REPORT

Study outlines reactor designs that may be ready for deployment by decade's end

A Department of Energy report examines the feasibility of deploying small, modular, nuclear reactors in remote communities.

After a review of designs of nine small modular nuclear reactors (SMR), the Department of Energy is optimistic that there are no substantive technical issues to hinder the development and availability for deployment of the most technically mature SMRs by the end of the decade.

The DOE, through its Office of Nuclear Energy, Science and Technology, in June released a study on SMRs, "Report to Congress on Small Modular Nuclear Reactors" (dated May 2001). The nine reactors analyzed in the report ranged from power levels of 16.4 MWe to 50 MWe, with refueling frequencies from about once per year to once every 15 years. The most popular fuel type was UO_2 pins or particles.

An issue requiring further study, the DOE noted, is the lack of supporting infrastructures for supplying fuel for the SMRs. Depending on the fuel type, suitable fuel fabrication facilities may not currently exist, and would need to be constructed and qualified. This may be of particular concern for gas-cooled reactors using graphite fuel, although potential pebble-bed reactor development might alter this situation. Further, since some of the new SMRs are gas- or liquid-metal-cooled, there may be licensing questions that are outside the traditional light-water reactor experience of the Nuclear Regulatory Commission. Also, the DOE questioned to what extent the elimination of a conventional containment would be acceptable to the NRC.

The report looks at the feasibility of deploying SMRs in remote areas that are deficient in transmission and distribution infrastructures. These areas, according to the DOE, pose special challenges in providing electric power because it is likely that there will be a higher expense in operating small reactors, difficulty in shipping and storing fuel, a shortage of trained personnel, and power requirements that are relatively small and variable.

The report contains the DOE's review of existing SMR designs and proposed SMR concepts from domestic and foreign sources.

Table I: Summary of Small Modular Reactor Designs and Concepts

		CAREM	ENHS	IRIS-50	KLT-40	MRX	MSBWR	RS-MHR	TPS	4S
Designer		CNEA	UCB	W	OKBM	JAERI	GE/ Purdue U.	GA	GA	CRIEPI
Type		Integral PWR	LMR	Integral PWR	PWR	Integral PWR	BWR	HTGR	PWR	LMR
Rating		25 MWe	50 MWe	50 MWe	35 MWe	30 MWe	50 MWe	10 MWe	16.4 MWe	50 MWe
Primary system pressure		12.3 MPa	N/A	-	13 MPa	12 MPa	-	-	3 MPa	N/A
Reactor vessel	Height	11 m	19.6 m	14-16 m	3.9 m	9.4 m	8.5 m	8 m	11.6 m	23 m
	Diameter	3.1 m	3.2 m	3.5 m	2.2 m	3.7 m	3.5 m	3.4 m	2.8 m	2.5 m
Reactor core	Height	1.4 m	1.25 m	1.8 m	0.95 m	1.4 m	1.9 m	3.6 m	1 m	4 m
	Diameter	1.3 m	2 m	1.5 m	1.2 m	1.5 m	3.1 m	3 m	1 m	0.8 m
Avg. power density*		55 kW/l	6 kW/m	13 kW/m	155 kW/l	42 kW/l	8.3 kW/m	4 kW/l	95 kW/l	61 kW/l
Fuel/type		UO_2 pins	U-Zr metal	UO_2 pins	U-Al alloy	UO_2 pins	UO_2 pins	UO_2 particles	UZrH pins	U-Zr metal
Fuel enrichment		3.4 %	13 %	4.95 %	-	4.3%	5 %	19.9%	19.9%	~15 %
Refueling frequency (percent replaced)		~ 1 year (50%)	15 years (100 %)	5-9 years	2-3 years (100%)	~ 4 years (50%)	10 years	6-8 years	1.5 years (50%)	10 years (100%)
Coolant flow rate		410 kg/s	0.51 m/s	-	722 kg/s	1250 kg/s	620 kg/s	-	419 kg/s	633 kg/s
Core inlet temperature		284 °C	400 °C	-	278 °C	283 °C	279 °C	500 °C	182 °C	355 °C
Core outlet temperature		326 °C	550 °C	-	318 °C	298 °C	14.3% quality†	850 °C	216 °C	510 °C

* the amount of power generated in a given volume of the reactor core kW per liter, or power in a given length kW per meter

† BWRs measure performance in terms of steam quality (percent by weight of vapor versus liquid) at the core outlet

"-"= Not Provided

N/A = Not Applicable

Source: DOE

Plant characteristics were evaluated on the basis of their ability to satisfy the criteria of inherent safety, cost-effectiveness, resistance to sabotage and diversion of nuclear materials, infrequent refueling, the level of factory fabrication, and transportability to remote sites.

The report is available on the Web at <www.ne.doe.gov/analysis/mod-small-reactors.html>.

Technical assessment

The SMRs all make greater use of inherent safety features than do existing larger commercial plants, the DOE noted. For example, inherent safety may be achieved through fuel designs that are able to withstand extreme temperatures without loss of the fuel's integrity. Almost all of the designs and concepts rely on natural circulation of the coolant in emergency modes, and many SMRs also rely on natural circulation for cooling of the core during normal operation.

The power plants would not be appealing targets for sabotage or diversion, the DOE stressed, because most of the SMRs studied use small inventories of low-enriched uranium (LEU)-based fuels (defined as less than 20 percent U-235 content of the total uranium).

Spent-fuel storage requirements vary, with some transportable SMRs being refueled at maintenance centers.

The degree of factory fabrication and modular construction varies greatly between the designs and concepts reviewed in the report. Factory fabrication of a power plant in modules results in shorter construction time, ease of transportability, and simpler onsite assembly in remote locations, the report said.

TABLE II: COST INFORMATION FOR GENERIC 50-MWE AND 10-MWE SMRS (YEAR 2000 DOLLARS)

ITEM	Minimum 50 MWe	Minimum 10 MWe	Maximum 50 MWe	Maximum 10 MWe
Unit capital cost, $/kWe	1 950	3 950	5 067	11 330
Levelization period, Yrs	20	20	20	20
Constant $ fixed charge rate, %	11.2	11.2	11.2	11.2
Levelized capital cost, M$/yr	10.9	4.4	28.3	12.6
O&M costs, M$/yr	5.5	2.6	9.4	5.6
Fuel costs, M$/yr	3.7	0.7	4.2	0.8

(Source: DOE)

Licensing, regulatory issues

The current NRC regulatory framework for ensuring plant safety has three main elements: reactor safety, radiation safety, and plant security. Because many SMRs use different approaches to satisfy areas of safety, including inherent safety characteristics, a more simplified licensing and regulatory process than that used for LWRs would be appropriate, the DOE advised.

Issues that could affect some SMRs' regulatory approval include pyrocarbon-coated particle fuel performance and reactor containment design.

Economic competitiveness

The report estimated and compared the economic competitiveness of a generic 50-MWe and 10-MWe SMRs with current generation costs of electricity in selected remote locations. For this comparison, the delivered cost of electricity charged to industrial customers by selected utilities in Alaska and Hawaii was used as a baseline for remote or isolated communities. For a generic 50-MWe SMR, the range of electricity cost was estimated at 5.4 to 10.7 cents per kilowatt-hour (¢/kWh). The range of cost for a 10-MWe SMR was 10.4 to 24.3 ¢/kWh. Since the industrial rate for electricity charged by selected Alaska and Hawaii utilities varied from 5.9 to 36.0 ¢/kWh, (depending on the location, type, and size of the

TABLE III: ESTIMATED MINIMUM COST OF ELECTRICITY (CENTS/KWH, YEAR 2000 $S)

	Capacity	
	50 MWe	10MWe
Capital	2.9	5.9
O&M	1.5	3.5
Fuel	1.0	1.0
Total	5.4	10.4

(Source: DOE)

TABLE IV: ESTIMATED MAXIMUM COST OF ELECTRICITY (CENTS/KWH, YEAR 2000 $S)

	Capacity	
	50 MWe	10MWe
Capital	7.2	16.1
O&M	1.5	7.2
Fuel	1.1	1.1
Total	10.7	24.3

(Source: DOE)

power plant, fuel cost, and ease of transporting fuel), SMRs could be a competitive option, according to the DOE.

SMR review

Following are reviews of the nine SMR designs examined by the DOE:

■ The **CAREM** project by the Argentinean National Atomic Energy Commission (CNEA) and commercial supplier INVAP is based on a simplified pressurized water reactor design with ratings of 100 megawatts-thermal (MWt) and 25 megawatts-electric (MWe). The design includes a helical steam generator above the core fed by natural circulation so the unit has no main coolant pumps or pressurizer. It has hydraulic control rod drives, a large-volume primary coolant system, and a negative temperature coefficient, which is an inherent effect that means the reactor power will automatically drop when there is an increase in temperature, bringing power and temperature back down.

The most innovative feature of this design, according to the DOE, is that the entire primary coolant system is contained within the reactor pressure vessel, so it is termed an "integral PWR." The integral reactor vessel contains the reactor core and support structures, steam generators, and the control rod system. The primary system is self-pressurized by the steam generated inside the vessel. A steam chamber, located near the top of the reactor vessel, is used to regulate pressure against variations in coolant temperature.

Pumps are eliminated in the primary system, and core cooling is accomplished by natural convection.

CAREM uses standard PWR fuel technology with 3.4 percent enriched uranium-oxide fuel contained in fuel pins. These materials, according to the DOE, are not attractive for use directly in weapons, and provide the first level of defense for diversion.

The possibility and consequences of a loss-of-coolant-accident (LOCA) are greatly reduced since the entire primary system is enclosed within the reactor pressure vessel. Vessel penetrations below the core, such as piping connections or instrumentation locations, are avoided, assuring that a large inventory of water is always available for passive cooling in the event of a break.

One of the main disadvantages of this design, according to the DOE, is its annual refueling requirements.

CNEA currently has legislative authorization from the Argentinean government to seek financing and provincial approval for a site to build a prototype.

■ The **Encapsulated Nuclear Heat Source (ENHS)** is a concept being developed under the NERI (Nuclear Engineering Research Initiative) program by a consortium led by the University of California–Berkeley. It is a liquid-metal-cooled reactor (LMR) that can use either lead (Pb) or a lead-bismuth (Pb-Bi) alloy as the reactor coolant. The lead-based coolants are chemically inert with air and water, have high boiling temperatures, and good heat transfer characteristics for natural circulation. The ENHS has a long core life.

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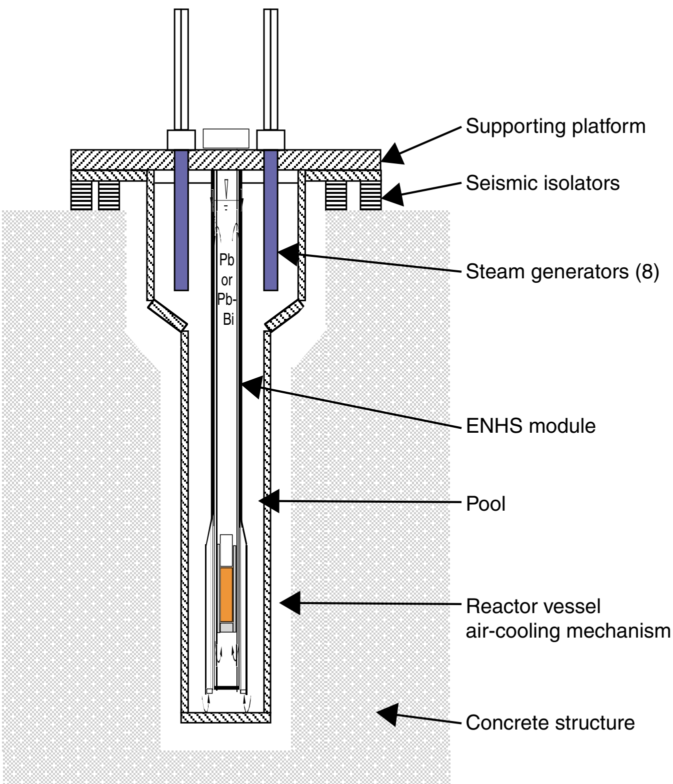
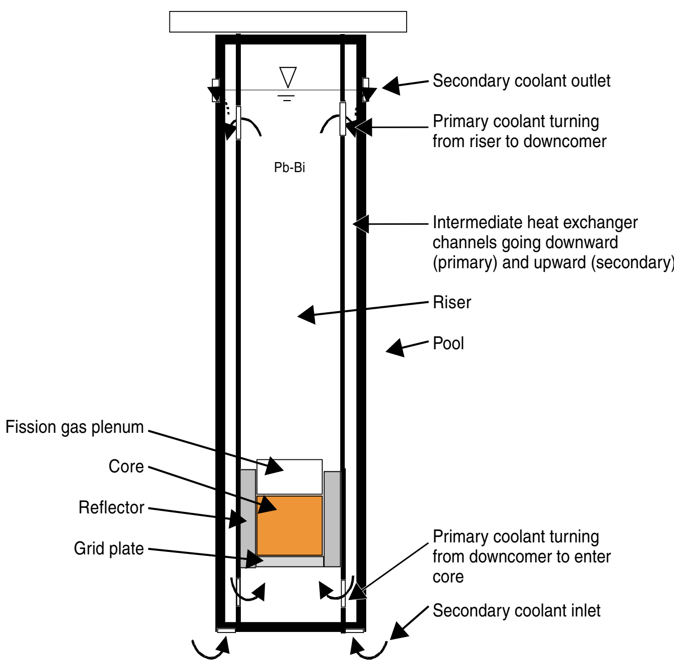


A schematic vertical view of an ENHS reactor (not to scale) (Source: UC-Berkeley)



A schematic vertical cut through an ENHS module (not to scale) (Source: UC-Berkeley)

The ENHS concept is based on the idea of encapsulating the reactor core inside its own vessel as a module, with no external piping connections. The steam generators are separate modules that are inserted into a pool. There are no pumps, valves or pipes located within the primary and intermediate cooling systems.

The ENHS fuel is a metallic alloy of uranium and zirconium (U-Zr) or, optionally, uranium, plutonium, and zirconium (U-Pu-Zr), and exhibits good stability under irradiation. The fuel is contained in cylindrical fuel pins. The reactor can operate at full power for 15 years using either U-Pu-Zr metallic fuel having about 11 percent plutonium, or U-Zr metallic fuel using uranium enriched to 13 percent U-235. The core consists of fuel assemblies having 217 rods in a hexagonal array with the central location reserved for a large safety element, which can assure complete reactor shutdown.

The ENHS module is manufactured and fueled in a factory, and shipped to the site as a sealed unit with solidified Pb (or Pb-Bi) filling the vessel up to the upper level of the fuel rods. With no mechanical connections between the reactor module and the secondary system, the module is easy to install and replace, similar to using a battery. After installation, hot coolant is pumped into the vessel to melt the solid lower part. At the end of its life, the ENHS module could be removed from the reactor pool and stored on site until the decay heat drops to a level that lets the coolant solidify. The module with the solidified coolant would then serve as a shipping cask. The total weight of an ENHS module, however, when fueled and when loaded with Pb-Bi to the upper core level is estimated to be 300 tons, which could pose a shipping challenge, especially to remote areas.

Since the ENHS is only at the conceptual stage, it is not likely to be ready for deployment in this decade.

■ The **International Reactor Innovative and Secure (IRIS)** concept is being developed under the NERI program by an international consortium led by Westinghouse Electric Company. IRIS is a PWR designed to address the requirements of proliferation-resistance, enhanced safety, improved economics, and waste reduction. **IRIS-50** is a concept variant using a low power rating (50 MWe) and natural circulation.

One of the notable IRIS-50 characteristics is the integral reactor vessel. The reactor vessel and other components are surrounded by a steel containment, spherical in shape and estimated to be about 16 to 18 meters (m) in diameter. The reactor core has 21 fuel assemblies and a diameter of 1.5 m. The active core height is 1.8 m. The IRIS-50 design features a five-year refueling interval using uranium oxide fuel with an enrichment of 5 percent U-235. Higher enrichment would allow a refueling interval nearly double (nine years), but the higher fuel burnup would require additional testing and analysis for licensing approval.

The fuel materials and the fuel assembly design are similar to current commercial PWR technology. Since the control rod system is partially contained within the reactor vessel, control rod ejection is eliminated as an accident initiator that would lead to an uncontrolled power increase. This is a safety improvement compared with existing PWRs, the DOE noted.

The probabilities of steam line breaks and steam generator rupture accidents are minimized by having the steam generator operate at the same design pressure and temperature as the reactor vessel. A small leak can be controlled with isolation valves, eliminating the need for steam line safety valves. The probability of a small-break LOCA also is reduced because of fewer reactor vessel penetrations.

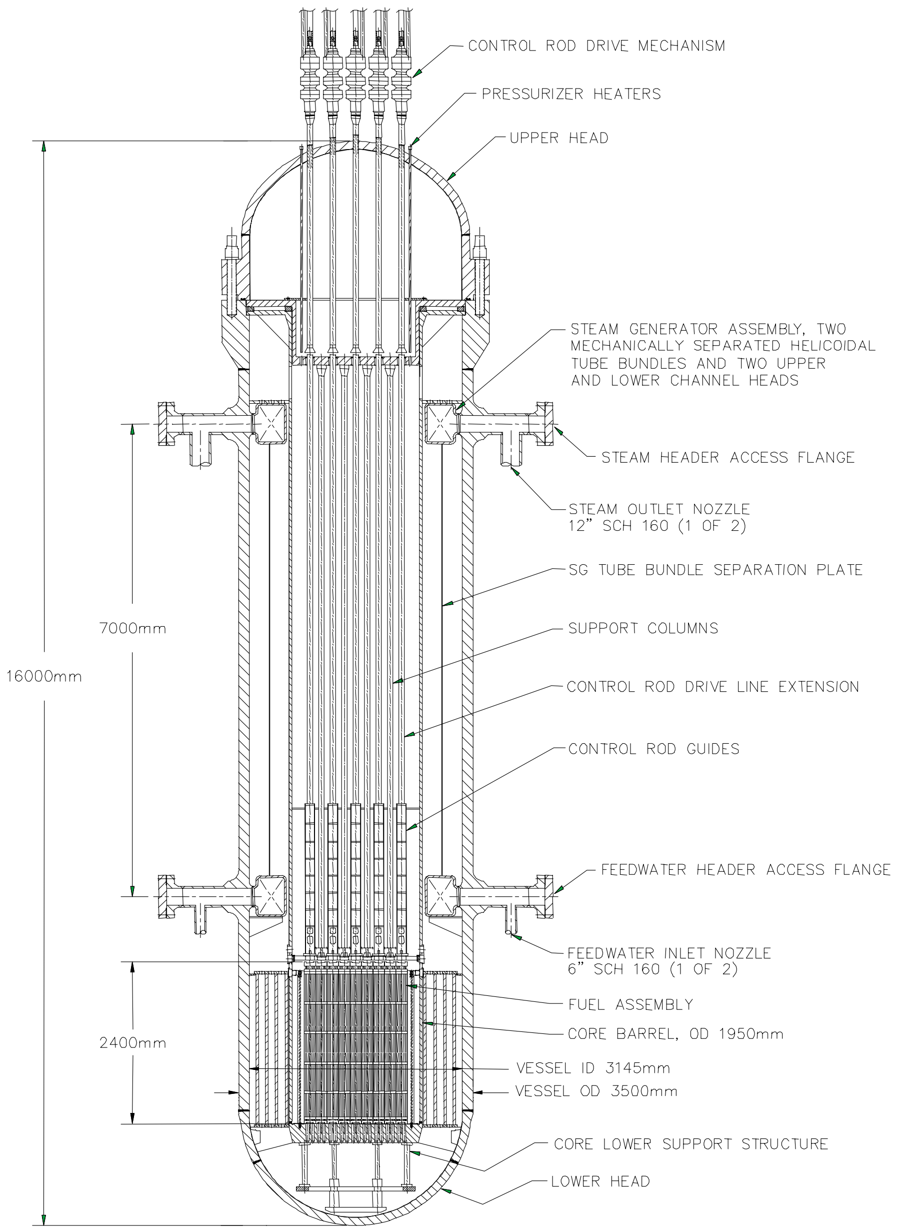
The IRIS-50 concept is sized to have the major components transportable, and its fuel is essentially the same as commercial PWR fuel and has the same diversion- and proliferation-resistance.

As part of another NERI project, barge-mounted construction also is under consideration for transportation and operation. In this case, the core would remain sealed and fully inaccessible at the remote location and returned to a center for refueling and maintenance.

IRIS-50 has the potential for deployment in this decade.

■ The DOE report called the **KLT-40** a proven, commercially available, small PWR system because its design is based entirely on the nuclear steam supply system used in Russian icebreakers. The unit is a portable, floating, nuclear power plant intended mainly for electric power generation, but it also possesses the capability for desalination or heat production. Although the reactor core is cooled by forced circulation of pressurized water during normal operation, the design relies mainly on natural convection in the primary and secondary coolant loops in all emergency modes.

The plant is mounted on a barge, complete with the nuclear reactor, steam turbines, and other support facilities. It is designed to be



IRIS-50 (50 MWe) natural circulation integral reactor vessel layout (Source: WEC) transported to a remote location and connected to the energy distribution system in a manner similar to the Mobile High Power nuclear power plant operated by the U.S. Army in the 1970s. The designer and supplier of the KLT-40 is the Russian Special Design Bureau for Mechanical Engineering.

Fuel is a uranium-aluminum metal alloy clad with a zirconium alloy. An inventory of 200 kg of U-235 gives a fairly high core power density of 155 kW per liter on average, and the fuel may be high-enriched uranium (U-235 content at or above 20 percent). The fuel assembly structure and manufacturing technology are proven, the DOE noted, and its reliability has been verified by the long-term operation of similar cores.

There are four coolant pumps in the primary circuit of the KLT-40, feeding four steam generators. The secondary system consists of two turbogenerators with condensate pumps, main and standby feed pumps, and two condensers. In the steam condensers, up to 35 MWt energy can be transferred to a desalination plant via an intermediate circuit.

The design of the KLT-40 includes a steel containment vessel capable of withstanding overpressure conditions. There also is a passive-pressure suppression system for condensing steam that escapes from the KLT-40 system into the containment building.

The inherent safety characteristics of the KLT-40 include a large negative temperature coefficient for the reactor core, where increasing core temperature lowers core power. This is achieved in the KLT-40 design without the use of soluble boron in the coolant water. Instead, a large quantity of burnable poison is used in the fuel and more control rods are incorporated in the design to ensure a cold shutdown.

Two features of KLT-40 are factory fabrication and transportability over water to remote locations. Although the KLT-40 requires refueling every two to three years, the transportability of the entire plant to maintenance centers provides enhanced proliferation-resistance, the DOE said.

■ The **MRX** design is a marine power reactor originally designed for an icebreaker and scientific observation ship. Like CAREM, it is an integral PWR with the steam generator and pressurizer installed inside the pressure vessel, although there are other major components of the primary coolant system that are outside of the reactor vessel. A large water inventory increases the thermal capacity of the primary system and reduces radiation damage to the vessel. The designer is the Japan Atomic Energy Research Institute.

The design of fuel elements is based on what the DOE called well-developed PWR fuel technology. The MRX uses 4.3 percent enriched uranium oxide fuel contained in fuel pins. The reactor core consists of 19 fuel assemblies, 13 of which contain control rods. Six of the control rod clusters are used for reactivity control and the other seven for reactor shutdown.

Since the reactor core has a low power density, the MRX responds slowly to temperature variations. The design of the reactor's core allows a cold shutdown to be assured without using a boron solution in the primary coolant water, even with one control rod cluster withdrawn.

The MRX design adopts a partially passive decay-heat removal system, where the residual heat is removed from the primary coolant by means of the steam generator.

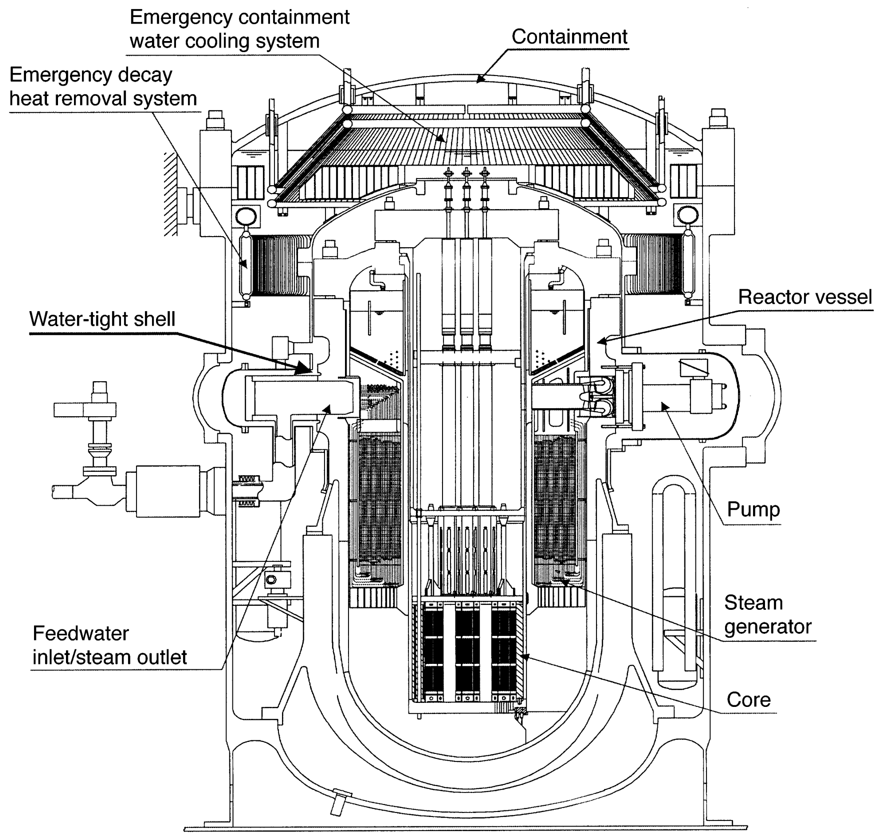
Similar to the Russian KLT-40, the advantage of the MRX design comes from its assembly-line fabrication of the entire plant, and its transportability as a self-propelled ship. Refueling is every 3.5 years, but the capability of performing refueling and maintenance activities at a central facility improves diversion- and proliferation-resistance.

■ The **Simplified Boiling Water Reactor (SBWR)** by General Electric is a concept that incorporates advances in existing, proven boiling water reactor technology at the 600-MWe power level.

As part of the licensing audit of the GE application to the NRC for design certification, Purdue University has been conducting a detailed study of the SBWR safety systems through integral tests, and developing a data bank for various transient scenarios. The **Modular SBWR** is Purdue's variation resulting from its systematic study of the SBWR. The Modular SBWR is being developed under the NERI program both at 200-MWe and 50-MWe levels.

The Modular SBWR design relies on natural circulation to cool the reactor core and produce the steam needed to drive the turbine and generator. As a result, the coolant pumps are eliminated from the Modular SBWR system.

The proposed 50-MWe Modular SBWR is a small, compact reactor concept with modifications on the fuel cycle and fuel type for extended core life and proliferation-resistance. The reactor vessel is 8.5 m in height and 3.5 m in diameter. The active core height is 1.9 m, and has an inherent negative temperature coefficient and a negative void coefficient; that is, if more of the coolant in the core is in the steam or vapor phase, the reactor power will decrease, since the neutrons are not slowed down as effectively as in the liquid-water phase. With uranium oxide fuel having an initial enrichment of 5 percent U-235, similar to commercial BWR fuel, it is estimated that the Modular SBWR would provide full power for an estimated 10 years of continuous operation before requiring refueling.



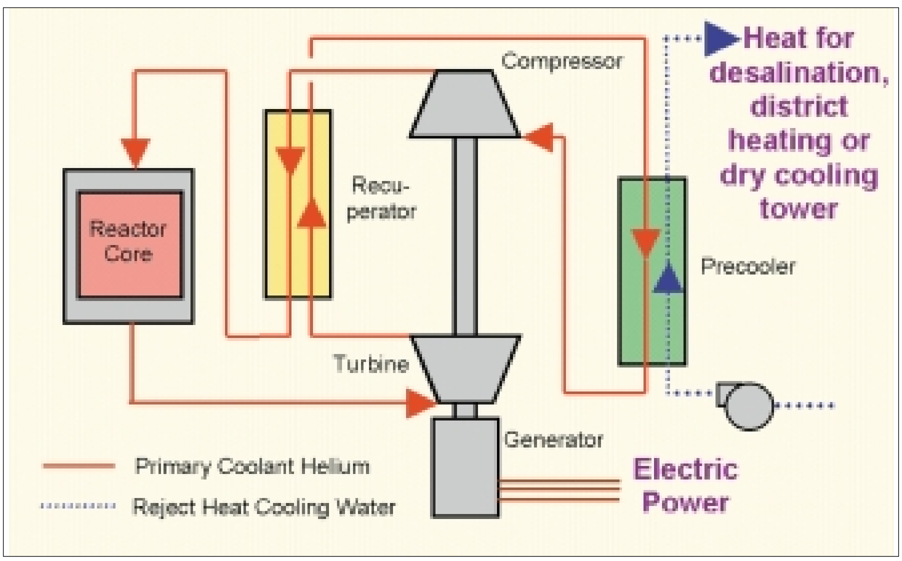
Conceptual drawing of the MRX reactor design (Source: JAERI)

The containment of the Modular SBWR is a cylindrical steel tank with overall dimensions of 14.6 m in height and 12 m in diameter. In addition to the reactor vessel, the containment includes compartments for various safety system components. The containment is isolated and placed in the steel-reinforced concrete cavity that provides an additional barrier against the leakage of contaminated coolant.

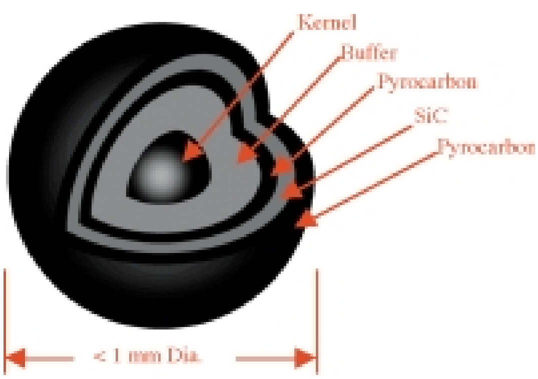
The passive reactor safety systems consist of the gravity-driven cooling system, suppression pool, containment cooling system, isolation condensers, and the automatic depressurization system.

The reduced core power density enables single fuel batch loading with 10-year core fuel life, which adds to proliferation-resistance.

■ The **Remote-Site Modular Helium Reactor (RS-MHR)** is a compressed-helium gas-cooled reactor proposed by General Atomics. It is contained in one vessel, while all of the power production and heat transfer equipment is in a second vessel, connected by a single coaxial pipe that carries the helium



Schematic of the RS-MHR reactor (Source: GA)



Ceramic-coated fuel particle of the RS-MHR, which contains uranium oxide. Layers of tough, high-temperature tolerant pyrolytic carbon and silicon carbide confine the radioactive fission products at their source, in the center of the fuel particle. (Source: GA) coolant between the two vessels.

The entire power plant and the support systems are housed in a building about four stories high at its highest, measuring about 18 by 24 m. The building is constructed entirely aboveground, eliminating the need for excavation. The reactor portion of the building is reinforced concrete—about 1 m thick for shielding purposes—and is about three stories high, measuring about 10 by 12 m. The power rating for the reactor ranges from 10 MWe to 25 MWe.

The RS-MHR uses compressed helium gas for the reactor coolant. The helium gas removes heat from the reactor core and directly drives a commercially available industrial turbocompressor. The helium turbocompressor both generates electricity and compresses the helium before it is sent back to the reactor core. The use of helium gas allows the reactor to be operated at much higher temperatures compared with a water-cooled system, leading to improvements in electricity generating efficiency.

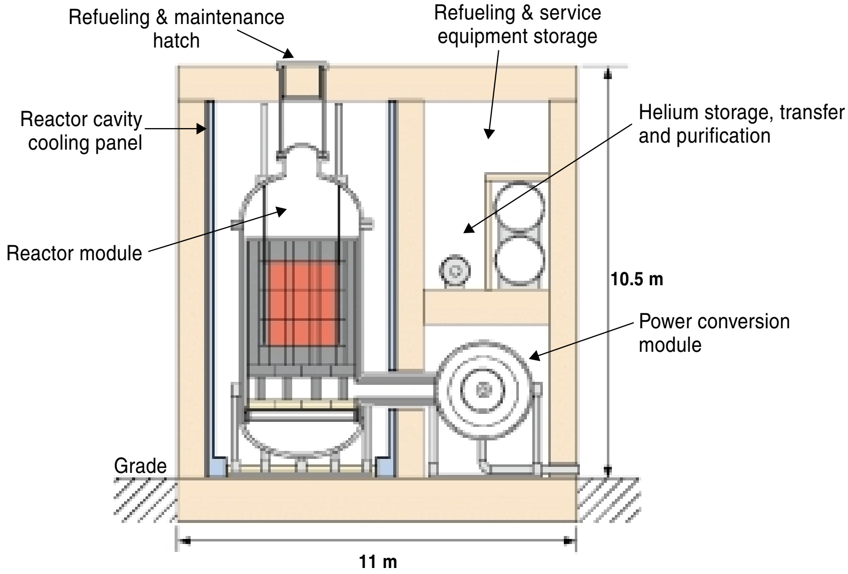
The RS-MHR uses uranium oxide fuel, similar to that used in most existing commercial nuclear power plants, but the fuel is contained in very small spherical particles approximately 1 mm in diameter rather than in the long fuel rods typically used in large power reactors.

The refueling interval is estimated at six to eight years of operation, ensuring that the reactor fuel is taken to a high burnup. Also, the fuel is contained in a graphite and silicon carbide, which provides protection in that it is somewhat difficult to remove or penetrate. The reactor is refueled at the site, with the spent fuel being stored in a small room adjacent to the room containing the reactor vessel. Estimates are that the spent fuel would need to be water-cooled for about six months, after it has been removed, and then it could be dry-stored, but with an active cooling system.

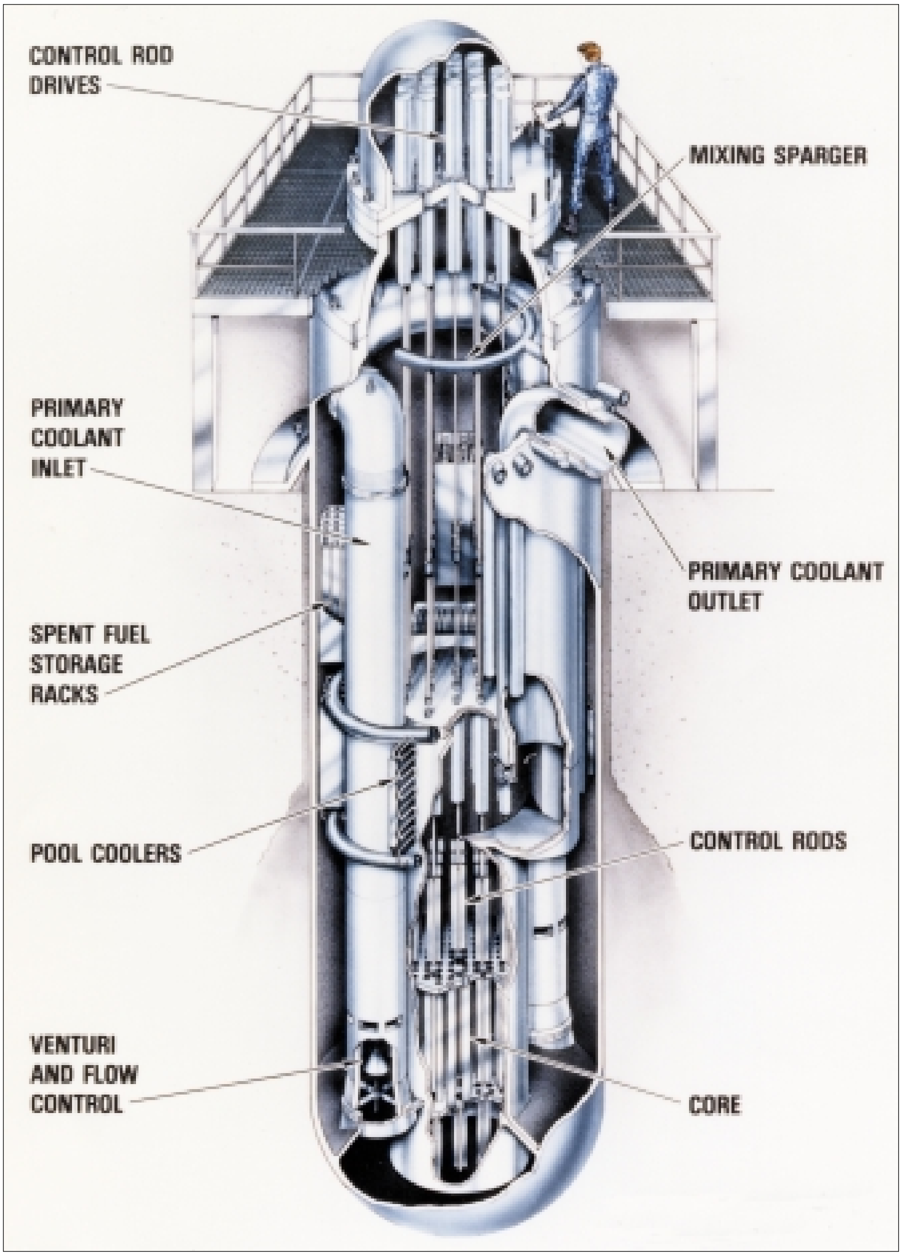
Parts of the RS-MHR concept represent a deviation from practice at existing nuclear power plants. Specifically, the RS-MHR does not have a containment building. The design may be ready for deployment at the end of this decade. The DOE noted that operational experience demonstrates that this is a reliable and proven fuel concept, with no unresolved technical issues.

■ The **TRIGA Power System (TPS)** from General Atomics is a PWR concept based on the TRIGA reactor design coupled with a commercially available organic power system. The TPS is designed for a power level of 64 MWt, 16.4 MWe. The TPS measures 40 by 60 m. There also is excavation required below grade, to a depth of about 10 m. Like standard PWRs, the primary coolant system of the TPS consists of the reactor core, primary circuit piping, pressurizer, coolant pump, and a heat exchanger. The reactor vessel containing the core and the primary heat exchanger where heat is transferred from the primary circuit to the secondary circuit constitutes two large factory-fabricated modules to permit a transportable system.

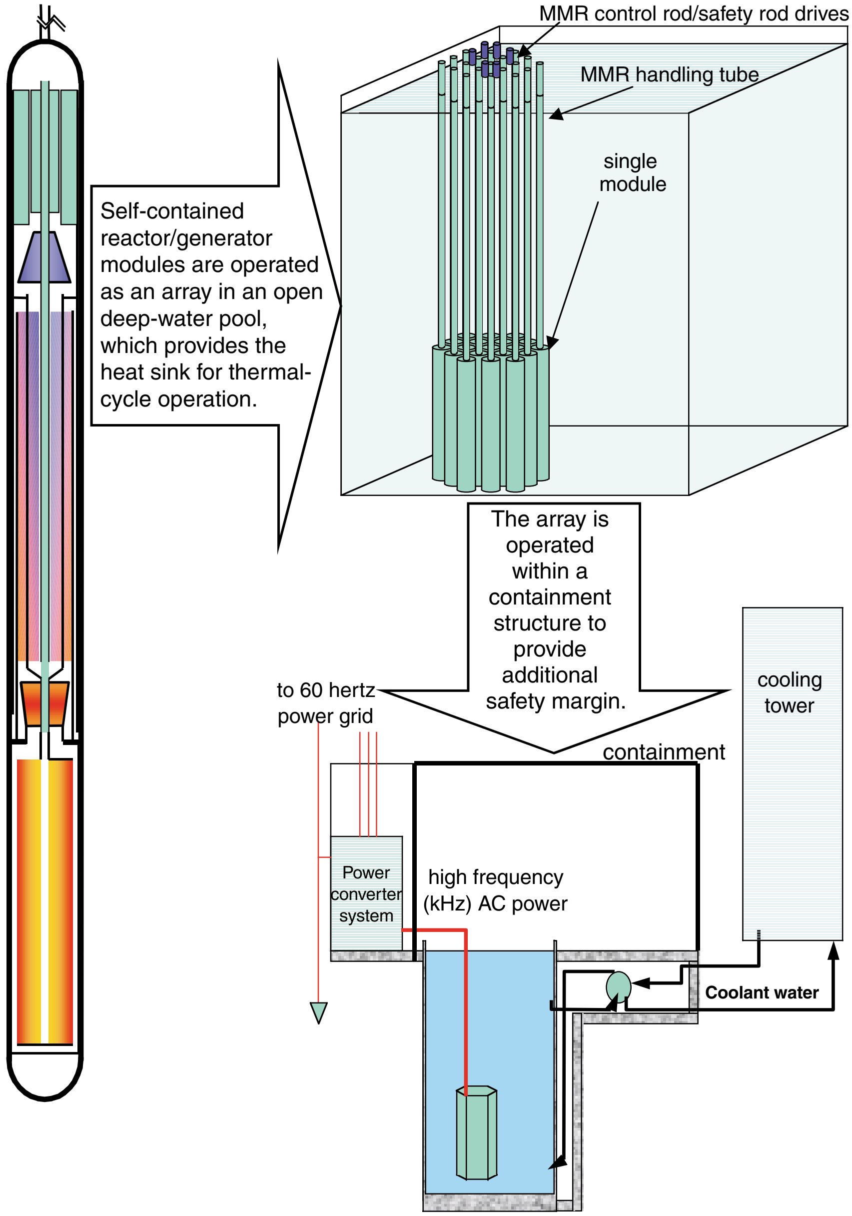
A substantial amount of auxiliary equipment and piping systems are needed to support the TPS reactor. The secondary system is housed in a room adjacent to the reactor room. The reactor in the TPS uses a pool design, in which the reactor core is located in, and phys-



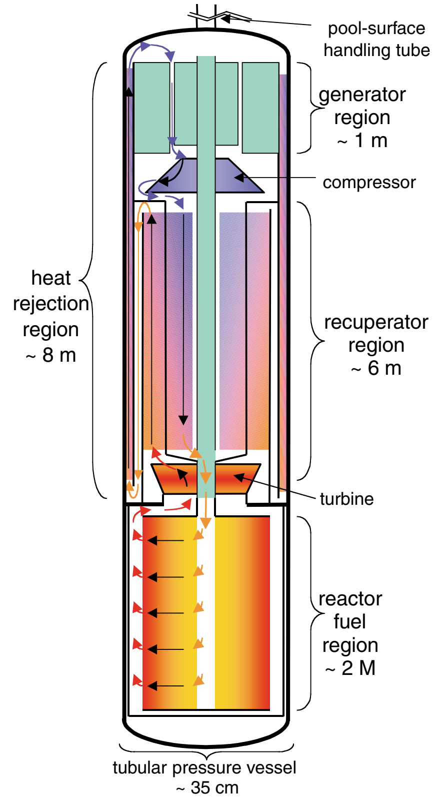
RS-MHR reactor system arrangement (Source: GA)



Triga Power System (TPS) reactor vessel (Source: GA)



The MMR reactor system design concept. As a 37-module system, the plant output would be 100–200 MWe. (Source: SNL)



The MMR reactor module design concept. This module is for 3–6 MWe output (depending upon the fuel outlet/turbine inlet temperature). (Source: SNL)

ically separated from, a larger pool of water. The TPS operates at a lower pressure than a standard large PWR. The water that cools the reactor is used to heat an organic fluid (a commercially available inert perfluorocarbon FC-72) on the secondary side of the plant, which in turn is used to drive a turbine.

In an effort to improve the inherent safety of the TPS reactor, the core has been placed in a pool inside the reactor vessel. During normal operation, only a small amount of the water in the reactor vessel is actually circulated to transfer heat to the heat exchanger. The remainder of the water in the reactor vessel, about 90 percent of the reactor vessel volume, is maintained at a constant temperature by the auxiliary cooling system that operates continuously using natural circulation. The innovative feature of this concept is the connection between the circulating primary coolant and the pool of water in the reactor vessel, termed the venturi pressure balancing system, that does not use any valves to control the coolant flow between the pool and the circulating coolant.

The reactor core uses standard uranium-zirconium hydride fuel, as in the TRIGA reactors. The fuel also contains a small amount of burnable poison. The TPS fuel uses low-enriched uranium, with U-235 at 19.9 percent enrichment, and currently is manufactured for the TRIGA research reactors currently in operation.

The TPS is susceptible to the accidents that are common to water-cooled reactors, according to the DOE. The TPS design, however, mitigates the effects of some of these accidents, by operating at a much lower pressure and temperature. To respond to some accident conditions, redundancy is built into the system.

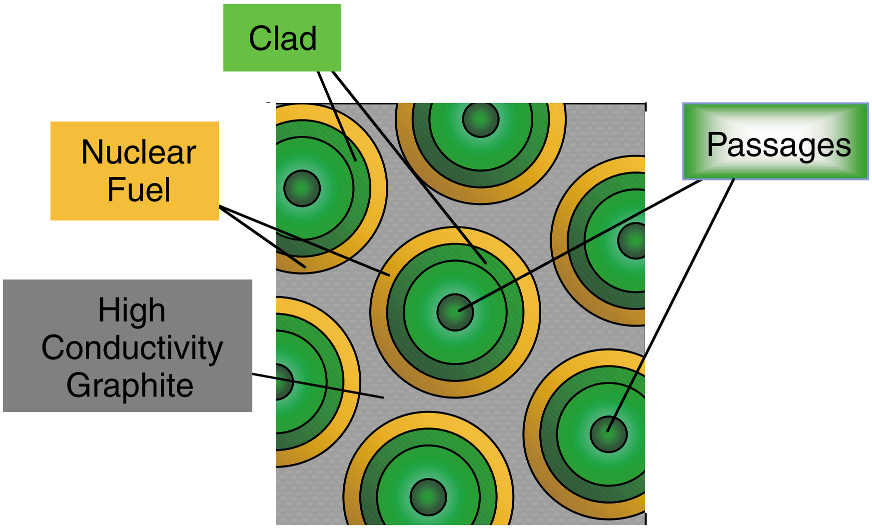
The LEU standard TRIGA fuel is not attractive for use directly in weapons, the DOE stated. Also, the reactor vessel is isolated from any routine operations. For the core design, one-half of the core is refueled on site every 18 months, which is considered a short refueling interval. The spent fuel is stored inside the reactor vessel until it is cold enough to be removed and shipped from the reactor site, making access to these materials difficult.

■ The **4S** is an LMR using sodium as the coolant. The 4S design is based on the principles of simplified operation and maintenance, improved safety and economics, and proliferation resistance. It combines infrequent refueling—about every 10 years—with a short construction period based on factory fabrication. The designer is Central Research Institute of Electric Power Industry, Japan.

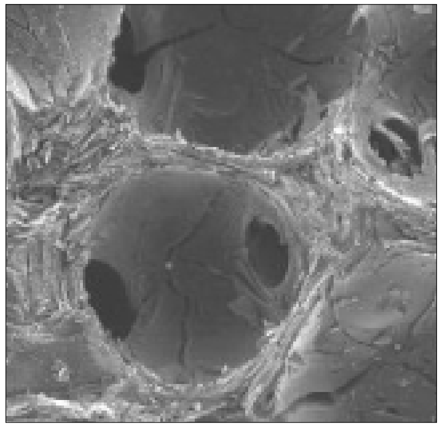
The primary coolant system includes an electromagnetic pump to pressurize the liquid-sodium coolant and an intermediate heat exchanger, both placed inside the reactor vessel and above the core. A containment vessel envelops the reactor vessel and the top dome.

The reactor fuel uses a metallic alloy (either U-Zr or U-Pu-Zr) that was developed in the United States and later in Japan. In the 4S reactor core, the steady-state power level is maintained throughout the core life primarily by slow vertical movement of a graphite reflector surrounding the core, rather than by using neutron-absorbing control rods.

The 4S is a small reactor designed to have totally passive safety systems that do not require power and may not require valve movements to initiate them. The use of a movable reflector to control neutron leakage and reactor power is what the DOE calls perhaps the most unusual feature of this concept. Liquid sodium is a coolant with excellent heat capacity, very high thermal conductivity, low operating pressure, and superb natural con-



SSR's core matrix (Source: ORNL)


Graphite-foam cell structure of the SSR reactor. The reactor core is made of fissile material embedded in advanced graphite-foam material. (Source: ORNL)

vection capability. Decay heat is removed from the core by natural circulation of the primary coolant, and discharged by a coil system placed above the intermediate heat exchanger. If the main pump fails, however, passive cooling also is provided, using natural circulation of air from outside the guard vessel.

The fuel is handled remotely, so that there is never any direct physical contact between the fuel and plant personnel. This physical separation enhances diversion-resistance. The 4S design is in the early stages of development and may not be ready for deployment in this decade.

Other designs

Two other small reactor concepts being developed under DOE programs are the Multi-Module Reactor (MMR) design from Sandia National Laboratories and the Solid-State Reactor (SSR) design from Oak Ridge National Laboratory. Both designs are at a preconceptual design stage:

■ The **MMR** concept consists of an array of self-contained, factory-built, transportable gas-cooled modules in a pool configuration. The modules consist of a reactor core and an integral direct cycle turbine-compressor-generator system, all contained in a single tubular pressure vessel. Each module is expected to have 1- to 5-MWt capacities.

■ The **SSR** is a concept to achieve demand-driven heat generation without the need for moving parts or working fluids. It is a self-regulating 3-MWt nuclear heat source for small power units based on advanced graphite-foam material. This foam material, currently being studied at ORNL, has enhanced heat transfer characteristics and good high-temperature mechanical properties. The SSR concept is being developed under the NERI program. NN