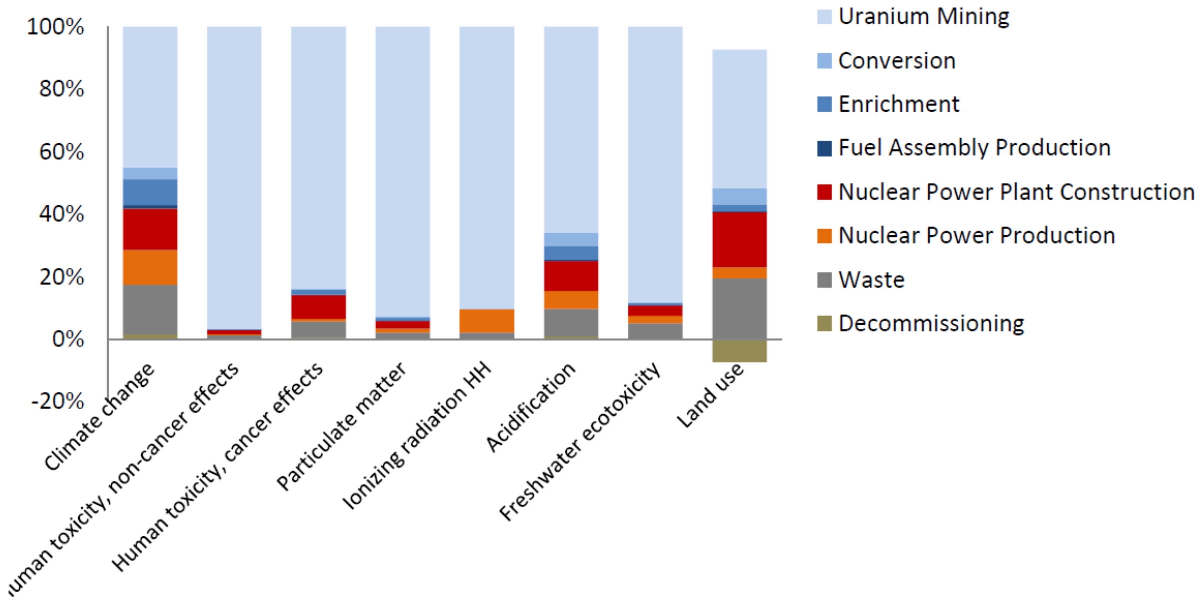

Figure 8.6 Contribution analysis for a range of environmental burdens per kWh of power generated by the Swiss PWR [8.5]; "uranium mining" includes milling processes.

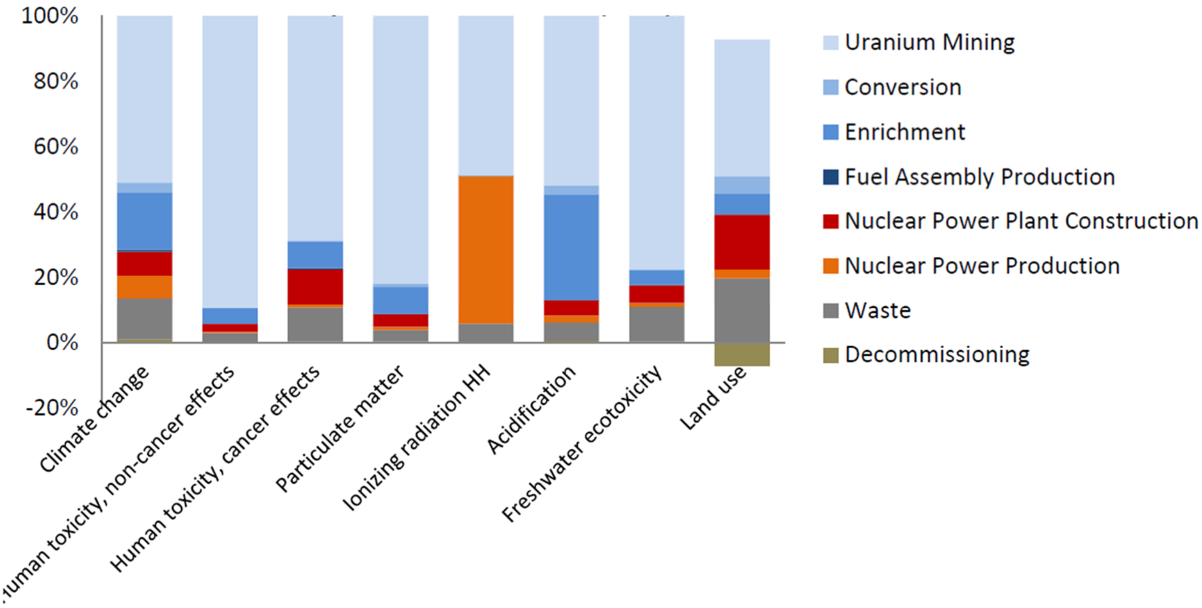

Figure 8.7 Contribution analysis for a range of environmental burdens per kWh of power generated by the Swiss BWR [8.5]; "uranium mining" includes milling processes.

Especially environmental burdens other than impacts on climate change are to a large extent caused by uranium mining and milling activities. Regarding land use, also activities taking place in Switzerland, namely power plant construction and radioactive waste management, are important.

As recommendations for future analysis, Zhang and Bauer [8.5] highlight the importance of the origin of uranium and the fact that only old inventory data with a low level of detail for a limited range of extraction sites are available for those processes. Further, the need to rely on assumptions, approximations and extrapolations, respectively, has been mentioned as a result of the lack of detail in available inventory data for uranium mining and milling – limiting the reliability of the LCA results. From today's point of view, assuming that uranium supply from Russia is no longer part of the Swiss fuel supply chain, an update of the uranium fuel supply chain representing a more diverse supply should be performed.

8.5 Comparison of nuclear power with other electricity generation technologies

To put the LCA results of nuclear power into context, they can be compared with those of other electricity generation technologies. As power generation technologies are most frequently (and often only) evaluated regarding their impact on climate change, a literature-based comparison is most meaningful and consistent regarding life-cycle GHG emissions. As shown in Figure 8.8, nuclear power is among the power generation options with the lowest impact on climate change. Similarly low GHG emissions can be observed for electricity production using hydro (in non-tropical regions) and wind power plants.

Figure 8.8 shows GHG emissions of nuclear power based on average parameter setting in the parameterized LCA model of Gibon and Hahn [8.6]. A comparison of Figure 8.8 and Figure 8.2 indicates that the variability of associated GHG emissions is higher, but also with a more "disadvantageous" setting hardly results in GHG emissions above 100 g CO_2eq/kWh, which is in the range of the renewables included in this comparison. Ranges for non-nuclear technologies shown here are due to regional variability of life cycle inventories of those technologies.

Other impacts than those on climate change are mostly dependent on location- or regional-specific aspects, like local water scarcity for water consumption or population densities for air pollutant related impacts. Such impacts should be assessed applying regionalized impact assessment factors instead of generic average ones, which is, however, not yet common practice in LCA.

Generic comparisons of environmental impacts other than those on climate change of a range of power generation technologies including nuclear power – without taking into account any regionalized impact assessment – have been performed though, e.g. by Bauer et al. [8.7], [8.9] Gibon et al. [8.12], and Zhang and Bauer [8.5]. Though they apply partially different methods for impact assessment und use different inventory data, these comparisons consistently show that the electricity generation technologies causing the lowest impacts in most categories are different types of hydro power plants, if not operated in tropical regions, where reservoir lakes can emit substantial amounts of greenhouse gas emissions. Among the technologies with low impacts in many impact categories are wind and nuclear power plants. These results can – despite of certain limitations in such generic comparisons – be interpreted as an indication of the overall environmental performance measured by common impact assessment methods. Such a generic comparison of environmental burdens, using data from [8.7], is shown in Figure 8.9, while Figure 8.10 summarizes the results from [12].

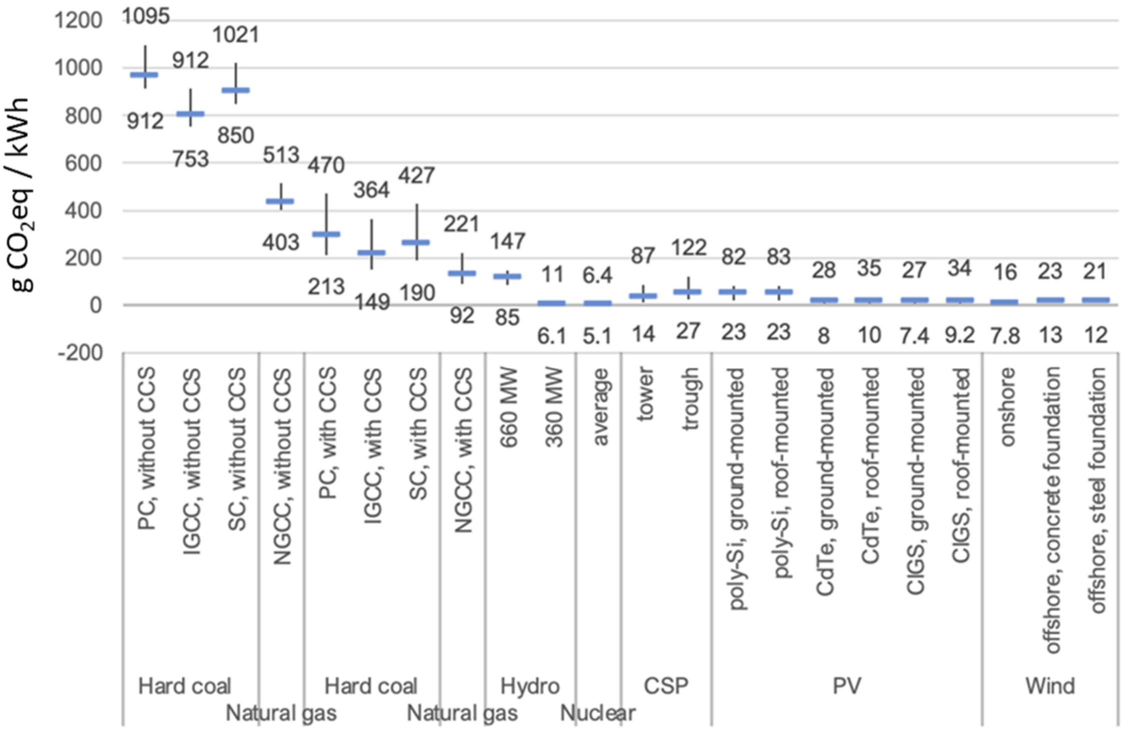

Figure 8.8 Impacts on climate change in terms of life-cycle greenhouse gas emissions of different power generation technologies, in g CO2eq per kWh of electricity generated [8.12]. Blue horizontal markers indicate average values for each technology. Error bars indicate regional variations. CSP: Concentrated Solar Power; PV: photovoltaics; PC: Pulverized Coal; IGCC: Internal Gasification Combined Cycle; SC: Supercritical; NGCC: Natural Gas Combined Cycle; CCS: Carbon Capture and Storage; poly-Si: polycrystalline Silicon; CdTe: Cadmium Telluride; CIGS: copper indium gallium diselenide.

Different environmental impacts can also be aggregated to a single indicator supposedly representing "total environmental impacts", using impact assessment methods such as "ReCiPe" [8.13] or "Ecological Scarcity" [8.14]. However, such an aggregation always includes subjective weighting of the different impacts and is thus not recommended in comparative LCA [8.2], [8.3]. Contrary, the product environmental footprint rules do recommend to provide normalized and weighted LCA results [15]. As there is, however, no commonly accepted and recommended method to so, and providing such single-score LCA results would need an extensive discussion on advantages, disadvantages, benefits and shortcomings of different approaches of aggregation, we consider such a single-score evaluation as out of scope of this report.

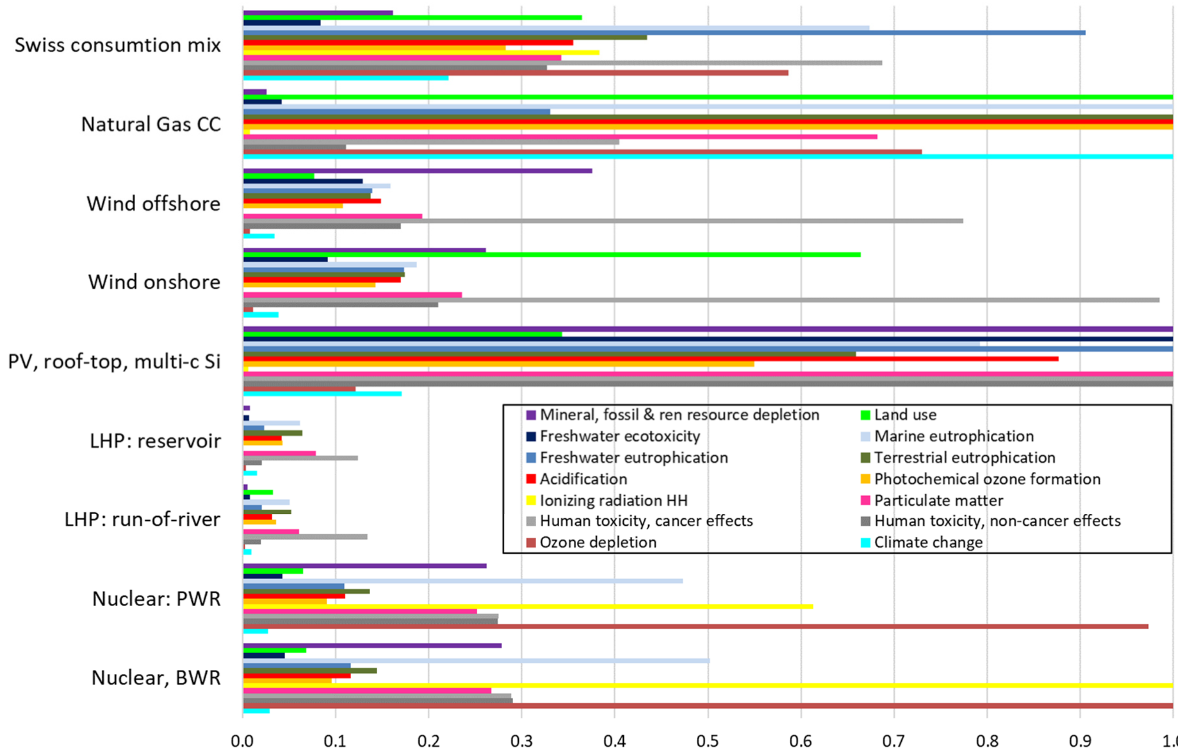

Figure 8.9 Relative environmental burdens of different power generation technologies, based on [8.7]. Environmental burdens are scaled relative to the technology with the highest score in each impact category. All results representative for power plant operation in Switzerland. LHP: Large hydro power; CC: Combined Cycle; PV: photovoltaics; BWR: Boiling Water Reactor; PWR: Pressurized Water Reactor; HH: Human Health.

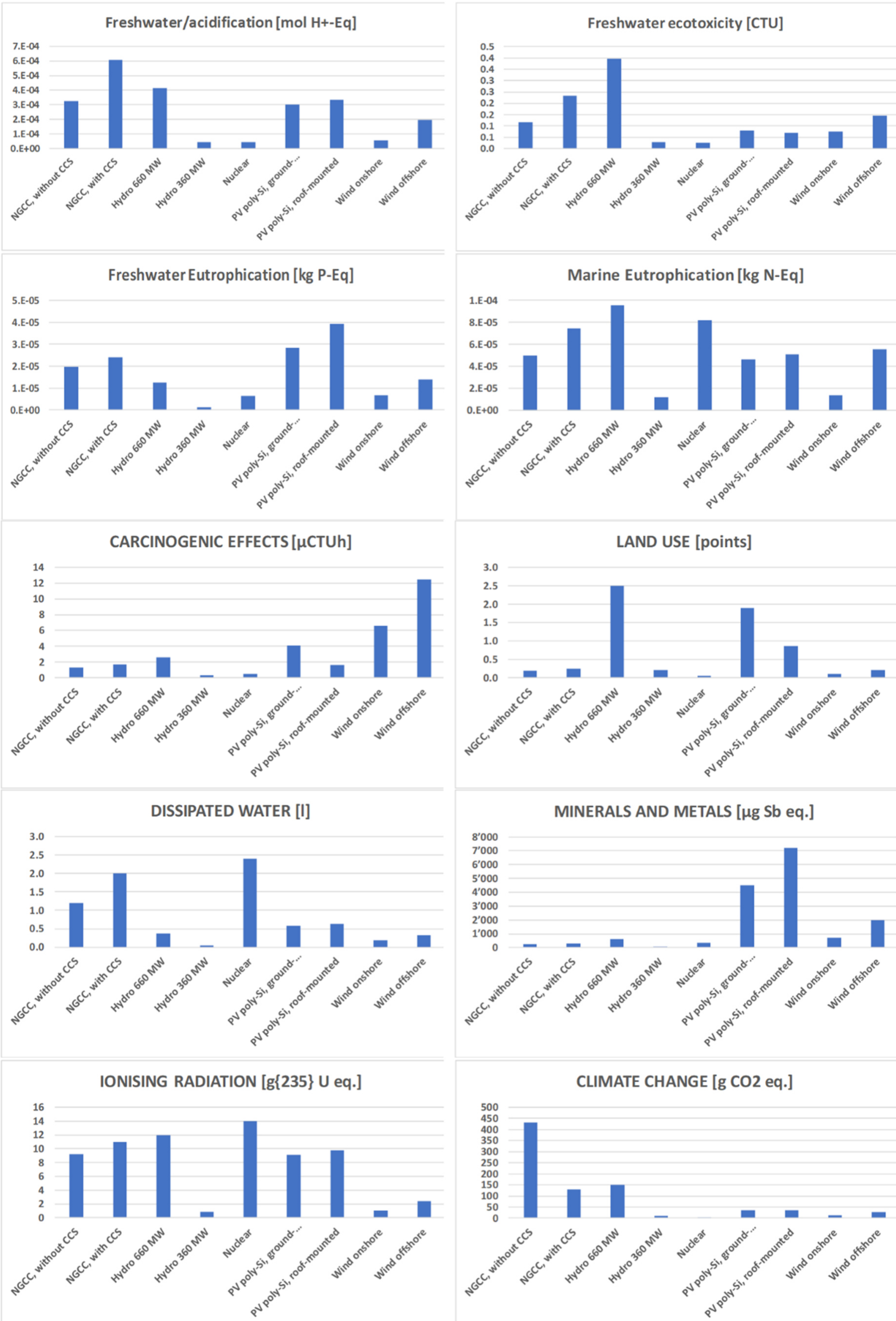

Figure 8.10 Environmental burdens of different power generation technologies based on Ref. [8.12]

8.6 Main uncertainties, data gaps and areas of further research

Both the Swiss-specific [8.5] and the generic LCA in Refs. [8.6] and [8.12] of nuclear power are consistent in terms of identifying key uncertainties and data gaps and thus provide rather similar recommendations for future research to reduce uncertainties and close gaps:

- Uranium mining and milling

Currently available life cycle inventory data for uranium mining (and milling) are old and based on very few uranium extraction sites. Energy requirements associated with these processes are extrapolated using models which are hard to verify due to the very limited data base. Reliable information regarding the handling of uranium tailings from mining and milling as well as data for emissions, both radioactive and non-radioactive, of these processes are scarce and only allow for representation of non-climate change related environmental impacts in a very generic way, while it can be assumed that in reality such emissions are case- and location specific. A broader coverage and a comprehensive update of those processes, ideally performed in a consistent way, could reduce uncertainties substantially, also in the context of a more diversified fuel supply and the exploitation of uranium deposits with potentially reduced ore grades in the future.

- Advanced and new reactor concepts and fuel cycles

Currently available LCA literature focuses on Gen II and III light water reactor designs and conventional uranium fuel cycles. LCA addressing both small modular reactor as well as Generation IV concepts and for example thorium fuel cycles are missing, but should be performed to contribute to an informed decision making process.

- Proper consideration of lifetime extension

Many nuclear power plants have originally been designed for lifetimes of 40 years. However, it seems likely that at least some of them will be operated for 60 or even 80 years, if economically viable. Such lifetime extensions are usually only possible with major upgrades of the reactors to ensure safe operation. Such upgrades and the associated (construction) measures have hardly been considered in detail in LCA so far.

- Representation of ionizing radiation in Life Cycle Impact Assessment (LCIA)

As briefly discussed by Gibon and Hahn [8.6], there are large uncertainties and a lack of consensus on how to quantify impacts of ionizing radiation on human health and ecosystems. When quantifying impacts of ionizing radiation, Radon-222 emissions and radiation integration time are very important. However, ionizing radiation is an indicator rarely addressed, or at least analyzed in detail in LCA. The latest characterization factors were published by Frischknecht et al. [8.16], using the so-called “linear no-threshold” (LNT) model that assumes that human health impacts occur from the first radionuclide emissions. Although precautionary and conservative, this LNT model has been criticized [8.17]. Similarly, in the context of integration time of environmental burdens, the question of whether radionuclide emissions will still be of concern in 80’000 years from now becomes relevant (while the time horizon for climate impacts is usually 100 years in LCA).

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9 Legal framework for the construction of new NPPs in Switzerland

Georg Schwarz (ENSI, retired)

9.1 Introduction

The current Nuclear Energy Act (NEA) is a compromise that was drafted as an indirect counter-proposal to the two anti-nuclear popular initiatives "Strom ohne Atom" and "Moratorium Plus". The "Strom ohne Atom" initiative called for the gradual closure of all nuclear power plants (NPPs) by 2014 at the latest. The "Moratorium Plus" initiative was less radical. For ten years after its adoption, no new nuclear power plants would be allowed to be built, nor would existing NPPs be allowed to increase their power. The continued operation of existing NPPs for more than 40 years would have been subject to an optional referendum.

On 18 May 2003, both initiatives were clearly rejected by 66% and 54% respectively. This meant that the indirect counter-proposal, the NEA, came into force. According to this law, the continued operation of existing NPPs and the construction of new ones are in principle possible. In the spirit of compromise, the counter-proposal took the concerns of the initiatives into account on several points. In particular, the safety requirements for new NPPs were significantly increased and the moratorium on reprocessing spent fuel has been converted into a definitive ban.

The decision on new nuclear facilities was made subject to an optional referendum. In addition, the possibility of appealing against construction and operating licences to the Federal Administrative Court and the Federal Supreme Court was created. The resulting licensing procedures have become so complicated and time-consuming that it would now take more than a decade to license a new NPP, not including construction time. The NEA entered into force on 1 February 2005.

Following the Fukushima nuclear accident, the NEA was further tightened as part of the Energy Strategy 2050. A ban on the construction of new NPPs came into force on 1 January 2018.

Given that the NEA originated as a counter-proposal to two anti-nuclear popular initiatives, it is not surprising that, in addition to the ban on new construction, the law contains numerous other restrictions on the construction of new NPPs. As a result, the legal framework in Switzerland is much less favourable than in the Western countries that are currently building new NPPs (Finland, France, the UK and the US).

For example, Swiss safety requirements are very strict by international standards. As they are also dynamically adapted to developments in the state of the art in science and technology, the Swiss regulations do not provide a stable regulatory basis for new construction projects. In addition, Swiss approval procedures are very lengthy and fraught with legal risks due to the extensive possibilities for public consultation and appeal.

This chapter sets out to provide explanations to the existing legal constraints and outlines the necessary legislative changes that would be required to improve the legal framework for the construction of new nuclear power plants in Switzerland.

9.2 Legal restrictions

9.2.1 Legal restrictions

According to Art. 12a of the NEA, no general licence may be granted for the construction of a new NPP. This means that existing NPPs may continue to operate, but their replacement at the end of their operating life or the construction of new plants is prohibited.

9.2.2 Level of safety

According to Art. 4 para. 3 of the NEA, all precautions shall be taken in the use of nuclear energy which are necessary in the light of experience and the state of the art in science and technology and which contribute to a further reduction of the hazard, provided that they are appropriate. This provision would also apply to the construction of new NPPs if the ban on new construction were to be lifted.

Instead of concrete safety standards, the NEA uses the dynamic term "state of the art in science and technology". As the state of the art is constantly evolving, new knowledge can be taken into account without the need to constantly adapt the law.

As an example of the state of the art in science and technology, the commentary to the NEA mentions reactors with a pronounced passive and inherent safety, which can control core melt accidents in such a way that emergency protection measures outside the plants are only required to a limited extent.

This is a high level of safety, but it is achievable. Today, all manufacturers offer Generation III reactor types that meet the Swiss requirements. A safe reactor type does not a priori lead to high costs. In China, two reactors of the latest generation, type AP-1000, which are also licensed in the USA, were built in Sanmen for 2,600 CHF/kW.

However, country-specific regulations that require changes to the standard design or to the manufacturing and verification procedures are unfavourable. This was the case, for example, in Finland, where unique regulations required complete separation of the the instrumentation and controlof each safety level. Such regulations have led to the need for Framatome to change the EPR design specifically for OL3 during the plant construction, a contributing factor to the construction delays and costs increase.

The Swiss regulations are largely based on the US and German regulations and contain hardly any country-specific requirements. The only difficulty could be meeting the Swiss requirements for seismic resistance. These are significantly higher than the standard design of most reactor types. Adapting reactor designs to the robustness required in Switzerland could therefore increase construction costs.

However, there does not seem to be any objective justification for reducing the safety requirements in this respect. It will be up to the project proponent to select a reactor design that can be adapted to Swiss seismic requirements at a reasonable cost.

9.2.3 Licensing procedure

In general, three licences are required for nuclear installations in Switzerland (see also Figure 9.1):

- **General licence:** The general licence is the first stage of the licensing procedure and is granted by the Federal Council. It confirms the fundamental suitability of the site and the project for a nuclear installation. It must be confirmed by Parliament and is subject to an optional referendum.

- **Construction licence:** The construction licence defines and approves the technical details of the installation. It is the second stage of the licensing procedure and is issued by the Federal Department of the Environment, Transport, Energy and Communications (DETEC). The construction permit can be appealed to the Federal Administrative Court and then to the Federal Supreme Court.
- **Operating licence:** The final level of licensing is the operating licence, which is also issued by DETEC. It lays down the conditions for the operation of the installation. Appeals against operating licences may be lodged with the Federal Administrative Court and then with the Federal Supreme Court.

For each stage of licensing, applicants must submit various documents such as safety analyses, environmental impact reports, emergency plans, etc. These are examined by the competent authorities. In addition, the licensing process must be made publicly accessible.It takes about four years to obtain a general licence. Of this, one year each is spent on the preparation of the application documents by the applicant and their review by the Swiss Federal Nuclear Safety Inspectorate (ENSI) and the Swiss Federal Nuclear Safety Commission (KNS). Consultation of the cantons and public notification will take at least another year. This is mainly due to the fact that several cantons are obliged to hold a referendum as part of the consultation procedure for NPPs. Discussions in the National Council and the Council of States, as well as the foreseeable optional referendum, will take a further two years. As the general licence is subject to an optional referendum, it also represents the democratic legitimisation of a project.

The applicant needs around two years to prepare the construction licence application. The assessment of the application by ENSI and the KNS takes presumably a further two years. Residents have the right to appeal against the decision. Appeals to the Federal Administrative Court and further appeals to the Federal Supreme Court can each take more than a year and have a suspensive effect. Given the strong opposition to nuclear power in Switzerland, it is likely that residents will make extensive use of this right of appeal.

Together with the time required by the authorities to examine the documentation, the granting of a construction licence can take up to six years in the event of an appeal. A legally valid construction licence is required before construction of a nuclear power plant can begin. Together with the general licensing procedure, it takes a total of eleven years to obtain such a licence.

In principle, it is possible to prepare and treat the construction license application in parallel with the general license procedure, so that the construction license could be issued immediately after the general license is granted. However, as the preparation of a construction license application is very expensive and there is no legal entitlement to the granting of the general license, such a procedure involves a very high financial risk and is purely theoretical. In practice, therefore, the building permit application is prepared and submitted only after the general licence has been granted.

In contrast to Switzerland, Finland and France have more streamlined permitting procedures. Excluding the time needed to prepare the application documents, the procedure for obtaining a site and construction permit took four years in Finland[40] and just over three years in France[41]. In neither case was the project delayed by legal proceedings.

The same duration as for construction licences, i.e. six years, must be foreseen for granting operating licences. However, unlike the construction licence, the operating licence can be processed in parallel with the construction work.

A total of eleven years is estimated for the general and construction licensing procedures. Assuming a construction period of seven years, it will therefore take at least 19 years to build a new nuclear power plant in Switzerland (see also illustration in Figure 9.1).. This long period drives up project planning and financing costs.

9.2.4 Stability of the regulatory framework

New safety requirements imposed during the course of a project can result in major design changes, long delays and additional costs.

An example of such delays is the construction project for reactors 3 and 4 at the Vogtle NPP in the USA. Although the planned AP-1000 reactor had a valid design certification and early construction approval was imminent, the US regulatory commission imposed a new requirement to consider aircraft crashes in the design (https://atomicinsights.com/nrcs-imposition-of-aircraft-impact-rule-played-a-major-role-in-vogtle-project-delays-and-vc-summer-failure/). The design adjustments and reviews required by the new regulation were responsible for much of the delay and additional cost to the project. In the case of the second AP-1000 project at VC Summers, which started in parallel, the new requirement even led to the cancellation of the project. The associated financial loss amounted to USD 9 billion.

In Switzerland, the regulatory framework tends to be less stable than in the US. As explained above, the NEA uses the dynamic term "state of the art" instead of concrete safety benchmarks. It is up to the regulator to define the term in detail. In doing so, it can take into account the latest technical and scientific developments, but also new findings from incidents and operating experience of existing plants.

Whether a project complies with the state of the art is assessed on a case-by-case basis at each licensing stage. Such changing safety requirements reduce the benefits of standardisation. This has a particular impact on the design of SMRs, which depend on standardised series production. It does not matter how standardised a design is if it has to be changed for each project.

However, new requirements can be imposed not only during the licensing process, but also during the course of a project. Changes in safety requirements during the design and construction phase result in costly adjustments to the plant design and delays. This was the case in both Finland (Olkiluoto-3) and France (Flamaville), where additional regulatory requirements led to significant construction delays.

The situation is aggravated by the fact that the general licence contains very few substantive specifications and thus does not provide a technical basis for the design of the plant that is

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[40] https://de.nucleopedia.org/wiki/Kernkraftwerk_Olkiluoto
[41] https://de.nucleopedia.org/wiki/Kernkraftwerk_Flamanville

accepted by the authorities. Most of the safety requirements remain open and will only be defined in the construction and operating licences, i.e. 8 and 15 years after the start of the project.

Due to the late stage at which additional safety requirements can be imposed, there is a risk of significant additional costs and delays. Under the current licensing framework is it highly unlikely that a company would take the financial risk of building a new NPP in Switzerland.

9.3 Options to accelerate the licensing procedures of new NPPs

As explained in the previous section, there are significant legal constrains to a return to nuclear power in Switzerland. First and foremost, of course, is the ban on the construction of new nuclear power plants. But even if this were to be lifted, the licensing procedures, which are very long by international standards, pose significant financial risks for potential project developers and investors. This is because objections and changes in safety requirements during the design and construction phase can result in costly project changes and delays. A nuclear power plant built under the current Swiss framework would have very high prime costs.

It would be conceivable to simply lift the ban on new construction, leaving the rest of the legislation unchanged. The resulting high production costs could be offset by subsidies or state guarantees. However, as these options do not reduce costs but merely shift them, they will not be discussed here.

Instead, two options are presented below that could be used to shorten the licensing process to an internationally standard timeframe and to limit project risks due to objections or changing safety requirements during the project planning and construction phase (see also illustration in Figure 9.2 and Figure 9.3). This reduces the financial risks for potential project developers and thus the construction costs of new nuclear power plants.

9.3.1 Waiver of the general licence

Waiving the general licence is the most straightforward way to shorten the licensing procedure. The general licence sets out the fundamental, politically important issues such as site and proof of disposal. It is a prerequisite for the other licences. The general licence application only specifies the approximate size and location of the main buildings, the reactor system, the power class and the main cooling system. It is clear that no sound safety assessment can be made on the basis of this sparse information.

Thus, from a safety point of view, the general licence is of only minor importance. In order to shorten and simplify the licensing procedure the general licence could therefore be cancelled without replacement. However, as the general licence is a "political" licence, it is foreseeable that a general waiver would be highly controversial.

Nevertheless, there are already two exceptions to the requirement for a general licence. The first exception applies to nuclear installations with a low hazard potential that do not require a general licence. These nuclear installations are designated by the Federal Council in an Ordinance. The significantly higher safety level of modern types of NPPs compared to existing plants could be used to justify to extend the current definition of nuclear installations with low hazard potential to very safe NPPs. This would at least allow the construction of new small modular reactors with a very low risk.

Existing nuclear installations are the second exception. None of the existing nuclear power plants has a general licence. Their operation is based on the transitional provisions of the NEA. This exemption could be extended to cover replacing existing NPPs.

Both options described above have the advantage that, apart from lifting the ban on new construction in Article 12a of the NEA, very few other legislative changes are required. There is no need for a time-consuming comprehensive revision of the NEA, which would speed up the legislative process.

As explained above, the exceptional waiver of the general licence would have no negative impact on safety requirements and would shorten the licensing procedure by at least four years. However, the problem of changing safety requirements during the design and construction phase remains, but is mitigated by the shorter licensing procedure.

By exempting nuclear power plants with low hazard potential or replacement NPPs at existing sites from the general licensing requirement, the opportunity for the public to vote on a new construction project would be removed. However the extensive opportunities for objection at the construction and operating licence stage would remain.

9.3.2 Streamlining of the licensing procedures

Thanks to the optional referendum, a general licence gives a project a high degree of democratic legitimacy. As outlined above, it is foreseeable that waiving the general licence would be politically controversial. As an alternative, proposals for streamlining the existing licensing procedures are therefore presented below. The focus here is not so much on saving time as on reducing the financial risks arising from objections or changing safety requirements during the project planning and construction phase.

The basic idea of the present discussion is to extend the democratically legitimised general licence so that it also specifies the essential safety features of a project. The applicant would thus have to commit to a specific reactor type earlier than at present.

In return, the subsequent construction and operating licences could be merged and the opportunities for appeals narrowed. This reduces the likelihood of costly project delays and changes

9.3.2.1. Requirements for the granting of a general licence

In particular, the requirements for the granting of a general licence will be extended. The requirement that a project complies with the principles of nuclear safety and security will now be a prerequisite for the general licence and not only for the construction licence.

Full demonstration of compliance with the principles of nuclear safety and security is currently part of the construction licence. It is clear that such proof cannot be provided in detail at the early stage of the general licence. However, a generic review of the safety concept is possible. This could be based on a generic design certification along the lines of the American (NRC design certification) or British (ONR generic design assessment) models. Such certificates have already been issued for several reactor types available on the market and could, in principle, be adopted by the Swiss authorities.

The other requirements of the general licence relating to the decommissioning concept, proof of disposal of the radioactive waste produced, environmental compatibility and coordination with spatial planning would remain unchanged.

9.3.2.2. Content of the general licence

With the procedure under §9.3.2.1, the content of the amended general licence would include the main features of the project, in which, in addition to the reactor type, the reactor capacity and the main cooling system, the safety concept is now also defined in the form of a design certification. In particular, the design certification will specify the safety requirements to be applied for the duration of the project in accordance with experience and the state of the art in science and technology, as well as the precautions to be taken to further reduce hazards. This would provide a stable assessment basis for subsequent licensing steps and avoid the need to take account of new safety requirements during the course of a project.

As is the case today, the general licence would specify the maximum permissible radiation exposure for people in the vicinity of the plant. In addition, all other requirements with a direct impact on the surrounding population that are currently part of the construction or operating licence will be integrated into the general licence. This applies in particular to the basic principles of emergency preparedness, the limits for the release of radioactive substances into the environment and the measures for monitoring the environment.

The other contents of the general licence, such as the licence holder, the site, the purpose of the facility, the maximum permissible radiation exposure for persons in the vicinity of the facility, and the information on deep geological repositories, would remain unchanged.

Since the proposed extended general licence sets out the essential safety requirements, the possibility of objecting to the combined construction and operating licence could be limited. By analogy with the recently adopted Federal Act on the Secure Supply of Electricity from Renewable Energy Sources, it could be stipulated that the interest in realising the nuclear power plant takes precedence over other national interests and conflicting interests of cantonal, regional or local importance. This would reduce the number of possible objections and thus the likelihood of costly project delays and changes.

9.3.2.3. Construction licence

To streamline the procedures, construction and operating licences could be combined. This is already principally possible today, provided that the conditions for safe operation can be conclusively assessed at the time the combined licence is granted. This requirement could be relaxed. The assessment of whether the conditions for safe operation are met would therefore not have to be finalised at the time of granting the combined licence. Any outstanding issues could also be addressed in a subsequent permitting procedure.

The final demonstration of safety, in particular that the generic type approval also covers the site-specific hazard assumptions, would then be the combined licence. New findings after the granting of the general licence do not have to be implemented immediately, but only after the commissioning of the nuclear power plant in accordance with the state of the art in backfitting technology.

Merging the construction and operating licences will reduce the design and construction time for a new nuclear power plant by two years, from 19 to 17 years. However, this relatively small time saving is not the main objective of the present proposal; it is much more important to establish the final requirements after the design phase and before the start of construction, and to remove the possibility of appeals that exists in the current procedure shortly before the nuclear power plant is commissioned. In addition, the proposed extension of the general licence reduces the number of reasons for appeal and thus limits the possibilities for objections to the combined construction and operating licence.

The examination of whether other grounds provided for in federal legislation, i.e. environmental protection, protection of the natural and cultural heritage and spatial planning, conflict with the project would be carried out conclusively as part of the general licence and is not to be repeated for the construction licence. In particular, this eliminates the need for an environmental impact assessment for the construction licence and thus the lengthy international approval process required under the Espoo Convention.

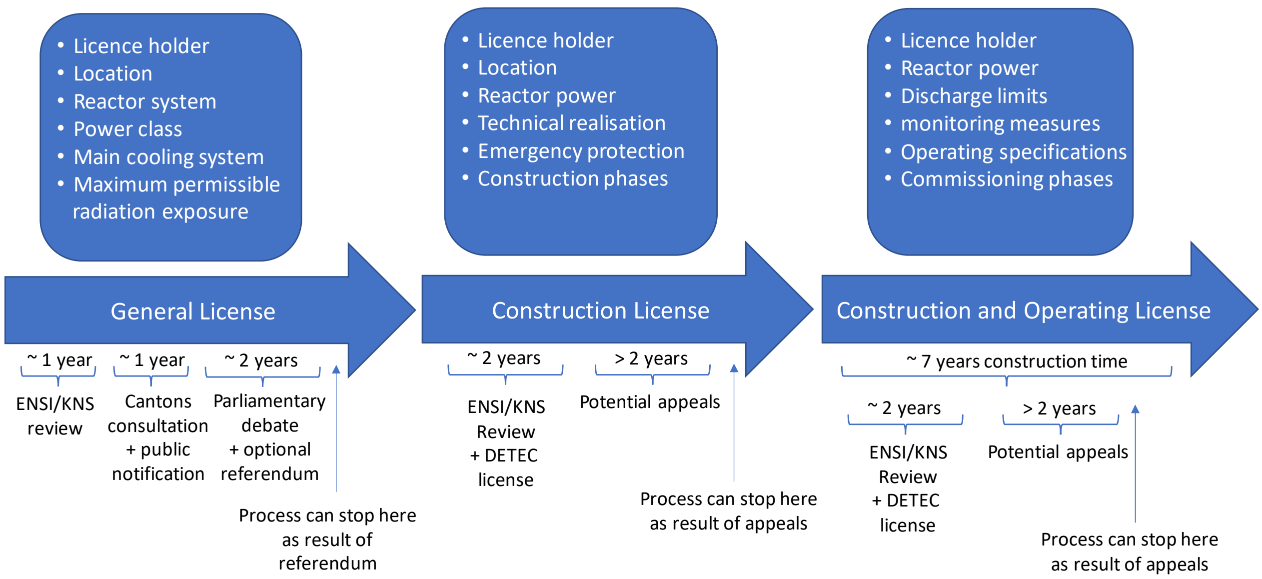

Figure 9.1 Current licensing procedure for a new NPP in Switzerland

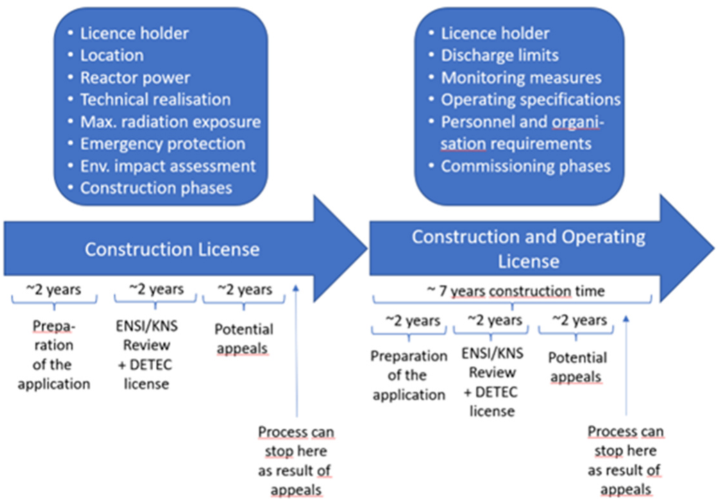

Figure 9.2 Streamlined licensing procedure for a new NPP without general license

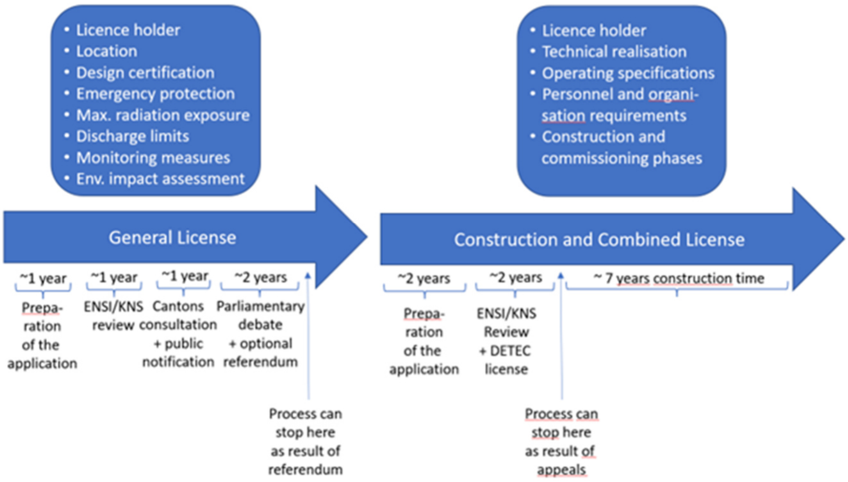

Figure 9.3 Streamlined licensing procedure for a new NPP with combined construction and operation license

10 State of fusion technology and main actors

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10.1 Introduction

Nuclear fusion is the process in which light elements fuse into heavier ones, thus releasing large amounts of nuclear binding energy. It is the opposite process to that used in today's fission plants, where heavy elements are split into lighter ones, and is the energy source of our Sun and all active stars. The goal of the worldwide fusion research effort is to master fusion on Earth and to develop it into a safe, clean, carbon-free, and essentially inexhaustible source for baseload electric power production to meet the needs of an environmentally- and climate-conscious society.

Figure 10.1 shows the basic principles of a future fusion reactor. It represents a reactor based on the tokamak concept, but its main principles apply to all concepts relying on the fusion of the hydrogen isotopes deuterium (D) and tritium (T). In the core of the reactor (the purple toroidal volume in Figure 10.1), D-T fusion reactions occur. Most of the released energy is carried away by high-energy neutrons. They then transfer their energy to a heat exchanger inside a so-called blanket surrounding the reactor core. The resulting heat is then used as in conventional reactors to run a steam turbine to generate electricity.

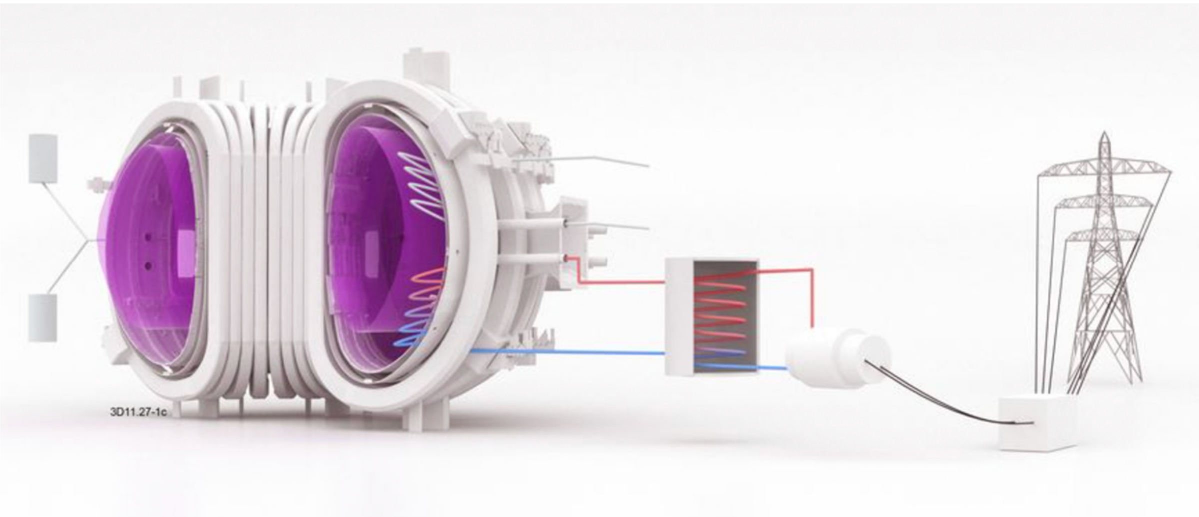

Figure 10.1: Scheme of a future fusion power plant based on deuterium-tritium fusion. High-energy neutrons produced in the reactor core (in this case based on the tokamak principle) transfer their energy to a heat exchanger within a meter-thick blanket surrounding the core. This heat is then used to produce electricity via a steam turbine (figure credit: EUROfusion).

For fusion reactions to occur, the positively charged nuclei of the atoms need to approach each other very closely, overcoming their mutual electrostatic repulsion. This requires the nuclei to undergo collisions at high energy, corresponding to temperatures of approximately 100 million degrees Kelvin. At such temperatures, matter is in the form of an ionized gas, the forth state of matter called a plasma. For this reason, fusion research is tightly linked to plasma physics[42].

The central task towards a fusion power plant is to generate and efficiently maintain a $\approx$100 million degree Kelvin plasma. This requires some sort of confinement, isolating the plasma from the surrounding structures. While gravity provides such confinement in the case of our Sun, the most advanced concepts in the laboratory rely either on the use of strong magnetic fields or on so-called inertial confinement, where small capsules of fusion fuel are strongly compressed and heated by symmetric irradiation, typically by powerful laser pulses, generating fusion plasmas that are solely confined (for a very short time) by the particles' inertia.

Since fusion research was declassified in 1958, tremendous progress has been made in increasing the fusion performance parameter called the triple product [10.2,10.3] and in other key aspects [10.4]. Significant difficulties in approaching fusion-relevant conditions have also been discovered, in particular related to plasma instabilities and turbulence. Many of these processes are today well understood and simulations to quantitatively model them on today's most powerful supercomputers are highly developed.

In magnetic-confinement fusion (MCF), the tokamak is generally considered the most advanced concept, demonstrating the highest performance to date. There is also a good understanding of the remaining challenges and risks, with a detailed roadmap towards a power plant relying on the construction and exploitation of the next step device ITER and research on smaller existing and future devices (see e.g. [10.4,10.5,10.22] for the European strategy towards a fusion power plant). In parallel, the stellarator is being pursued as a key competitor to the tokamak, featuring some important benefits. Inertial fusion energy (IFE) has also strongly progressed with an enormous increase in the laser-driven power and in particular the completion of the National Ignition Facility (NIF) in 2009, but the path towards a reactor seems less clear to date [10.6]. The temperatures required for fusion are routinely achieved and significant levels of fusion power have been demonstrated, notably in 1997 on the JET tokamak, reaching a peak thermal (not electric) power of 16 MW. To date, however, the total power invested to achieve fusion reactions is still significantly higher than the power released by the reaction. Key recent milestones in fusion research include the record fusion energy production achieved in the JET tokamak [10.7], confirming key ITER predictions, and the positive plasma energy balance achieved in NIF [10.8]. Furthermore, a record for long-pulse (1056 seconds), high-parameter plasma operation was achieved on the EAST superconducting tokamak [10.9] and a 1.3 gigajoule energy turnover (the product of heating power and plasma duration) was reached on the W7-X stellarator [10.10].

In parallel to these publicly funded activities, the fusion landscape has strongly evolved over the past few years with the emergence of a large number of private fusion initiatives. According to the 2023 report of the Fusion Industry Association (FIA) [10.11], there are currently 43 private fusion companies that have to date attracted a total of over $6 billion in investment. These private initiatives pursue a far more diverse set of concepts than public research, including compact tokamaks, spherical tokamaks, stellarators, field-reversed configurations,

[42] A brief, non-technical overview of this fascinating field of modern physics can be found in [10.1]

magnetic mirrors, Z-pinches, laser-driven inertial fusion, shock-driven inertial fusion, magnetized target fusion and other approaches, some of them even considering fuels other than D-T. A key driver for a number of these initiatives are recent developments in high-temperature superconductors (HTS), promising a significant increase in the achievable strength of the confining magnetic field, which increases the efficiency of plasma confinement. An important milestone in this direction has recently been reported by the company Commonwealth Fusion Systems (CFS) in collaboration with MIT, with the construction of a record 20 Tesla HTS magnetic field coil [10.12]. Recently, there has also been a development more and more towards public-private partnerships (see e.g. [10.4, 10.6]).

Clearly, fusion is an extremely attractive future energy source. It has a high level of safety (reactions can be stopped at any time and only ≈1g of fuel will be present in a reactor), it is clean (no $C0_2$, no long-term radioactive waste), the fusion fuels deuterium and lithium-6 are readily available for thousands of years, likely much more, and power production is continuous and independent of meteorological conditions. At the same time, 60-70 years of research have shown that it is extremely difficult to bring a plasma to the conditions necessary for a reactor and a number of technological challenges remain to be addressed, related in particular to tritium production and material lifetime in the presence of high-energy neutrons.

This chapter provides an overview of the current state of fusion science and technology (including a dedicated section on HTS developments), the ongoing worldwide activities in the public and private sector, recent developments, the remaining challenges and projections towards a reactor. A brief overview of the role of Switzerland in fusion science and technology and upcoming opportunities are also provided. Throughout this document, cross-reference is provided to recent assessments with a similar purpose conducted in the EU [10.6] and the US [10.13].

10.2 Fusion reactions of interest and fuel resources

Figure 10.2 shows the reaction cross-sections (essentially the reaction probabilities) as a function of ion energy for the most relevant fusion reactions. The fusion of the hydrogen isotopes deuterium (D) and tritium (T) stands out as a clear winner. Its maximum cross-section is at least five times higher than for other reactions and occurs at the lowest energies (1keV corresponds to approximately 10 million degrees Kelvin). For this reason, all public fusion activities and most private ones focus on this reaction, which generates a high energy helium particle (an alpha particle) and a high energy neutron, as follows:

$D + T \rightarrow {}^4He$ (3.5 MeV) + n (14.1 MeV) (1)

While deuterium is widely available (≈0.02% of hydrogen atoms on Earth are deuterium isotopes) and easily extractable from seawater, tritium is unstable with a half-time of 12.3 years and does, therefore, not exist naturally in relevant quantities. It will need to be produced directly inside a reactor, in a *blanket* surrounding the fusion plasma, using the neutrons produced in the D-T reactions. The most interesting reaction for this purpose is the following (to be combined with reactions to multiply, by a certain fraction, the neutrons from reaction (1))

$n + {}^6Li \rightarrow {}^4He$ (2.1 MeV) + T (2.7 MeV) (2)

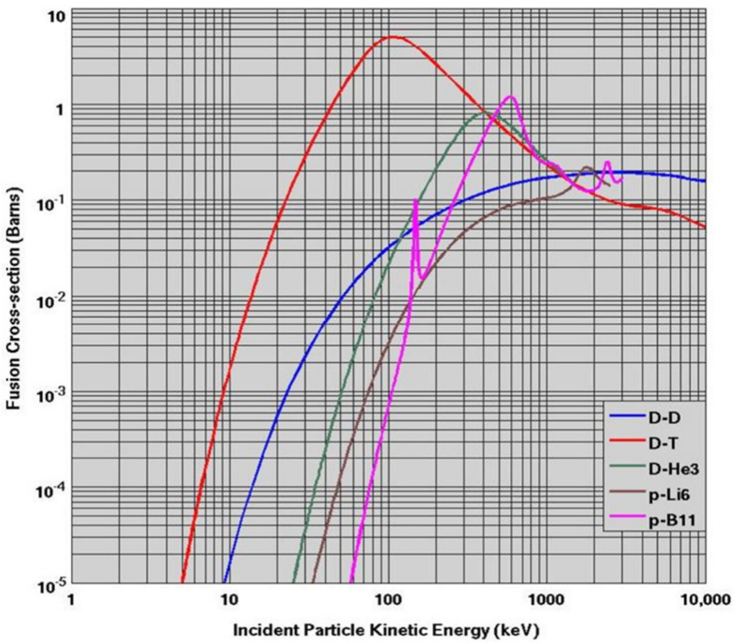

Figure 10.2: Cross-sections of various fusion reactions, as a function of the energy of an incident D or proton on a stationary target. The data for the D-D, D-T, D-He3 and p-^{6}Li curves are taken from the ENDF B-VII database [10.33] for incident deuterium/proton, while that for p-^{11}B is taken from Ref. [10.34]. The curve for D-D represents a sum over the cross-sections of the reaction branches. Figure reproduced from [10.35] with permission by the IAEA. The copyright remains with the IAEA.

Therefore, the fuels for D-T fusion are deuterium and lithium-6, both widely available on Earth. A future, 1GW fusion power plant will only consume approximately 130 kg deuterium and 400 kg lithium-6 per year[43]. Lithium, the limiting factor as far as fuel supply is concerned, is currently mined in the Earth's crust and reserves could meet the current worldwide energy demand for one to a few thousands of years [10.3,10.15]. With lithium extracted from sea water, this number would increase to millions of years.

The next most probable fusion reactions are D-^{3}He and p-^{11}B. These reactions have the advantage that they do not directly produce any neutrons[44], whose interaction with the reactor components constitutes a considerable challenge for D-T fusion, as discussed in Sec. 10.6. Furthermore, producing primarily charged particles, these reactions could potentially allow for a direct energy conversion that is more efficient than the conversion of heat into electricity. The important drawback of these fuels is that the required reactor conditions are much more challenging to achieve. Not only do they require considerably higher plasma temperatures, they also require much higher values of the already extremely challenging-to-achieve triple

[43] In comparison, a 1GW coal plant uses more than ten-thousand times as many kilograms of coal per day

[44] If the full reaction chain is considered, D-^{3}He fusion releases approximately 10 times less energy in the form of neutrons than D-T fusion, but strictly speaking is not aneutronic [10.14].

product for D-T fusion, as discussed in the next section. It should furthermore be noted that ^3He does not exist in relevant amounts on Earth (it is fairly abundant on the moon) and losses due to bremsstrahlung complicate p-^{11}B fusion further tremendously [10.3,10.30].

10.3 Figure of merit for fusion performance – the triple product

While huge amounts of fusion reactions can be easily achieved in the laboratory, e.g. by colliding two energetic particle beams, the key challenge consists in obtaining net energy gain. In particular, the energy needed to enable the fusion process should certainly be smaller than the energy released by fusion. The appropriate figure of merit to assess the closeness of a given fusion concept to reach net energy gain is the fusion triple product $n \cdot \tau_E \cdot T$, the product of ion density n, energy confinement time τ_E, and ion temperature T. If the fusion rate is high enough, the high-energy, charged fusion products (in the case of D-T fusion the alpha particles) can themselves provide sufficient plasma heating to guarantee fusion-relevant conditions without the need of external heating. Such a state is called ignition. For D-T fusion, assuming the alpha particles transfer all their energy to the D-T fuel mix, ignition is reached for the following condition on the triple product [10.3]:

$$n \cdot \tau_E \cdot T \geq 5 \cdot 10^{21} \text{ keV} \cdot \text{s/m}^3 \qquad (3)$$

with n, τ_E, and T expressed in m^{-3}, s and keV, respectively.

Ignition will not necessarily quite be needed in a power plant, at least for non-pulsed processes typical for MCF. Actually, some remaining heating requirements can give the operator an additional control knob. It is therefore useful to introduce the scientific power multiplication factor Q_{Sci}. It is defined as $Q_{Sci} = P_{fusion}/P_{external}$, with P_{fusion} the fusion power and $P_{external}$ the injected power to keep the plasma hot. $Q_{Sci}=1$ is called (scientific) breakeven and, for a non-pulsed approach, requires approximately 15% of the triple product value needed for ignition [10.3]. For $Q_{Sci} \geq 5$, alpha heating exceeds the external heating and one speaks of a burning plasma. As ignition is approached, Q_{Sci} goes to infinity. It should be mentioned that $Q_{Sci}=1$ has no practical meaning for a reactor, as efficiencies of the plasma heating systems, of the conversion from heat into electricity, and of other auxiliary systems need also to be considered in the total energy balance. Realistically, for a tokamak power plant, for instance, $Q_{Sci} \approx 30$ will be needed, corresponding to approximately 85% of the triple product for ignition [10.3].

Table 10.1 Summary of the achieved vs required triple product for various fusion concepts

	Achieved Triple Product [10^{21}keV·s/m³]	Triple Product required for reactor [10^{21}keV·s/m³]
Inertial fusion energy (NIF)	≈ 10	$\approx 50 - 500$
Tokamak (JET, JT-60U,…)	≈ 1	≈ 5
Stellarator (W7-X)	≈ 0.1	≈ 5
Others: FRC, Z-pinch,…	≈ 0.0001	≈ 5
Private sector, tokamak	≈ 0.01	≈ 5
Private sector, other approaches	≈ 0.0005	$\approx$ 5 for steady-state DT $\approx$ 50-1000 for steady-state alternative fuels

While the triple product is a robust figure of merit, the values of the individual factors can greatly vary depending on the concept (see also Table 10.1). A tokamak power plant is foreseen to operate at a temperature of ≈10keV, an energy confinement time of ≈5s and an ion density of ≈10^{20}m^{-3}, while IFE will operate at similar temperatures but at much lower confinement time of ≈10^{-9}s, which is compensated by a much higher ion density of ≈10^{31}m^{-3}.

Figure 10.3 presents, as a function of time, the progress in the achieved fusion triple product in publicly funded, magnetic confinement devices. An impressive progress is apparent for the tokamak from ≈1970 - ≈2000, starting from the T3 tokamak, one of the first tokamaks ever built and which by itself constituted a breakthrough at the time. Indicated is also the projection for ITER and the stellarator progress, which to date achieved peak values approximately a factor 10 below those of the tokamak. The highest values of the triple product have so far been achieved with inertial confinement at NIF, Figure 10.4, roughly 10 times above the tokamak. It should be noted, however, that IFE will require 10-100 times higher triple product values than magnetic confinement fusion, due to its pulsed nature, as the energy needed to get to fusion relevant conditions in the first place is not negligible [10.3].

Apparent from Figure 10.3 is that the impressive increase in the achieved triple product before 2020 could not be maintained beyond, with ITER targeting an increase of "only" a factor ≈5 and substantially later than past progress might suggest. A reason for this is the substantially larger device size when moving from JET (torus radius of 3m, copper magnets) to ITER (radius of 6.2m, with superconducting toroidal field magnets 24m in height), and also the substantially more ambitious scope of ITER on other key reactor-relevant scientific and technological aspects beyond the triple product. The flattening of the triple product curve in Figure 10.3 should also not be interpreted as a lack of progress towards fusion since 2000. As a matter of fact, tremendous progress has been achieved in the past 20 years in terms of plasma scenario development, in addressing the plasma-wall interaction challenge and the control of plasma transients, plasma modelling, design and fabrication of key elements (such as e.g. plasma facing components), design integration etc., with many of these aspects directly due to the ITER construction or driven by ITER needs.

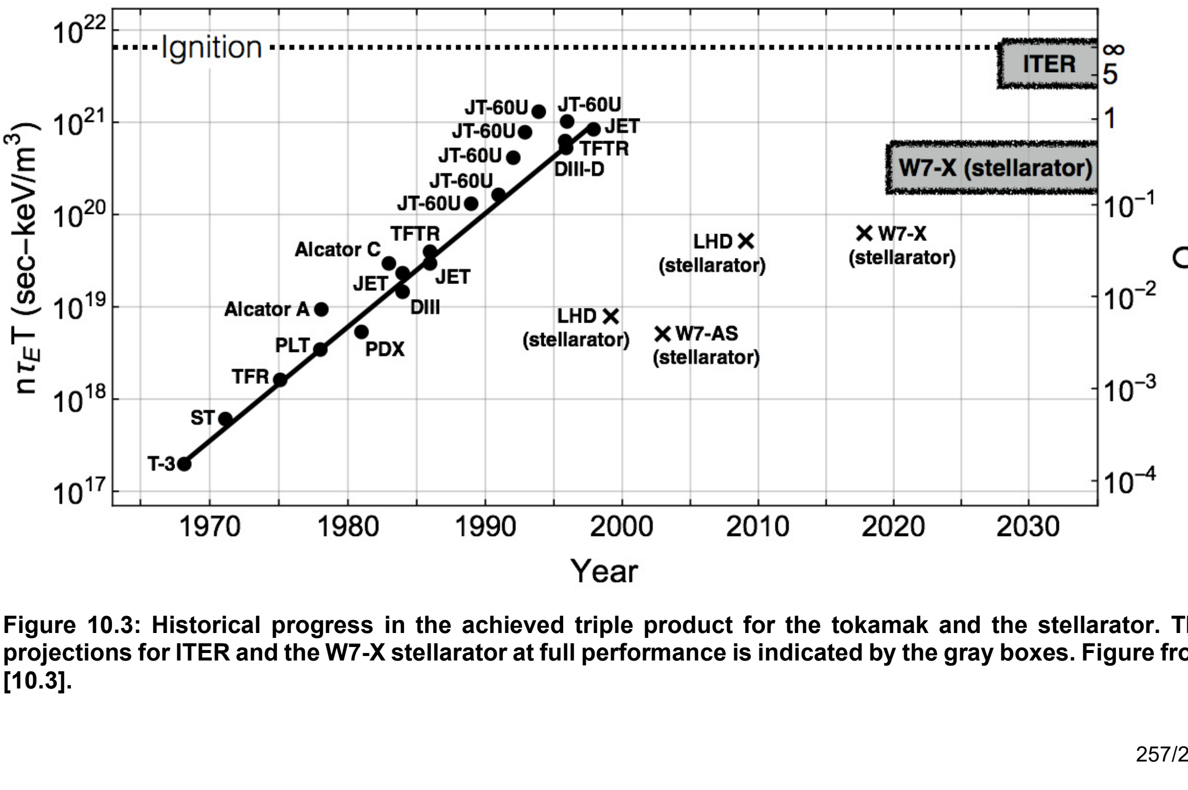

Figure 10.3: Historical progress in the achieved triple product for the tokamak and the stellarator. The projections for ITER and the W7-X stellarator at full performance is indicated by the gray boxes. Figure from [10.3].

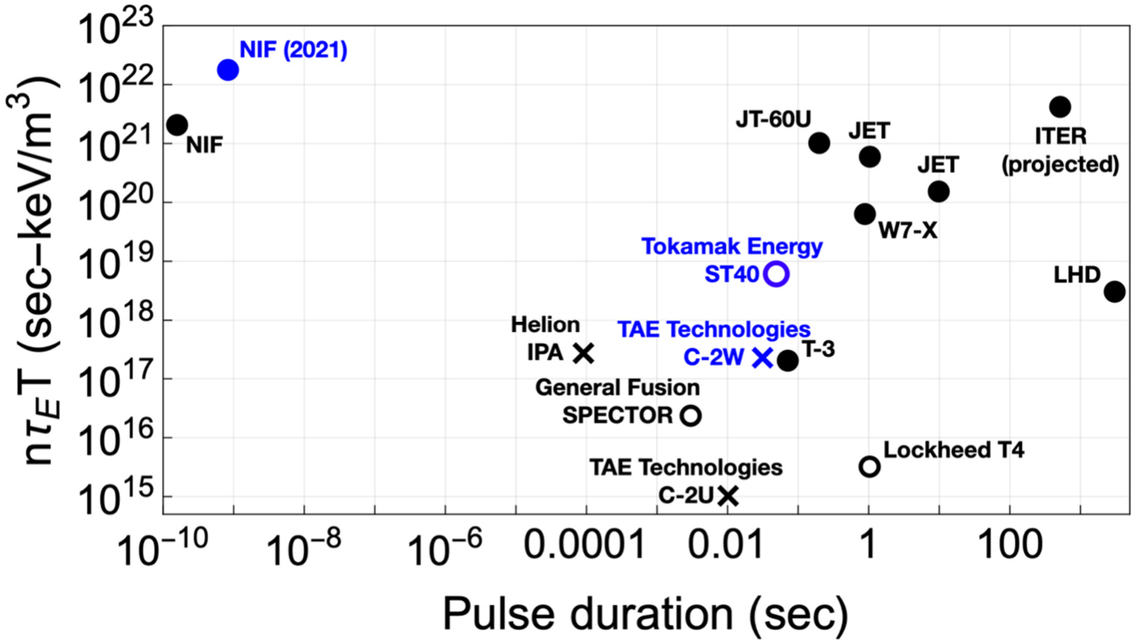

Figure 10.4: Triple product values published by some of the fusion startups compared to more conventional devices. Full circles represent publicly funded devices pursuing D-T fusion, open circles indicate private companies pursuing D-T fusion, and crosses indicated private companies pursuing alternative fuels. Devices with short pulse durations and alternative fuels require significantly higher triple products for energy production [10.3]. Recently published triple products [10.16-10.18] are marked in blue. Figure adapted from [10.3].

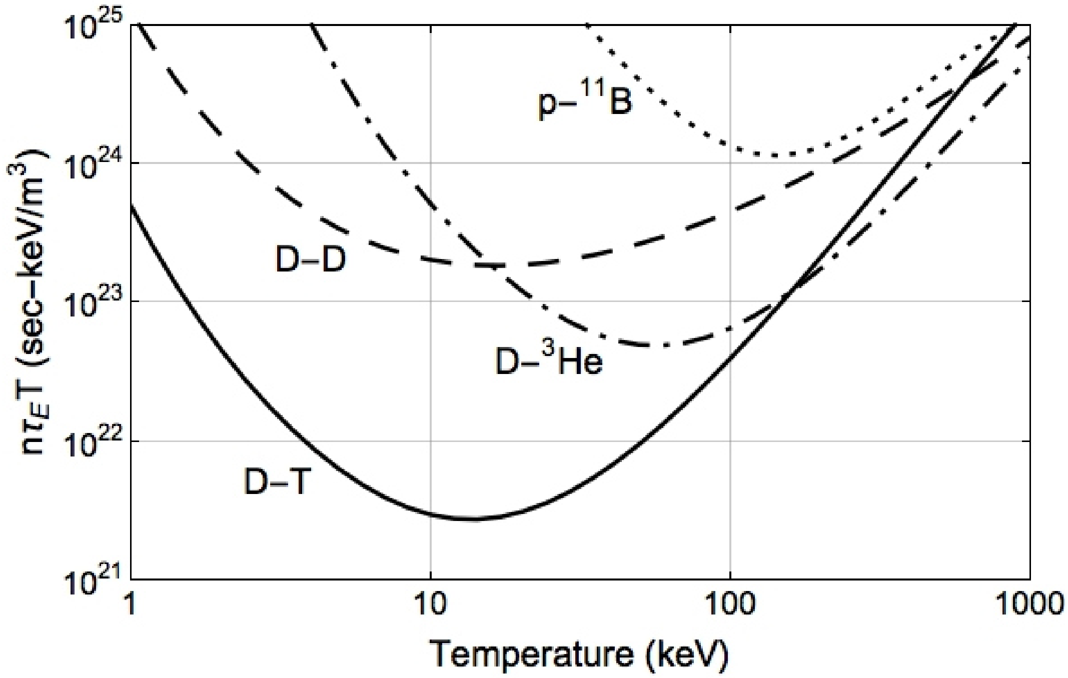

Figure 10.5: Required triple product for ignition as a function of ion temperature and for different fusion fuels. Bremsstrahlung losses are neglected here, which is not a good assumption for p-^{11}B fusion. Figure from [10.3].

To date, tokamaks and stellarators have clearly outperformed other magnetic confinement concepts, such as field reversed configurations and Z-pinches, which fall short of the tokamak in terms of triple product by a factor of ≈10’000 [10.13].

In Figure 10.4, we show triple product values from private companies (open circles and crosses), together with some key examples from government supported tokamaks and stellarators and from NIF. Tokamak Energy to date holds the record in triple product by a private company of $6 \cdot 10^{18}$keV·s·m^{-3}, achieved in its compact, spherical copper coil tokamak ST40.

Finally, we briefly discuss the triple products requirements for ignition with alternative fusion fuels, see Figure 10.5. Not only do the D-^{3}He and p-^{11}B reactions have their minimum at considerably higher temperature, they also require much higher values of the triple product. Furthermore, if bremsstrahlung losses are included (an effect neglected in Figure 10.5 and well justified for D-T fusion), net energy gain from p-^{11}B fusion becomes even more difficult [10.3,10.30]. Together with the fact that ^{3}He does not exist naturally on Earth in relevant quantities makes the D-^{3}He and p-^{11}B fusion reactions look unattractive at present.

10.4 Brief overview of different fusion concepts

There exists a variety of concepts pursued to achieve controlled nuclear fusion. They can mostly be classified into Magnetic Confinement Fusion (MCF), Inertial Fusion Energy (IFE), and Magneto-Inertial Fusion (MIF).

In MCF, strong magnetic fields are used to confine hot fusion plasmas for an extended time, essentially in steady state. The fusion triple product is typically targeted with a temperature of $\approx$10keV, an energy confinement time of $\approx$5s and an ion density of $\approx 10^{20}$m^{-3}. The pursued concepts include but are not limited to tokamaks, spherical tokamaks, stellarators, compact tori (field-reversed configurations (FRC) and spheromaks), Z-pinches, and magnetic mirrors. A tokamak confines the plasma in a torus-shaped volume, using a magnetic field generated by external magnetic field coils and by a strong current induced in the plasma by transformer action. The external field is directed along the toroidal direction (the long way around the torus). The smaller poloidal (the short way around the torus) field component, generated by the plasma current, provides twisted magnetic field lines, a necessity for a stable configuration. The spherical tokamak is a tokamak with a large cross-section relative to the device size. As such, it resembles more an apple with its core removed than a donut. The stellarator is a toroidal device where no current in the plasma is required to generate the twisted field lines. Instead, both toroidal and poloidal field components are generated by complex-shaped external magnetic field coils. Compact tori are toroidal configurations that are not interlocked by toroidal field coils. In a sense, they are extreme spherical tokamaks, where the hole in the torus is eliminated entirely and toroidal and poloidal fields are generated internally by plasma currents. If toroidal and poloidal field components are of the same order, one speaks of a spheromak. In an FRC, there is no toroidal field. In a Z-pinch, a strong current flows along a cylindrical plasma volume. This current produces an azimuthal magnetic field that compresses the plasma due to the **j**x**B** force. This compression heats and densifies the plasma, resulting in plasma confinement without the need for external magnetic field coils. Finally, magnetic mirrors are linear devices. A large fraction of the charged particles that would be lost along the magnetic field lines at both ends of a mirror device are instead reflected back by an increasing magnetic field at both ends. The fraction of particles trapped by this mirror effect depends on the mirror ratio, the ratio of the maximum and minimum magnetic field strength along a given field line.

A very different approach is used in IFE, which does not rely on magnetic fields. Instead, a small capsule filled with fusion fuel is strongly compressed to reach ignition, targeting for a temperature of $\approx$10keV, a confinement time of $\approx 10^{-9}$s, and an ion density of $\approx 10^{31}$m^{-3}. Different methods are explored to compress the capsule, such as precisely focused lasers (either directly or via a hohlraum producing symmetric X-ray radiation), ion beams, high-energy projectiles, or Z-pinches.

MIF are pulsed approaches targeting plasma density and energy confinement times intermediate between MCF and IFE. A magnetic field is used to confine a plasma, called the target. This target is then heated further by compressing it magnetically, with lasers, or using liquid or solid walls called a liner. As plasma target, FRCs or spheromaks are used, for instance.

Brief descriptions of these different concepts can be found in Ch. 2.4 of [10.13], Ch. 3.2 of [10.6], and [10.19], with more details provided in Ch. 9 of [10.3] and Ch. 9 and 10 of [10.14].

10.5 Overview of current fusion research and R&D activities and breakthroughs

This section presents an overview of the main public and private fusion programs and strategies. For complementary information, the reader is referred to Sec. 3.1 of [10.6]. A comprehensive list of the worldwide public and private fusion devices currently in operation, under construction or being planned, as well as technical data of these devices, is available in [10.20] and a yearly updated list of private fusion companies is published in [10.11].

10.5.1 Public programs

The next big step in public magnetic confinement research is the ITER tokamak [www.iter.org], under construction in the south of France. Its mission is to demonstrate the scientific and technological feasibility of fusion energy, in particular by achieving burning plasma operation for 400s continuously with a fusion power of 500MW and Q_{sci}=10, as well as producing a steady-state plasma with Q_{sci}=5. ITER will thus allow studying burning plasma operation and to test heating systems, diagnostics, cryogenics and remote handling at unprecedented scale. This also involves the constructing of a significant number of first-of-a-kind components and plans to test tritium breeding. ITER is an international endeavor between the EU, U.S., China, Russia, Japan, Korea, and India. With a major radius of its toroidal plasma volume of 6.2m, it will be twice as large, in linear dimension, as today's largest tokamak, the Joint European Torus (JET), located in the UK. ITER is widely considered to be the key step towards a first demonstration power plant, usually referred to as DEMO. The formal agreement to build ITER was signed in 2006 and construction started in 2010. Since 2016, first plasma was expected for 2025, but recent technical difficulties are expected to delay this date by several years [10.21].

Closely related to ITER is the JT-60SA tokamak in Japan [www.jt60sa.org], like ITER utilizing superconducting magnetic field coils. It is a joint effort between Japan and the EU under the Broader Approach program and has a major radius of 3.4m. Completed in 2020, one of the superconducting coils caused damage, resulting in a few years delay. JT-60SA is currently being commissioned and will support ITER operation and investigate how to optimize operation of future fusion power plants. Key research questions for ITER and DEMO are currently being explored on a number of existing tokamaks. The largest and most powerful tokamak to date is the JET device [euro-fusion.org/devices/jet/] with a major radius of 3m. It was the first device to operate with a 50-50 mix of D-T and in 1997 achieved a record Q_{Sci} of 0.67, with 16MW of fusion power produced for 24MW of power injected into the plasma. JET is scheduled to reach end of operation by December 2023. Other operating tokamaks include the superconducting devices EAST (China), K-STAR (South-Korea), WEST (France), and SST-1 (India) and the copper-coil devices DIII-D (U.S.), NSTX-U (U.S.), AUG (Germany), MAST-U (UK), TCV

(Switzerland), Compass (Czech Republic, currently undergoing a large upgrade), HL-2M (China), and ADITJY-U (India). For a full list, see [10.20].

The largest stellarator devices are W7-X (Germany) [www.ipp.mpg.de/w7x] and LHD (Japan), both superconducting. They are complemented by a number of smaller devices, such as TJ-K (Germany) and TJ-II (Spain), see [10.20] for a full list. W7-X is operational since 2015. It is the most advanced stellarator and holds the world record for the stellarator fusion triple product. Its mission is to study the suitability of the stellarator concept for a power plant and aims eventually for 30 min long plasma operation.

The leading inertial fusion device is the National Ignition Facility (NIF) in the U.S. [lasers.llnl.gov/about/what-is-nif]. It indirectly drives D-T capsules with 192 lasers, providing a total of 1.9 megajoule of energy. In 2022, it achieved the first ever scientific breakeven with Q_{Sci}=1.5 [10.8]. Contrary to magnetic confinement fusion research, which targets peaceful use of fusion for mass energy production without any military interest, the main mission of NIF is to support nuclear weapon maintenance, with the fusion part constituting a relatively small fraction of the program.

As far as country strategies and programs are concerned, the **European strategy** is based on the EUROfusion program and its roadmap to the realization of fusion energy [10.5]. This roadmap lays out in detail how research on ITER and on present and planned facilities, analysis, and modelling should enable to resolve the key remaining challenges for the construction of a complete demonstration fusion power plant DEMO. These present facilities include in particular JT-60SA, JET, and the medium sized tokamaks AUG, MAST-U, TCV, and WEST. It also includes the divertor tokamak test facility DTT [dtt-project.it] under construction in Italy, with the main aim being the exploration, in near-reactor-like conditions, of alternative solutions for the exhaust of the plasma particles and heat generated by the fusion process. On the material side, the roadmap relies heavily on the International Fusion Materials Irradiation Facility - DEMO Oriented Neutron Source (IFMIF-DONES, ifmif-dones.es) for material testing, in addition to other dedicated test facilities, such as EDIPO/SULTAN in Switzerland for research and qualification of superconductors for fusion (more on this in Sec. 10.9). IFMIF-DONES is about to be launched in Spain to provide a center for testing, validation and qualification of materials for fusion subject to the irradiation with 14MeV neutrons. A key part of the roadmap is also the continued assessment of the stellarator concept with WX-7, as a long-term alternative to the tokamak. Currently, the EUROfusion roadmap is being revised [10.22], in particular as "*the urgency to develop DEMO before the middle of the century, after which a fusion reactor economy can begin, requires us to proceed as much as possible parallel to ITER, rather than adopting a sequential approach that fully depends on ITER milestones*" [10.4] and as "*It is crucial to strike a balance between consolidated knowledge and innovation, to explore higher-risk higher-potential solutions than have been undertaken so far, and to ramp up public-private partnerships, as DEMO will inevitably be built in an industrial frame with fully industrial practices.*" [10.4]. The European program has also a dedicated DEMO design team.

European fusion research is organized by the EUROfusion consortium, consisting of $\approx$30 member research organizations and associated entities across Europe. The European contribution to ITER (Europe finances 45% of the costs) is provided by the EU Domestic Agency Fusion for energy (F4E).

Regarding the strategies of other countries, a brief overview is presented in the following, referring to Sec. 3.1 of [10.6] and references therein for more details. The **United Kingdom**, besides participating to the EUROfusion program and hosting JET and MAST-U, pursues its own national fusion strategy. This involves the development of the Spherical Tokamak for Energy Production (STEP), managed by the UK Atomic Energy Agency (UKAEA), which will act as a pilot plant based on the spherical tokamak approach. The UK moves also quickly in terms of governance and policy coordination, generating favorable conditions for both public and private programs [10.6]. The **United States**, besides its contribution to ITER, is operating NIF, the DIII-D and NSTX-U tokamaks, and additional, smaller devices located at universities and national labs. The new strategy of the U.S. was presented at the first Fusion White House Summit in March 2022, where the Department of Energy announced an agency-wide initiative to accelerate the development and commercialization of fusion energy and to stimulate public-private partnerships [10.6,10.13]. Significant increases in the annual funding of the Fusion Energy Science program have also been appropriated [10.6,10.13]. The U.S. approach to public-private partnerships foresees a large part of the responsibility and leadership to be taken up by the private sector, with support from the public institutions, including in particular access to their IP. The instrument for this, called INFUSE (Innovation Network for Fusion Energy), is already in place [https://infuse.ornl.gov]. **China**, besides also participating in ITER and operating EAST, pursues additional initiatives with CRAFT (a facility serving a comparable purpose as IFMIF-DONES) and the tokamak devices BEST and CFETR. BEST, currently in the design phase, is intended to demonstrate a burning plasma and to support CFETR [10.6], the Chinese equivalent to the EU-DEMO. **Japan** pursues a similar roadmap to fusion as Europe [10.6], with JT-60SA (jointly with Europe) and LHD (the second largest stellarator) its main experimental facilities. The **South-Korean** fusion roadmap is also quite similar to that of Europe [10.6], with its current facility K-STAR, ITER and a future Korean DEMO. **Russia**, aside from participating in ITER, proposed a fission-fusion hybrid. **India** is operating the SST-1 and ADITYA-U tokamaks, participates in ITER, and plans for a medium-sized fusion reactor (SST2), which is in the design phase. It is the next device to be realized in the Indian fusion roadmap and planned to be built with existing technologies and materials. The tritium fuel cycle handling is planned to be established in this device and it should serve as the testbed for qualifying various reactor concepts and technologies which will eventually be considered for an Indian DEMO.

10.5.2 Private initiatives

Over the past years, there has been an impressive increase in the number and size of private fusion initiatives. The report of the Fusion Industry Association (FIA) of July 2023 [10.11] includes a total of 43 companies, having together attracted over $6 billion in investment. 25 of these companies, combining the bulk of the investment, are located in the U.S., but there are companies with large investments also e.g. in the UK, China, and Germany (see [10.11] for a brief description of each company identified by the FIA). Some companies exist for approx. two decades, such as TAE Technologies (founded in 1998) and General Fusion (2002), while others are much more recent, such as Commonwealth Fusion Systems (2018).

These companies pursue a much broader set of approaches to fusion than the public sector, with a number of them considering even fuels other than D-T. Fig 10.6, reproduced from [10.6], presents an overview of the different fusion approaches, with key private initiatives indicated in blue. While some of them build on well-established concepts, where strengths and drawbacks are rather well understood, and others count more on new discoveries, the majority of concepts in MCF and MIF have in common that they rely on the potential of new, high-temperature superconductors (HTS) to operate at significantly higher magnetic field strengths (see chapter 10.8 dedicated on this topic).

To date, none of the private initiatives have demonstrated operation near what is needed for a fusion reactor. In particular, the record triple product achieved by a private company so far ($6{\cdot}10^{18}$keV·s·m^{-3} by Tokamak Energy [10.17]) is still a factor of $\approx$1000 below what is needed for a power plant. In light of this, the fact that according to an FIA poll (page 11 of [10.11]), 19 fusion startups anticipate their company to put fusion power to the grid by 2035, appears over optimistic. Indeed, such timelines should certainly be met with a lot of care, as confirmed also by the FIA's UK director of communication who, according to [10.23], stated that "*Timelines that companies project should be regarded not so much as promises but as motivational aspirations*". With the caveats regarding the timeline of private fusion companies in mind, it is clear that a fast-emerging industry, coupled to an increase in public-private partnerships, can develop its own dynamics and potentially significantly accelerate progress.

Highlights in experimental successes of private companies to date include successful demonstration by Commonwealth Fusion Systems and the MIT Plasma Science and Fusion Center of a first HTS magnetic field coil producing 20T [10.12] and the achievement of 100 million degree ion temperature plasmas by Tokamak Energy in their latest prototype ST40 (a copper coils device) and by Helion Energy in their pulsed FRC device, as well as 75 million degrees in the FRC of TAE Energies [10.6]. Referring to the results from Tokamak Energy, Ref. [10.6] remarks that "*Whilst the temperature demonstration is unremarkable for public programmes that regularly work with such regimes, it was a major first for a private fusion firm, highlighting the significant and rapid progress being made*".

In the following, we list some of the key private initiatives, very briefly mentioning benefits and challenges associated with the chosen concepts.

Approach	Initiatives
MFE / MCF	
Tokamak	ITER, JET, JT-60SA (JP-EU), K-STAR (KO), DIII-D (US), EAST (CN) CFS
Spherical tokamak	MAST-U (UK), STEP (UK), NSTX-U (US), Tokamak Energy, ENN
Stellarator	W7-X (EU), LHD (JP), Renaissance Fusion, Type One Energy
Z-Pinch	Zap Energy, MIFTI
Compact Toroids (Field Reversed Configurations [FRC], Spheromak)	TAE Technologies, CT Fusion
IFE Including various direct, indirect and target-based approaches	NIF, Marvel Fusion, First Light Fusion, HB11, Focused Energy
MIF Including Magnetised Target Fusion (MTF), FRC-based, and other approaches	Helion Energy, General Fusion

Figure 10.6: List of different approaches to fusion, with main public (black) and private (blue) devices or companies indicated for each approach. Figure from [10.6].

Commonwealth Fusion Systems (CFS, cfs.energy), established in 2018 in Massachusetts (U.S.) by a group of academic fusion scientist from MIT, is pursuing the approach probably closest to the main public fusion research line of any private company. Building upon decades of tokamak research, the company's goal is to use new HTS to develop higher field, more compact fusion reactors based on the tokamak concept. After their successful demonstration of a first HTS magnetic field coil at 20T, CFS has attracted over $2 billion in funding for the construction of its first device SPARC, foreseen to start operation in 2025 and to demonstrate operation well beyond scientific breakeven ($Q_{Sci}\approx10$). In a next step, CFS foresees to build a first fusion power plant in the 2030s, called ARC, with a size similar to JET. By many colleagues in the field, CFS is considered to be the most promising private fusion initiative. An approach similar to that of CFS seems to be pursued by Energy Singularity [energysingularity.cn/en/], founded in Shanghai, China in 2021 with a total funding to date of $112 million. An (advanced) tokamak concept is also pursued by General Atomics, which is operating the DIII-D tokamak and started a commercial fusion effort in 2022 with $113 million of declared funding to date [10.11].

Advantages of the compact tokamak are that the higher magnetic fields achievable with HTS should allow significantly smaller, cheaper, and easier to build devices. However, the technology and the mastering of the enormous stresses at these large fields still need to be proven. The so-called heat exhaust challenge might also be more challenging that in an ITER-like device.

Two companies listed in the FIA 2023 report pursue another more conventional concept, the spherical tokamak. The first one is Tokamak Energy [tokamakenergy.co.uk] in Oxford, UK, founded in 2009 and with a declared funding to date of $250 million, complementing the public focus in the UK on the spherical tokamak mentioned above. The second company is ENN [10.11] in Langfang, China, founded in 2018 and with a declared funding to date of $400 million. While Tokamak Energy aims at D-T fusion, ENN focuses on p-^{11}B fusion. Both companies work with HTS. As mentioned above, Tokamak Energy holds the record in fusion triple product achieved by a private company. While the company reports progress in HTS prototyping, the above record was achieved in a copper-coil device.

Advantages of the spherical tokamak approach are that this geometry is known to improve the efficiency of the magnetic field which, if further combined with high-field HTS magnets, promises more compact devices. The same HTS-related difficulties as for the compact tokamak above apply. In addition, it is an open question if the “donut hole” can really be made small enough and still incorporating all the necessary components, such as breeding blanket etc. [10.3], or if a full spherical tokamak reactor will rather look more like an ordinary tokamak.

There are a number of companies pursuing the stellarator approach, also heavily relying on HTS magnets and other technological advances as well as exploring the wide geometrical parameter space of possible stellarator solutions. Aside from Renaissance Fusion (Fontaine, France, 2021, $17 million) and Type One Energy (Wisconsin, U.S., 2019, $30 million) listed in Figure 11.6, there are also Stellarex (Princeton, U.S., 2022), THEA Energy (Princeton, U.S., 2022, $23 million), Helical Fusion (Tokyo, Japan, 2021, $6.5 million), NT-TAO (Hod Hasharon, Israel, 2019, $28 million), and Proxima Fusion (Munich, Germany, 2023, $8 million).

The key advantages of the stellarator over the tokamak are the fact that no plasma current is needed to ensure plasma confinement. This makes the stellarator intrinsically steady-state and avoids current-driven instabilities, such as violent plasma disruptions. A challenge is the confinement of fast ions and that triple product values demonstrated by public devices are still a factor $\approx$10 below those of the tokamak. There are also significant engineering challenges in constructing the complex stellarator coils and geometry. All these companies are very recent and, to our knowledge, no device has been built yet.

The leading company exploring the Z-pinch seems to be ZAP ENERGY (Washington, U.S., 2017, $208 million), relying on a pulsed approach in a “shear-flow-stabilized” Z-pinch.

Z-pinches have the compelling benefits of requiring neither superconducting coils nor lasers, eliminating some of the challenges of other concepts. They have been investigated since the 1950s and were generally found to be subject to strong plasma instabilities, resulting in a quick radial loss of the plasma [10.14], such that significant progress is clearly needed before reactor-concepts can be envisaged.

The largest company pursuing research on compact tori is TAE Technologies (California, U.S., 1998, ≥$1.2 billion). Their approach is to sustain an FRC in steady state via current drive with neutral beam injection. While pursuing p-^{11}B fusion, other fuels can also be accommodated. In their current prototype C-2W, temperatures of approx. 75 million degree Kevin were achieved. The triple products in C-2W seem still to be of the order of the T-3 tokamak from 1968, see Figure 10.4. The goal of their next step device Copernicus is to reach the D-T breakeven point [10.11]. Compact tori can make much better use of the magnetic field (allowing for higher plasma beta, the ratio of plasma pressure to magnetic field pressure) than tokamaks. They are, however, difficult to sustain and confinement has, to date, been much lower than in tokamaks, but they have so far also not been investigated very intensively.

Another concept that has recently gained traction again is the magnetic mirror concept. Realta Fusion (Wisconsin, U.S., 2022, $12 million) aims to leverage new HTS technologies to confine the plasma with higher magnetic fields and mirror ratios than previously possible.

Private companies pursuing the IFE approach with lasers primarily focus on direct drive, so without a hohlraum as used in NIF. Both D-T and P-^{11}B fusion are considered. The largest companies are Focused Energy (Texas, U.S. and Darmstadt, Germany, 2021, $82 million) and Marvel Fusion (Munich, Germany, 2019, $112 million). Both explore new approaches aiming for more efficient compression, namely a two-staged approach (Focused Energy) and pellets with a special nanostructure (Marvel Fusion). The company First Light Fusion (Oxford, UK, 2011, $97 million) is exploring the use of high-velocity projectiles rather than lasers to compress the fuel capsule. Their next device, targeting construction in 2024, aims at demonstrating net energy gain [10.24].

In general, an advantage of IFE is that the approach is completely different from MCF, thus constituting essentially an independent second approach to fusion. In particular, no powerful magnets and the associated cryogenics are needed. Challenges are the coupling of sufficient energy to the target, in an efficient manner (considering also the typically low electric efficiency of the lasers), the need for very high gain factors (triple products 10-100 times larger than for MCF), the cost of the precisely manufactured targets (which will need to drop by many orders of magnitude), and the high repetition rate that is needed, going from order one pulse per day in NIF to 10 pulses per second. Overall, comparing the status of MCF and IFE, Ref. [10.6] concludes that “The medium and long-term view of the system is less clear for IFE than MFE based approaches” and “…it would be fair to say that the starting point is much less developed for IFE compared to tokamaks”.

The largest companies pursuing MIF are General Fusion (Vancouver, Canada; London, UK; Tennessee, U.S., 2002, $300 million) and Helion Energy (Washington, U.S., 2013, $577 million). General Fusion aims to strongly compress a magnetized target (e.g. a spheromak) by a liquid metal, a mix of lithium and lead, at high repetition rate. A very compelling feature is that the lithium-lead would also act as the blanket (heat absorption, neutron shielding, tritium breeding). A challenge is to achieve sufficient compression and plasma lifetime and the risk that the lead contaminates and dilutes the fusion fuel.

Helion aims to generate two FRCs and to collide them and further compress them magnetically. Their aim is to do D-^{3}He fusion. Helion, according to [10.6], achieved ion temperatures of 100 million degrees Kelvin. Achieved triple products, according to Ref. [10.3] (Figure 10.4) are currently at the level of the T-3 tokamak from 1968. As [10.6] notes, “*Details on the MIF-FRC approach are somewhat limited, as the main proponent, whilst highly ambitious is also quite secretive, therefore there are considerable uncertainties surrounding the concept*”. Unusual is also the use of D-^{3}He fusion, with ^{3}He being very scarce on Earth, and ^{3}He production from D-D appears very challenging, considering the difficulty of D-D fusion, Figure 10.5.

While most of the above-mentioned private fusion companies pursue their own technology developments, there are also companies specifically focusing on the development of advanced technologies for fusion. In particular, the FIA report [10.11] lists Kyoto Fusioneering (headquater in Tokyo, Japan, established in 2019, investment $91 million) focusing on technologies for plasma heating, the fusion fuel cycle, and energy conversion.

We conclude this section by mentioning that many of the private fusion companies have collaborations and ties to public institutions [10.11]. Ref. [10.11] also reports on significant new public-private partnership (PPP) programs (18 companies reported that they are or will be involved in PPP).

10.6 Key challenges (approach-independent)

In this section, we briefly discuss what we consider to be the most important remaining, approach-independent challenges towards a fusion reactor. These challenges apply to all the approaches to fusion discussed above that rely on D-T fusion. We do not, here, discuss D-^{3}He, p-^{11}B or D-D fusion, as they are much harder to achieve and have significant additional drawbacks/roadblocks, see Secs. 10.2 and 10.3.

More information on the remaining challenges can e.g. be found in [10.4,10.5,10.6,10.13].

Demonstration of $Q_{Sci} \gg 1$. Despite large progress in the fusion triple product over the years in both MCF and IFE, no approach has yet proven $Q_{Sci} \gg 1$, which is required for net energy gain, including conversion efficiencies of all the auxiliary systems of a power plant. NIF has so far reached the highest Q_{Sci} of $\approx$1.5, yet with difficulties to routinely reproduce similar success as in their record shots. The tokamak record achieved in JET is at $Q_{Sci} \approx 0.67$. In case of the tokamak, the recipe to reach $Q_{Sci} \gg 1$ is probably the clearest, requiring a larger device than JET, higher magnetic field and/or improved plasma scenarios. These paths are pursued in particular with ITER, relying on LTS magnets and a size twice that of JET, and with CFS, STEP, Tokamak Energy and other initiatives aiming for HTS magnets and more compact designs. The development of improved plasma scenarios is a large focus also on current day devices.

Plasma instabilities, specifically in the presence of fast particles. Fusion plasmas are complex, self-organized systems and although different from approach to approach, they are all subject to instabilities and turbulence, making it difficult to optimize and predict plasma confinement. Numerical models are very advanced but still do not fully predict the experiment. Furthermore, little is yet known about the behavior of a burning plasma, i.e., a plasma primarily heated by the energetic alpha particles, and plasma might behave differently in these conditions.

Materials, in particular in direct contact with the plasma. These materials have to withstand harsh environments for extended periods. This includes thermal stresses that can cause cracking and particle bombardment which can lead e.g. to surface modifications or brittleness, such as in the case of helium implanted in tungsten.

A particular challenge is also the behavior of **structural materials under bombardment of the high-energy neutrons** produced in D-T fusion. Degradation of the mechanical and thermal properties of the materials by atomic displacement cascades need to be limited while assuring transmutation into short-lived radioactive isotopes only. This limits the pool of possible material choices. Candidate materials exist, such as EUROFER, but extensive experimental testing is needed under 14MeV neutron bombardment. This requires dedicated facilities, such as IFMIF-DONES discussed in Sec. 10.5.

A particularly technologically complex component is the **breeding blanket** surrounding the fusion plasma. It will need to have a thickness of at least one meter to provide adequate neutron shielding. It will furthermore need to assure tritium breeding at sufficient rate and to harness the fusion power. Urgent R&D is needed to develop the concept of a blanket [10.4], in particular considering that the worldwide supply of tritium is very limited, requiring a fusion power plant to start with a full breeding blanket in place.

Finally, in addition to these scientific and technological challenges, key challenges are also related to the successful **integration** of all the aspects required to build and operate a fusion power plant as well as **staffing**, **supply chains**, and **fusion-specific licensing regulations**.

10.7 Key challenges (approach-dependent)

While the key challenges discussed in Sec. 10.6 apply to all D-T fusion approaches, we briefly discuss in this section the key approach-dependent challenges. We limit the discussion to the concepts that are, in our opinion, the most advanced and for which the whole set of challenges has been assessed in detail, namely the tokamak and the stellarator. Specific challenges related to other concepts have briefly been alluded to in Sec. 10.5.2., but are less well assessed to date. For more details related to the specific challenges associated with tokamaks and stellarators, we refer the reader to [10.4,10.5].

A key challenge for tokamaks, stellarators, and most magnetic confinement concepts is **the concentrated heat flux to localized wall components**, as a result of the continuous loss of particles and heat from the confined region and subsequent rapid transport along the magnetic field lines intercepting reactor wall components. These heat fluxes are dealt with in a dedicated region called the divertor, whose role is also to assure adequate pumping of the spent fuel and to avoid excessive contamination of the core plasma by plasma species other than D-T, such as the wall material. The associated heat fluxes to the wall structures in the divertor are generally acceptable in today's devices, but the problem becomes considerably more challenging when extrapolated to a reactor with higher power and larger device size and/or higher magnetic field strength. A *detached* divertor, where most of the power is radiated in an isotropic manner by volumetric processes, is the foreseen solution, but extrapolation to a reactor remains highly uncertain. As a result, this is one of the key challenges actively studied on today's devices and with simulations, including the investigation of alternative geometrical solutions.

Significant tokamak-specific challenges are related to the need for a strong plasma current. A result is the existence of violent **plasma disruptions**, where the entire plasma becomes globally unstable and is lost on a millisecond timescale, and subsquent **run-away electron beam** generation. Disruptions are significantly more challenging to deal with in next-step devices and need to be avoided or their damage mitigated. This is also a very active area of research on contemporary devices and several possible solutions exist. The plasma current is further the source of other plasma transients that need to be controlled and, for steady-state operation, current drive methods other than via transformer action need to be further developed.

The stellarator circumvents all issues related to the plasma current, including the risk of disruptions. Instead, aside from the need to yet reach the performance of tokamaks,

stellarators face the challenge of assuring **adequate fast ion confinement** in their complex, 3D geometry, and the complexity in construction and maintenance.

10.8 Superconducting magnet technology and prospects of HTS

The plasma in a tokamak or any other fusion device, heated up to ≈100 million Kelvin, must not touch the walls of the reactor. In MCF, the plasma is therefore held in the center of the fusion reactor by strong magnetic fields. In small and medium size tokamaks, this can be achieved by conventional copper magnets. However, the resistive nature of copper leads to Ohmic heating in the coil winding, which limits the pulse duration. Once the temperature of the coils exceed some reasonable limit, the fusion pulse must be stopped. The limitations become more and more severe with increasing size of the reactor. In a medium-size tokamak, even if the copper coils are pre-cooled by liquid nitrogen to 77 K, the pulse duration is typically limited to just a few seconds. For the largest machines, like ITER or EUROfusion DEMO, it is impossible to generate the required magnetic field with copper coils. There is simply not enough space around the reactor chamber to put sufficient amounts of copper windings. The electromagnets must therefore be manufactured out of superconductors. Even the medium scale machines benefit from the usage of superconductors, as the duration of the plasma discharge is then not limited by the increasing temperature of the coils, and the pulse duration can be extended to hours instead of seconds (the pulse duration is then limited by other reasons, not anymore by the magnets).

The main challenges of the fusion superconductor magnets are:

1. necessity to cool them down to very low temperatures (4-20 K) and to shield them very well from the heat and neutron flux arising from the fusion reaction.
2. necessity to monitor the magnets for quench (the transition from the superconducting state into the normal state) and having a quench protection system that switches off the current in the magnet as soon as quench is detected (within a few seconds), otherwise the magnet could be damaged.
3. extremely high electromagnet forces acting on the magnets in operation, for which very complex mechanical support structures are needed.
4. high price of the superconductors, high manufacturing complexity and high manufacturing precision (tolerances of ≈1mm on ≈10m size components).
5. electrical insulation. Even though there is no voltage in the superconducting coils during normal operation, several kV voltage is induced on the coil during the fast current switch off when quench is detected. The electrical insulation must be robust to withstand this high voltage, enormous forces (in some coils repetitively) and thermo-mechanical stresses during quench.

In the following, a brief overview of the current state of superconductor technologies will be presented. Particular focus is put on recent developments in high temperature superconductors, which have the potential to allow for higher field, more compact fusion devices. These prospects include but are not limited to tokamaks and stellarators. For tokamaks, the expected benefits are in particular related to a general increase in the plasma operational limits with magnetic field strength B, the fusion power P_{fusion} that can be generated in a machine of a given size that increases as $P_{fusion}=B^4$ [10.25], and the fusion triple product that scales as B^a, with $a\geq 2$ [10.26]. Challenges are associated with mastering the enormous stresses occurring at such high magnetic field.

10.8.1 LTS magnets

In the presently existing machines (JT60-SA, Wendelstein W7-X, …) and most machines that are under construction (ITER), the superconductors of the magnets are the so-called Low Temperature Superconductors (LTS), which operate at the temperatures close to the temperature of liquid helium, say 4.5-6K. The employed superconductors are NbTi and Nb_3Sn. The maximum magnetic field that can be reached with LTS is 5-6T on the plasma axis, which corresponds to 12-13T in the superconductor inside the coil winding pack. Even though some R&D is still ongoing in order to improve the performance and reliability of LTS magnets, the LTS technology is considered to be mature and manageable by industry (though only a few companies world-wide are able to build ITER-like fusion magnets).

10.8.2 HTS magnets

The name High Temperature Superconductors (HTS) suggests that these materials are superconducting at higher temperatures compared to LTS. Indeed, the critical temperature T_c of HTS is mostly above the boiling temperature of nitrogen, 77 K. However, at these temperatures, HTS cannot withstand high magnetic fields and high electric currents. Only when cooled down to temperatures below or equal to approx. 20 K, the HTS coils can operate at fields of 20 T or more in the winding pack, corresponding to $\approx$10 T on the plasma axis, i.e. at twice higher magnetic field compared to the LTS magnets. This feature makes the HTS materials superior to LTS in fusion machines, as tokamaks of much smaller size, the so-called compact tokamaks, become conceivable. The question is, why all previous machines were made of LTS and not from HTS? There are several challenges of the HTS technology:

1. low production capacity of HTS materials. The most practical HTS are produced in the form of tapes, and until $\approx$2010, the manufacturing lengths of the HTS tapes were limited to a few hundreds of meters, and the world yearly production of the tapes was a tiny fraction of the amount that would be needed to construct a fusion-relevant tokamak.
2. related to the previous point was the high price of the HTS material. Expressed in price to carry a given current per meter length, the HTS materials used to be more than an order of magnitude more expensive than LTS.
3. quench detection turned out to be much more challenging compared to LTS magnets. The quench propagates much slower in the HTS magnets, which may lead to creation of a very localized hot-spot somewhere in the coil that might remain undetected until it is too late, and the coil gets irreversibly damaged.
4. AC losses – in a steady operation (constant magnetic field and constant current), superconductors have zero resistance and no heating in the coil. However, the situation changes when the current or field is changing (alternating current, AC) – in that case some losses are induced in the coil, leading to a heat dissipation. In an extreme case, these losses may lead to the quench of a coil. The AC loss in HTS tapes is an order of magnitude higher compared to the AC loss in LTS magnets. Fortunately, HTS materials are more stable (have higher temperature margin) compared to LTS, and the higher AC loss can therefore be overcome.
5. performance degradation as a result of repetitive load (repetitive coil charging). This fatigue issue is known also from LTS conductors, however in HTS it seems to be even more pronounced, due to the larger stresses in the high-field regions of the winding pack.

In the past $\approx$10 years, we have witnessed a rapid development in most of the points above [10.27]. The industry is capable to deliver much higher quantities of HTS tapes, and also the price dropped significantly. This triggered interest in the usage of HTS in the research labs as well as in the industry, and the increased demand for the HTS tapes in turn encouraged the HTS manufacturers to increase their manufacturing capacity. The most promising material became the REBCO tapes, which became some sort of a standard HTS product produced by several companies all over the world.

A big milestone in the development of HTS fusion magnets was the manufacturing and testing of the toroidal field (TF) coil prototype made of HTS tapes by the US company Commonwealth Fusion Systems (CFS) in autumn 2021, reaching a magnetic field of 20 T [10.12], see Figure 10.7. The uniqueness of the coil was the usage of unprecedented amounts of HTS material, as well as its non-insulated winding pack, which means that during the coil charging or discharging, electric current can flow not only along the superconducting coil winding, but also transversely through the steel jackets of the coil turns and layers. In a steady state situation, the transverse paths are inconvenient (they are resistive) and current flows along the superconducting winding. In case of a local quench, current can bypass the quenched region, and the coil does not need active quench protection. Even if the first TF CFS prototype coil has not validated the concept of the passive quench protection yet, it triggered a big attention in the fusion community and confirmed the trend of going towards HTS conductors for fusion magnets. The strategy of CFS is to make a compact tokamak as discussed in Sec. 10.5.2, where the HTS coil might be very expensive, however there will be enormous savings on other components of the tokamak, significant reduction in the construction time, and therefore allowing for more room in design errors or technical failures, as the next machine can be rebuilt relatively quickly, if needed.

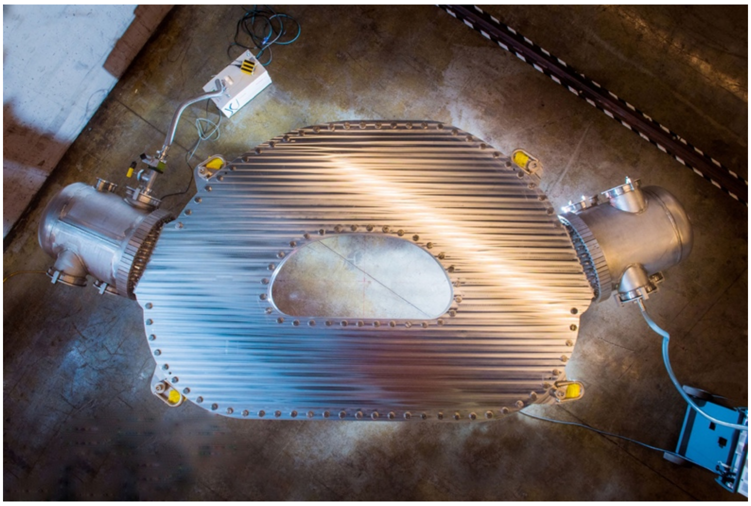

Figure 10.7: This large-bore, full-scale high-temperature superconducting magnet designed and built by Commonwealth Fusion Systems and MIT's Plasma Science and Fusion Center (PSFC) in 2021 has demonstrated a record-breaking 20 tesla magnetic field. It is the strongest fusion magnet in the world.

Since several years, there is an increased interest of states as well as private companies in developing fusion power plants. Most of these demonstration plants count on the usage of HTS materials, either in a hybrid setup (some coils made of HTS others from LTS), which is the case of e.g. one of the design options of the EUROfusion DEMO tokamak, DTT in Italy, and the BEST and CFETR tokamaks in China, or as purely HTS machines, such as the SPARC tokamak of CFS in the US or the Spherical Tokamak of Tokamak Energy in the UK. The advantages of HTS have been recognized, and the price of the HTS tapes is expected to further drop in the coming years. The R&D on quench detection and protection is getting very intense with a lot of innovative approaches. The HTS materials thus have a potential to shorten the construction times and to make fusion power plants economically more attractive.

In Europe, the main developers of fusion-size conductor prototypes made of HTS tapes are EPFL-SPC and ENEA (Italy). The third important player used to be Karlsruhe Institute of Technology (Germany), however they have recently abandoned this research line.

10.8.3 Conclusion of this section

The ITER tokamak is the world-leading fusion project. It is under construction in France since 2013 and relies on well-established LTS technology. Without affecting the pivotal role of ITER for the development of fusion energy, HTS technology developed dramatically, potentially opening up a new field of more compact, faster-to-build fusion. The world leader in this effort presently seem to be the CFS company, already constructing their first demonstration (a sub-size) HTS tokamak SPARC, as well as Tokamak Energy. Other, more conservative actors are also implementing HTS materials in the designs of their fusion machines in the form of hybrid magnets, in order to increase the magnetic field compared to ITER and consequently reduce §the machine size. The HTS technology still requires further R&D to get fully mature, however the progress is very fast and promising. As discussed in the next section, the superconductivity group of EPFL-SPC is one of the leading labs in the development of new fusion conductors, both LTS and HTS, and operates the world unique facilities for the fusion superconductor testing – SULTAN (in operation) and EDIPO (in reconstruction).

10.9 Fusion Science and Technology in Switzerland and upcoming opportunities

EPFL's Swiss Plasma Center (SPC, https://spc.epfl.ch), formerly Centre de Recherche en Physique des Plasmas (CRPP), is the competence center for plasma and fusion science and technology in Switzerland. It provides the fusion community with world-renowned expertise and unique facilities. The Center's main missions are to contribute to the success of ITER, to the development of the science and technology basis for DEMO, to the preparation of the ITER/DEMO generation of scientists and engineers, and to explore plasma and fusion spinoffs for industry and society.

The SPC operates the Tokamak à Configuration Variable (TCV), Figure 10.8, a device renowned for its unique magnetic shaping capabilities and flexible heating systems. It is dedicated, among other things, to the optimisation of plasma performance, the plasma-wall interaction challenge, and plasma control including protection against plasma disruptions. TCV is one of the key facilities within the EUROfusion research program and the European fusion roadmap [10.5,10.22], operated partly (≈40%) as a shared European facility. The SPC also includes a mm-wave laboratory with the FALCON and T-REX facilities, providing leading expertise in the field of mm-wave heating systems, such as design and qualification of high-power RF sources (gyrotrons) for Electron Cylcotron (EC) Resonance Heating and current drive and extensive testing of mm-wave components. This is complemented by strong mechanical and electrical design and analysis teams, involved in ITER (e.g. high-voltage power supplies and design of EC wave launchers) and DEMO activities, featuring critical skills that are rather scarce in the fusion community. The SPC further hosts the superconductor test facility SULTAN, Figure 10.8, located at the Paul-Scherrer Institute (PSI). SULTAN is a world unique facility, where conductors of all large fusion devices with superconducting magnets built in the past (JT60-SA, Wendelstein) and under construction now (ITER, DTT, CRAFT, CFETR, BEST, SPARC) are being tested. No other facility can provide a similar combination of low temperatures (4.5 K), available sample space (94x144 mm2), current (100 kA) and magnetic field (11 T), allowing for tests of fusion conductors in tokamak operating conditions.

The SPC is also at the forefront in the simulation and theoretical understanding of fusion plasmas, with a large spectrum of state-of-the-art recognized modelling tools and a leading expertise in High-Performance Computing (HPC). As part of a wider EPFL effort including SCITAS (Scientific IT and Application Support), the applied mathematics department (MATHICSE), the Swiss Data Science Center (SDSC), the laboratory for experimental museology (eM+), the SPC theory group further operates one of the five EUROfusion Advanced Computing Hubs, dedicated to develop, verify and validate cutting-edge modelling algorithms with the aim to create certified standard software for fusion. Finally, the SPC operates a number of smaller plasma devices, in particular RAID and TORPEX, to explore specific issues for fusion and applications in industry, including a recently established bio plasma lab in collaboration with the EPFL school of Life Science. Such activities also include interaction with industry on focused projects, e.g. with the support of the Innosuisse. Aside from the SPC, the university of Basel is also contributing to the ITER project and a number of companies is providing components for its construction, in particular related to cryotechnology, high-voltage grid components, plasma diagnostics, metrology and vacuum technology.

Currently (status as of November 2023), the SPC workforce comprises 53 PhD students, 32 postdoctoral researchers, 28 engineers, 22 technicians, and 38 senior scientists and 5 professors. The Center offers a wide range of introductory and advanced courses on plasma and fusion, including two online courses with over 40'000 enrolments from all over the world since their start in 2015. Members of the SPC also engage in a wide variety of activities to communicate with the public and typically welcome over 2'000 visitors per year.

Looking ahead, the SPC appears in the 2023 Swiss Roadmap for Research Infrastructures [10.31]. In order to bring the role of Switzerland in fusion to a next level, the Roadmap proposes the creation of the Swiss Fusion Hub within the SPC, by substantially upgrading its infrastructure. This includes upgrades to TCV and rebuilding, with improved performance, the coil of its second superconductor test facility EDIPO, damaged in 2016, to reach 15 T and thus outperforming SULTAN. As such, the aim of the Swiss Fusion Hub is to "*…leverage the key competencies of the SPC, including its long-lasting leading role in the optimization of the core plasma, the extensive high-impact recent investigations of advanced boundary plasma geometries, its world-renowned expertise in theory and modelling and the 30-year experience in superconductor testing on the SULTAN facility.*" [10.31].

Due to the non-association of Switzerland to the current EU Framework Programme for Research and Innovation, Horizon Europe, and coincidentally with Euratom, Switzerland's contribution to the ITER construction, ongoing since 2007 via the membership in Euratom and F4E, is suspended since 2021. The EPFL can still play some role within ITER through an inter-institutional cooperation agreement between EPFL and F4E. Previously established commercial contracts awarded to Swiss companies continue, but new ones can in the current situation not be established. However, EPFL's participation in the EUROfusion program is guaranteed through an associated partner agreement with the coordinator of the EUROfusion consortium, the Max Planck Gesellschaft.

Following this brief overview of the current fusion activities in Switzerland, upcoming opportunities are briefly discussed below.

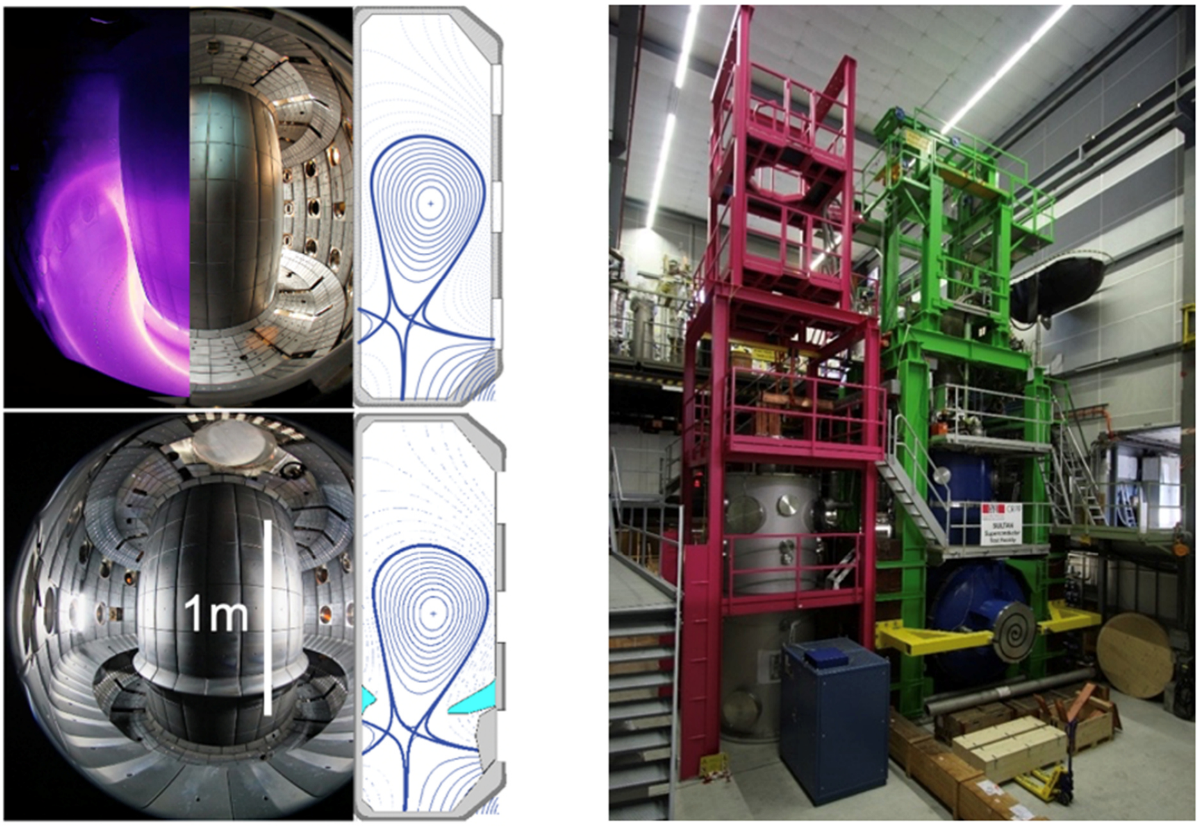

Figure 10.8: left: Picture of the TCV vacuum vessel in its "open" configuration (top) and its "closed" configuration. The latter is a recent key feature of TCV, made possible by the 2017-2020 federal ERI (Education, Research, and Innovation) funds. The colored image in the top left is taken during a plasma discharge. To the right of the photographs, example cross-sections of a TCV magnetic equilibrium are shown. Right: EDIPO (pink structure of the left) and SULTAN (green structure on the right) test facilities for testing of superconducting prototype conductors for fusion. The facilities are operated by the Superconductivity Group of EPFL-SPC, physically located at the PSI premises in Villigen.

10.9.1 Upcoming opportunities for public and private Swiss institutions in the next 20 years

The new, accelerated approach to the European roadmap to fusion energy [10.22] (including a parallelization of activities between ITER and DEMO and a stronger involvement of industries through Public-Private Partnerships (PPP)), the significant number of large devices in construction or in planning (Sec. 10.5), and an emerging private sector open up opportunities for public and private institutions in Switzerland. The highly flexible and versatile TCV device, with its direct inclusion in an institution of higher education, is well suited for key contributions to a wide set up topics of **fusion plasma physics and control**. This includes continued developments of key issues for ITER and innovative solutions for DEMO, such as developments of improved tokamak scenarios, advanced control schemes using conventional and machine learning techniques, disruption prevention and mitigation strategies, rigorous experimental validation of the fast-improving numerical tools developed at the SPC and internationally, and improved power exhaust solutions. In the area of plasma control, the collaboration with private initiatives has already been initiated, e.g with Google Deepmind in the UK and is bearing fruit [10.32]. TCV and the SPC can furthermore play an even more important role as a **training center for the operation of tokamaks and their ancillary systems** and serve as a testbed for new actuators, diagnostics, robotics, and materials, before integration in a larger and more complex device.

Great opportunities exist also in **mm-wave (microwave) technologies and the development of gyrotrons**. Most of the next step devices will require a large number of gyrotons in the coming years, far above current production capacities. Switzerland, with its extensive expertise at the SPC-EPFL and international network is well positioned to strengthen its position in this field. Indeed, the last generation of the gyrotrons built by the unique European manufacturer (THALES), has been designed by SPC scientists. As such, microwave technologies constitute an ideal candidate for a Public-Private Partnership (PPP).

A key fusion technology also featuring extensive Swiss expertise is **Fusion magnet R&D**. The rebuild of EDIPO with a higher magnetic field will be able to respond to the steady increase in the demand to test HTS conductor samples in SULTAN over the past three years, namely for the SPARC (CFS, US), CFETR (China), BEST (China), and EU-DEMO tokamaks. The EPFL-SPC superconductivity group is also pursuing activities in conductor development, with many opportunities for further future developments. In case of LTS superconducting cables for the EU-DEMO, SPC proposed a flat, react&wind Nb_3Sn type of conductor [10.28] shown in Figure 10.9 (left), which has been chosen as the main design option for the DEMO TF coil. The advantages of this design are significant savings of Nb_3Sn and simplification of the coil manufacture compared to ITER-like coils. In 2015, the first-ever-built full-size fusion conductor prototype made of HTS was manufactured and tested at the SPC [10.29], see Figure 10.9 (right). The conductor reached the design performance of 60 kA at 12 T field, however turned out to degrade its performance when exposed to the repetitive cycling loading. Since then, the R&D on HTS superconductors for fusion continues worldwide. Two new HTS conductor prototypes are being manufactured at EPFL-SPC and will be tested this year. In addition, quench experiments on five various HTS conductors were performed by EPFL-SPC in the past few years. EPFL-SPC is also investigating various innovative methods for HTS fusion coils, namely quench detection using optical fibers or electrically insulated superconducting strands, cryogenic electrical high-current switches (components that do not exist so far), non-insulated coils, and conductors for stellarator magnets (low bending radius is required). Considering the large interest in HTS in the fusion community and beyond, further pursuing and extending these activities holds a great potential, also for the private sector or as part of a PPP.

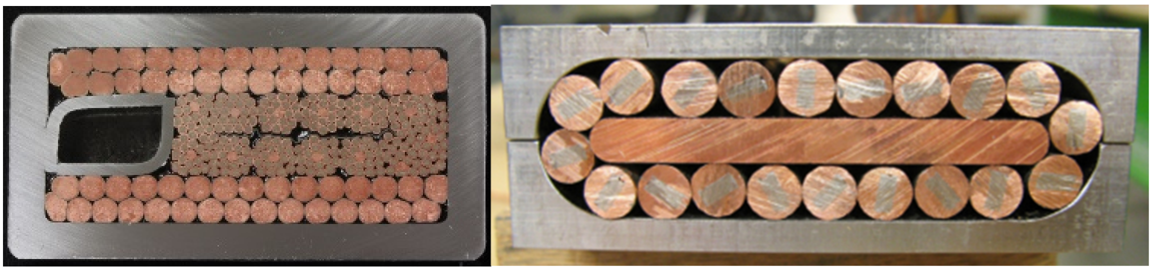

Figure 10.9: Examples of conductors for fusion magnets: Cross section of the Nb_3Sn react-and-wind prototype (left) and cross section of an HTS conductor made of stacked REBCO HTS tapes inserted into round copper profiles (right). Both conductors were developed at EPFL-SPC.

Material research for fusion, in particular the study of the behavior of **materials in high heat-flux and high neutron fluence conditions**, has only recently been initiated in Switzerland through a collaboration between the SPC theory group and the EPFL's laboratory of Theory and Simulation of Materials. There is clearly large potential to extend these efforts substantially and to leverage the know-how and expertise from the nuclear engineering community in Switzerland, in particular the Laboratory of Reactor Physics and Systems Behavior at the EPFL and the PSI and the Nuclear Systems and Multiphase Flows Laboratory at ETH Zurich. Such activities could be extended to address the probably largest knowledge gap in fusion technology, the blanket design.

These are just some of the opportunities expected to arise in the next years for Swiss academia and industry. Additionally, important opportunities for **contributions to the construction** of ongoing projects, such as DTT in Italy, or upcoming ones, such as a possible Volumetric Neutron Source [10.22] for blanket testing, EU-DEMO, and potentially even a large new device built in Switzerland are expected to arise.

Last but not least, SPC presently hosts the Swiss Industry Liaison Office (ILO), an initiative sponsored by the Swiss State Secretariat for Research, Education and Innovation with the aim to strengthen the relation between Swiss companies, research institutes and international research projects, for example ITER. The ILO might contribute to the setup of PPP in fusion.

10.10 Conclusions and timeline towards a power plant

Nuclear fusion has a tremendous potential for a future safe, clean, carbon-free, dependable and essentially inexhaustible energy source. While great progress has been achieved towards this goal over the past decades, a number of challenges still need to be overcome, requiring R&D in particular with respect to plasma scenario optimization, heat exhaust, and control of plasma transients, as well as material research for plasma-facing components, materials in high neutron flux conditions, and breeding blanket technologies.

The magnetic confinement fusion approach based on tokamaks and stellarators is generally considered to be the most promising path and most national and international programs pursue this route. Inertial fusion energy has also demonstrated significant progress recently and some ideas exist to make this route more reactor-relevant. The big next steps for most public fusion programs are the ITER tokamak and the IFMIF-DONES material test facility. The UK with STEP and China with CRAFT, CFETR and BEST pursue also very ambitious national efforts, and an increasing commitment in such directions is also manifesting in the U.S. In addition, a dynamic scene of private fusion companies has emerged over the past years, initially primarily in the U.S., but including now also to the UK, China, Germany, and other countries. These companies explore or re-explore a wide range of approaches to fusion based on magnetic confinement fusion, inertial fusion energy, and magneto-inertial fusion concepts. The companies focusing on variants of the tokamak have thus far reported the best performance and the largest advances in technology, in particular related to HTS magnet developments. Other companies, especially those focusing on alternative fuels, rely more on un-proven concepts and breakthrough discoveries. The private sector has also the potential to play an important role in establishing the critical need for a powerful fusion industry, and both public and private sector move more and more towards Public-Private Partnerships to combine the strengths of the two worlds.

The timeline towards a first fusion power plant is currently associated with significant uncertainties. While many private companies promise power to the grid by as soon as 2035 or even earlier, these statements should be met with a lot of care, considering, among others, the approach-independent challenges discussed in Sec. 10.6. As such, these promises should rather be seen as motivational aspirations and in the context of the need to attract private investors. The European fusion roadmap [10.5,10.22], which accounts for all aspects needed for the development of a fusion power plant, targets for a first demonstration reactor DEMO by 2045. The recent, independent report published in [10.6] discusses four different scenarios, projecting the first fleet of commercial fusion power plants to enter operation between the 2040s to the 2070s.

Clearly, progress depends on funding levels and decisions taken today. The European fusion roadmap [10.5,10.22] lays out a detailed strategy for swift progress towards DEMO, Ref. [10.6] makes a number of recommendations to accelerate progress and Ref. [10.13] developed four policy options to help address the remaining challenges. In this context and considering the urgent need to develop carbon-free energy sources not only by 2050 but also beyond, fusion science and industry is expected to grow substantially over the coming years. As discussed in this report, Switzerland has high potential to get inserted in the global fusion power sector where research and business opportunities will grow.

10.11 References

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