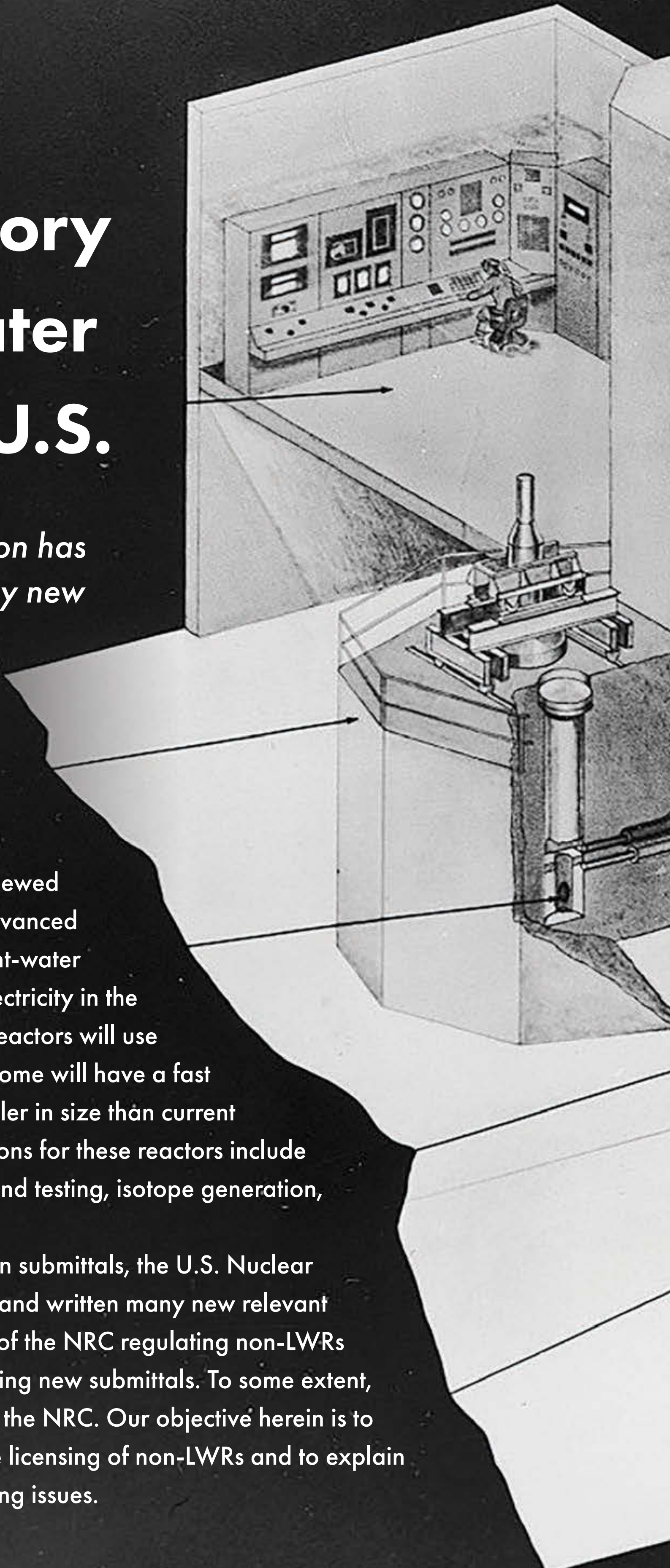
Regulatory history of non-light-water reactors in the U.S.

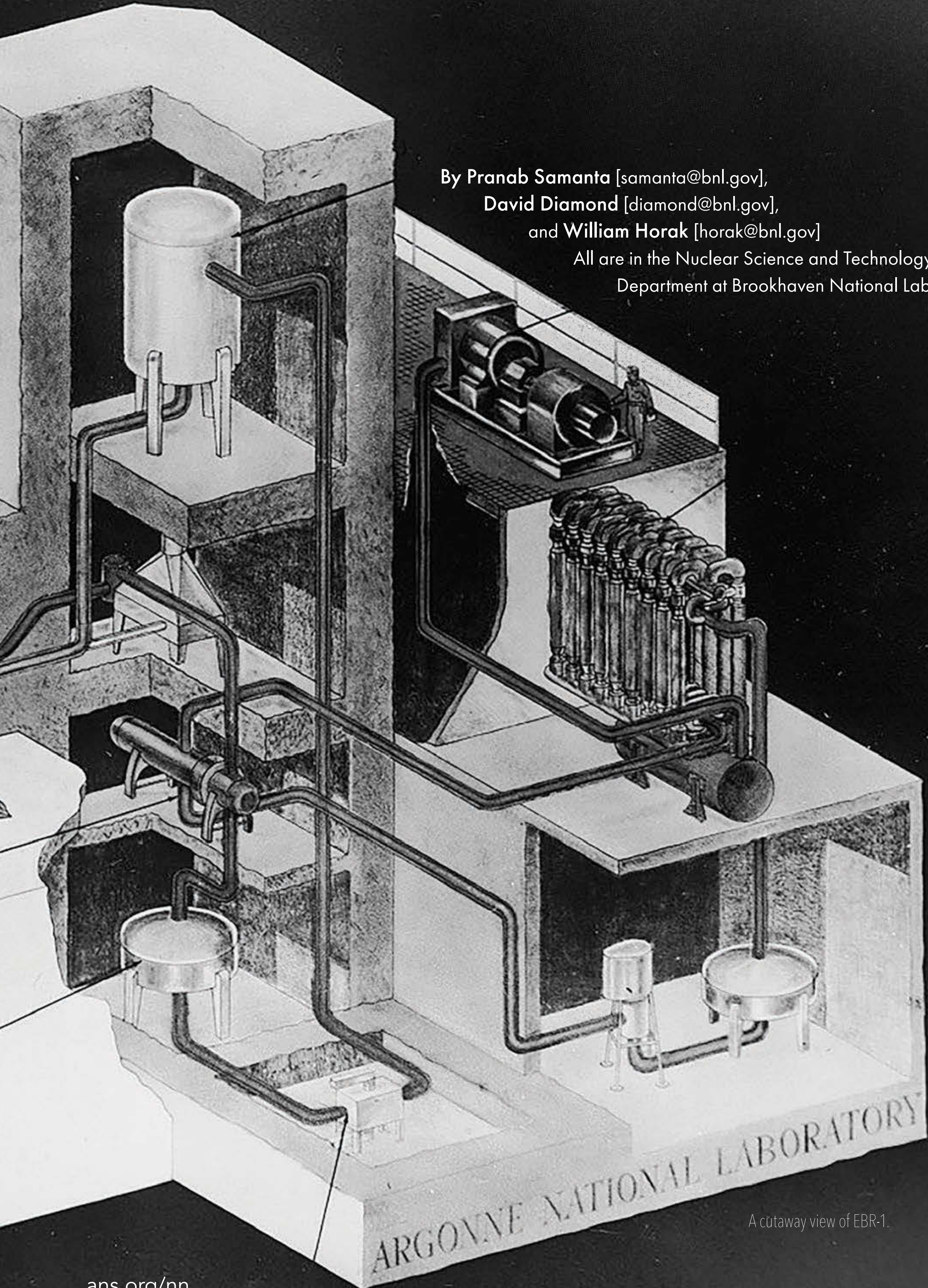
The Nuclear Regulatory Commission has studied issues and has written many new relevant documents to prepare for potential application submissions for non-LWRs.



Over the past several years there has been renewed interest in the development and licensing of advanced reactors that will be very different from the light-water reactors that are currently used to generate electricity in the United States. For example, some advanced reactors will use gas, liquid metal, or molten salt as a coolant, some will have a fast neutron spectrum, and some will be much smaller in size than current generation LWRs. The many possible applications for these reactors include electricity production, process heat, research and testing, isotope generation, and space applications.

To prepare for potential non-LWR application submittals, the U.S. Nuclear Regulatory Commission has studied the issues and written many new relevant documents. In addition, there is a long history of the NRC regulating non-LWRs that might be useful to study to help in addressing new submittals. To some extent, this has been chronicled in general histories of the NRC. Our objective herein is to describe the NRC's history specifically with the licensing of non-LWRs and to explain some of the most salient regulatory and licensing issues.

By **Pranab Samanta** [samanta@bnl.gov],
David Diamond [diamond@bnl.gov],
and **William Horak** [horak@bnl.gov]
All are in the Nuclear Science and Technology
Department at Brookhaven National Laboratory



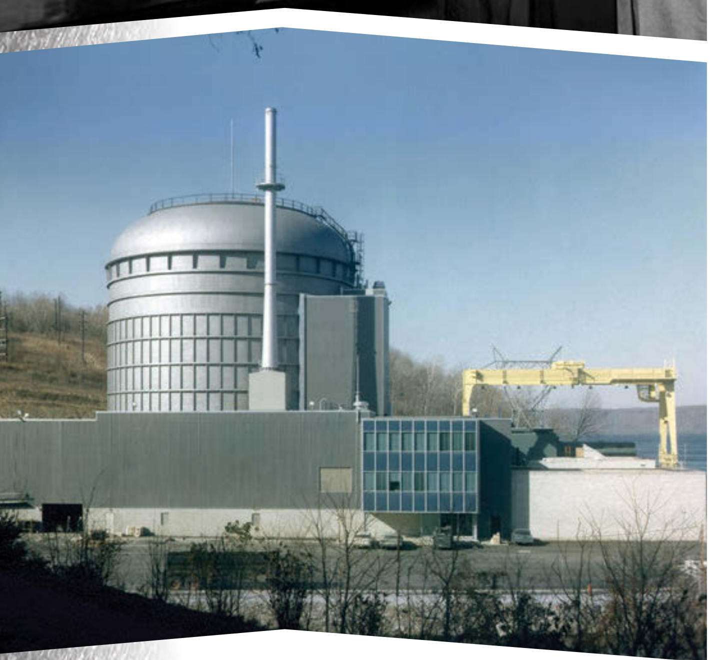
A cutaway view of EBR-1.

Early non-LWR history

The Atomic Energy Commission (AEC) was created by Congress via the Atomic Energy Act (AEA) of 1946. By the beginning of the 1950s, the AEC began to get industrial participation and initiate work at national laboratories that led to the building of many research, test, and prototype or demonstration reactors, including non-LWRs that were liquid sodium, organic liquid, heavy water, or gas-cooled.

During the 1950s, safety was regarded to be important, but the corresponding regulatory infrastructure was minimal. Safety evaluations involved writing hazards-summary reports that were evaluated by committees. Improved safety was one of the considerations in the revision of the AEA in 1954. The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory committee that was (and still is) to provide oversight on safety and report directly to the commission. The concerns over safety also led to the establishment of a separate Reactor Hazard Evaluation Staff within the AEC in 1955. Although the process was defined, the evaluation of hazards was difficult due to all the technical uncertainties. For example, there was limited experience in how properties of materials changed with irradiation and high stress levels, or how coolants would interact with metals at high temperature, or the impact of uncertainties in nuclear properties.

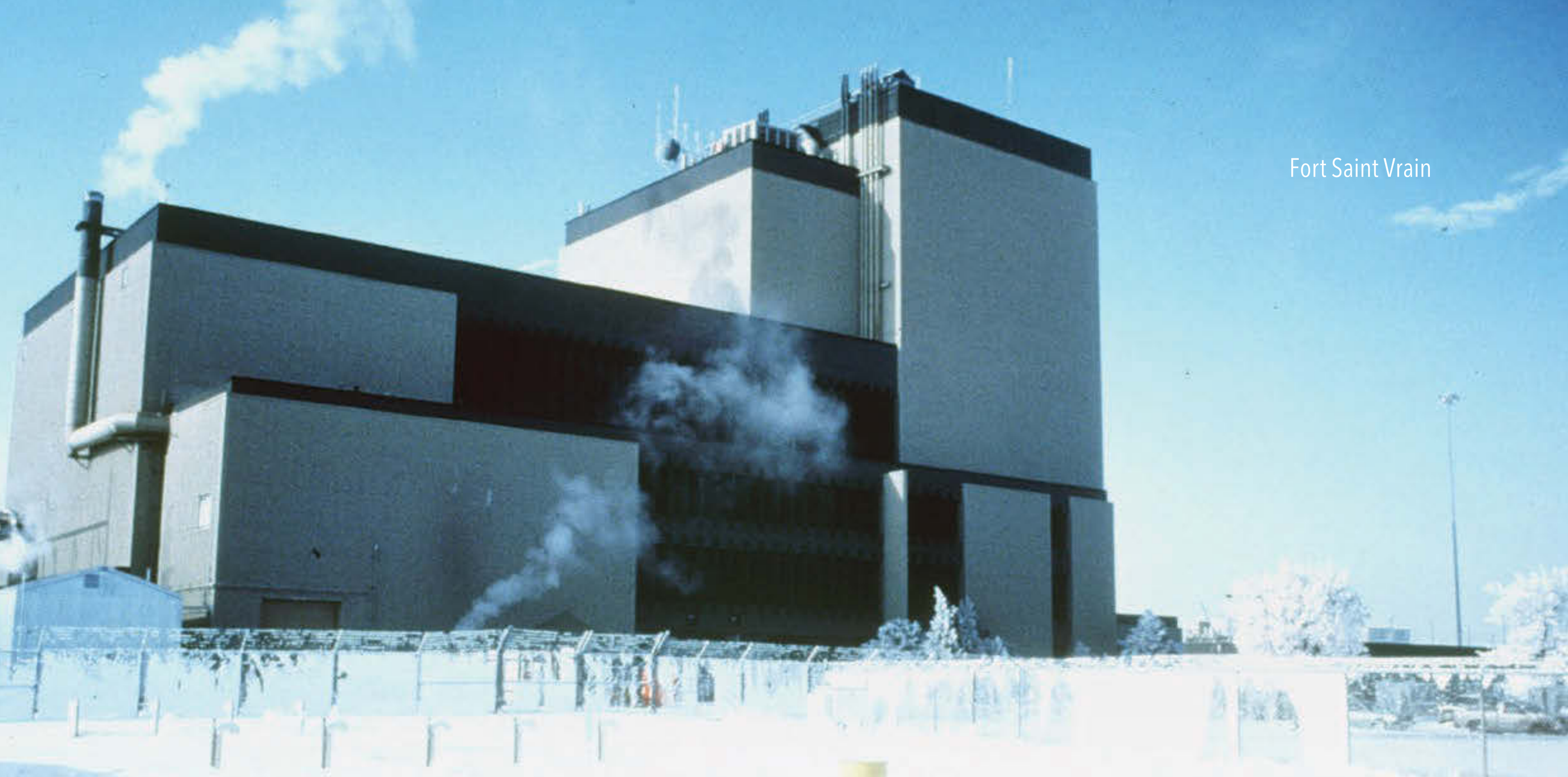
The Energy Reorganization Act (ERA) of 1974 marked the end of the AEC and the founding of the NRC to carry out the independent licensing and regulation of nuclear reactors. It was clear by the time the ERA was passed, and signed into law, that a body for regulation of nuclear safety needed to stand on its own.



Top: President Truman signs AEA
Bottom: Peach Bottom Atomic Power Station

Licensing gas-cooled reactors

Peach Bottom Atomic Power Station (PBAPS) Unit 1 was the first high-temperature gas reactor (HTGR) built in the United States. It was a 40-MWe demonstration plant, which operated at 2.4 kPa primary system pressure with a core inlet temperature of 350 °C and outlet temperature of 750 °C. The reactor went critical on March 3, 1966, and operated successfully until permanent reactor shutdown near the end of 1974, completing its demonstration mission. The goal of this plant was to demonstrate production of 538 °C steam from a reactor with good neutron economy and high fuel burnup.



Fort Saint Vrain

Fort Saint Vrain (FSV) was constructed based on the design of Peach Bottom and went into commercial operation on July 1, 1979. The reactor, with an output of 330 MWe, used high-temperature helium as the primary coolant to produce superheated and reheated steam at approximately 538 °C. The reactor fuel elements were a prismatic block design containing a mixture of carbides of uranium and thorium with tristructural isotropic (TRISO) coatings. FSV remained in commercial operation for a little more than 10 years. When the plant was shut down to repair a stuck control rod pair, numerous cracks were discovered in several steam generator main steam ring headers, and operation was terminated.

FSV was licensed under the provisions of the *Code of Federal Regulations*, Title 10—Energy, Part 50, Section 21 (10 CFR 50.21), "Class 104 Licenses; for Medical Therapy and Research and Development Facilities." FSV was considered by the AEC to be a "research and development reactor that could be shut down immediately if there were any real safety problems." It had a different oversight structure than that used at its contemporary LWRs. The Class 104(b) operating license issued for FSV and the NRC cognizant staff interpretation of the statutory basis for that license meant that FSV regulatory requirements were tailored to allow more flexibility than perhaps was afforded other contemporary plants that were licensed differently (under Section 103 of the AEA).

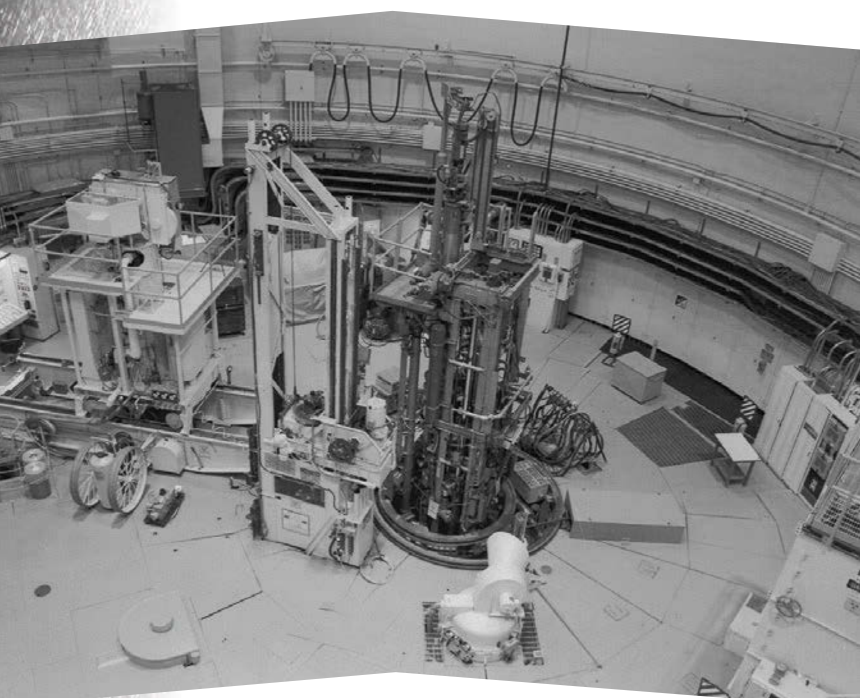
Reviewing the licensing and regulatory experience of PBAPS and FSV provided insights for reviewing later license applications. For example, developing clear safety analysis reports addressing principal design criteria that meet the safety functions underlying the NRC's general design criteria, and seismic and environmental qualifications for the cooling systems, among other equipment, were seen to be important. There was also concern over: the need to address industry codes and standards in a consistent manner to the new and innovative designs; defining a fire protection program and the associated mechanisms for responding to a fire to achieve hot and cold safe shutdown that is consistent with regulatory requirements; and maintaining detailed documentation of how calculations are done, how measurements are made (with all uncertainties accounted for), and how analytical and experimental results are reconciled. Some of the issues arose because FSV had a Class 104(b) license that didn't require such information.

The goal of the modular high-temperature gas-cooled reactor (MHTGR) was to develop a passively safe HTGR plant that was also economically competitive. To maintain the coated-particle fuel temperatures below damage limits during passive decay heat removal, the core's physical size had to be limited; hence, the maximum reactor power was to be about 200 MWt for a solid, cylindrical core geometry. This rating, however, was projected to not be economically competitive for electric power generation. This judgment led to the development of an annular core concept to enable larger cores with increased power capacity. Licensing activities included preapplication interaction with the NRC and submittal of numerous documents, including a preliminary safety information document.

The NRC conducted and documented a preapplication safety evaluation of the MHTGR. As stated in the safety evaluation, the general safety advantages of the MHTGR, like those of other HTGRs, were its slow response to core heat-up events, because of the large heat capacity and low power density of the core, and the very high temperature that the fuel can sustain before the initiation of fission-product release (~1,600 °C). Also, like other HTGRs, its major potential vulnerabilities derive from the need to protect metal components from continued exposure at elevated temperatures to hot helium during postulated transients and to prevent uncontrolled access of air or moisture to hot graphite and fuel particles.

The preapplication safety review defined policy issues that needed commission guidance for resolution. One was the definition of four event categories (abnormal operating experience, design-basis accidents, severe accidents, and emergency planning) that must be considered in a design. Other issues were a proposed mechanistic means of source term calculation and its use in assessing the need for conventional containment structure. Lastly, the NRC staff also discussed the emergency planning requirements and stressed that the need will depend on, but may not necessarily directly follow from, the acceptance of the mechanistic source term.

EBR-II



The mission of the Next Generation Nuclear Plant (NGNP) project was to develop, license, build, and operate a prototype MHTGR that would generate high-temperature process heat for use in hydrogen production and other energy-intensive industries while generating electric power at the same time. The Energy Policy Act of 2005 directed the Department of Energy to develop the NGNP prototype for commercialization and provided the licensing authority to the NRC. The DOE and the NRC jointly developed a licensing strategy and carried out activities that provided useful input for the regulatory basis for non-LWRs. The DOE decided in 2011 not to proceed into the detailed design and the license application phase of the project was not pursued.

Licensing liquid metal–cooled reactors

EBR-I was a 1.4-MWt test reactor that began operation in 1951. A loop design, it used electromagnetic pumps in the primary loop. It was cooled by a eutectic alloy of sodium and potassium and used a metal fuel. The plant suffered a partial core meltdown in 1955 during a series of reactivity tests; the reactor was unstable under certain flow conditions. A second core was designed and installed which addressed the stability problems, and was used until the program was terminated in 1963.

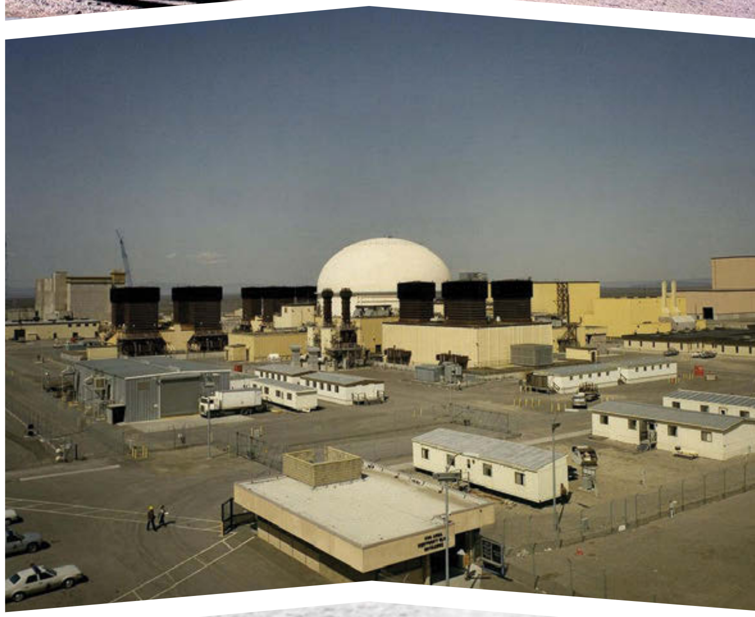
EBR-II was designed and built as a follow-on to the EBR-I project. EBR-II operated for more than 30 years, a record for a liquid metal–cooled plant in the United States. The plant was a pool design with metal fuel and used centrifugal pumps augmented by a single electromagnetic pump. Initially focused on further refining the "breeding cycle," it also demonstrated the inherent safety of the design and at the end of its life was used to test advanced metal fuel.

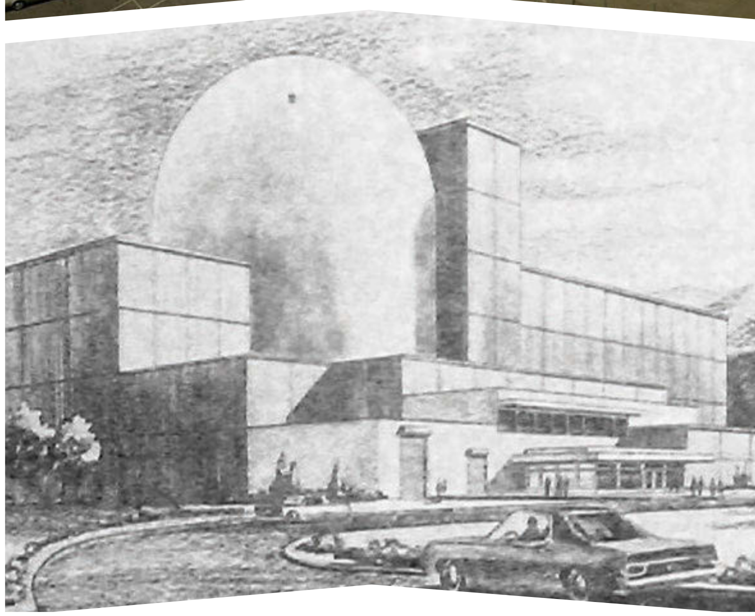
Fermi-1 was a three-loop sodium-cooled fast reactor designed for a nominal power of 300 MWt (100 MWe). The fuel was a uranium-molybdenum alloy placed in 105 core (or driver) subassemblies and 531 radial blanket subassemblies. In 1966, the plant suffered a partial meltdown of two subassemblies when flow was blocked to two channels. The damage was repaired, and the plant operated until 1972.

The Southwest Experimental Fast Oxide Reactor was a 20-MWt reactor fueled with mixed oxide fuel. The plant was built as a test reactor mainly to measure the Doppler reactivity coefficient that is an important contributor to the overall negative power coefficient in fast sodium-cooled reactors. The plant operated from 1969 until 1972, when the test program was completed.

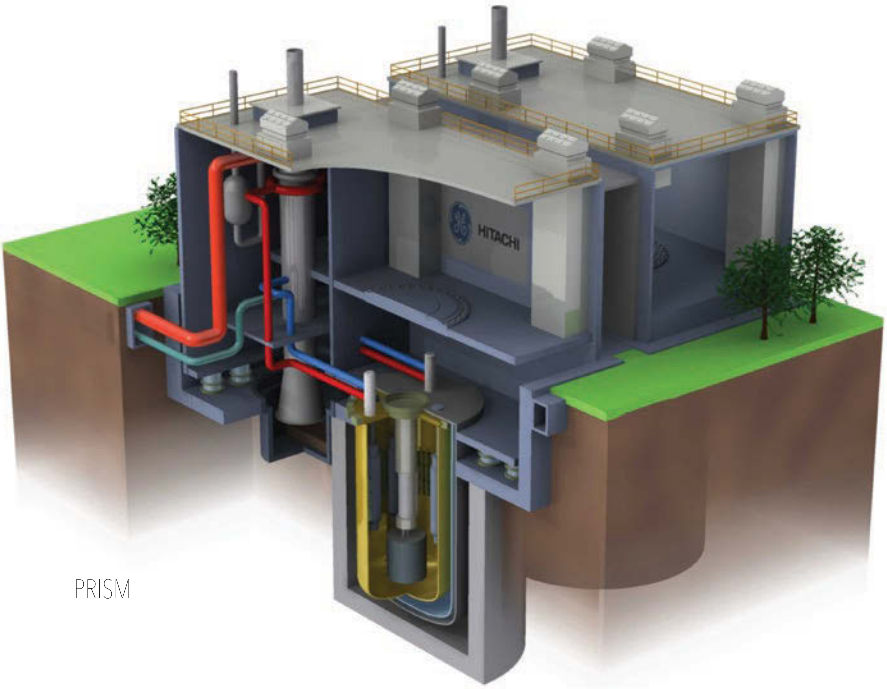
The Fast Flux Test Facility was a 400-MWt loop design that operated from 1982 to 1992. Although its primary mission was irradiation of materials for advanced reactors, much data was obtained on safety tests conducted as part of the program, including natural circulation decay heat removal and transients and loss-of-primary-coolant flow without reactor trip. Although the plant was not licensed by the NRC, a review was conducted by the NRC and the ACRS and a formal safety evaluation report (SER) was written. This was the last liquid metal–cooled reactor placed into operation in the United States.

The Clinch River Breeder Reactor (CRBR) was a 1,000-MWt (350-MWe) reactor that was to be constructed and operated under contract initially to the AEC (and later to the DOE). An SER for the application for a construction permit for the CRBR was issued in March 1983. Because of the extremely conservative nature of the principal design criteria, the staff concluded "that core disruptive accidents can and must be excluded from the design-basis accidents for the plant." A Memorandum of Findings, issued by the Atomic Safety Licensing Board in lieu of a construction permit in January 1984, resolved all outstanding issues regarding the construction permit, but the project was canceled.





Top to bottom: Fermi-1, Fast Flux Test Facility, Clinch River Breeder Reactor

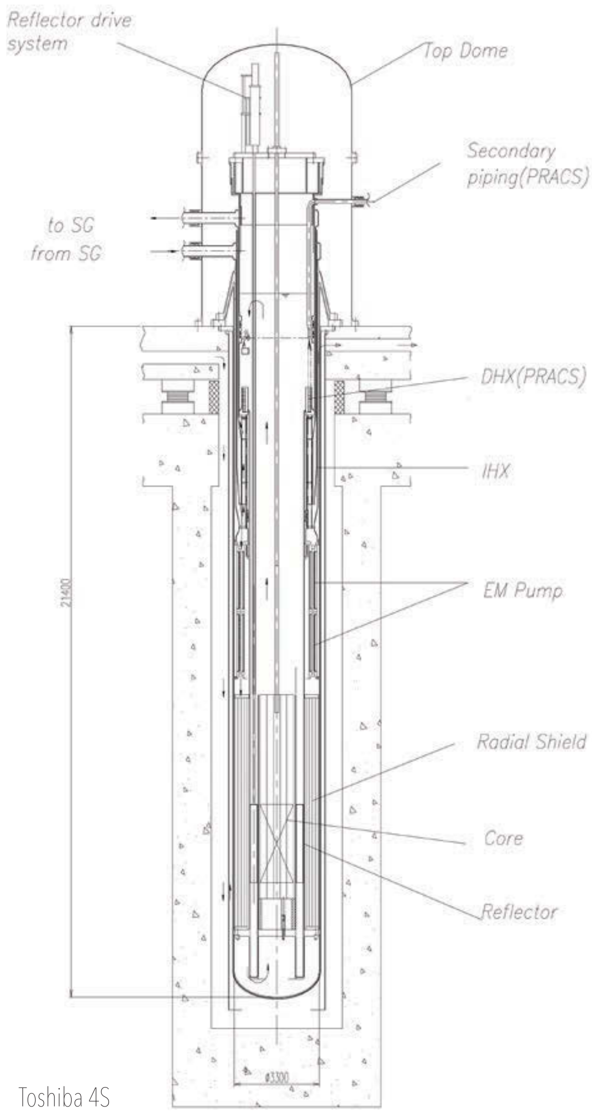


PRISM

After the cancellation of the CRBR, the DOE funded the development of several liquid metal–cooled reactor designs. The most developed of these were SAFR (canceled following the development of conceptual design and partial review by the NRC) and PRISM.

The PRISM design continued to evolve over the years and eventually included a number of variations ranging in power from 425 MWt to 1,000 MWt, with the standard design being 840 MWt. The reactor was a pool design using metallic fuel, with solid oxide fuel as a backup design. The design has different arrangements of fuel, driver, and blanket elements depending on whether the core is optimized for breeding, actinide burning, plutonium burning, or long life (so-called break-even). All designs have two intermediate heat exchangers that connect to a single steam generator.

The NRC conducted a thorough review of the 475-MWt design between 1986 and 1994. The NRC staff identified eight areas where the design deviated from LWR guidance, only one of which (Control Room and Remote Shutdown Area Design) was considered not eligible for a departure from LWR regulations. After revisions to the design, the staff, with ACRS concurrence, concluded that there were "no obvious impediments to licensing the PRISM design."

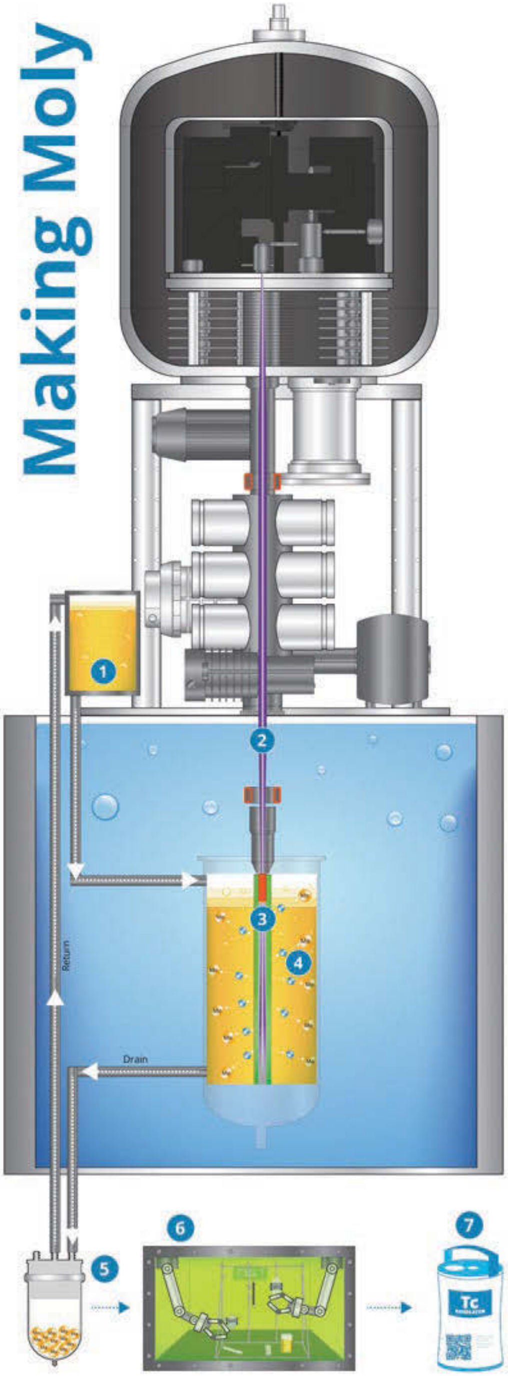


Toshiba 4S

The Toshiba 4S (Super-Safe, Small, and Simple) was a 30-MWt (10-MWe) pool-type reactor designed for remote locations with small grids. The reactor was designed with a long-life core (30 years with no refueling) and utilized metallic fuel. A single loop, with electromagnetic pumps, was used for steam generation to a single turbine. This would meet the current NRC definition of a microreactor. From 2007 to 2013, Toshiba submitted a series of technical reports, but the review ceased in 2013 without any review documents.

As a result of all the aforementioned experience, in 2012, Sandia National Laboratories led a Sodium Fast Reactor Safety and Licensing Research Plan that proposed "potential research priorities for the [DOE] with the intent of improving the licensability of the Sodium Fast Reactor." The report recommended that in all areas a structured knowledge management program was needed to effectively maintain and access the operational knowledge obtained during the U.S. fast reactor program prior to 1994.

Making Moly



SHINE molybdenum-99 generator: 1. Target solution; 2. Accelerator; 3. Fusion Chamber; 4. Fission target; 5. Moly extraction; 6. Purification; 7. Distribution

Licensing liquid-fuel reactors

In the early days of the NRC, there were 12 liquid-fuel reactors licensed with fuel in an aqueous solution and thermal power levels of from 5 W to 50 kW. More recently, there have been two licensing activities for aqueous liquid fuel for isotope generation, and this experience might prove to be relevant to liquid-fuel molten salt reactors and to microreactors as well. One was the 220 kWt Aqueous Homogeneous Reactor (AHR), for which Babcock & Wilcox Technical Services Group submitted preapplication material in 2010. The second was the application for a construction permit in 2013 by SHINE Medical Technologies for an accelerator with an aqueous target. In both cases an aqueous solution of uranyl sulfate with low-enriched uranium was used. The objective of these projects was primarily to generate the fission product molybdenum-99, an extremely useful medical isotope, which would be separated from the fuel at the plant site.

The NRC convened a panel to produce licensing guidance taking into account the unique features of an AHR: the fuel being in solution; the fission product barriers being the vessel and attached systems; the production and release of radiolytic and fission product gases and their impact on operations and their control by a gas management system; and the movement of fuel into and out of the reactor vessel. An interim staff guidance (ISG) report for "Radioisotope Production Facilities and Aqueous Homogenous Reactors" was then written.

The ISG was applicable to the SHINE facility, which applied for its construction permit after it was written. Although the accelerator target in the facility is not a nuclear reactor, "its safety analysis must consider phenomena analogous to those of an AHR."

The AHR never submitted a license application, and so the NRC never did a formal review of the reactor. The SHINE facility received a construction permit after the NRC staff issued an SER and after an environmental impact statement was written by the NRC.

Licensing heavy-water reactors

During the period from 1989 to 1995 the NRC reviewed documents for the CANDU-3 reactor, and during the period 2002-2005 there was a preapplication review of the ACR-700. Both reactor designs were based on the CANDU reactors that had been built, and successfully operated, in Canada and other countries.

The NRC staff documented the policy issues for the CANDU-3 along with those for several other new reactors and wrote a safety assessment report for the ACR-700.

To the present and beyond

"Regulation of Advanced Nuclear Power Plants; Statement of Policy" was published by the NRC in 1986 and revised in 2008 to include consideration of security, and it continues to provide the overall guidance of all activities relating to advanced nuclear power plants. The commission defined its expectation for advanced reactors as part of the policy statement: "Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the environment and public health and safety and the common defense and security that is required for current generation LWRs. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions." Details about the development and utilization of the policy statement on the regulation of advanced reactors can be found in NUREG-1226.

The NRC has initiated rulemaking to revise regulations and guidance for emergency preparedness (EP) for small modular reactors and other new technologies, such as non-LWRs and medical isotope facilities, for a consequence-based approach to establishing requirements, as necessary, for offsite EP, and is pursuing a limited scope rulemaking effort that would evaluate possible performance criteria and alternative physical security requirements for advanced reactors.

NRC staff is developing a "technology-inclusive regulatory framework" for optional use by applicants for new commercial advanced nuclear reactor licenses, as required in Section 103 of the 2019 Nuclear Energy Innovation and Modernization Act.

Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," presents design criteria addressing two specific design concepts, sodium-cooled fast reactors and MHTGRs, as well as generally applicable criteria for lead-cooled fast reactors, gas-cooled fast reactors, fluoride-salt high-temperature reactors, and liquid-fuel molten salt reactors.

NRC staff prepared a number of documents (SECY) recommending positions for resolving issues related to non-LWR designs that were approved by the commission. NRC staff has developed functional performance criteria for containment and stated its belief that a mechanistic approach could be applied to non-LWR designs for accident source terms and siting subject to availability of adequate tools and analysis approaches, allowing future applicants to consider reduced distances to exclusion area boundaries.

In preparing to review and regulate a new generation of non-LWRs, the NRC developed its vision and strategy for mission readiness in assuring safe, effective, and efficient licensing of non-LWRs. The vision and strategy, when implemented, is developed to address potential inefficiencies in the current licensing process based on LWR criteria and provide for regulatory certainty for non-LWR applicants.

The NRC has also published a regulatory review roadmap for non-LWRs—"A Regulatory Review Roadmap for Non-Light Water Reactors," December 2017 (ML17312B567)—providing guidance to staff reviewers and applicants.

This article is a condensation of a more detailed report funded by the Nuclear Regulatory Commission. The report is available at nrc.gov/docs/ML1928/ML19282B504.pdf.