Quadrennial Technology Review 2015

Chapter 4: Advancing Clean Electric Power Technologies

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U.S. DEPARTMENT OF ENERGY

Light Water Reactors

Chapter 4: Technology Assessments

Past, Present, and Future of the Technology

The world's first full-scale nuclear power plant (NPP) devoted exclusively to peacetime uses came online in 1957. Light water reactors (LWRs) are now a mature technology, with over 350 operational LWRs worldwide (Figure 4.M.1) and over 60 under construction (Figure 4.M.2).[1] Note that the Fukushima accident adversely affected nuclear power operations in Japan (and other countries throughout the world) and that 43 reactors are currently operational or able to restart in Japan and 24 are in the process of restart approval. The World Nuclear Association[2] and the International Atomic Energy Agency[3] are good sources for status of reactors internationally, and the Nuclear Regulatory Commission (NRC)[4] is a good source for domestic information.

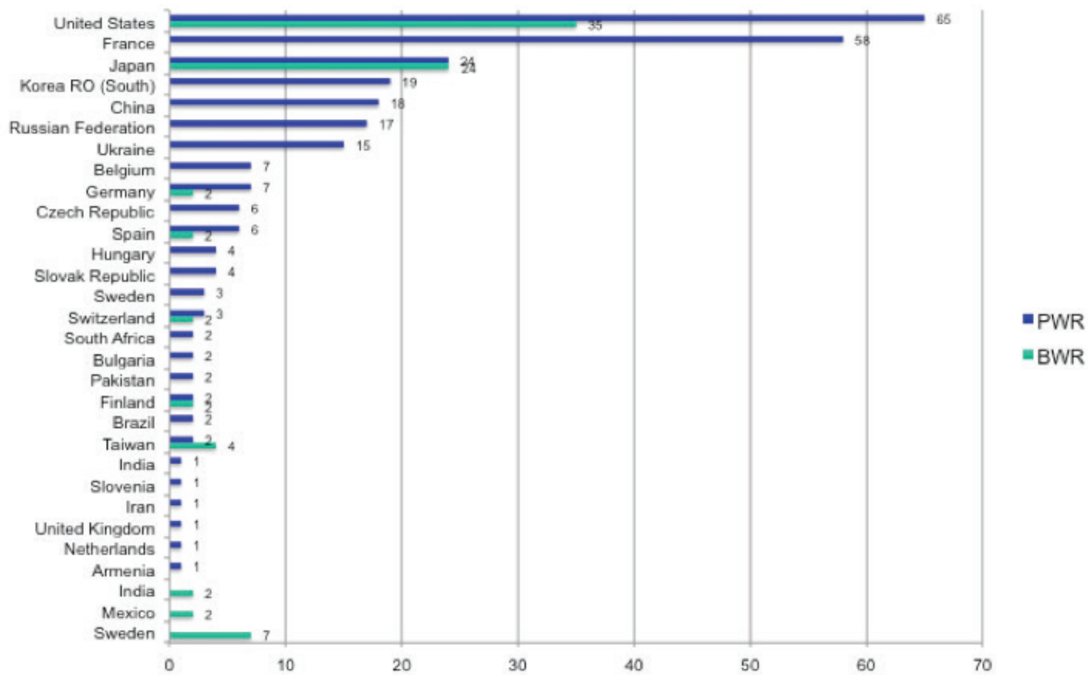
There are two major types of LWRs: pressurized water reactors (PWRs) and boiling water reactors (BWRs); the large majority of operating LWRs are PWRs (see the later section, "Description of the Technology", for an overview of these technologies). The nuclear industry is very much like the automobile industry (also a mature technology) in that safety and economic improvements are continuously made, and an LWR can undergo many significant changes in its lifetime (component change-outs, power uprates, addition of safety systems, etc.).

Commercial LWRs are used primarily for electricity production, with some examples of use in district heating and desalination (they are also used extensively to power ships and submarines used for national defense). The coolant outlet temperature in conventional LWRs is limited to about 300°C, precluding process heat applications that require much higher outlet temperatures (such as steam reforming of natural gas to produce hydrogen). This outlet temperature also limits thermal efficiency to about 30%. The current fleet of LWRs relies on "active" safety features, while advanced LWRs (ALWRs) typically rely on "passive" safety features (active safety contrasts to passive safety in that it relies on operator or computer automated intervention, whereas passive safety systems rely on the laws of nature to make the reactor respond to dangerous events in a favorable manner).

The United States operates the largest fleet of commercial nuclear reactors, with 99 operating reactors (Figure 4.M.3) as of 2015; about two-thirds of them are PWRs, and one-third are BWRs. Nuclear energy provides almost 20% of U.S. electricity, and over 60% of non-carbon-emitting electricity, providing secure and stable base load, independent of factors such as severe cold weather. The average capacity factor of U.S. nuclear plants is over 90% and has been since 2002 (up from 50% in the early 1970s and 70% in 1991). In January 2013, 104

Figure 4.M.1 There are over 350 light water reactors operational worldwide (2014 data). While some reactors (in particular those in Japan) are "operational," they are not currently operating.

Credit: International Atomic Energy Agency



nuclear power reactors operated in 31 states. However, since then, five reactors have been shut down due to various economic and operational reasons. Nuclear plants face ongoing competitive pressures from low-cost natural gas, as well as from near-zero marginal cost wind power.

Figure 4.M.2 Total Number of Reactors Under Construction (August 2015)

Credit: International Atomic Energy Agency

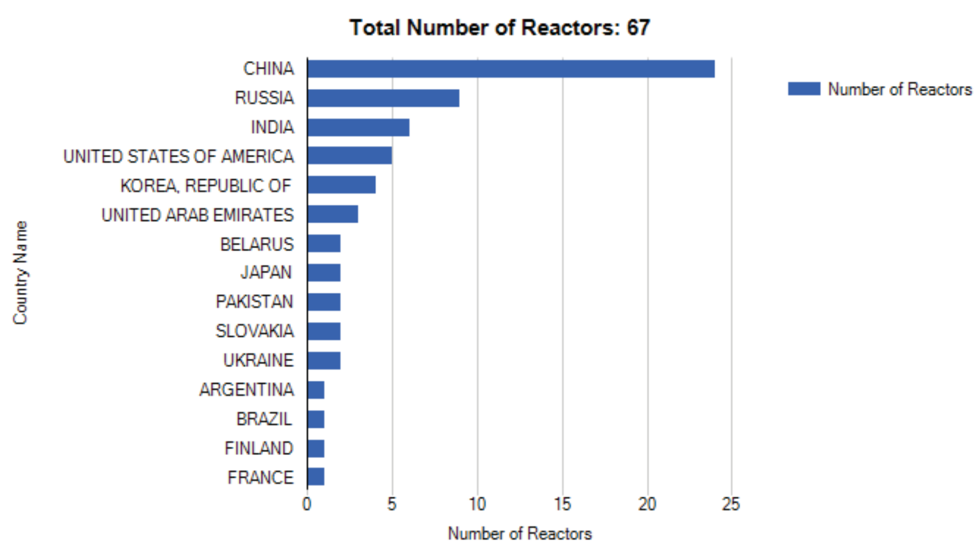
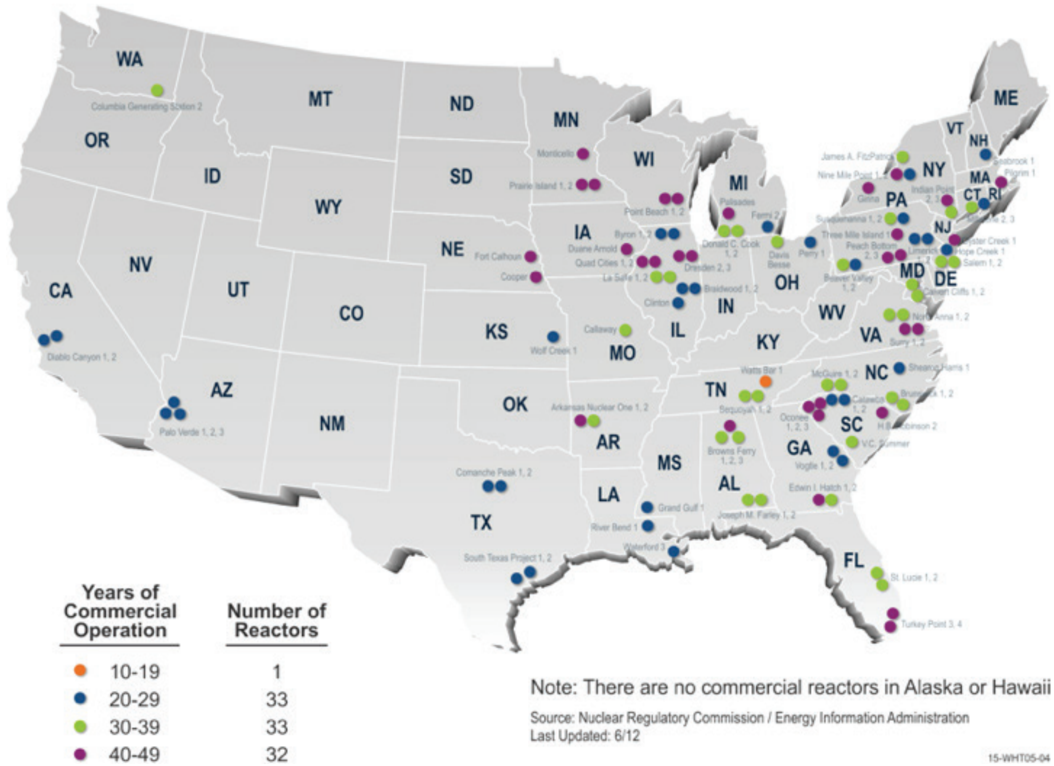


Figure 4.M.3 There are 99 operational reactors in the United States (August 2015).

Credit: U.S. Nuclear Regulatory Commission, U.S. Energy Information Administration



In the United States, there are several NRC certified designs for LWRs.[5] An NRC design certification indicates that the NRC approves a particular nuclear power plant design, independent of an application to construct or operate a plant. NRC has certified five designs for ALWRs as follows:

- Westinghouse AP1000
- Westinghouse AP600
- Westinghouse System 80+
- GE Hitachi Nuclear Energy Advanced Boiling Water Reactor
- GE Hitachi Nuclear Energy Economic Simplified Boiling Water Reactor

The three following designs are currently undergoing review for certification:

- Areva U.S. EPR
- Mitsubishi Heavy Industries Ltd. U.S. Advanced Pressurized-Water Reactor
- Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd. Advanced Power Reactor 1400

There are five reactors under construction in the United States, all of them PWRs. Four of them are the AP1000 design (two in Georgia and two in South Carolina), and one is the completion of a Westinghouse PWR (in Tennessee) that received its construction permit in 1973. The new reactors are scheduled to begin operation in 2019 and 2020, with the reactor in Tennessee scheduled to begin operation by 2016.

In addition to the "conventional" size LWRs, there are small modular reactors (SMRs) under consideration worldwide (see NuScale,[6] mPower,[7] Holtec,[8] and Westinghouse websites[9]). These reactors are, for the most part, small versions of the conventional reactors, with some modifications to the designs appropriate to the smaller size (for example, different steam generator designs or one control room per multiple SMR units). In general, the LWR SMRs are in the demonstration phase with some research and development (R&D) under way, mostly design optimization and development of validation bases for licensing and design tools. The economics of small reactors have yet to be demonstrated. For example, SMRs offer smaller size for smaller markets, or can be grouped together for larger markets, but may provide electricity at a higher cost than conventional reactors. Particularly promising are better economics associated with modular construction, for example, with "factories" that produce multiple units. Standardization on a small number of designs also benefits economics (as has been demonstrated in France in the past with their first build-out, for example), but because power production in the United States is a private-sector rather than government-sector responsibility, it is difficult to mandate standardized designs. Because construction of nuclear power plants is very highly capital–intensive over a five-year or longer period, delays in construction are very costly; modular construction and standardization on a small number of designs could have a significant impact. The Department of Energy (DOE) SMR Licensing Technical Support Program is providing important economic support for SMR research and deployment.

There is another variant on the LWR that has been the focus of some R&D work, particularly under the Generation IV International Forum, and that is the super critical water reactor (SCWR).[10] This reactor is a higher-temperature, higher-pressure version of the conventional LWR and is still in the R&D phase (see the later section, "Description of the Technology," for an overview of the technologies, including temperatures and pressures).

Nuclear reactors are an important part of the portfolio needed to realize the administration's goal of reducing greenhouse gas emissions to 80% below 1990 levels by the year 2050 as well as greenhouse gas emission reduction goals worldwide. Small reactors have the potential to penetrate markets that conventional-size reactors have difficulty penetrating, primarily owing to the smaller capital investment needed for a small reactor. However, that must be accompanied by favorable economics for small reactors to penetrate more than just niche markets, such as remote locations or locations with an electrical grid that cannot handle the amount of power put out by a conventional size reactor.

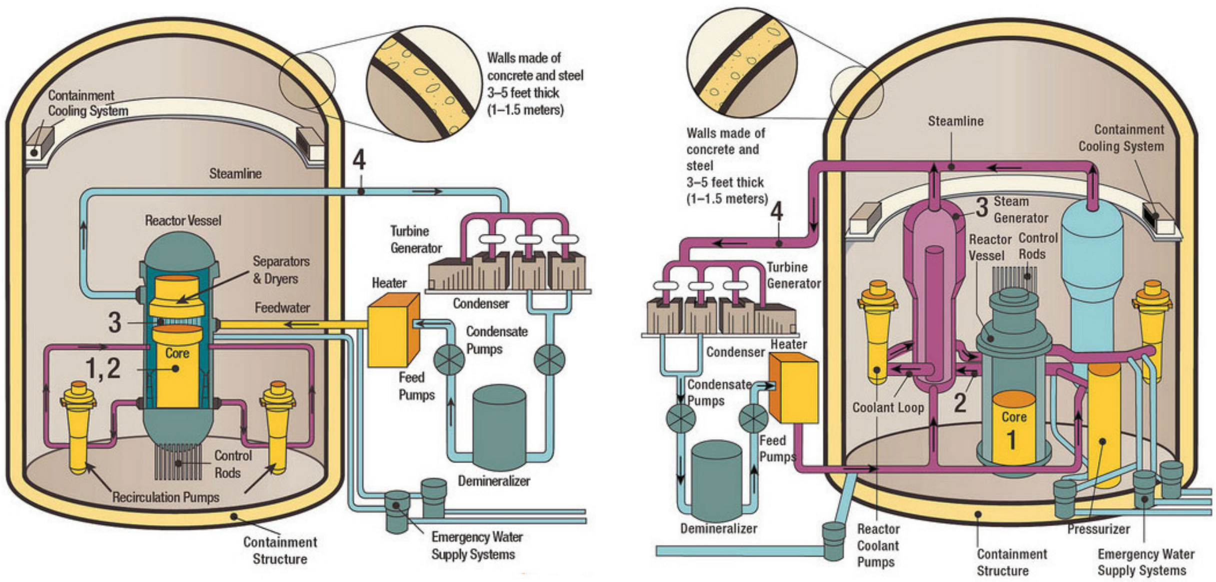
France derives about 75% of its electricity from nuclear (including for electrification of their train system), showing that penetration of nuclear energy could be much larger than it currently is in the United States, accompanied by the appropriate controls to reduce proliferation risk to an acceptable level. Nuclear plants could, for example, replace many fossil plants that have relatively slow ramps, and older plants could be replaced with newer plants if the economics are favorable. Desalination, currently done primarily with fossil fuels, could be done with LWRs.

Description of the Technology

The combination of pressure and temperature of the coolant is the primary difference between BWRs and PWRs. This difference enables major design differences between the two types of reactors. Figure 4.M.4 shows a high level comparison of the BWR and PWR.

Figure 4.M.4 Comparison of Generic BWR and PWR (RPV is Reactor Pressure Vessel)

Credit: U.S. Nuclear Regulatory Commission



BWR

- RPV Pressure ~7MPa
- RPV Temperature ~288°C
- Steam Generated in RPV (with Separator and Dryer)
- Bulk Boiling Allowed in RPV

PWR

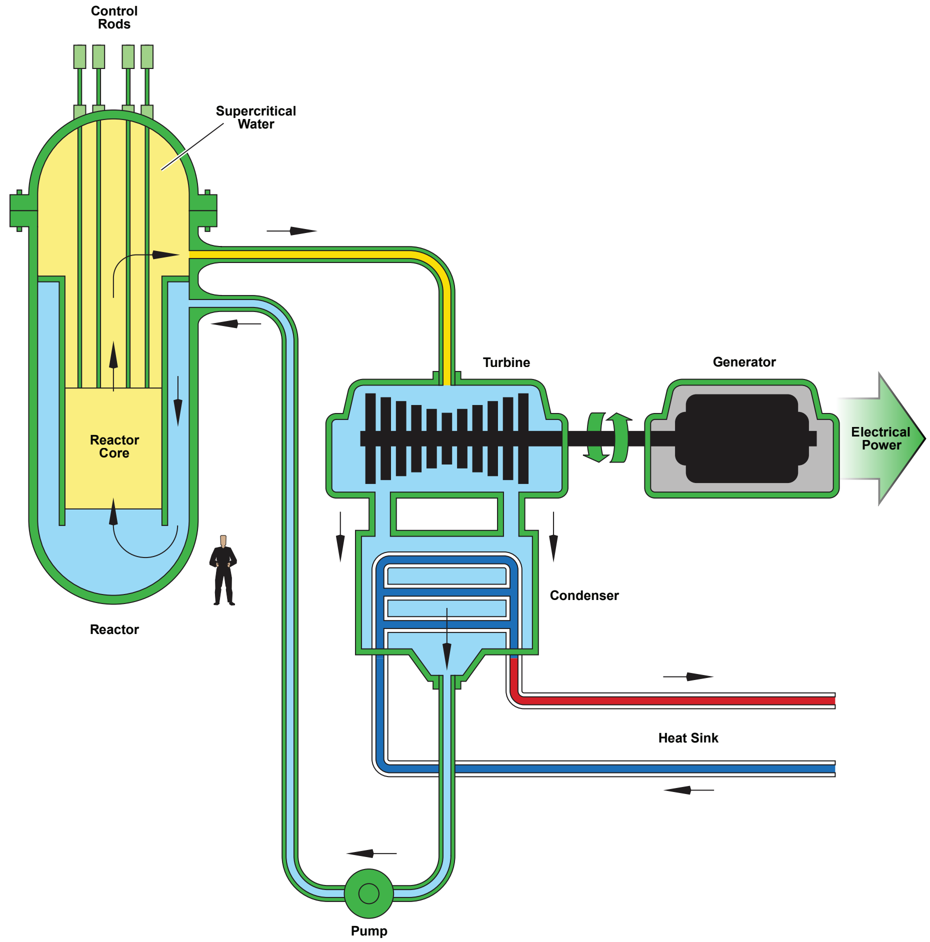
- RPV Pressure ~15MPa
- RPV Temperature ~326°C
- Steam Generated in Steam Generator (via Second Loop)
- No Bulk Boiling in RPV

A generic SCWR is shown in Figure 4.M.5. Characteristics of the SCWR include the following:[11]

- High temperature, high-pressure, light-water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa)
- Reactor core possibly having a thermal or fast-neutron spectrum, depending on the core design
- Outlet temperature up to 625°C, pressure ~25MPa
- High thermal efficiency (>44%) and simpler design than conventional LWR (no steam generators, no steam separators and dryers)

The high outlet temperature of the SCWR enables a broader mission (including process heat applications beyond desalination); however, there are significant materials and safety challenges. The SCWR environment is high temperature and high pressure, which necessitates structural materials that can withstand these conditions. The safety challenge is that the reactor will likely require an active safety system, a reliance that is reduced in ALWR designs. Because of the materials challenge that this design poses, as well as the safety challenges, there is little R&D currently under way. A description of materials and design challenges can be found in Buongiorno and MacDonald.[12]

Figure 4.M.5 A Schematic of the SCWR Design

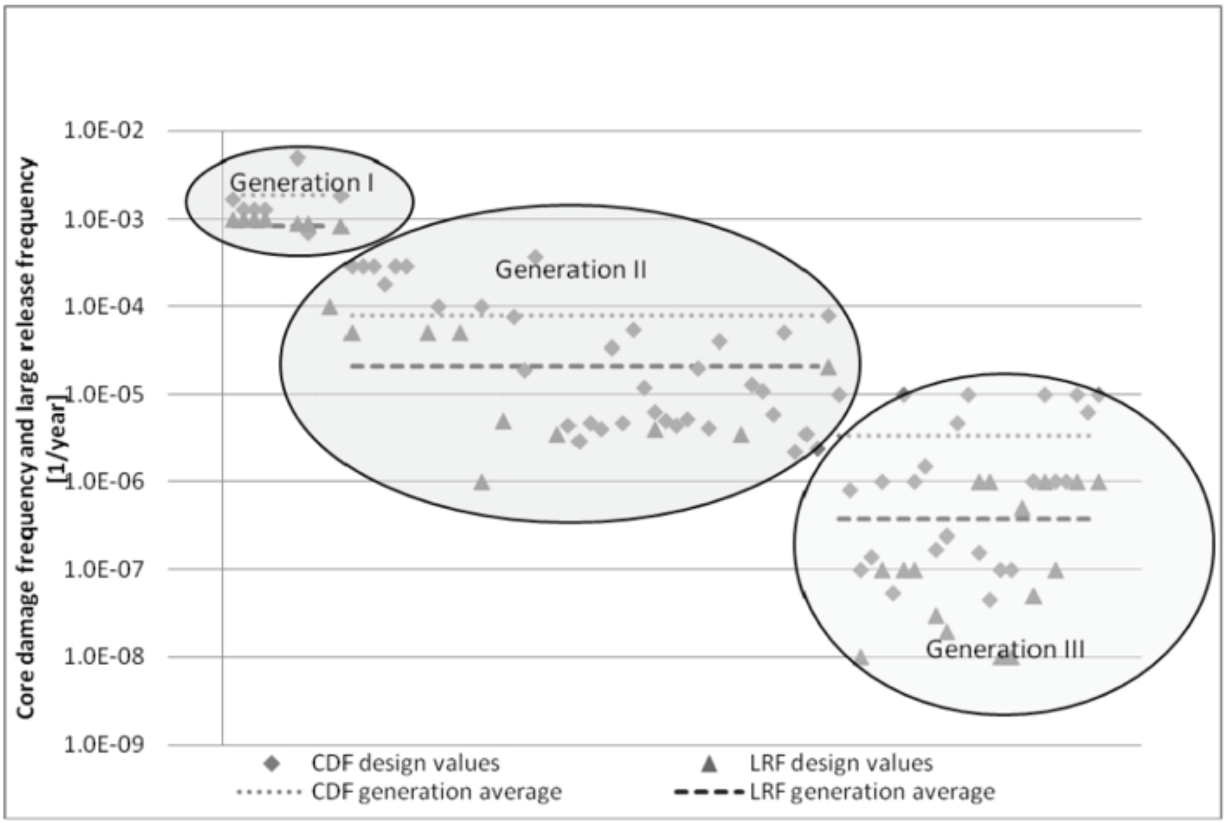


The oldest operating commercial reactors in the United States began operation in 1969. Continued operation requires an understanding of aging and degradation of materials and components. The current fleet of LWRs worldwide has undergone many changes since initial deployment, primarily to enhance safety and economics. Lessons learned from major accidents (Three Mile Island, Chernobyl, and now Fukushima) have influenced technology and operational changes. Modern PWR designs (including SMRs) incorporate passive safety features. The events at Fukushima have motivated a global reexamination of severe accident prevention and mitigation, with some changes already implemented (additional changes are under consideration, many of which could have a large negative impact on plant economics). In the United States, plants have added equipment (additional generators, batteries, water pumps, and other emergency equipment) needed to respond to extreme natural events. Regional response centers (part of the FLEX strategy implemented by the U.S. commercial nuclear industry) have been established to maintain more emergency equipment that can be dispatched quickly to any facility that needs it. Other changes are under discussion, both domestically and worldwide.

Measures of nuclear reactor safety can include calculations of core damage frequency (CDF) and large release frequency (LRF). Figure 4.M.6 shows the calculated CDF for the early LWR prototype reactors (Generation I), the current LWR fleet (Generation II), and ALWRs (Generation III). The figure illustrates how safety has improved across generations of reactor design (from early prototypes through today's reactors), primarily by using new methods in the design of nuclear reactors, incorporating prior operating experience, and employing enhanced safety features to mitigate anticipated accidents.

Figure 4.M.6 Core Damage Frequency (CDF) and large Release Frequency (LRF) for Early Prototype Reactors (Generation I), the Current U.S. Fleet (Generation II), and ALWRs (Generation III). The x-axis represents time (increasing to the right).[13]

Credit: International Atomic Energy Agency



Deployment of nuclear power requires the development of a regulatory infrastructure and fuel cycle infrastructure that may be too costly or undesirable (from a proliferation risk perspective) for smaller nations considering the introduction of the technology. Global expansion of nuclear energy requires international oversight and collaboration/cooperation. In addition, the large capital cost of conventional reactors can be a deterrent in deployment.

There are several estimates of overnight costs for LWRs.[14] The range of overnight costs for nuclear technologies in OECD (Organisation for Economic Co-operation and Development) countries is large, from a low of $2,021/kWe in Korea to a high of $6,215/kWe in Hungary.[15] A paper by Hezir and Davis estimated capital cost of GW-scale LWRs at $4,210/kWe.[16] Ultimately, cost of money and regulatory factors have a very large impact on the cost of a nuclear plant. Compared with other nuclear power plant types, the overnight cost is relatively well known because several hundred have been built worldwide. The average cost to fuel a nuclear plant in the United States in 2014 was 0.76 cents/kWh.[17]

Getting from the Present to the Future

Continued operation of the current fleet of U.S. reactors requires proof that the aging degradation will not adversely impact the safety of the reactors and that economics will continue to be favorable (including making commercial reactors more labor efficient).

The Electric Power Research Institute (EPRI) is the primary R&D arm of the U.S. nuclear industry and has several programs focused on issues important to the long-term operation of the commercial nuclear fleet. For example, the EPRI Long-Term Operation Program and various materials R&D programs (such as the BWR Vessel Internals Program and the PWR Materials Reliability Program) perform R&D focused on materials aging and degradation. Other EPRI programs, such as the Instrumentation and Controls Program, focus on modernization needs in nuclear reactor instrumentation and control systems, ultimately benefiting plant economics and safety. Many of the R&D areas in the DOE program described in the next paragraphs are common R&D areas between EPRI and DOE.

In coordination/collaboration with the U.S. commercial nuclear industry, primarily through EPRI, the DOE Light Water Reactor Sustainability (LWRS) Program[18] is developing the fundamental scientific basis to understand, predict, and measure changes in materials and systems, structures, and components (SSCs) as they age in environments, apply this knowledge to develop and demonstrate methods and technologies that support safe and economical long-term operation of existing reactors, and research new technologies that enhance plant performance, economics, and safety.[19] The program includes research in the four following areas:

- Materials aging and degradation
- Risk-informed safety margin characterization
- Advanced instrumentation, information, and control systems technologies
- Reactor safety technologies

The R&D program has been developed with input from international subject matter experts, such as from industry, national laboratories, universities, and regulators. The materials aging and degradation R&D activities are developing the scientific basis for understanding and predicting long-term behavior of materials in nuclear power plants. This work will inform long-term operation decisions by providing data and methods to assess the performance of SSCs essential to safe and sustained nuclear power plant operations, including methods for monitoring and assessing degradation via nondestructive techniques and advanced strategies for mitigating the effects of aging. Research areas include the following:

- Reactor metals
  - High fluence effects on reactor pressure vessel (RPV) steels
  - Material variability and attenuation effects of RPV steels
  - Nondestructive evaluation of RPV degradation
  - Mechanisms of irradiation-assisted stress corrosion cracking
  - High fluence irradiation-assisted stress corrosion cracking
  - High fluence phase transformations of core internal materials
  - High fluence swelling of core internal materials
  - Cracking-initiation in Ni-base alloys
  - Nondestructive evaluation of cracking precursors
  - Environmentally assisted fatigue
  - Nondestructive evaluation of environmentally assisted fatigue degradation
  - Thermal aging of cast stainless steels

- Concrete
  - Concrete and civil structure degradation
  - Nondestructive evaluation of concrete and civil structures
- Cables
  - Mechanisms of cable insulation aging and degradation
  - Nondestructive evaluation of cable insulation
- Mitigation technologies
  - Advanced weld repair
  - Advanced replacement alloys
  - Thermal annealing

The purpose of the Risk-Informed Safety Margins Characterization (RISMC) Pathway R&D is to support plant decisions for risk-informed margins management with the aim to improve economics and reliability as well as sustain safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by NPP decision makers as part of risk-informed margin management strategies; (2) create an advanced RISMC toolkit that enables more accurate representation of NPP safety margins. Included in this toolkit are the next generation reactor systems-analysis code (RELAP-7), a probabilistic-based scenario simulation code (RAVEN), and a component aging and damage evolution mechanism simulation application (Grizzly). The RISMC methodology can optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into the safety analysis. R&D activities include the following:

- Development of the technical basis for risk-informed margins management
- Delivery of the RISMC toolkit, including a systems-level safety tool, component aging tool, and external hazards tools

The advanced instrumentation, information, and control systems technologies research supports safe and efficient modernization of the current instrumentation and control technologies used in nuclear power plants through development and testing of new instrumentation and control technologies and advanced condition monitoring technologies for more automated and reliable plant operation. The R&D products are used to design and deploy new instrumentation, information, and control technologies and systems in existing nuclear power plants that provide an enhanced understanding of plant operating conditions and available margins and improved response strategies and capabilities for operational events. The goals are to enhance nuclear safety, increase productivity, and improve overall plant performance. Pathway researchers work with nuclear utilities to develop instrumentation and control technologies and solutions to support the safe and reliable life extension of current reactors. Areas of research include the following:

- Human performance improvements for nuclear power plant field workers
- Outage safety and efficiency
- Online monitoring
- Automated plant
- Hybrid control room

The fourth research area, reactor safety technologies (RST), provides scientific and technical insights, data, analyses, and methods that can support industry efforts to enhance nuclear reactor safety during and beyond design basis events. The RST activities evolved from an initial coordinated global effort to assist in the analysis of the Fukushima accident progression and accident response into the following areas:

- **Accident tolerant components:** Analysis or experimental efforts for hardware-related issues with the potential to prevent core degradation or mitigate the effects of beyond-design basis events
- **Severe accident analyses:** Analyses using existing computer models and their ability to provide information and insights into severe accident progression that aid in the development of Severe Accident Management Guidelines
- **Fukushima forensics and examination plans:** Provide insights into the accident progression at Fukushima through data collection, visual examination of in situ conditions of the damaged units as well as collection and analysis of samples within the reactor systems and structural components from the damaged reactors

Details on R&D needs for the first three areas can be found in McCarthy et al.[19] The RST activities are currently under development, and the 2015 update to the LWRS Program plan will include this information. Much of the R&D under way in this program is also relevant to ALWRs, SMRs, and in some cases, other types of reactors.

The DOE Nuclear Energy Enabling Technologies (NEET) Program develops crosscutting technologies that benefit a broad range of nuclear reactors, including LWRs. Areas of research include reactor materials, advanced sensors and instrumentation, advanced methods for manufacturing, proliferation and terrorism risk assessment, and advanced modeling and simulation. Under the NEET Program, the Consortium for Advanced Simulation of Light Water Reactors (CASL)[20] is developing modeling and simulation tools that will assist the nuclear industry with analyzing operational issues in current LWRs, and the Nuclear Energy Advanced Modeling and Simulation Program[21] is developing modeling and simulation tools that will support analysis of current and future reactor systems. Through a peer-reviewed proposal process, the Nuclear Scientific User Facility[22] provides external research teams cost-free access to reactor, post-irradiation examination, and beamline capabilities at Idaho National Laboratory and a diverse mix of affiliated partner institutions at universities, national laboratories, and industry facilities located across the country.

Demonstration of the economics of SMRs is one of the primary hurdles to deployment of SMRs. The DOE SMR Licensing Technical Support Program[23] was established to support the first phase of deployment of SMRs. Ultimately, adequate funding (from a source or variety of sources either public, private, or both) will be needed to support SMR development and demonstration.

Programmatic risks to the continued deployment of ALWRs include successful construction and operation of the four new reactors currently under construction and a long-term plan for used fuel management. The cost and schedule adherence in construction of the new reactors will have an impact on the willingness of U.S. industry to build additional reactors. The delay in identifying a final repository for used nuclear fuel is also an impediment to the construction of new reactors; DOE R&D on used fuel disposition and advanced fuel cycles are working to reduce the uncertainty associated with used fuel disposition. The U.S. government loan guarantee program is supporting the deployment of new nuclear plant construction.

Endnotes

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Acronyms

ALWR	Advanced light water reactor
BWR	Boiling water reactor
CASL	Consortium for Advanced Simulation of Light Water Reactors
CDF	Core damage Frequency
EPRI	Electric Power Research Institute
LRF	Large Release frequency
LWR	Light water reactor
LWRS	Light Water Reactor Sustainability Program
MPa	Megapascals
NEET	Nuclear Energy Enabling Technologies Program
NNP	Nuclear power plant
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
PWR	Pressurized water reactor
R&D	Research and development
RPV	Reactor pressure vessel
RELAP 7	Reactor Excursion and Leak Analysis Program-7
RISMIC	Risk-Informed Safety Margins Characterization
RST	Reactor safety technologies
SMR	Small modular reactor
SCWR	Super critical water reactor
SSC	Systems, structures, and components

Glossary

Active and Passive Safety	Passive safety in the context of nuclear systems means that the system or reactor relies on natural phenomena such as the laws of nature or behavior inherent in the reactor's materials and design to cool the reactor in the event of an upset condition. The reactor will be able to cool the core and shut down in an emergency without actions from an operator or electronic system. Active safety systems instead have a reliance on external mechanical and/or electrical power, signals or forces to operate.
Base Load	Base load power plants are those that provide the minimum level of electric power needed over a given period. Base load power is typically provided by plants that run steadily for months at a time. Base load power is often provided by nuclear, coal, and hydroelectric facilities.
BWR	BWR or boiling water reactor is a common nuclear power reactor design in which water flows upward through the core, where it is heated by fission and allowed to boil in the reactor vessel. The resulting steam then drives turbines, which activate generators to produce electrical power. BWRs operate similarly to electrical plants using fossil fuel, except that the BWRs are powered by 370–800 nuclear fuel assemblies in the reactor core.[24]
Capacity Factor	The ratio of the available capacity (the amount of electrical power actually produced by a generating unit) to the theoretical capacity (the amount of electrical power that could theoretically have been produced if the generating unit had operated continuously at full power) during a given time period.[24]
CDF	CDF or core damage frequency is an expression of the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.[24]
FLEX	FLEX is the nuclear industry's strategy to enhance safety at the nation's 67 plant sites to better equip them for unexpected events. The strategy—known as the "diverse and flexible mitigation capability," or FLEX—addresses many of the recommendations set forth by the Nuclear Regulatory Commission's Fukushima task force and takes into account some of the early lessons from the Fukushima accident on the need to maintain key safety functions amid conditions where electricity may be lost, back-up equipment could be damaged, and several reactors may be involved.

Fluence	Fluence or neutron fluence is the neutron flux integrated over a period of time where the neutron flux is the number of neutrons traveling through a unit area in a unit time, often cm^2 per second.
Generation II-IV Reactor	Generation IV reactors are the next generation of reactors that are currently being researched for potential deployment in the future. Reactors operating today are primarily Generation II and III designs. New reactors under construction in the United States are considered Generation III+.
MOOSE, BISON, RAVEN, Grizzly, etc.	These are series of software codes and platforms developed at Idaho National Laboratory to support advanced modeling and simulation for nuclear applications. MOOSE (Multiphysics Object Oriented Simulation Environment) makes modeling and simulation more accessible to a broad array of scientists. BISON is a thermo–mechanical code that models nuclear fuel performance at the engineering scale. The RAVEN software tool will provide a user interface for RELAP–7, INL's premier reactor safety and systems analysis tool. Grizzly models degradation that can build up after years of use in reactor pressure vessels and other components.[25]
LWR	Light water reactors are the standard reactor design deployed today. They use normal water (H_2O) as the coolant and neutron moderator to lower the energy of the neutrons to thermal levels. The fuel is typically uranium dioxide pellets that are placed into cladding of a zirconium alloy. The system can operate with low-enriched uranium. In the United States and a number of other countries, LWRs are operated using a once-through fuel cycle, but some countries also deploy a limited recycle option.
Moderator (neutron)	Material used to lower the energy level of neutrons (from fast to thermal) that are generated from fission. Moderators are materials like natural water, heavy water, or graphite. The energy of the neutron is lowered due to collisions with the moderator atoms.

PWR	PWR or pressurized water reactor is a common nuclear power reactor design in which very pure water is heated to a very high temperature by fission, kept under high pressure (to prevent it from boiling), and converted to steam by a steam generator (rather than by boiling, as in a boiling-water reactor). The resulting steam is used to drive turbines, which activate generators to produce electrical power. A PWR essentially operates like a pressure cooker, where a lid is tightly placed over a pot of heated water, causing the pressure inside to increase as the temperature increases (because the steam cannot escape) but keeping the water from boiling at the usual 212°F (100°C). About two-thirds of the operating nuclear reactor power plants in the United States are PWRs.[24]
Ramp Rate	Ramp rate is the speed at which an electricity generator can increase or decrease generation.
RELAP 7	RELAP 7 or the Reactor Excursion and Leak Analysis Program-7 is the nuclear reactor system safety analysis code currently under development at Idaho National Laboratory for the Risk Informed Safety Margin Characterization Pathway as part of the Light Water Reactor Sustainability Program. It is an evolution in the RELAP-series reactor systems safety analysis applications. The RELAP-7 code development is taking advantage of the progresses made in the past three decades to achieve simultaneous advancement of physical models, numerical methods, coupling of software, multi-parallel computation, and software design.[26]
RISMIC	RISMIC or Risk-Informed Safety Margins Characterization is being developed to support plant decisions for risk-informed margins management with the aim to improve economics, reliability, and sustain safety of current nuclear power plants (NPPs). Goals of the RISMC Pathway are twofold: (1) Develop and demonstrate a risk-assessment method coupled to safety margin quantification that can be used by NPP decision makers as part of their margin recovery strategies, and (2) Create an advanced "RISMC toolkit" that enables more accurate representation of NPP safety margin.
SMR	SMR or small modular reactor are nuclear power plants that are smaller in size (300 MWe or less) than current generation base load plants (1,000 MWe or higher). These smaller, compact designs are factory-fabricated reactors that can be transported by truck or rail to a nuclear power site.

SCWR	The super critical water reactor or SCWR is a Generation IV advanced reactor design (https://www.gen-4.org/gif/jcms/c_42151/supercritical-water-cooled-reactor-scwr). The concept may be based on current pressure vessel or on pressure tube reactors, and thus use light water or heavy water as moderator. Unlike current water-cooled reactors, the coolant will experience a significantly higher enthalpy rise in the core, which reduces the core mass flow for a given thermal power and increases the core outlet enthalpy to superheated conditions.