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World status of the SMR projects

L. Paparusso, M.E. Ricotti, M. Sumini

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WORLD STATUS OF THE SMR PROJECTS

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***UNIVERSITA' DI BOLOGNA**

DIENCA - Dipartimento di Ingegneria Energetica, Nucleare e del controllo ambientale

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INDEX

EXECUTIVE SUMMARY

The document presents the current status of development and, in some cases, deployment of Small-Medium size, Modular Reactors worldwide.

In the recent years, an increasing interest has grown around the concept of "smaller size" reactors. Albeit this concept is already known, i.e. nuclear reactors of few hundreds MWe power were built in the 60's and the 70's and also in Far East (e.g. India) in recent past times, the paradigm of the economy of scale hence of "larger size" reactors has largely overwhelmed this original approach since long time. That led, for example, to conceive-develop-deploy 1400-1600 MWe NPPs (e.g. APR-Korea, EPR-France).

Some issues call into question the assumption of the general and absolute validity of the paradigm: the availability of funding to sustain the construction effort of such a big endeavor, the associated risk of costs increase, e.g. in case of construction-operation delays, the suitability of such large size plants for poorly interconnected electrical grids, the availability of manufacturers-suppliers of large components or critical elements (e.g. tube bundles).

Moreover, the concept of modular construction already adopted in some Generation III reactors (e.g. AP1000-USA) and very likely leading to lower costs and better quality and safety (e.g. due to in shop manufacturing), is supposed to be better implemented in SMRs than in Large Reactors.

These are the assumptions or the believed factors underpinning the SMR approach, together with a claimed superior safety and, in some cases, a better proliferation resistance.

Several countries (including France in the very recent months) begun studying, developing and in few cases deploying, their domestic SMR project. IAEA as well is active in this field, with several coordinated research programmes under way.

The list and essential description of the different SMRs, herein reported, show the increasing interest towards this type of reactors, covering several technologies (from PWR to FBR and MSR). The feasibility, affordability and success of such an approach are yet to be proven. The financial and safety constraints, recently strengthened by the global economic crisis and the Fukushima event, could give to the SMR industrial strategy a real opportunity.

1 INTRODUCTION

Light water reactor systems for propulsion were the forerunner of commercial nuclear power systems we see nowadays. Light-water reactors (LWR) were chosen because of their simplicity and compactness at this small scale. U.S. Air Force and Army also started a nuclear power program. From 1946, the Air Force studied the use of small nuclear reactors to power long-range bombers, but this application proved too difficult and politically unattractive and was terminated in. The Army Nuclear Power Program ran between 1954 and 1976 and led to the construction of eight reactors. These included six 1–2MWe pressurized water reactors (PWR), one 10MWe barge-mounted PWR reactor and one 0.5MWe gas-cooled reactor (GCR).

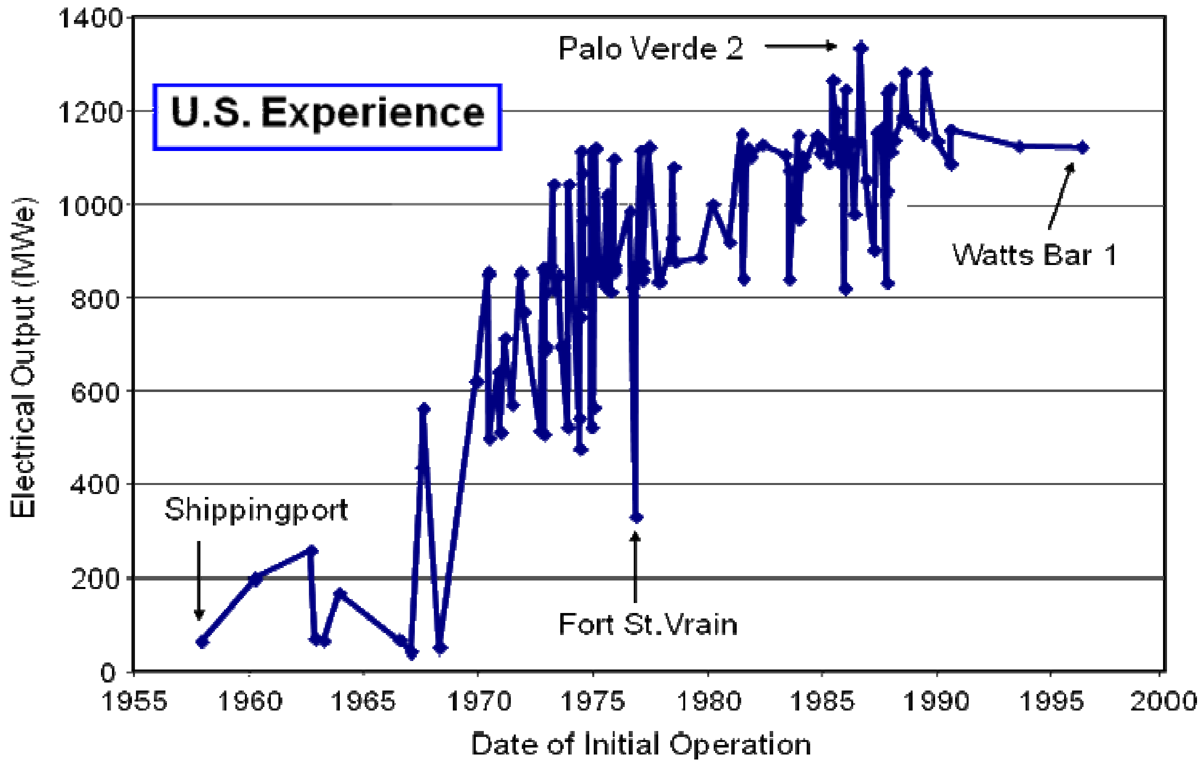


Fig.1 - Progression of power level for the commercial nuclear power plants built in the United States (Energy Information Administration, 2008).

The Army program was discontinued because of the poor economics of the nuclear plants compared to cheaper alternative fuels available at that time. The early commercial reactors commissioned in the late 1950s and early 1960s were essentially scaled-up versions of the naval power plants. The 60MWe Shippingport plant began operation in 1957, the 200MWe Dresden plant in 1960, and the 250MWe Indian Point Unit 1 plant in 1962. Due to the rapidly growing demand for electricity, the high level of confidence in the safety of nuclear plants, and the economic principle of ''economy of scale,'' reactor size began to grew up till 1300MWe. Much of this growth occurred over a 15-year period without the benefit of operating experience from smaller predecessors of these new large-size reactors. Fig. 1 shows the progression of power level for the commercial nuclear power plants built in the United States (Energy Information Administration, 2008). Analyzing this progression one can notice that power plants commissioned before 1973 were SMRs by IAEA's definition.

The apparent anomaly in the growth trend in 1976 was the startup of the demonstration gas-cooled reactor, Fort St. Vrain. No subsequent gas-cooled reactors have been built in the U.S. As plant sizes grew and as operational issues began to moderate the industry’s confidence in the ultimate safety of the plants, more stringent safety requirements were imposed. This fact led to a growing complexity in the plant designs, adding redundant safety and auxiliary systems. This escalation of plant complexity contributed to rapidly increasing costs, construction and operational delays, licensing delays, and eventually decreased confidence by the owners and lenders in the profitability of the plants. Almost every reactor was built to accommodate the interests of individual customers, making every reactor a “one of a kind” construction process. Obviously this contributed to increased licensing, construction, and operational complexities. These and many other factors contributed to the eventual demise of the first nuclear era, which was punctuated by the accident at the Three Mile Island (TMI) plant in 1979.

Interest of many countries to the development and application of SMRs reactors continued as continued operation, construction of new small power plants, and progress in design and technology development. SMRs designers explored innovative design approaches to reach a higher level of plant safety, economics, and proliferation resistance. These facts ensure that such reactors could competitively meet the needs of potential users in those markets that cannot be effectively served by the economy of scale nuclear deployments. The potential SMRs users are diverse, spacing from small towns and industrial sites in off-grid locations to growing cities in developing countries.

There’s also the possibility to use such reactors for non-electrical applications in deregulated markets. The requirements of these user groups are also diverse, ranging from small capital outlay and incremental capacity increase to autonomous operation, advanced cogeneration options and long refueling interval. To facilitate SMRs development, the IAEA is carrying out new activities for SMRs that include:

-Design and deployment strategies to overcome loss of economies of scale, for example, advantages in reduced design complexity, modularity and accelerated learning;

-Definition of investor requirements for innovative SMRs and consolidation of methodologies to help public and private investors in developing countries assess the overall potential of innovative SMRs;

-Continuous re-examination and quantification of needs for SMRs based on a country-independent model which will be developed to support such quantification;

-Dynamic simulations of energy systems with innovative SMRs.

The potential for small and medium size reactors, SMRs is under study in the USA, Japan, Russia and other countries. France's naval construction firm DCNS together with Areva, Electricité de

France, EDF and the Commissariat a L'Energie Atomique, CEA research organization decided to set up a joint study of DCNS' submerged reactor. it could provide wide energy for coastal locations all over the world.

The concept is called Flexblue and involves a cylindrical vessel about 100 meters long and 15 meters in diameter that encase a complete power plant producing from 50 to 250MWe.

The cylinder with the power plant inside would be lowered to the seabed at a depth of 60 or 100 meters, at a site between 5 and 15 kilometers from the shore.

Undersea cables would bring the electricity to customers at the coast much like offshore wind turbines. It is estimated that ¾ of the world's population lives within 80 km of 'the sea.

A submerged power plant unlike a floating one would not be vulnerable to earthquakes, tsunamis, or floods, and would be less vulnerable to voluntary attack.

It would also have an "unlimited" source of coolant due to the sea all around and the plant's footprint would be minimal.

The cylindrical vessel obviates the need for civil engineering ,which has proved challenging at Areva's and EDF’s under construction nuclear plants in Finland and France and it means the plant can be built in a factory in a modular way with standardized components. For safety issues it can be brought till the surface and taken to a DCNS’ shipyard for repair.

It could be refueled in the same way, and at end of life be repatriated to the shipyard for decommissioning, which would resemble the decommissioning of nuclear submarines, Areva has already begun developing, a Small Modular Reactor, or SMR, of about 100MW based on the experience of its Technicatome in building reactor plants for submarines and France's nuclear-power aircraft carrier: the Charles De Gaulle. Such a reactor could be embarked in a Flexblue power plant.

The market for SMRs is estimated at about 200 units worldwide over the next 20 years, Flexblue could grab a significant share of that market.

DCNS’ shareholders are the French stare at 74 percent. defense firm Thales at 25 percent and employees at 1 percent.

AREVA, a world leader in nuclear energy has launched a program to study small reactors rated at 100 MWe with a view to rounding out its range of third generation reactors comprising EPR, ATMEA and Kerena types. This study draws-on AREVA's expertise in small shipboard reactors to assess the product's feasibility and market potential.

U.S. government nowadays gives to nuclear energy an important role. Nuclear power’s objective is to assist in the revitalization of the U.S. industry through R&D. Developing these technologies through R&D could help accelerating the deployment of new plants in the short term, supporting development of advanced concepts for the medium term, and promoting design of revolutionary systems for the long term. This target will be achieved in partnership with industry to the maximum extent possible. Elements of nuclear energy’s strategy in this area include:

- Assist industry to improve light water reactors using existing technologies and designs.
- Explore advanced LWR designs with improved performance.
- Research and develop small modular reactors that have the potential to achieve power’s objective is to assist in the revitalization of the U.S. industry through R&D.

Smaller reactors have the possibility to be built in modules. This might help reduce the capital costs associated with large plants. It’s always possible to incrementally “step up” to larger electrical capacities while generating revenue and repaying initial debts.

To help SMRs’ development President Obama has earmarked $500 million over the next five years for SMR research and demonstration projects. Moreover Energy Secretary Steven Chu predicts that an SMR will be producing electricity by the end of this decade.

Congressmen also included in their speech to the Senate issues about SMRs (e.g. Sen Mark Udall**,** Colorado; Jeff Bingaman, New Mexico; Congressman Jason Altmire, Pennsylvania).

IAEA published a document (Design Features to Achieve Defence in Depth in Small and Medium Sized Reactors, Vienna 2009) showing SMRs future and describing the most important project.

Looking at Fig.2 it is possible to understand which prospective exist for small-size reactors, having an idea of the time scale connected to the major projects.

Not all of the project showed in figure are going to be licensed. Brown color indicates project in a more advanced stage of development.

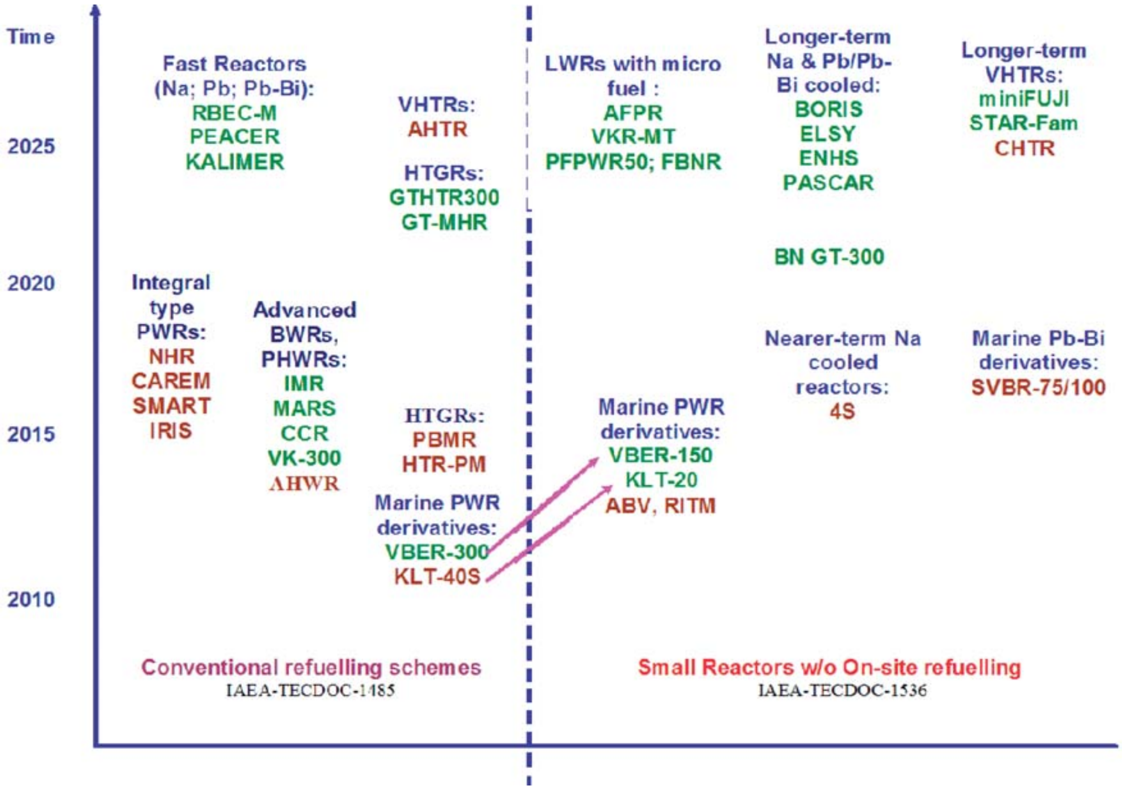


Fig.2 – Time schedule for the development and possible deployment of innovative SMRs, with and without on-site refuelling.
(IAEA, 2010)

SMR PROJECTS - CLASSIFICATION BY TYPE

1.1 LWR (PWR)

Pressurized water reactors (PWRs) constitute a majority SMRs. In a PWR the primary coolant, high pressure light water, flows in the reactor core where it is heated and, passing through a steam generator, it transfers its thermal energy to a secondary system where steam is generated. This steam flows to turbines which provides electrical energy. Small PWRs were originally designed to serve as nuclear propulsion for nuclear submarines.

In this category we can find: Nuscale, mPower, IRIS, SMART, KLT-40

1.2 LMFBR

LMFR reactor uses liquid metal as primary coolant. Liquid metal cooled reactors were first adapted for nuclear submarine use but have also been extensively studied for power generation applications. They don't need to be kept under pressure, and they allow a much higher power density than traditional coolants. Difficulties of inspection and repair of a reactor immersed in opaque molten metal, corrosion, production of radioactive activation products are the most discussed issue.

The most significant LMFBR SMR project is Toshiba's 4S.

1.3 "Exotic" or "Unconventional" Projects

Under the name of "Exotic" projects it is possible to find out innovative and non conventional reactor projects as PbBi Hyperion (US) or CANDLE (JP) or the new French project of an underwater Integral PWR named Flexblue. These projects usually have different fuel cycle process (e.g. sealed core).

1.4 MSR

In the MSR, the fuel is a molten mixture of lithium and beryllium fluoride salts with dissolved enriched uranium, thorium or U-233 fluorides. Heat is transferred to a secondary salt circuit and thence to steam. It is not a fast neutron reactor, but with some moderation by the graphite is epithermal. The fission products dissolve in the salt and are removed continuously in an on-line reprocessing loop and replaced with Th-232 or U-238. MSRs have a negative temperature coefficient of reactivity, so will shut down as temperature increases beyond design limits.

2 SMR - CLASSIFICATION BY PROJECT

2.1 Nuscale (PWR-Integral)

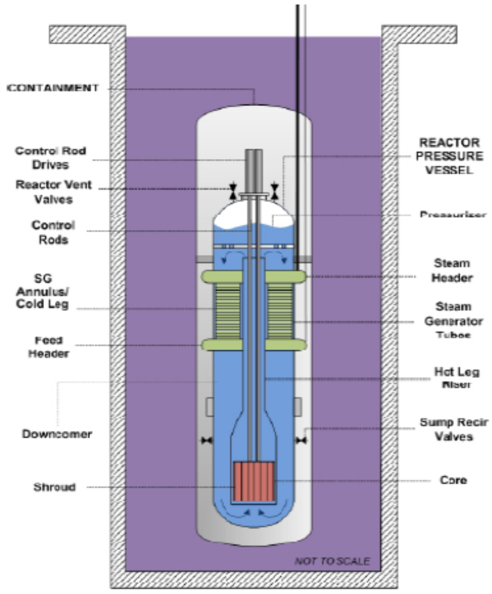


Figure 3.1.1 Nuscale module layout

	NuScale
Power	160MWt 45MWe
Dimension	2.7m Diam x13.7m Height
Vessel Thickness	7.6cm
Primary Pressure	<12.7MPa
Primary Flow	600kg/s
Layout	12x in pool
Weight	300 tons
Transportation	Barge, truck or train
Fuel	Standard LWR fuel in 17 x 17 configuration
Enrichment	4.95%
SG Length	22.3m
Secondary Flow	70kg/s
Feedwater Temp.	150°C

	NuScale
Secondary Pressure	3.1MPa
Refueling	24 month

A single NuScale module produces 45,000 kilowatts of electricity. Heat is transferred from primary circuit, the core, to the secondary one by steam generators, integrated in the vessel itself. Produced steam is sent to a steam turbine connected by a single shaft to the electrical generator. NuScale power plant will operate at full power for about 95% of the time. This makes it a really reliable generation system.

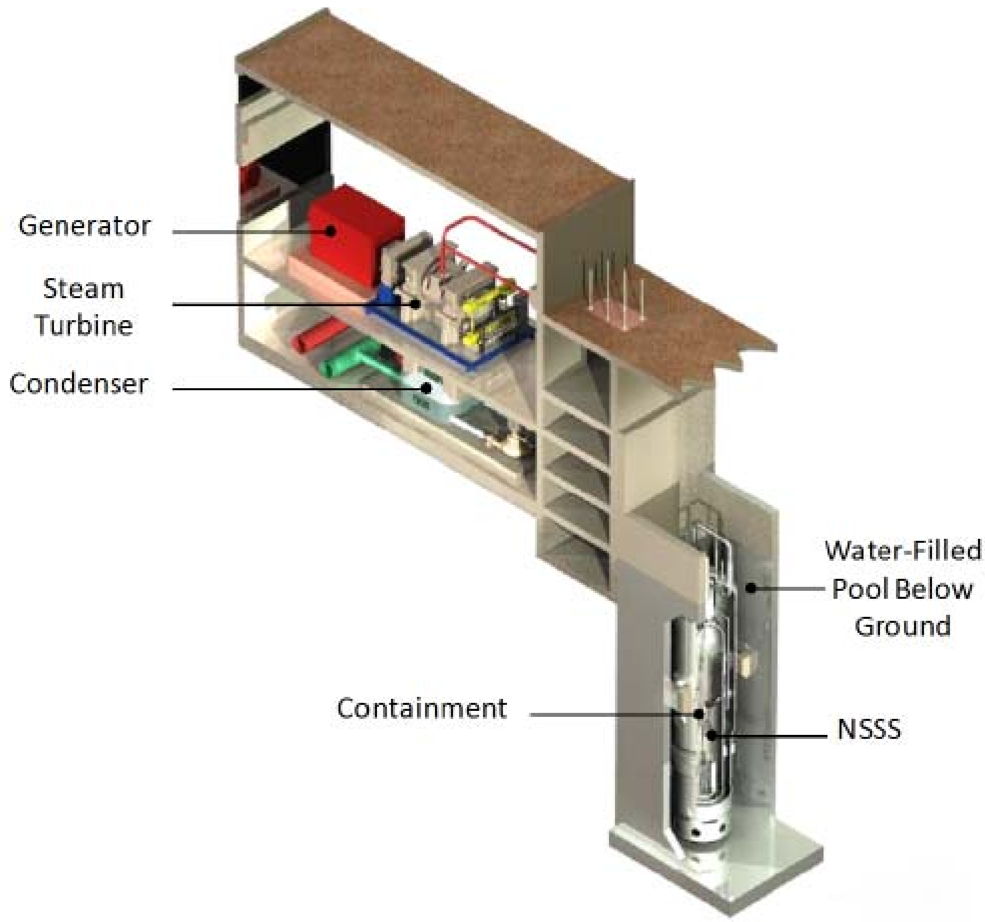


Fig 3.1.2 NSS and BOP of a Nuscale module

Because of its modular design it is possible to join each NuScale, self contained anyway and independent from the others, in a multi-module configuration. However, all are managed from a single control room.

A plan view of a layout made of a 12 module array with a total capacity of 540 MW(e) is shown in Figure 3.1.4. The design layout shows a building which houses the pool containing the modules, a turbine building, and a separate refueling building which contains an area used as the spent fuel storage pool.

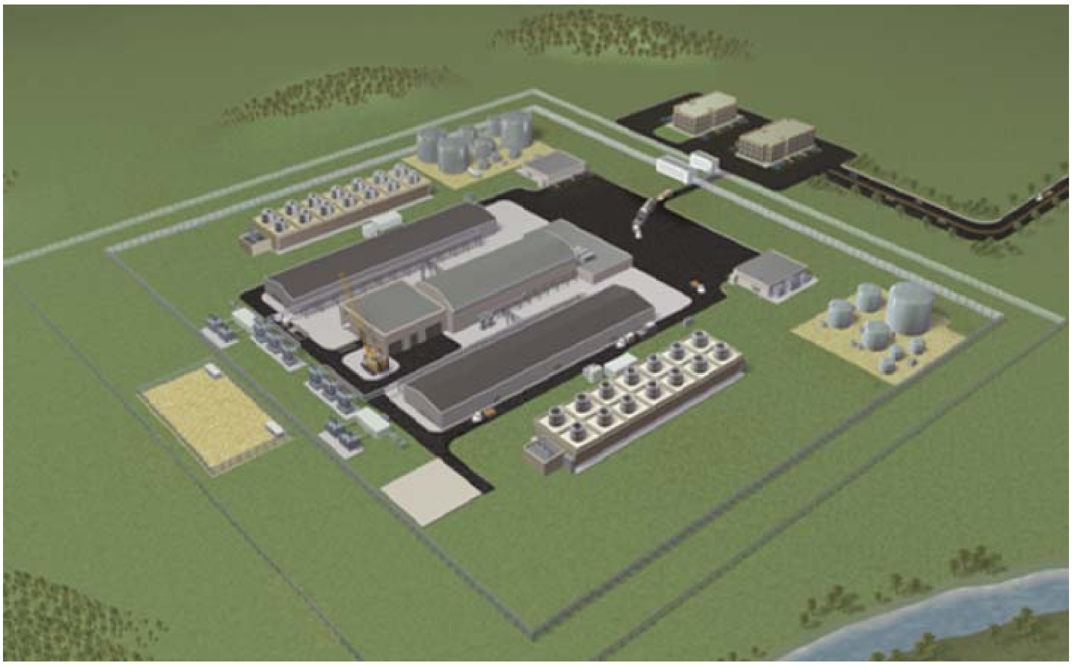


Figure 3.1.3 Nuscale 12 unit layout

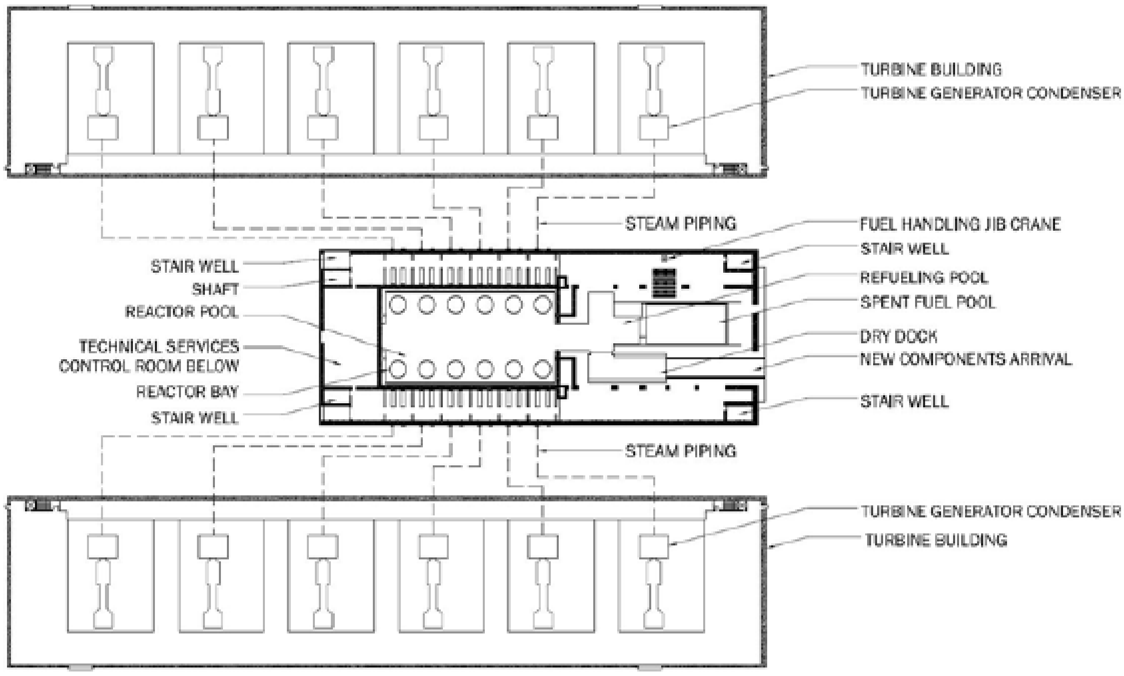


Figure 3.1.4 Schematic view of 540MW layout

There are multiple barrier between fuel and environment, starting from cladding, to the reactor pressure vessel measures that sits in a containment vessel. This entire module operates inside a pool built below grade. No pumps are needed to move water inside the reactor, because of natural circulation. This enhances safety and cut off the possibility of pump failures.

Secondary circuit is a standard 45MWe cycle, so, after steam passes through the turbines, it is cooled in a condenser and returns to the steam generator inside the reactor. There is the possibility to use steam, after it passes through the turbines, for low temperature, low pressure applications requiring heated water.

It is possible to use NuScale system only to produce steam, using its 160MW thermal, for industrial applications, such as district heating for communities, large facilities and installations, or to synthesize fuels.

Enhancing safety means, in a NuScale module, working with passive safety systems using natural circulation for emergency feedwater cooling, decay heat removal, and containment cooling. In this way, primary pipes and pumps are avoided as well as failures associated with pipe breaks and pump failures. This systems also operate without external power and there’s no need for emergency power on site or off site. In case of a simultaneous rupture of any or all of the reactor piping internal the containment vessel is capable of resisting to deriving pressure transient.

For what concerns earthquakes, the pool grants particular resistance to seismic. The possibility of a big radioactive material releasing is very low compared to the large-scale reactors: each 45 MWe NuScale power module uses about 4% of the fuel inventory of a big-size nuclear reactor.
the reduced amount of piping, low pressure and simpler design are a contribution to safety enhancing.

Security must also be stressed in a NuScale plant. The most important features are:

- Lower reactor building profile.
- The reactor and containment vessel are located in a water-filled pool underground creating a low profile and protected target.
- NuScale high pressure containment vessel is capable of seven times the internal pressure of conventional containments.
- Submerging the reactor further reduces post-impact jet fuel fire concerns.
- No external power is needed to cool the core, which limits plant vulnerability and loss of off-site power is not an issue.

The NuScale containment vessel has several characteristics distinguishing it from other existing containment systems designs.

During standard power operation, an insulating vacuum is maintained in the containment providing a big reduction of heat loss from the reactor vessel. Thanks to this solution, the reactor vessel does not need surface insulation.

Furthermore, when safety valves vent steam into containment atmosphere, the deep vacuum improves steam condensation rates. Further, in case of a severe accident, eliminating containment air grants a security margin against the creation of a combustible hydrogen mixture (no need for hydrogen recombiners), and eliminates corrosion and humidity problems inside containment.

Plus, thanks to the reduced dimensions, it can sustain a pressure greater than 3.4 MPa (500 psia). In this way, the final pressure in the event of a small LOCA will always be below the containment design pressure.

Every NuScale module has its own set of passive safety systems and it is immersed in a pool that can absorb decay heat after a shutdown for 72 hours granting a bulk fluid temperature of 93°C.

The pool in built entirely below grade: it’s made of concrete with a stainless steel liner.

Decay heat must reach the pool, so each NuScale is designed with two redundant passive systems providing a path to the containment pool: the Decay Heat Removal System (DHRS) and the Containment Heat Removal System (CHRS).

To transfer heat generated to the containment pool, the DHRS uses the two steam generator tube bundles. Before natural circulation starts the feedwater accumulators provide initial water flow.

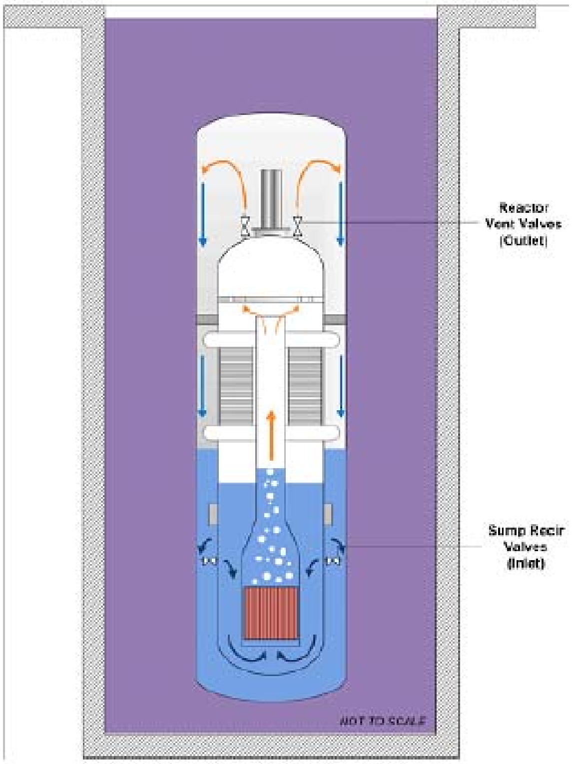


Figure 3.1.4. Schematic CHRS scheme

The CHRS, shown in Figure 3.1.4, acts in case the steam generator tube are not available. It works opening the vent valves on the reactor head. Steam of the primary system is vented into the containment and it condenses on the containment surfaces. Recirculation valves are then opened when the liquid level rises above the top of the recirculation valves, to start natural circulation from the sump through the core and out of the reactor vent valves.

The effect of these systems combined together eliminate Large Break Loss of Cooling Accident (LOCA) by design. Even in case of design basis small break LOCA, there is no scenario in which the core is exposed, as it will be under water all the time. Thus cooling pathways are always available to remove decay heat.

2.2 mPower (PWR-Integral)

Figure 3.2.1 mPower reactor core

	mPower
Power	425MWt 125MWe
Dimension	4.5mx22m reactor vessel
Reactor Containment	28m Diam x 46m H; 1.5m thickness; concrete
Foundation	47m
Reactor Building	85mx73mx15m
Weight	500 tons
Transportation	Barge, truck or train
Fuel	Standard LWR fuel, 17 x 17
N° Fuel Assemblies	69

	mPower
Core Flow Velocity	2.5m/s
Enrichment	4.95%
Plant Footprint	170000 m^2
Refueling	4.5 years

Each 125MWe reactor is produced in a factory, cost about half a billion dollars, and could be built and installed, in multiples of two or four reactors, in only three years. mPower initial site designs show that the reactor should be installed in group of two or four modules, for a total of 250-500 MWe of generation capacity with a footprint of 170000 square meters for the twin configuration.

mPower is designed with an integral layout, that means the vessel holds all the components of the nuclear steam supply system. Fuel rods are on the bottom of the reactor, to make refueling easier. The reactor provides 425MWt, or 125MWe and it is designed to be air-cooled, for a cycle efficiency of 31%. In case of a water-based heat sink, cycle efficiency increases and power generation reaches 136MWe.

A difference between mPower and conventional PWRs occurs in SG configuration: in conventional PWRs primary coolant flows inside the tubes and secondary coolant flows all around them. In mPower, the primary coolant flows outside while secondary coolant is in the tubes. This is necessary thinking at its layout, and comes from experiences in naval propulsion.

Between SG and the reactor, are the control rod system; there's one control rod per fuel assembly and there's no soluble boron to control reactivity, to make the whole system simpler.

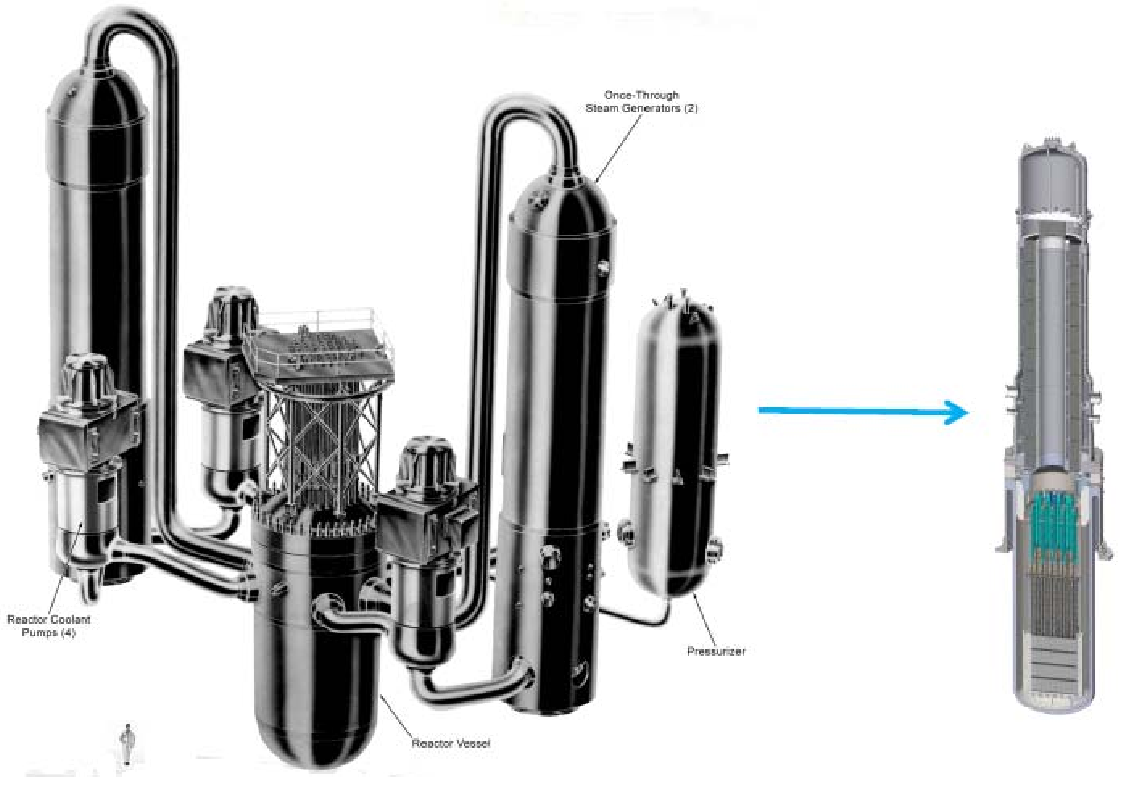


Figure 3.2.2 mPower integration concept

The integrated layout makes the safety case simpler as there are no primary loop penetrations, except for a 2in-diameter clean-up valve at the top of the reactor. In this way we find no large piping going put of the primary, so LOCA possibilities are lowered by design. It must be stressed that due to the height of the unit, a design-basis accident would not drain the reactor core. Gravity fed systems are proposed to remove decay heat from the reactor.

The fuel has a single five-year burn, instead of the standard three-burns as it happens in PWRs; at the end of fuel life the entire core is replaced in one load. Refueling would be expected to last about a week. A nearby spent fuel pool can store 12 cores, enough for a 60-year lifetime. During refueling it would also be possible to substitute the steam generator and inspect it while a new steam generator is put in operation, without lose time and money. This could be done alternatively every 5 years.

The project requires the core and reactor containment to be built entirely underground, to enhance security.

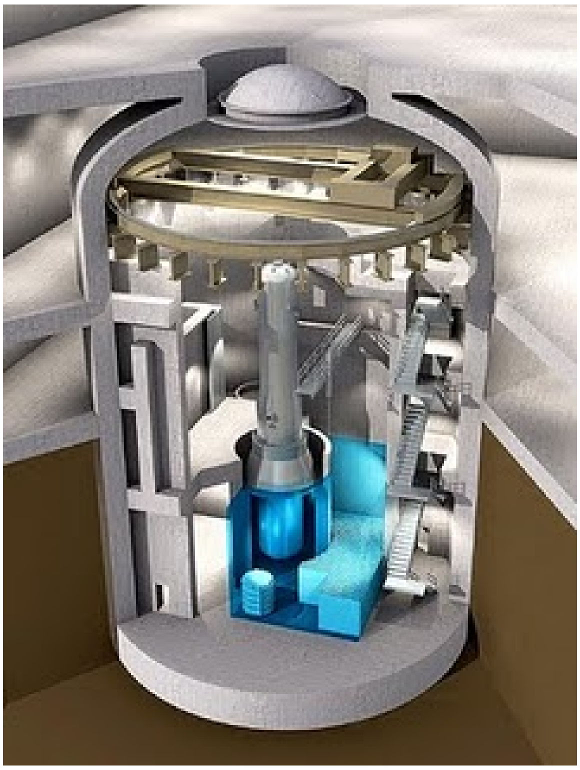


Figure 3.2.3 mPower containment

Transportation is a key point: mPower vessel size is the largest unit that can go by rail from the factory, plus the reactor is small enough to be forged in North America, instead of Europe or Japan . almost the whole unit would be assembled in factories, rather than in situ, granting higher standards and low costs; the construction process results more similar to a combined-cycle gas turbine.

Designers plans are to invert the standard nuclear construction process.. the new approach is to build the power plant first and then bring the reactor on site and connect it to the plant. So it’s possible to build modules in parallel with field activity to shorten construction times. On the contrary in a large plant it’s necessary to build the reactor first and then the rest of the power plant.

Different mPower reactors can be joined in a single power station to provide multiples of 125MWe power. Single modules can be twinned to drive a single turbogenerator this process gives the possibility to fit electricity layout on customer needs: 500-750MWe, 125-250MWe etc. It's possible to go up till 1000MWe or above. The capacity can be added in steps, thanks to the modularity of the base project, rather than all at once, allowing stepwise capital investment

After licensing process, according to estimated time schedule, first startup could be around 2018-2019.

2.3 IRIS (PWR-Integral)

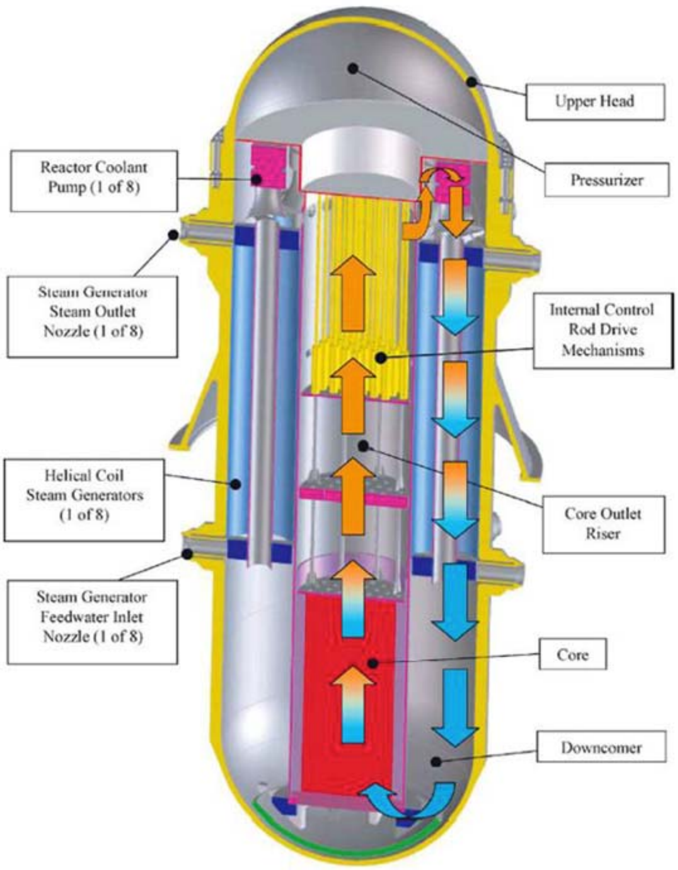


Figure 3.3.1 IRIS core and primary flow path

	IRIS
Power	1000MWt 335MWe
Reactor Vessel	6.2m x 22.2m H; 25cm thickness
Outlet Condition	330°C
Coolant	Light water
Weight	1070 ton
Reactor Containment	25m D; 4.4cm thickness; steel
Reactor Building	50mD x 39mH
Steam Generator	1149m^2*8 units; 8,5m height
Steam Pump	1600 kg, 1800 rpm, 4500 kg/s
Steam Flow	502.8kg/s
Steam Temperature	223-317°C

IRIS	
Steam Pressure	5.8MPa
Condenser Pressure	0.005MPa
Transportation	Barge or special truck
Fuel Design	17x17 assemblies
Enrichment	4.95%
Plant Footprint	358080 m^2
Refueling	3-3.5 years

IRIS is a light water reactor with a modular integral primary system configuration with a net electrical output of about 300 MWe/module. Its design is characterized by four milestones: enhanced safety, improved economics, proliferation resistance and waste minimization.

Integral design means that steam generators, pumps, and pressurizer are located inside the reactor vessel. In this way is it possible to reach enhanced safety standard, due to the elimination of external loop piping, the source of accidents involving a large loss of coolant. Thanks to this configuration it is allowed the use of a small, high design pressure, spherical steel containment resulting in a great reduction in the size of the nuclear island. Safety-by-design approach aim to eliminate some accident initiators starting from the very beginning, the design stage, or when elimination is not possible, lowers accident consequences and probability. This enhances defense in depth and lowers core damage frequency for example; it also allows IRIS to claim no need for an emergency response zone. There are also active and passive safety systems.

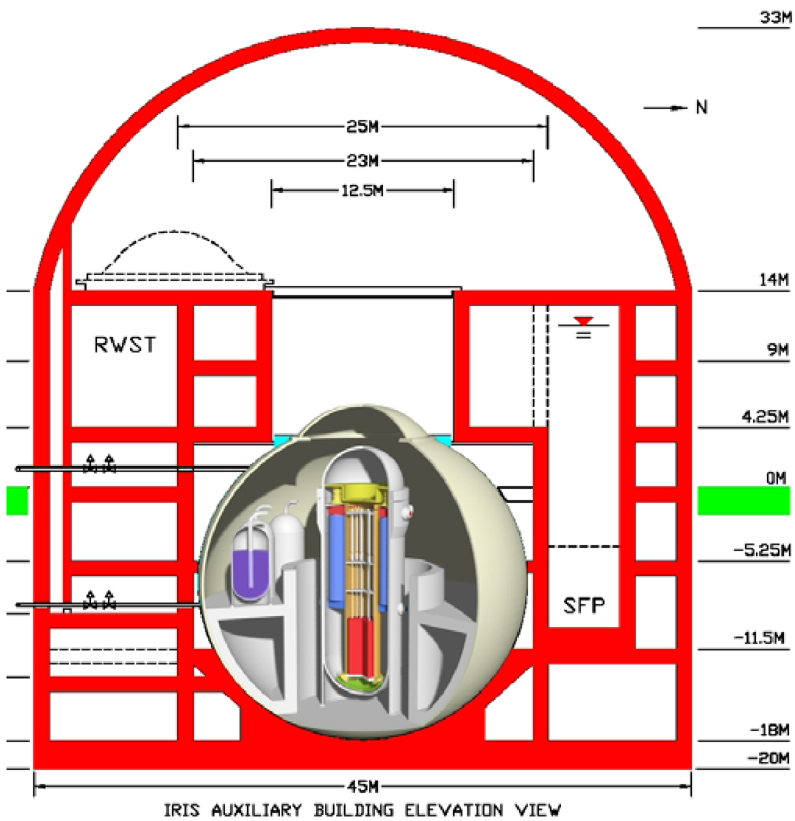


Figure 3.3.2 Schematic view of IRIS reactor building

Among active systems it's possible to find: stand-by diesel generators, startup feedwater system to fill the SG to remove heat from the core, boron injection systems. Passive systems are simpler and more economic: pressure suppression system, emergency heat removal system (natural circulation+ heat exchanger), automatic depressurization system.

This new safety approach eliminates accidents scenarios like: LOCA, control rod ejection, feed line break, steam line break, SG tube rupture.

The entire reactor is the pressurizer; pressure is maintained using sprayer and the core heat. Each reactor has eight once-through helical steam generators, placed inside the reactor vessel near the walls.

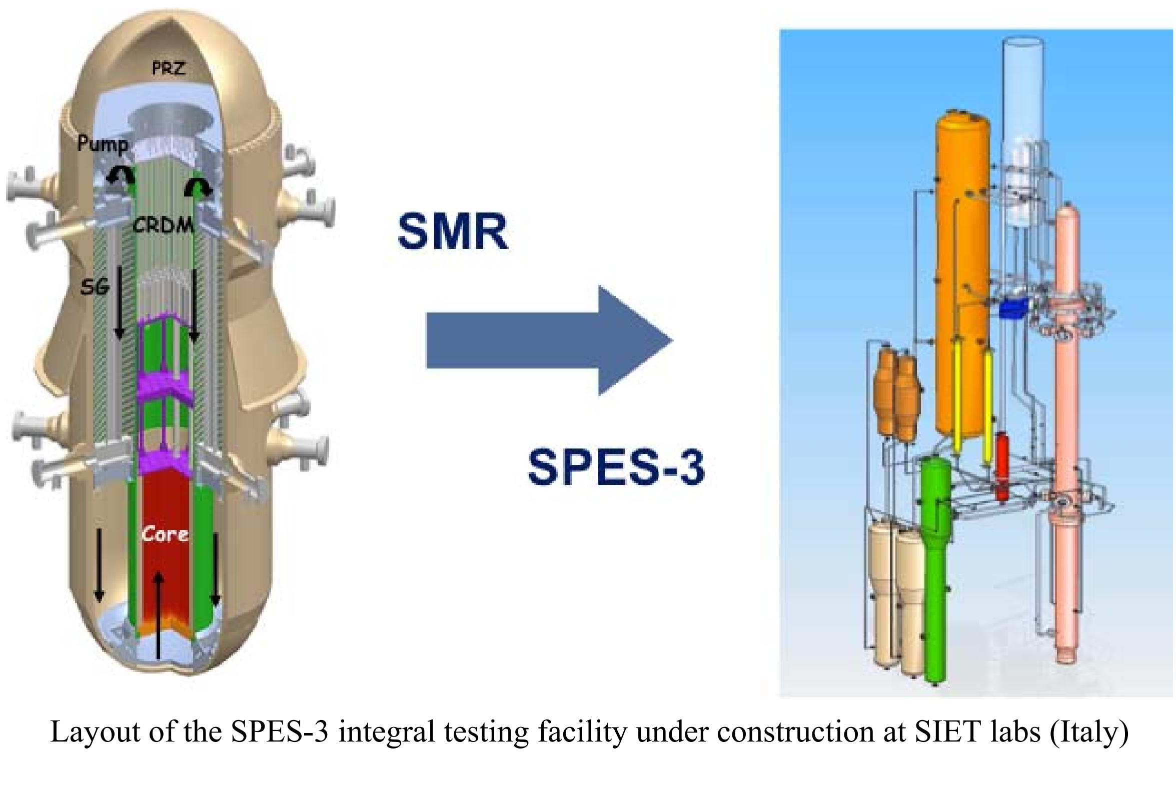
Reactivity coefficients remain negative throughout all reactor life. Burnable poisons are added to the fuel to flatten neutron flux. Reactivity is controlled both with boron and control rods

Shut down maintenance is scheduled every four years because of its simplified design, with less pumps, valves, pipes, and other components. There's also the possibility to operate maintenance while reactor is operating thanks to, modular, easily replaceable components.

The basic feature of a modular reactor is to match the construction of generating capacity to a utility's future power requirements. IRIS offers flexibility, with a defined construction time of two to three years. This makes IRIS a good economic option to produce electricity power required, instead to have bigger power plants with the consequent higher investments and difficulty in injecting big electrical power on the grid.

It is also possible to establish a process lead to desalination of water. The development of a region is usually based on two main components: water and energy. An analysis was set up to study the possibility of building three IRIS modules to produce the amount of energy needed plus 7 reactor used for desalination of water in the Sonora region.

A key step in the R&D phase for SMR concepts as well as for IRIS, is the testing phase of the reactor safety features. This effort is currently under way in Italy: the SPES-3 facility will represent a reference facility worldwide for such a new type of reactors.



Layout of the SPES-3 integral testing facility under construction at SIET labs (Italy)

2.4 SMART (PWR-Integral)

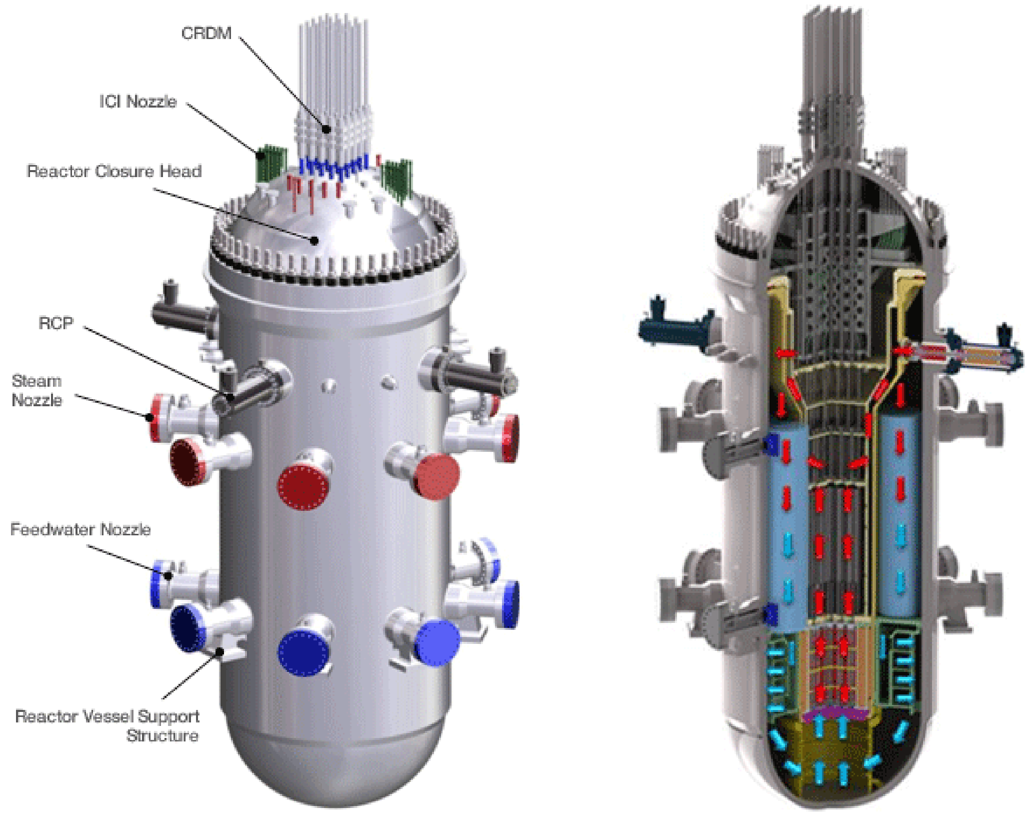


Figure 3.4.1. SMART reactor

	SMART
Type	Integral PWR
Power	330MWt 100MWe
Core Dimensions	5.3m D x 15.5m H
Desanalination (ton/day)	40,000
Design life	60 years
Assembly type	17x17 square FA
Fuel material	UO2
Active core length	2.0 m
Design pressure	17MPa

	SMART
Operating pressure	15MPa
Design temperature	360 °C
Core outlet temp.	323 °C
Core inlet temp.	296 °C
Minimum flow rate	2090 kg/s
Enrichment	Max 5%
Core Damage Frequency	10^-6
N° Steam Generator	8
Stram Generator Inlet	323 °C (30°C superheating)
Refueling	36 months

In 1996 Korea Atomic Energy Research Institute launched a project to develop an SMR based on numerous preceding studies. These studies suggested to focus on an integral pressurized water reactor (PWR) with a thermal power of 330 MWt and electric output of 100 MWe. This reactor is called System-Integrated Modular Advanced Reactor or SMART.

Korea wanted to set nuclear power industry as one of the first economy growth engines. SMART was addressed to developing countries for which small reactors are the best option, either because their power grids need to be geographically scattered or because their power grids are small.

SMRs designs can be inspired easily to new technologies and new concepts, due to their size. Safety can also be highly enhanced by using systems that couldn’t be used in large scale reactor so easily, such as passive safety systems. They can also contrast diseconomies of scale suffered by SMRs reactors by pursuing innovative approaches that lower costs simplifying systems, component modularization, factory fabrication and reduction of the construction time. The SMART reactor is characterized by an enhanced safety standard and the possibility to be used for: electrical power generation, desalination and district heating. One SMART reactor can supply power and water to a city with a population of 100,000.

In the SMART reactor core, steam generators, reactor coolant pumps and a pressurizer are inserted in the pressurized vessel creating an integral layout. This integral design makes it possible to remove the large-size pipe connections going outside the vessel, thus essentially excluding the occurrence of large LOCA accidents. During design basis events the in-vessel pressurizer maintain the system pressure at a constant level.

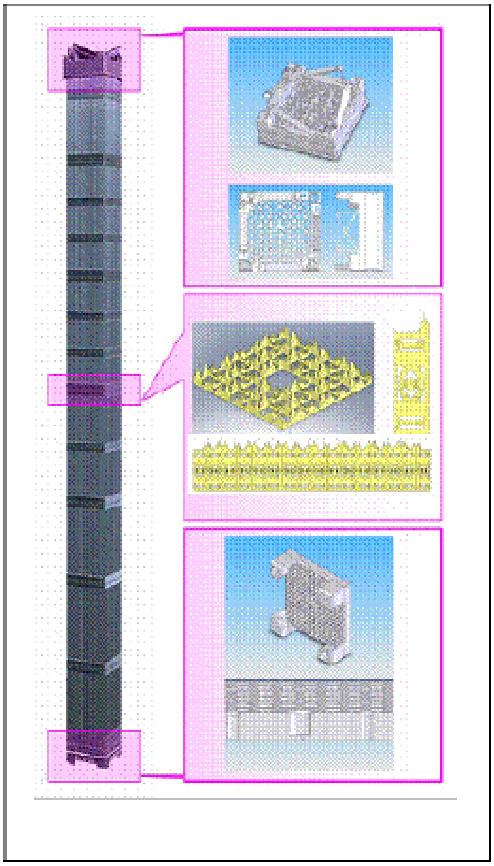


Figure 3.4.2. Typical 17x17 Fuel Assembly

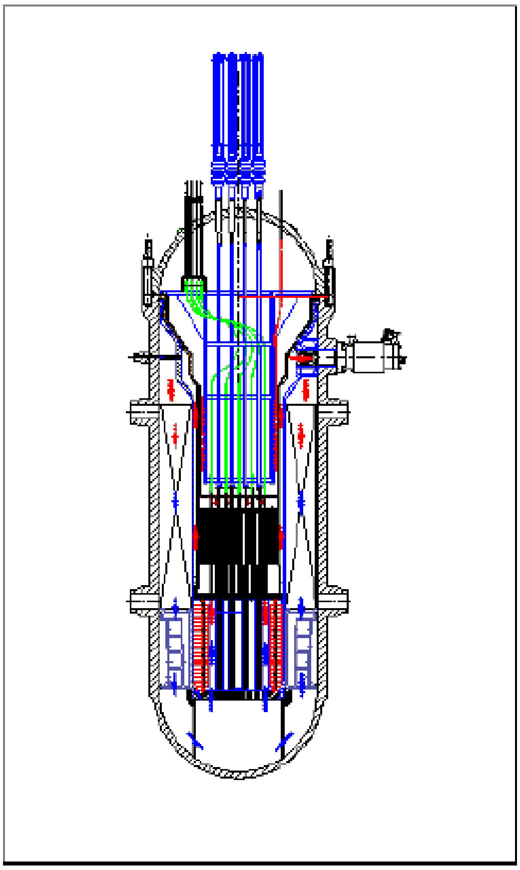


Figure 3.4.3. Primary water flow inside the core

Other important design features in SMART are the simplified and improved safety systems. Passive safety systems such as passive residual heat removal system (PRHRS) are installed. These systems prevent and/or mitigate the effects of accidents such as over-pressurization and excessive heating of the primary system carrying away the core decay heat during an accident without the use of pumps, but only through natural circulation.

In detail, these are the main safety systems in the SMART reactor:

- Reactor shutdown system (RSS):

  The RSS starts a reliable and rapid shutdown if it spots a deviation out of the permitted range of monitored variable. This systems is made of control rods and their drive mechanism (CRDM). The shutdown signal makes the control rods drop into the reactor core by the use of only gravity force and consequently stops the neutron chain reactions

- Safety injection system (SIS):

  The SIS is acts to prevent core damage when a small break LOCA occurs. So its function is to cover the core with a large quantity of primary water. If the pressure drops below 10 MPa the SIS is actuated automatically and injection of water into the reactor coolant system from IRWST starts immediately.

SIS is made of four independent trains: each train has a 100% capacity. The aim of that system is to provide vessel refilling so that the decay heat removal system can work properly also in a long-term scenario following an accident.

- Passive Residual Heat Removal System (PRHRS)

  In case of an emergency such as a station black out, the PRHRS removes the core decay heat using natural circulation. Alternatively, PRHRS can be used if a long-term cooling is needed, for example in case of repair or refueling.

  The PRHRS is designed with four independent trains with a 50% capacity each. Each train is made of one emergency cool-down tank (containing the water needed for cooling), a heat exchanger and a makeup tank.

  A system made this way is able to maintain the core un-damaged for 36 hours without any other external actions by operators. In the case of a standard shutdown, so not in case of emergency, the residual heat is removed through the steam generators setting up a turbine bypass system.

- Shutdown Cooling System (SCS)

  The SCS is used in together with the PRHRS to decrease the temperature of the RCS after the shutdown, from the hot shutdown temperature to the refueling one.

  Steam generator or PRHRS act during the initial phase of the cool-down process. After the reactor coolant temperature and pressure have been reduced, the SCS, using heat exchangers and pumps, operates from now on to reduce the temperature, finally reaching and maintaining the refueling one.

- Containment Spray System (CSS)

  CSS purpose is to reduce containment pressure and temperature due to a main steam line break (MSLB) or LOCA and to clear containment atmosphere from fission products. The CSS uses the in-containment refueling water storage tank (IRWST) and has two independent trains. The CSS provides orated water to the containment atmosphere by the use of sprayers from the upper containment parts.

- Reactor Overpressure Protection System (ROPS)

  ROPS function is to reduce the pressure inside the reactor considering design basis accidents. The system is made of two pressurizer safety valves (PSVs), located in the upper part of the reactor. The piping discharging steam of PSVs are linked to the containment atmosphere through the reactor drain tank (RDT). In case primary system pressure increases over the limit, PSVs are opened and the steam is discharged into the RDT.

- Severe Accident Mitigation System (SAMS)

  SAMS prevents the molten corium possibly resulting from a severe accident to go out of the containment. Characteristics of the design of cavity and containment and the safety systems together can prevent egress of corium can be avoided due to. In case of a severe accident water from CSS fills the small air gap under the RPV. External cooling is granted by the water in the IRWST preventing an egress of the corium out of the RPV. To prevent hydrogen explosion the containment has some hydrogen.

All this systems can be summarized in picture 3.4.4.

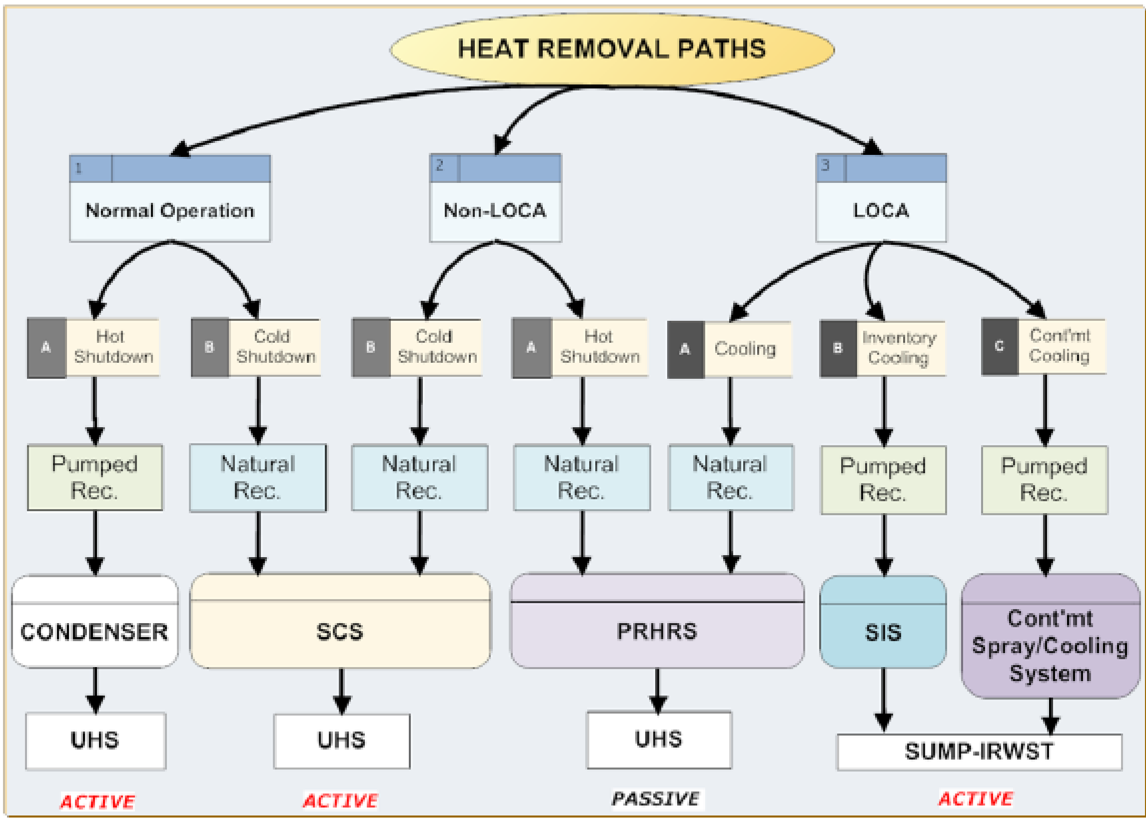


Figure 3.4.4 Security systems in SMART reactor

A thermal margin of about 15% , accommodating any transient regarding heat flux, is ensured by the low power density design with slightly enriched (<5 w/o) UO2-fueled core. This feature ensures the core thermal reliability under normal operation.

Soluble boron and control rods are the systems used to control reactivity during normal operation. To obtain a radial flat neutron flux burnable poison rods are introduced; this results in the increased thermal margin of the core. Constant average coolant temperature program is adopted in SMART reactors to improve load follow operation performance having stable pressure and water level within the pressurizer.

8 modular type once-through steam generator are designed as helically coiled heat transfer tubes and produce superheated steam at 30 degrees in normal operating conditions.

Among other improved design features is it possible to find the canned motor reactor coolant pump, that has no pump seals, preventing loss of coolant in case of a seal may suffer a failure. System reliability is also enhanced by four channel control rod position indicators.

2.5 KLT-40 (PWR-Loop, Barge)

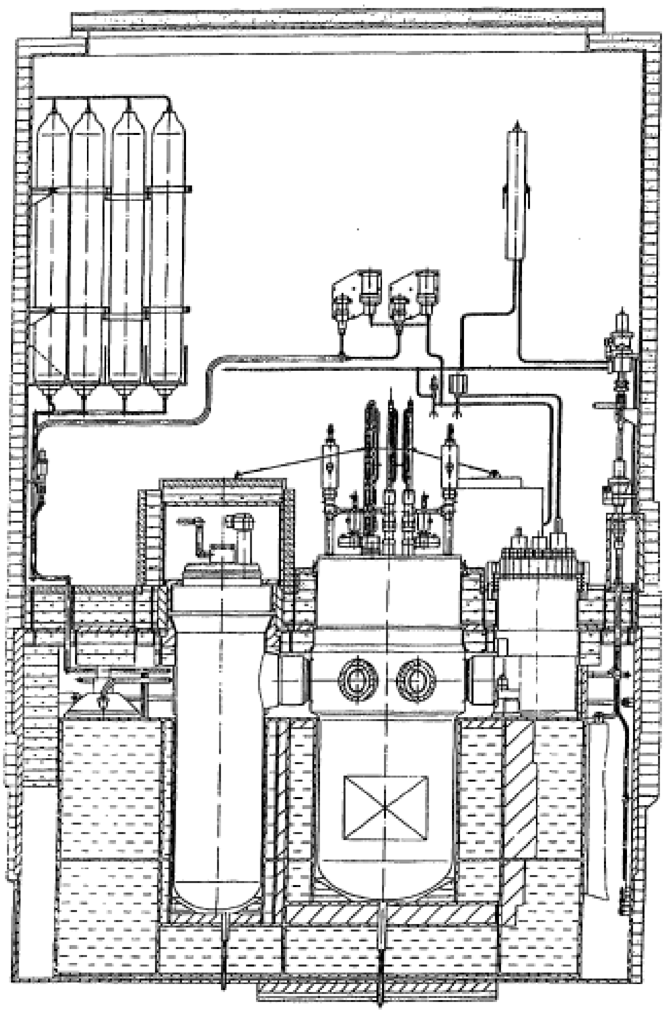


Figure 3.5.1 KLT-40 schematic view of the reactor

	KLT-40
Type	PWR
Power	150MWt 35/19.4MWe
Weight	21500 ton
Fuel Assembly	121
Vessel Base Material	Steel, 15Cr2NiMo, VA-A
Primary Pressure	12.7MPa
Core outlet temp.	316 °C

	KLT-40
Core inlet temp.	280 °C
Steam Output	240t/h
Superheated Steam Pressure	3.72MPa
Superheated Steam Temperature at SG Outlet	290°C
Feedwater Temperature	170°C
Tube Material	Titanium Alloy
SG Weight	23 tons
Refueling	3.5 years

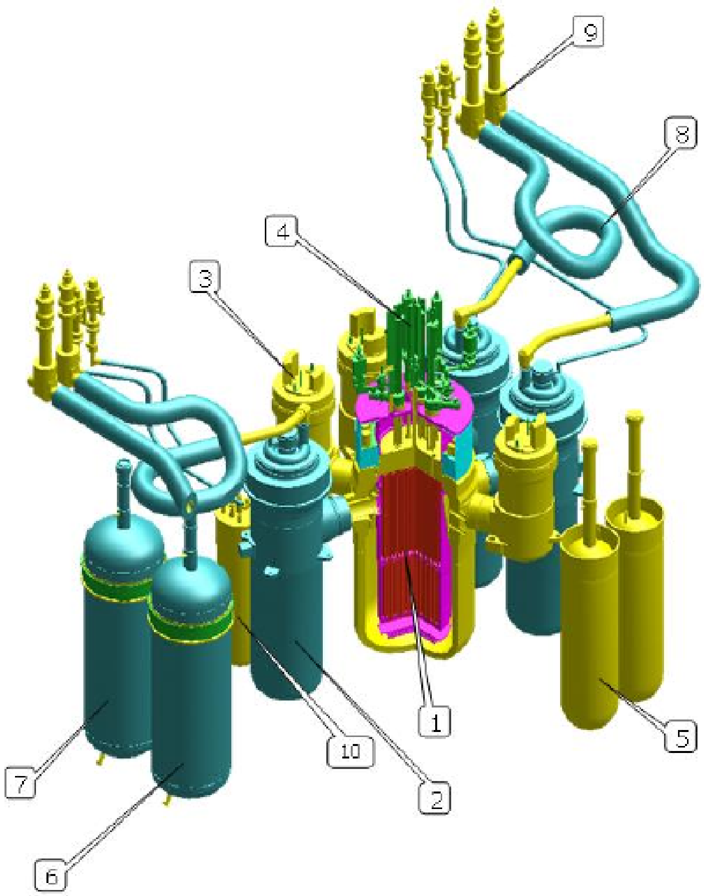


Figure 3.5.2 Primary circuit layout

The KLT-40S is a variant of the KLT-40 used to power icebreakers and it is used in the Russian floating nuclear power station.

Floating nuclear power stations are projected by Rosatom self-contained and with low-capacity. The stations are to be mass-built in factories or naval-building facilities and then moved to the destination point in coastal waters near a city, a town or an industrial enterprise. By 2015, at least seven of the vessels are supposed to be built.

This kind of systems are non-self-propelled vessels with a length of 144.4 metres (474 ft), width of 30 metres (98 ft), height of 10 metres (33 ft), and draught of 5.6 metres (18 ft). The vessel has a displacement of 21,500 tonnes and a crew of 69 people.

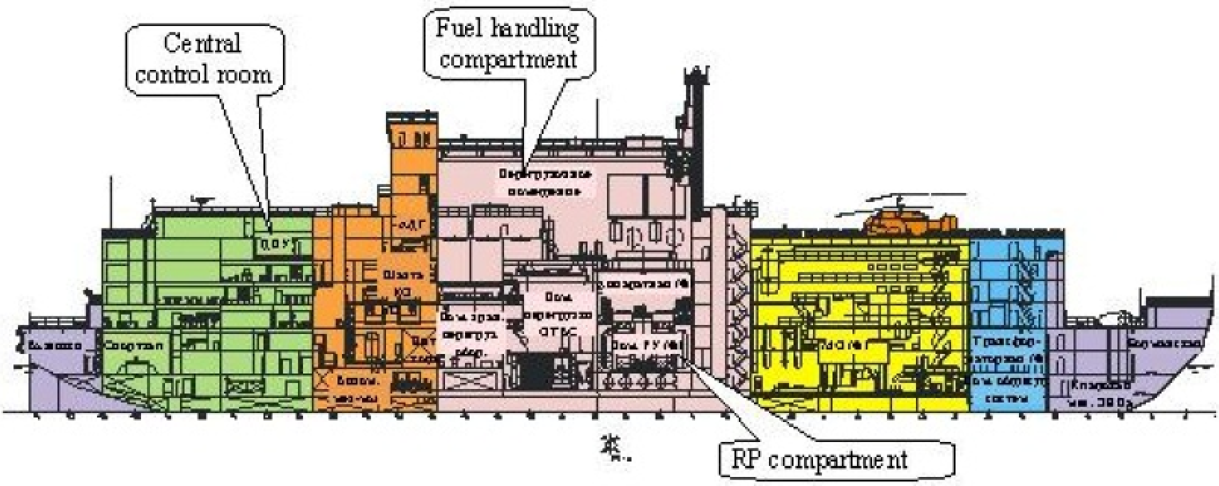


Figure 3.5.3 Floating plant layout

The core contains uranium dioxide fuel rods with high corrosion resistance cladding made of zirconium alloy; uranium fraction is also increased used a closely packed pattern, providing an high density of fuel compared to the volume core available.

Each reactor has its own containment that is a physical barrier projected to limit the spread of radioactivity and to localize fission products in case of a loss of coolant accident (LOCA), using emergency containment cooling systems.

Some features provide self-defense in the KLT-40: negative reactivity coefficients (fuel and coolant temperature, coolant specific volume, steam density and integral power); high heat capacity of the primary coolant and metal structures (safety margin provided by the design for the depressurization pressure of the primary system under emergency); limited outflow rate in case of a break, due to restriction devices in connection nozzles; once-through steam generators, limiting the rate of heat removal via the secondary circuit during steam line break accident.

Two KLT-40 naval propulsion reactors (modified) together provide up to 70 MW of electricity or 300 MW of heat, enough for a city with a population of 200,000 people. It can also be used for desalination plant producing 240,000 cubic meters of fresh water a day.

It must be pointed that floating plants could be more vulnerable to accidents and terrorism than land-based stations, considering also a history of naval and nuclear accidents in Russia and the former Soviet Union, including the Chernobyl disaster of 1986.

The 2011 Japanese nuclear accidents due provide a sharp contrast to some comparative safety advantages of floating nuclear plants. Land based nuclear facilities are designed to resist to severe ground accelerations. Sea water is needed for cooling, so nuclear power plant usually are located on the coast. Coastal locations tend to be the areas of maximum tsunami damage, requiring protective design against this phenomena.

A floating facility, near a coast but not in shallow water, can avoid the worst problems of earthquakes and tsunamis. In the event of an accident, terrorist attack, or other calamity, it is essential to keep the core cooled, usually by covering it with water. An emergency measure can be to lower the core into the sea. Finally, standard nuclear power plants’ decommissioning can be difficult and expensive.

Safety of the KLT-40 reactor is based on the defense-in-depth principle. This principle set up accident prevention and mitigation procedure and strategies, such as a number of physical barriers preventing diffusion of radiation and radioactive materials into the environment, and a system of technical and organizational procedures to protect barriers and retain their effectiveness, as well as protection of the personnel, population and environment.

There are lots of defense levels of technical and organizational measures under the defense-in-depth principle: prevention of abnormal operation and failure (based on negative reactivity coefficients and high thermal conductivity of the fuel); control of abnormal operation and detection of failure (active systems of control); control of accidents within the design basis (emergency protection rods, passive emergency heat removal system and self-actuating devices in emergency during shutdown); control of severe plant conditions (passive reactor vessel bottom cooling system and passive containment cooling system); mitigation of radiological consequences of significant release of radioactive materials (mainly organizational measures).

At present, fabrication of main equipment for the KLT-40 reactor and turbine-generator sets is near completion.

2.6 PHWR-220

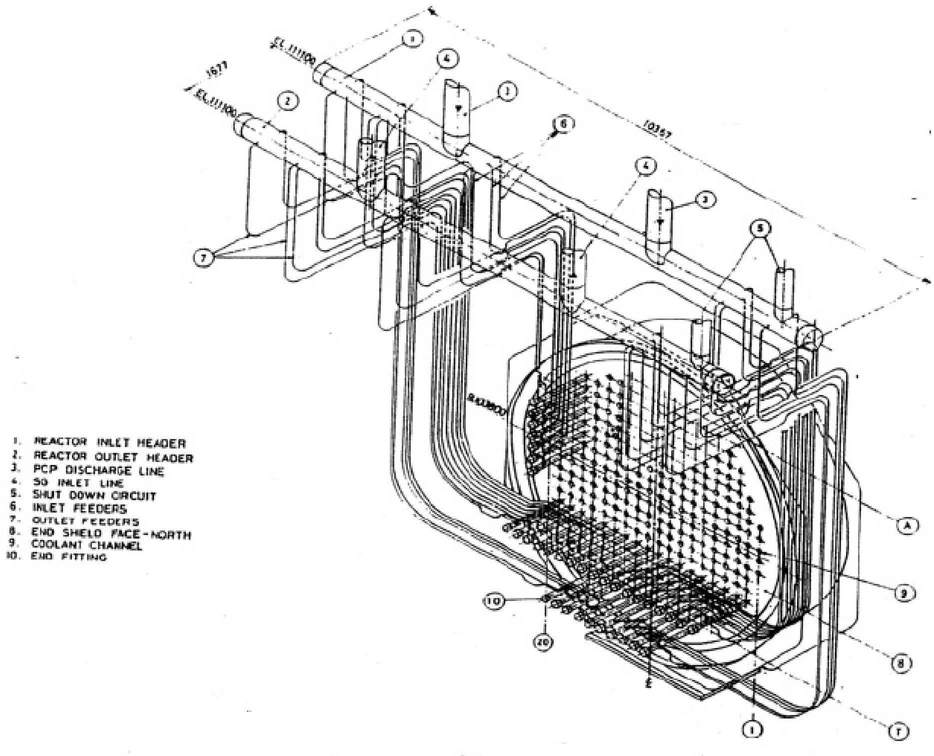


Figure 3.6.1 Schematic plant layout

	PHWR-220
Power	754MWt 211MWe
Active Core Dimensions	4.51m D x 4.95m H
Shell Dimensions	6m D x 5m; 25mm Thickness
Design Pressure/Temperature	.23 MPa/100 °C
Shell Weight	21.3 tons
Efficiency	26.5%
Primary/Secondary Coolant	High/Light Water
Core Damage Frequency	10^-5
Fuel Rods Diameter	15.22 mm
Primary Coolant Flow Rate	221000 kg/s

	PHWR-220
Core outlet temp.	293 °C
Core inlet temp.	249 °C
Steam Output	2216 kg/s
Steam Pressure	4.03 MPa
Steam Temperature at SG Outlet	250°C
Feedwater Temperature	171°C
Refueling	2 years
Plant life	40 years

Different size of water reactors are included in the Indian Pressurized Heavy Water Reactors consists of unists of 220 MWe, 540 MWe and 700 MWe.

Natural uranium dioxide and heavy water reactor are used in the PHWR-220. The reactor is made of a calandria filled with water and an integral assembly of two end shields. Pressure tubes containing the fuel are made of 306 Zr-2.5%Nb, in a square pattern design of 22.86 cm pitch. At the ending parts of pressure tubes it is possible to find a sort of cap made of AISI 403 modified stainless steel going through the shields and extending into the fueling machinary to facilitate power fueling. The calandria is a horizontal vessel containing the coolant channel, moderator, reactivity control mechanisms. Diameter and length of the main outer shell are 6.05 and 4.16 m. Calandria stands atmospheric pressure, so wall thickness is only 25 mm and made of austenitic stainless steel type 304 L.

Shield is very important, above all, during refueling process providing shielding to limit the dose rate in the fuelling machine vault, protecting the operator. They also have a support function for the fuel channels. A baffle palate separates the space inside the end shield into two. Water fills the front compartment and the rear one is full of carbon steel balls and water.

On-line refueling is an important feature of Indian PHWRs. Pressure tubes must be opened and resealed during reactor operation and the refueling machine must stand the high temperature and high pressure of the primary. A key spot in this process in the fuel handling system. It's composed by two fuelling machines working in unison During refueling operation , fueling machines are connected on the upstream end and on the downstream side of the pressure tube that needs refueling. In this way the upstream machine loads the new fuel and the downstream one receives the spent fuel assemblies.

The primary heat transport circuit has four coolant pumps with three mechanical seals that, singularly, can sand full system pressure, providing good boundary sealing.

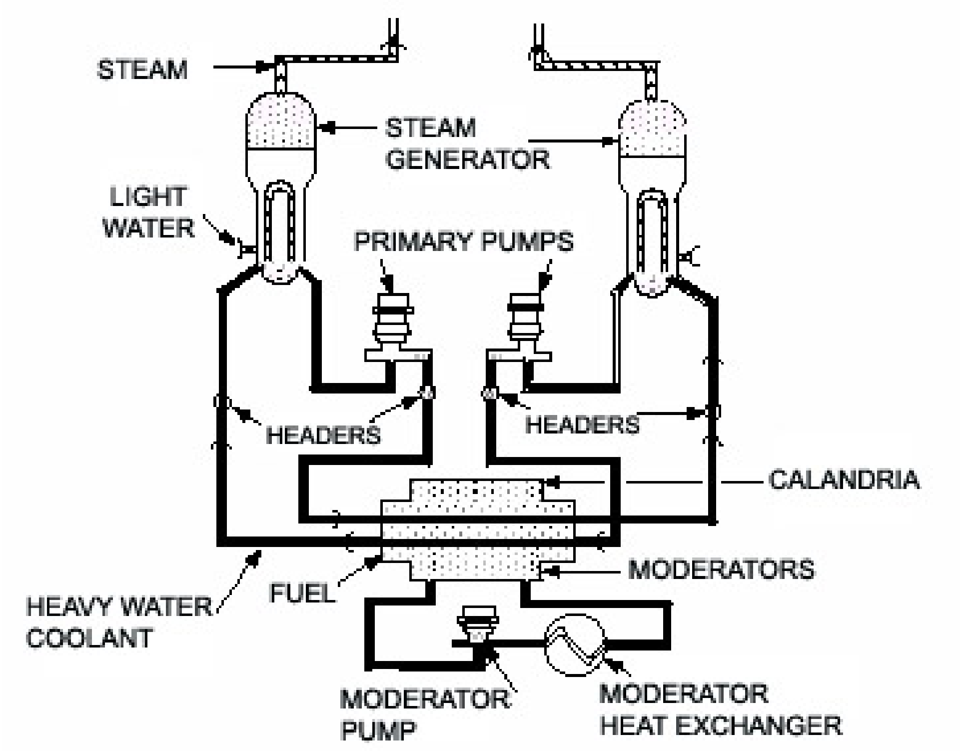


Figure 3.6.2 Schematic of the primary heat transport system PHWR

PHWR-220 is designed with 4 recirculation steam generators, vertical mushroom type, with U-tubes in which flows the primary coolant. Steam generators are located in two concrete zones, on the opposite sides of the reactor. Hot water if the primary circuit flows from the reactor to the inlet of the steam generator; the feedwater flows back to the SG inlet at 171°C. Steam separator are placed in the upper part of the steam generator to separate water from dry steam.

Indian PHWR's design is projected to reach fundamental safety objectives according to regulatory requirements and standards. Key spots for safety design are the following: defense in depth, physical and functional separation of systems relevant for safety, detailed safety analysis using both deterministic and probabilistic methods, redundancy of safety systems, routine testing.

Secondary containment surrounds the primary one and it's made of pre-stressed, while the primary one is built using reinforced concrete. The gap between inner and outer containment is maintained at a negative pressure to minimize the possibility of a possible release to the environment during accident conditions. In case of accidents all lines opening to the containment atmosphere are automatically closed; this process is triggered by pressure sensor or activity growth inside the containment. The containment is provided with some systems designed to start after an accident in order to cool down the containment atmosphere, limiting the maximum pressure and to maintain a clear atmosphere inside the containment.

There's the possibility to increase fuel burn up beyond 15000 MWd/TeU using higher fissile content materials like enriched uranium, instead of natural one, MOX or TOX fuel. This enhancement is under investigation. Actually the maximum burn up that has been studied is 30000 MWd/TeU.

Slightly Enriched Uranium Bundles (SEU), MOX and thorium dioxide bundles and deplete uranium bundles were designed, and successfully irradiated in different PHWR-220. In order to flatten the flux thorium bundles and reprocessed depleted uranium dioxide bundles were in the initial fuel load

Then MOX-7 fuel evolved in a 19-element cluster, with inner seven elements made of MOX pellets and outer 12 elements consisting only in natural uranium dioxide pellets.

The SEU bundle design differs from the previous one, in fact it’s composed by a 19-element fuel bundle with a 0.9% enrichment.

2.7 VBER-300 (PWR-loop)

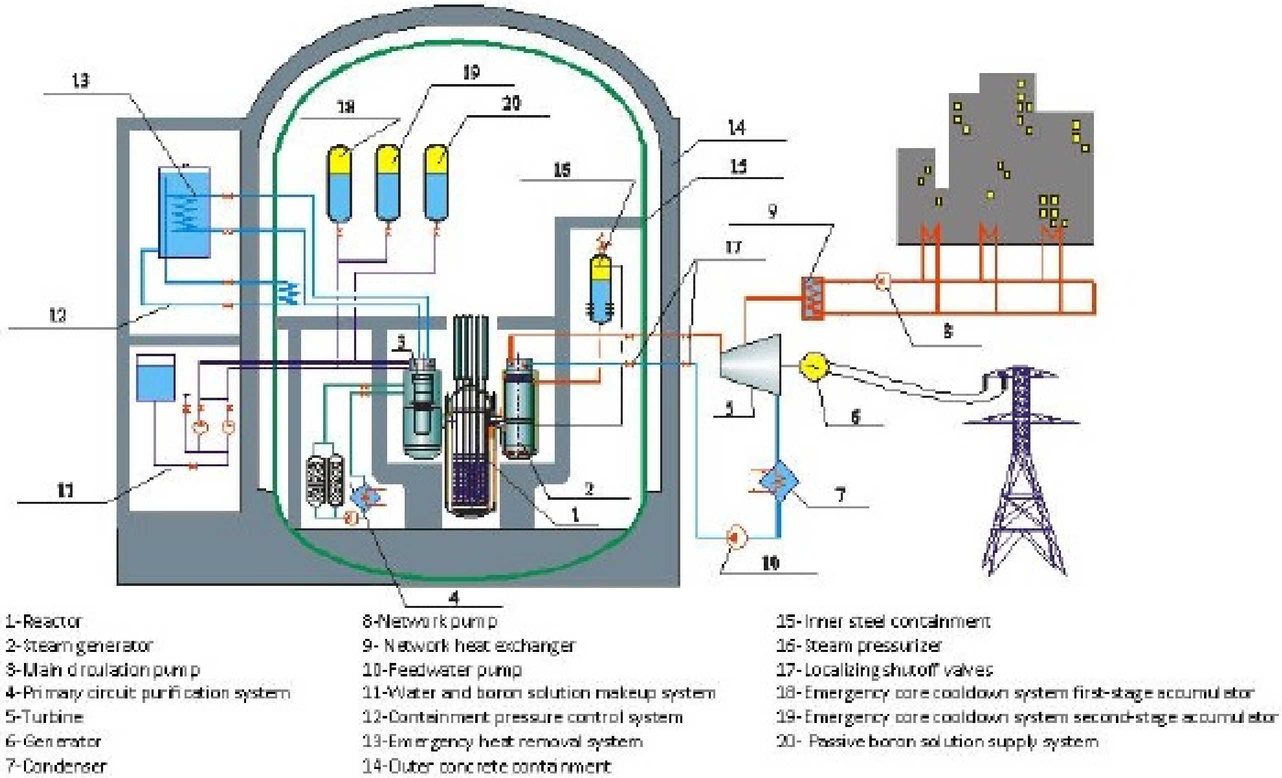


Figure 3.7.1 VBER-300 Plant layout

VBER-300	
Power	917MWt 335MWe
Core Size	2.29m D x3.53m H 20.5cm thickness
Core Weight	306.6 tons (1300 working)
Coolant	Light Water
Coolant temperature	400-503°C
Core Damage Frequency	<10^-6
Operating Pressure	16.3 MPa
Coolant flow rate	4483kg/s
Inlet/Outlet Core Temperature	292-327.5 °C

	VBER-300
Steam Flow Rate	472 kg/s
Steam Pressure	4.37 MPa
Inlet/Outlet SG Temperature	220-305°C
Fuel	UO2
Enrichment	4.95% U-235
Refueling	72 months
Structural Material	HT-9 ferritic martensitic steel
Efficiency	33%
Plant Design Life	60 years

The VBER-300 reactor is designed to be a power source, ground-based, and nuclear also a cogeneration plant, using its thermal power for desalination purpose. So, considering reactor design and layouts, the most suitable applications are: electrical power generation, seawater desalination and cogeneration of electricity and heat for district heating.

As most of Russian SMRs the VBER-300 RP is the result of the evolution of naval propulsion reactors. Increasing mass increases the thermal power output but the reactor plant layout and main design solutions are kept as similar as possible to those of marine propulsion reactors. The final design is the result of matching knowledge from naval systems and from that granted by the experience on VVER-type reactors operation and successes in the field of nuclear power plant safety.

The vessel consists of a reactor vessel and four steam generators and associated main coolant pumps connected to the reactor vessel by coaxial nozzles. The reactor vessel has a cylindrical body with an elliptical bottom. The modular design reduces reactor unit mass and volume, so that the specific capital investments decrease, and safety is enhanced by the exclusion of main circulation pipelines and possibly associated large and medium break LOCAs.

In the VBER-300 light water acts both as primary coolant and moderator. The hot primary water flows in a once-through steam generator that produces slightly superheated steam and sending it to the turbine it is possible to produce up to 335MWe of power. It is also possible to take off a small amount of steam to heat up the district heating circuit fluid.

Steam generator results in a modular coil-type vertical heat exchanger. Primary water lows inside the tubes, and the secondary outside. The tubes modules are grouped on the feedwater and steam sides into three independent sections.

Main coolant pump consists of an axial flow pump moved by canned electric motor constituting. Other systems enhance pumping efficiency, such as a guide flange, an axial-type console impeller and a guide vane. Guide flange and vane shape the flow at the impeller inlet and push the coolant in the pressure chamber.

In the VBER-300 project reliability and safety of the reactor play an central role, as well as high economic indicators of the fuel cycle that lead to the cassette design of the reactor core, as successfully applied previously in operation of VVER reactors.

Pellets 7.6 mm in diameter are used as fuel. Uranium enrichment is up to 5% (maximum licensed enrichment).

Shroudless fuel assemblies (FA) are used in VBER-300 as in VVER-1000 where they have proved a high load-carrying capacity and high resistance to deformation.

VBER-300 fuel is handled and transported as follows: spent FAs moved from the reactor into the storage pool and then into special container to be transported; to reload the fuel and in-vessel equipment it is used the dry method; a protective tube houses the reloaded FA; a shielded transportation container grants biological protection for the servicing personnel during dry materials transportation.

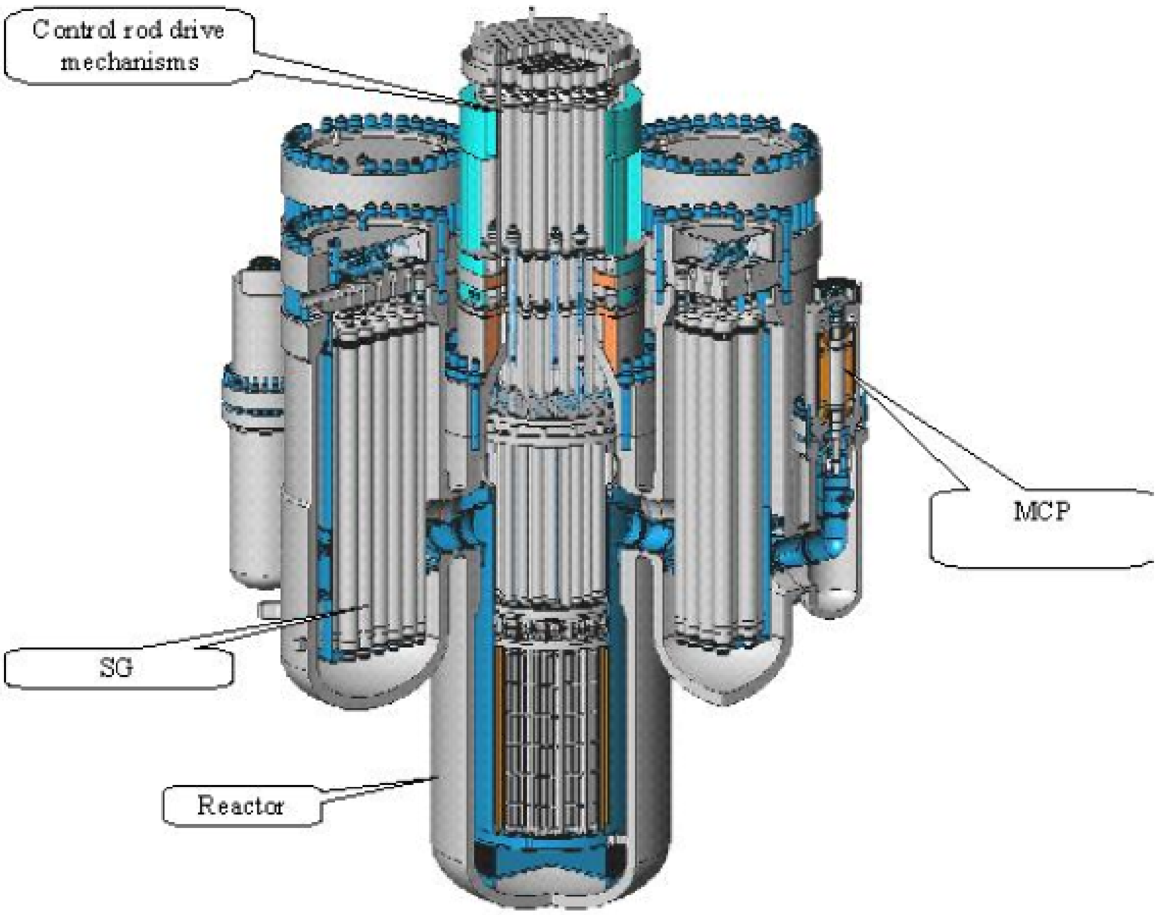


Figure 3.7.2. General view of the VBER-300

Transportation container is moved by the reactor compartment crane thanks to a double lift system.

The fresh fuel storage provides acceptance and storage of fresh fuel, and space for fuel preparation, before it is inserted into the core.

A special railway transportation is set up to deliver fresh fuel to the site, that can accommodate a 20% fuel in excess compared to the quantity needed to load the core. The fresh fuel storage and the reactor compartment are connected with an internal railway.

Dismounting the control rod drive mechanism is necessary before refueling process can begin; once the reactor cover is opened, the top structure is removed.

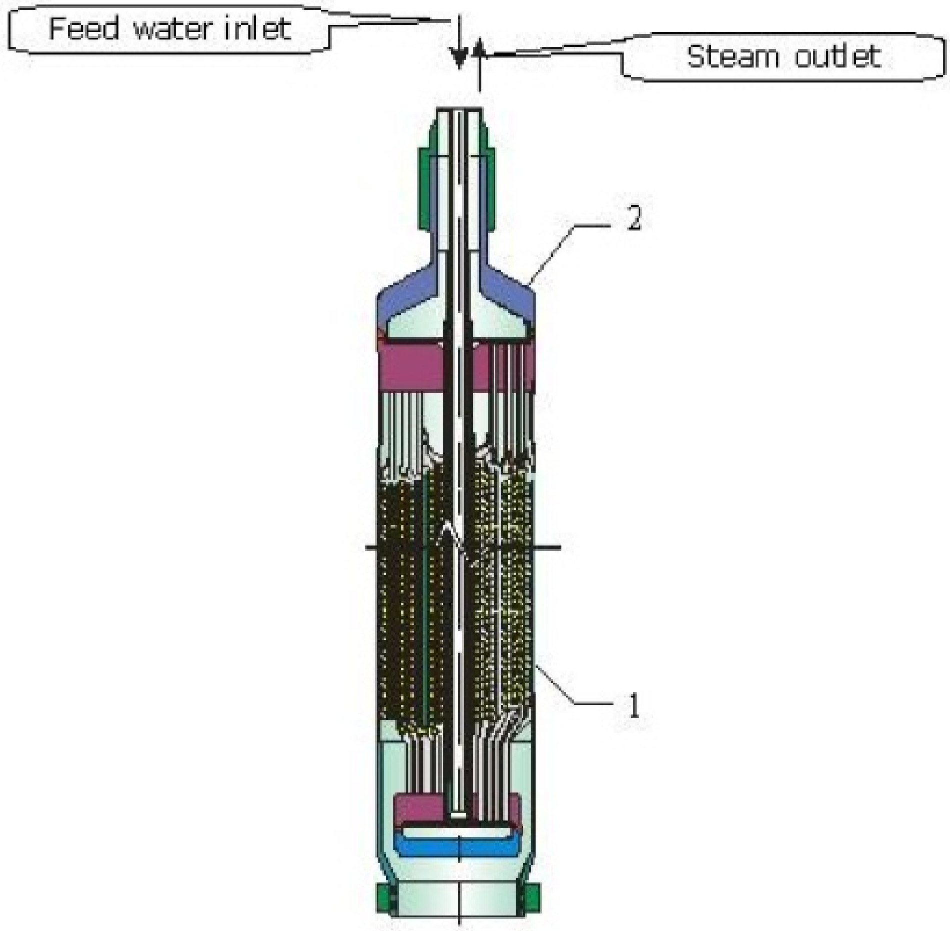


Figure 3.7.3. Steam generating module
(1 · tubing, 2 · steam generating module)

Refueling is made by the fuel handling machine handling, moving one FA at a time. The fuel handling machine is composed by a refueling tube housing one FA with the coolant (in this way it provides the biological shielding and heat removal).

Refueling process starts unloading spent FA from the reactor, transporting them to the decay storage pool and putting them onto an assigned storage rack shelf. After this is done, the fuel handling machine loads one fresh FA into the reactor and installs it into the assigned core cell. One of the most important side aspects of the refueling process is that in that period of time it is possible to inspect the integrity of fuel elements cladding and of the core itself.

The decay storage pool houses spent fuel for about six years and can accommodate the entire core inventory.

A nuclear power plant must provide personnel and population protection against the consequences of the design basis and severe accidents, so a containment system has been studied. These are the basic features and systems of the VBR-300 containment solution: fuel retention system in the reactor vessel during accidents with severe core damage; passive heat removal system limiting containment pressure in LOCAs; separation of functions granting protection against internal emergency impacts and external impacts both natural or human-caused; iodine and aerosol air purification system in order to clean air from radioactive leaks during accidents with containment standing a situation of overpressure.

In case of a ground-based nuclear cogeneration plant the containment structure is double, made of of an internal steel shell and an outer concrete shell (non pre-loaded).

The cylindrical steel shell dimensions are 28.0 m in diameter and 34 m high. The concrete shell is made of concrete with external diameter of 34 mm and height of 42.2 m.

Vessel system design service life is 60 years

2.8 ENHS (LMFBR- PbBi cooled)

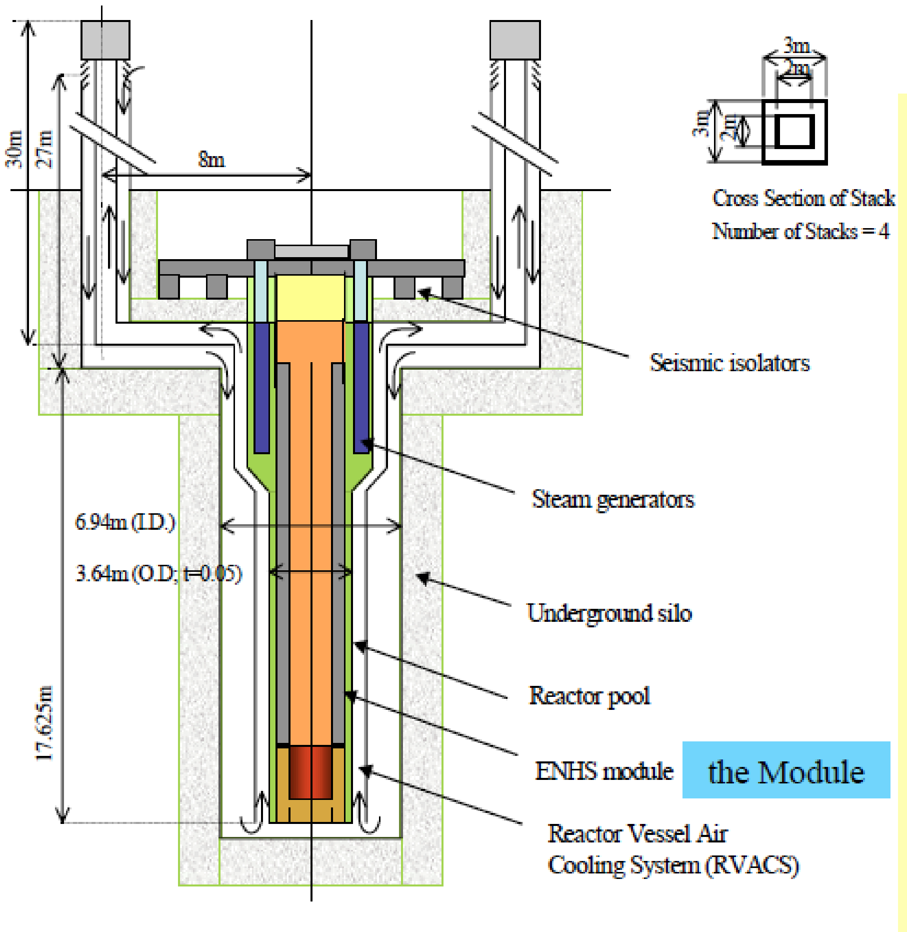


Figure 3.8.1. Schematic representation of the ENHS reactor.

ENHS	
Power	125-180MWt 50-75MWe
Core Size	3.64m x10m 5cm thickness
Coolant	PbBi
Coolant temperature	400-503°C
Coolant flow rate	8320kg/s
Inner/Outer Eff Radius	16.41/111.83cm
Fuel Rod Clad Diam	1.56cm
Transportation	Barge, truck or train

	ENHS
Fuel	Stainless Clad U-Pu
Enrichment	12-13% Pu
Refueling	20 years
Structural Material	HT-9 ferritic martensitic steel
Efficiency	38.4-40.7%

The ENHS (Encapsulated Nuclear Heat Source) is based on the lead-bismuth technology developed for Russian most advanced nuclear submarines. The ENHS is factory produced and it's delivered fueled and sealed. It has to be installed in a reactor pool and it's able to provide energy for about 22 Equivalent Full Power Years without refueling. At the end of the core life it would be substituted with a replacement "battery" and transported to a center for fuel cycle services. It may be an option for developing countries as well as for industrial countries. This reactor concept was studied by Ehud Greenspan and David Wade, its feasibility has been studied during 1999 through 2002 with the support of the DOE NERI program.

This DOE NERI sponsored project consisted of researchers from four institutions: The University of California at Berkeley (the lead organization), Lawrence Livermore National Laboratory, Argonne National Laboratory and Westinghouse. Three Korean organizations joined the project at the beginning of the second year; they are Korea Atomic Energy Research Institute (KAERI), Korea Advanced Institute for Science and Technology (KAIST) and Seoul National University. The Korean organizations were supported by the Nuclear Energy Research Initiative program of the Korean Ministry of Science and Technology (MOST).

Throughout the project there was useful interaction with CRIEPI, and to a lesser extent, TOSHIBA researchers that were involved in the design of the 4S reactor. In the second part of the project CRIEPI undertook to perform an independent evaluation of the transient behavior and safety characteristics of the ENHS reactor.

Most of the research done for this project and the obtained results are described in more than 40 publications that are openly available. University of Bologna has been involved in some final analysis of the ENHS core design in the years 2005-2009.

Designers working on that project focus their attention on some characteristics: natural circulation cooling, 20 full power years with no refueling, simplicity in construction operation and maintenance, transportability, autonomous load following capability.

Using small steam generators allows the plant to operate with supercritical steam increasing efficiencies up to about 40%.

There are no particular decay heat removing systems other than a reactor air cooling system to maintain the vessel at an appropriate temperature.

One of the most important feature is the possibility to ship the core "ready to use" and sealed. This is possible thanks to the absence of mechanical connection between reactor and secondary system. This sealed core is another barrier against proliferation. In fact transportation can be easier as the module with solid metal stored in the primary system can be used as shipping cask.

Except for control and safety (7 pieces total) there are no moving components; there are no fuel assemblies as every element is tied up to the grid plate.

ENHS reactor is designed to serve developing countries or to be placed in areas difficult to reach. It can be used to provide electricity, desalinization or district heating.

There's also the possibility of a different layout using heat pipes. This way lead temperature can be risen to 1040K and the entire module can be reduced in size. This design enhance passive heat removal and mechanical resistance of the core.

Factory fabrication and fuelling of the module makes construction time extremely short, about 2 years; this has a great influence also upon economical aspects.

The basic features of the original reactor concept did not change during the years, but the ENHS module is made more robust and more practical to fabricate. It is designed to have natural circulation of both the primary and secondary coolants. The variant design that uses cover-gas lift-pumps for both the primary and the secondary coolants will not be analyzed here.

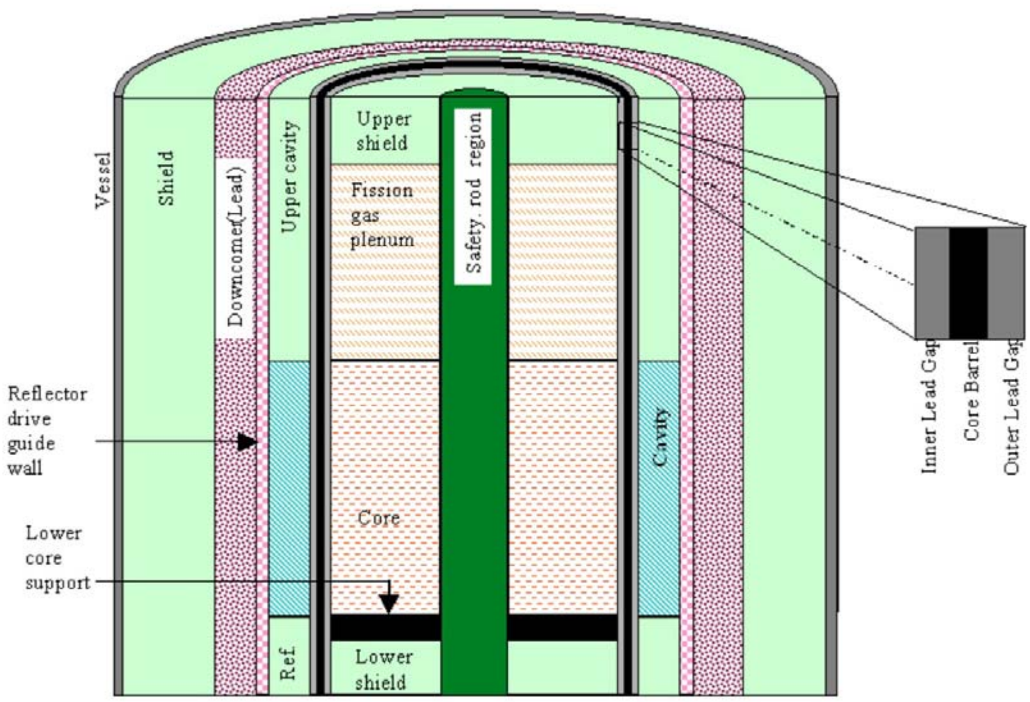


Figure 3.8.2. Schematic representation of the ENHS core

The ENHS core, Figure 2, is an annular cylinder made of uniform lattice of fuel rods that are individually tied-up to the lower grid plate; there are no fuel assemblies. There is are no blanket and no solid reflector assemblies. The core would be supported vertically from the bottom by the lower grid spacer plate. The pins would be held down against upward hydraulic flow and buoyancy forces by tying each pin to its pedestal on the lower grid plate in a way that allows for 2π steradian degree of freedom. The pins float in the coolant like an array of buoys, but are pinned in position at the bottom, and are positioned radially and azimuthally near the mid elevation by the upper grid spacer. The bottom pinning is done in a redundant fashion so that a single failure will not release a pin. The use of pin pedestals allows to eliminate a bottom grid spacer and to thereby avoid its pressure drop, this is an important consideration for the natural circulation cooling. The coolant enters the lower plenum from the downcomer via large holes in the sides of the core barrel. Then it distributes itself across the core radius, flowing radially between the pedestals, and turns and flows upward through the macro pin lattice. The large coolant gaps of the lattice and the pedestal placement facilitate the radial distribution both in the plenum and throughout the pin lattice.

The Control and Shutdown Systems consist of a single neutron absorber assembly located centrally in the core also referred to as the “Central Safety Assembly” and six segments of an annulus that surrounds the core sometimes referred to as the “peripheral absorbers”. The former is located in the coolant-filled cavity at the core center; the diameter of this cavity is determined so that the reactivity worth of the safety element will be adequate. The central cavity can also be used for flattening of the radial power distribution across the core. The central absorber has an electromagnetic latch that does not engage until the start-up temperature of 400◦C is achieved. At this temperature the assembly can be withdrawn. Normal operational shutdowns can be accomplished with the peripheral absorbers. The reactor is brought critical by a hydraulic system that moves the peripheral absorbers up at 1 mm/s to compensate for the negative temperature coefficient of reactivity. At the full power position, the peripheral absorber segments are stopped from further upward movement by mechanical stoppers whose movement is established by high-reliability gear drives. The height of the peripheral absorbers will be adjusted once a year or two to compensate for a slight drift in reactivity due to fuel burn-up. During shipping and reactor installation the absorbing elements are securely latched in place. The active element for both central and peripheral absorbers is B4C and tungsten; being heavier than LBE tungsten can scram by gravity.

The ENHS module, Figure 1, is designed to be as simple, robust and prolifera- tion resistant as possible. There are no moving components except for the control and safety elements drives. This module will be manufactured and fuelled in the factory and shipped to the site as a weld-sealed unit with solidified LBE filling the vessel up to above the fuel rods. A unique feature of LBE, that makes it possible to embed the fuel rods and core structure in solid LBE without damage, is its nearly zero coefficient of volumetric expansion upon phase change. At the end of its core life the module will be removed from the reactor pool and it will be stored on site until the decay heat drops to a level that will permit to solidify the coolant and to convert the module into a shipping cask. A schematic description of the design concept of the ENHS reactor is depicted in Figure 1. The nuclear steam supply system (NSSS) consists of one ENHS module and eight small steam generators. There is no mechanical connection between the module and the steam generators. Both primary and secondary coolants flow by natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the Intermediate Heat Exchanger (IHX) and flows back into the coolant plenum underneath the core. In a vertical counterflow arrangement the secondary coolant flows from the pool outside of the vessel into the bottom of the IHX and exits back to the pool near the top of the IHX. The IHX consists of rectangular channels that are connected at their top and bottom to a tube sheet. The 4 mm thick rectangular channel walls provide the barrier between the primary and the secondary coolants whereas the inner and outer walls provide the structural support. More conventional IHX made of circular tubes could be used as well. Relative to circular tube IHX the rectangular channel IHX features close to an order of magnitude smaller number of channels and smaller friction losses due to elimination of the grid spacers.

The eight steam generators (SG) are anchored to the support structure that

covers the pool and are not mechanically connected to the Module. They are designed to meet several unique requirements that are dictated by the ENHS reactor layout:

• effective utilization of the pool volume surrounding the Module;

• minimum friction losses so as to enable 100% natural circulation of the inter- mediate coolant;

• having no mechanical connection with the Module;

• minimum flow rate of water into the intermediate coolant pool in case of a breach in steam generator tube or failure of other water-containing component;

• accommodation of a large thermal expansion;

• ease of inspection and maintenance;

• modular design that is easy to install and replace.

The steam generator is of a once-through tube-in-tube design; feed-water flows in via the inner tube and the steam is generated in the shell between the inner and outer tubes. The liquid metal coolant flows outside the tubes. Also, the steam and feed water piping and nozzles are located outside the ENHS pool, and the feed water to each steam tube is inherently orificed by a small diameter feed tube. These features all act to minimize the quantity and or mass flow rate of water or steam that can be introduced into the pool due to a postulated steam line break, or feed line break, or tube rupture.

It is possible to construct a power plant made of multiple ENHS modules. There are a couple of general approaches to the design of multiple ENHS module plant like, for example, to install several ENHS modules in a single pool of secondary coolant or to use as many independent single module ENHS reactors as desirable in a single power plant. A Schematic representation of this second approach is reported here in Figure 3: the power plant consists of 12 ENHS reactors, each including its own reactor pool, steam generators and turbine-generator. The total capacity of this power plant is 600 MWe. This arrangement provides the utmost level of uniformity and modularity. Additionally, this arrangement is most suitable for a gradual increase in the installed capacity of the power plant so as to best fit the increase in demand for electricity.

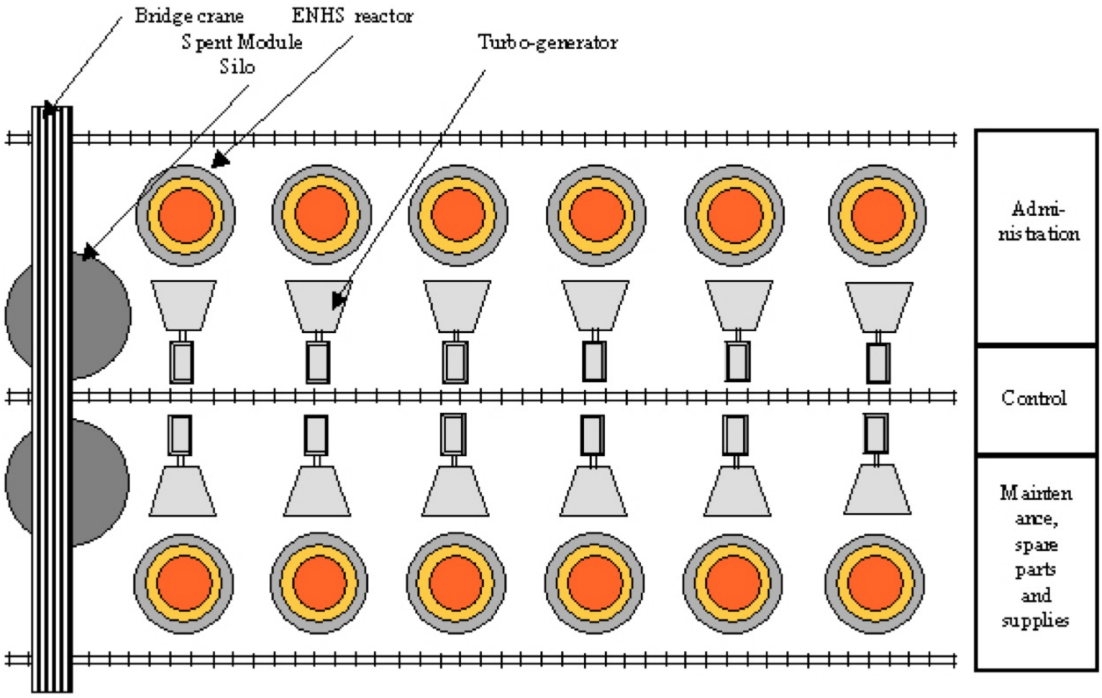


Figure 3.8.3. Schematic representation of a 12 ENHS reactors power plant for 600 MWe

The design philosophy of the ENHS has different features conceived to meet the technology goals of Generation IV reactors:

Sustainable Energy

The long-term fuel cycle envisioned for the ENHS reactor is a closed, fuel self

sustaining (FSS) Pu-U cycle. The ENHS core is designed to maintain the fissile-fuel content nearly constant with burn-up; that is, to have a breeding ratio that is few percent above unity. There is a slight build-up of fissile fuel with burn-up that is used to compensate for the negative reactivity effect of the fission products that accumulate during the cycle. The fuel is discharged from the FSS core when reaching its radiation damage limit. What is necessary for reusing this discharged fuel is to remove all or part of the fission products, mix the HM with makeup fuel and re-fabricate fuel elements. The resulting multi recycle fuel cycle features a high fuel utilization (an order of magnitude higher than in LWR), along with a great reduction in the inventory of the high level waste that need be disposed of in an underground repository.

Inherent Safety and Reliability

There are no pumps, valves or pipes in both the primary and secondary coolant systems of the ENHS reactor. Moreover, as the ENHS reactor pool is located inside a silo and the coolant level is nearly 15 m above the fuel level, it is inconceivable to have a LOCA or a LOFA. High heat capacity and very high boiling temperature of the low pressure single phase coolant make core-voiding accidents via coolant boiling inconceivable. Very small excess reactivity available at any time once the reactor is at full power. Accidents involving large reactivity insertion are inconceivable too. Accident analysis done so far revealed that the deviation of the system from nominal operating conditions is very slow as compared to other LMR or LWR reactors. All the postulated events were mitigated by naturally occurring phenomena – negative reactivity feedback, natural circulation as well as very high heat capacity. For these reasons, temperature changes due to accidents are relatively slow and small. No operator intervention is necessary. Moreover the reactor is very tolerant to operators’ errors. The Reactor Vessel Air Cooling System (RVACS) is a passive system which enables decay heat removal in case of an accident. It is sufficient to remove the decay heat and keep all plant components at below damage level temperatures.

There is no fuelling hardware on site. The module is removed and shipped in a

special cask with fuel frozen in the primary coolant; the fuel-handling accidents are eliminated. The ENHS reactor is expected to offer very high reliability due to the following features: no pumps or valves in the primary and secondary systems, nearly zero burn-up reactivity swing, very long core life, low power density. In fact, after the reactor is brought to nominal power, it can operate autonomously. Once a year or so there may be a need for a slight adjustment of the height of the peripheral absorber segments so as to compensate for small reactivity changes with burn-up. The reactor can follow the load autonomously over a wide range of power levels. Also, being disposable and having relatively few components, the ENHS module is not expected to require much maintenance. There is easy access to inspect and maintain the steam generators.

Proliferation Resistance

The ENHS offers a unique combination of technological barriers and material barriers that, along with adequate institutional barriers can make the nuclear energy system extremely proliferation resistant.

Technological Barrier 1: No Access to Fuel

The combination of long-life core and nearly constant Keff makes it possible to eliminate on-site refueling altogether. The ENHS module is designed to be factory fueled and to be disposed or recycled after 20 EFPY of operation. It is envisioned that the ENHS factories and recycling facilities would incorporate stringent international safeguards and security controls.

The fuel is to be sealed inside the ENHS module from the time the module leaves the factory until the spent module is returned to the waste disposal site or to a regional or international recycling center. It is envisioned that the ENHS power plants will not even have on-site hardware for refueling. The absence of on-site fuel handling, combined with the small number of components inside the ENHS module, enables designing the module in a way that will make it unnecessary to ever open this module. The components inside the module are robust and will be designed to operate reliably for 20 EFPY without a need to access them. The ENHS module is envisioned to have the fuel sealed inside a welded enclosure that could serve as a shipping and disposal container or would only be recycled at a secure, internationally controlled, regional recycling center. Even if individuals in the client country were to break into the top cover of the ENHS module, they would not be able to easily remove the fuel. The fuel is effectively secured in the core support structure since it is not designed for replacement on site. When outside the factory and outside of the secondary coolant pool, the fuel is imbedded in solid LBE. It is practically impossible to steal the ENHS module with the fuel: The module is 20 m long and 3 m in diameter and weighs 300 tons. The fact that the fuel is shipped imbedded in solid LBE or Pb makes it even more difficult to steal the fuel; it will take long time and special mechanical and heating equipment to destructively separate the fuel from the module. Any attempt to break into the module could be immediately detected by the IAEA by using automatically operating monitors that are connected to wireless transmission devices. The long time it will take even a trained team of people to break into the fuel will give the international community ample time to take measures to prevent diversion of the ENHS fuel.

Technological Barrier 2: No Access to Neutrons

There is no access to the neutrons in the client country. There are very few components inside the module; none requires maintenance. Hence, no need to open the module in the client country for operation and maintenance. Moreover, the module is sealed in the factory so that efforts to open it in the client country can be detected almost immediately. Even if there were ways to open the module in the client country undetected, it would be physically impossible to insert fertile material for irradiation into or in the vicinity of the core. This is because the fuel rods fill all the space inside the core barrel and there is no way to remove fuel from the top of the core. There is no blanket fuel in or around the core as is common in designs of sodium-cooled fast reactors. The current of neutrons in the pool in the vicinity of the ENHS module vessel is too low to be useful for any strategic material production application.

Technological Barrier 3: No Facilities Suitable for Military Applications

Installing and operating ENHS reactors will not require the country to obtain sensitive technologies that can be used for clandestine production of strategic nuclear materials. Specifically, no fuel fabrication or handling facilities and no fuel reprocessing capability are needed in the client country.

Material Barrier 1: Isotopics of the Fuel

It is technically possible to fuel the ENHS with uranium enriched to approximately 13% 235U. This fuel has little value for development of nuclear explosives. There are, however, a couple of concerns related to proliferation with the use of low enriched uranium fuel: the resulting spent fuel will contain significant amounts of plutonium which can be of weapons grade if the operation of the ENHS is stopped after a short period of time. In addition to this, there will be a continuous build-up of the global plutonium inventory. On the other hand, fuelling with LWR spent fuel plutonium has the following proliferation resistance attributes: the Pu is mixed with many minor actinides (MA); it is never separated from the uranium and MA. The concentration of the MA keeps building up. The in-core inventory of Pu increases very slightly; there is essentially no accumulation of Pu. The loaded fuel has a significant radiation barrier, due to the MA.

Material Barrier 2: Radiation Barrier

A unique feature of the ENHS is the possibility to seed in the core a very effective radiation barrier other than the fuel. This is because the fuel is loaded in the factory and is shipped to the site imbedded in LBE or Pb. Thus, after loading the fuel into the ENHS vessel and before pouring in the LBE or Pb it is possible to insert into the core or its close vicinity strong sources of gamma rays. After filling the module with LBE or Pb up to the top of the fission gas plenum, the radiation level outside of the ENHS module will be very low; the radiation sources can be designed so that they will not interfere with the shipment and installation of the ENHS module. However, access to the fuel will be deterred by the potential exposure to the high radiation fields of the seeded radiation sources when trying to remove the LBE or Pb.

2.9 BREST-300 (LMFBR, Lead cooled)

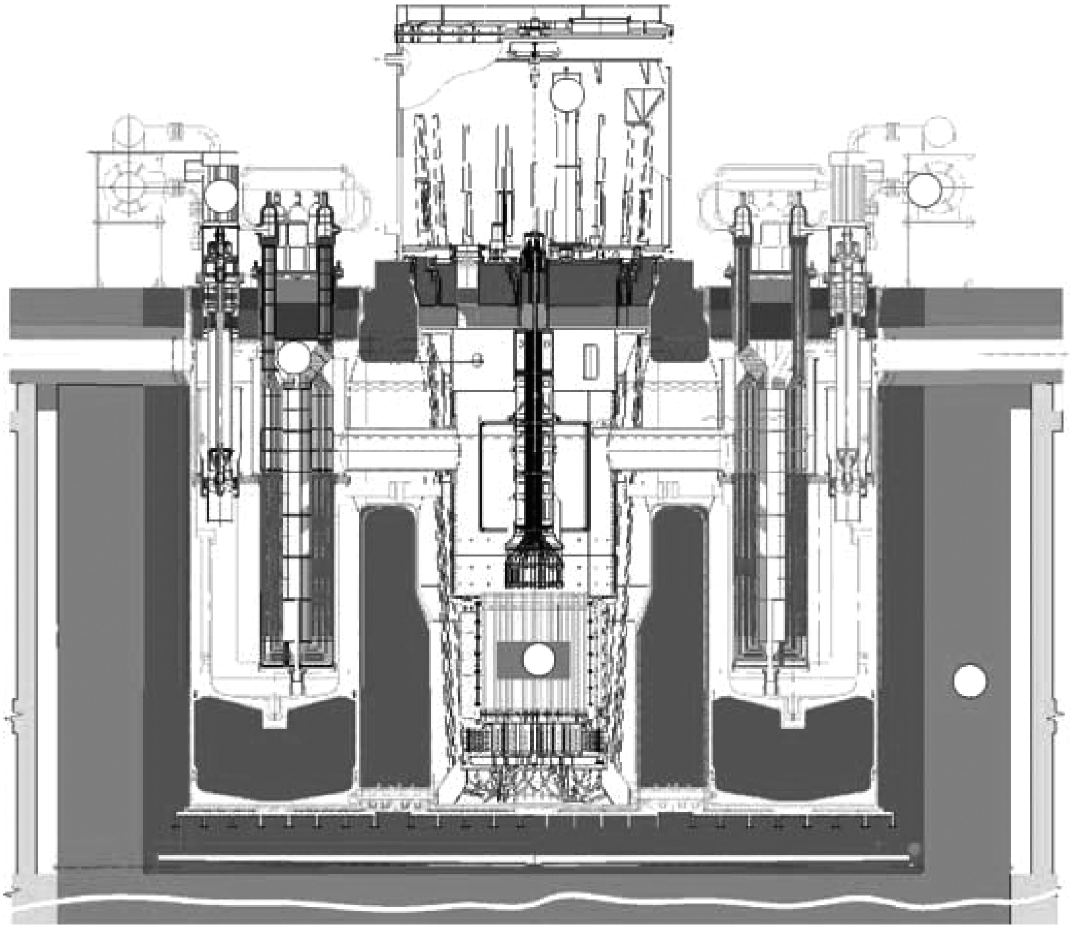


Figure 3.9.1. BREST-300 Plant layout

	BREST-300
Power	700MWt 300MWe
Core Size	2.3m D x1.1m H
Coolant	Pb
Coolant temperature	420-540°C
Coolant flow rate	4000kg/s
Inner/Outer Eff Radius	16.41/111.83cm
Fuel Rod Clad Diam	1.56cm
Max Cladding Temperature	650°C

	BREST-300
Fuel	UN-PuN
Fuel Density/Conductivity	14.3 g/cm^3 / 20Wm/K
Fuel Load	16 tons
Refueling	20 years
Structural Material	HT-9 ferritic martensitic steel
Efficiency	43%
Inlet/Outlet Steam Temperature	340-520°C
Refueling	1 year
Lifetime	60 years

BREST-300 is a fast neutron reactor cooled by lead. The fuel used in core is an high density and conductive nitride mixed fuel due to its high compatibility with lead and fuel cladding made of chromium ferrite−martensite steel.

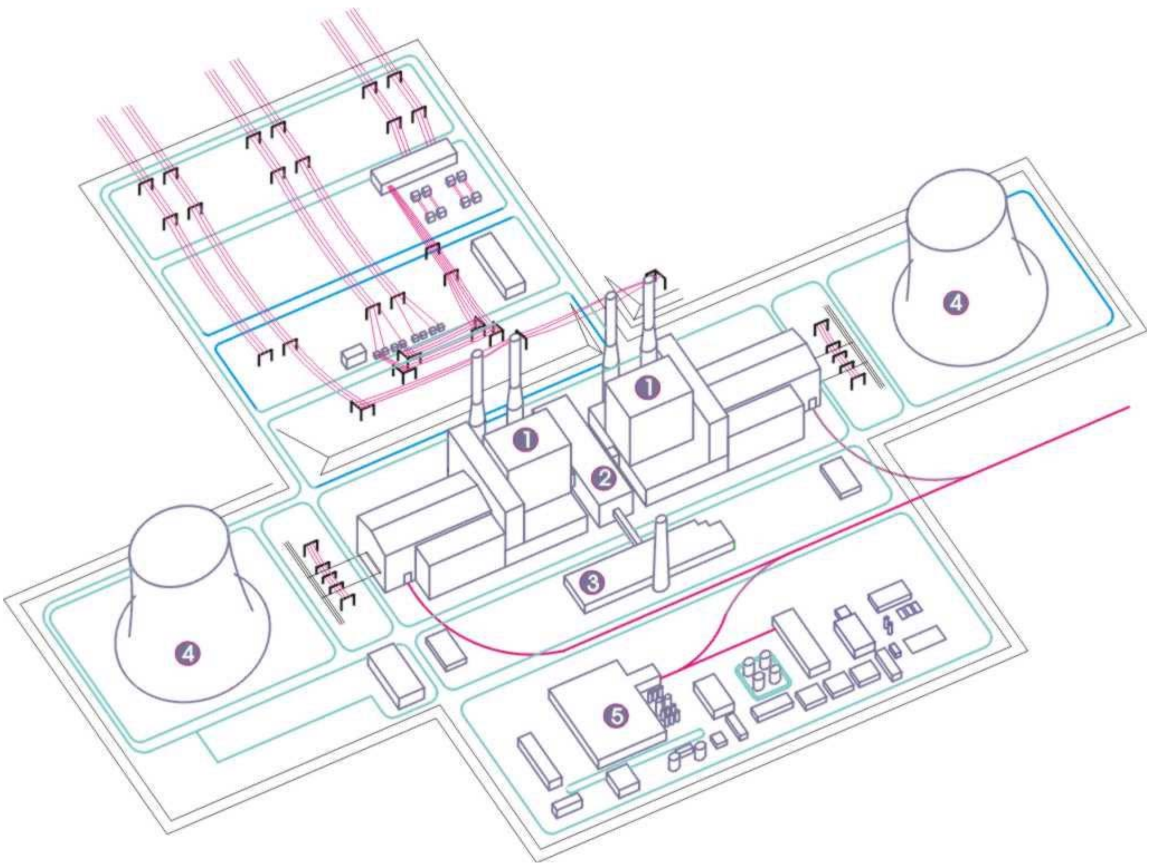


Figure3.9.2 BREST-Schematic nuclear plant layout

A slight gap between fuel and cladding is filled with lead which ensures a very good thermal fuel−coolant thermal interaction. This characteristic enhance thermal conductivity, lowering fuel temperature and consequently fission gas release. Can-free fuel assemblies ensure high heat removal rate from fuel to coolant avoiding local coolant blockage providing large cross section for coolant flow passage. Burnout is also prevented with such a fuel design. Leak-tight cans are instead installed on fuel assemblies used as reactor reflector. In fuel assemblies lines 2 3 and 4 it’s possible to put I and Tc for transmutation, but also Sr and Cs as stable heat generator.

Lead reflectors substitute standard uranium screens due to their albedo characteristics, better than those of uranium dioxide; this solution lowers neutron leakage.

High quality lead, chemical inert, used as coolant, enables to use just a simpler dual−cycle cooling system with superheated steam. Lead coolant is pumped forcedly to a 2 m height suction chamber and then the lead comes down to core support grid, flowing from the bottom upwards, heating lead till 540°C. Hot coolant enters then in the steam generator where it will be cooled to 420°C and steam will be produces in the secondary loop. Secondary water is preheated up to 340°C taking off some steam from the secondary circuit.

The emergency cooling system is in hot standby when the reactor is operated at normal conditions; to start immediately the system at design power, the temperatures of outlet circulation circuit is maintained at an optimal level.

An active or passive alarm signal may the trigger to start the emergency cooling system. Power output used by this cooling system is about the 1% of full power.

Different reactor core characteristics have been studied for the BREST-300 reactor.

Three critical assemblies had been used: BFS−61, BFS−61−1 and BFS−61−2; differences between this modules are related to side reflector. The critical assembly height was 86.7 cm, while the core radius was related to the side reflector configuration changing, ranged from 44.6 cm to 49.8 cm.

Measures of critical parameters were taken: mean cross section ratio, doppler reactivity effect, reaction rate distribution, reactivity effect in hydrogen insertion, void effect in Pb, reactivity effect in fuel melting simulation, fission neutron weights distribution, and delayed neutron effective fraction. Monte Carlo methods were used to verify and support BREST−300 reactor analysis.

2.10 Hyperion (LMFBR, PbBi cooled)

Figure 3.10.1. Hyperion core

	HYPERION
Power	70MWt 25MWe
Core Size	1.5 D X 2.5H [meters]
Coolant	PbBi
Weight	<50ton
Transportation	Barge, truck or train
Fuel Cladding	Stainless Clad
Fuel	Uranium Nitride
Enrichment	<20%
Refueling	8-10 years

The reactor, only a few meters in diameter, will be delivered on the back of a lorry to be buried underground. It must be refueled every 7 to 10 years and it's designed to last 50 years. It must be stressed that Hyperion is small enough to be transported on a ship, truck or train, and its modules are about 1.5 meters wide. Hyperion power modules are buried far underground and guarded by a security detail. Like a "nuclear battery", its modules have no moving parts and are delivered factory sealed. The core cannot be opened on site for safety and no proliferation resistance.

Due to the fact that the material inside the core would not be appropriate for proliferation purposes, proliferation issues will not arise in case of damages to the core and theft of nuclear materials. The waste produced after five years of operation is approximately the size of a softball and is a good candidate for fuel recycling.

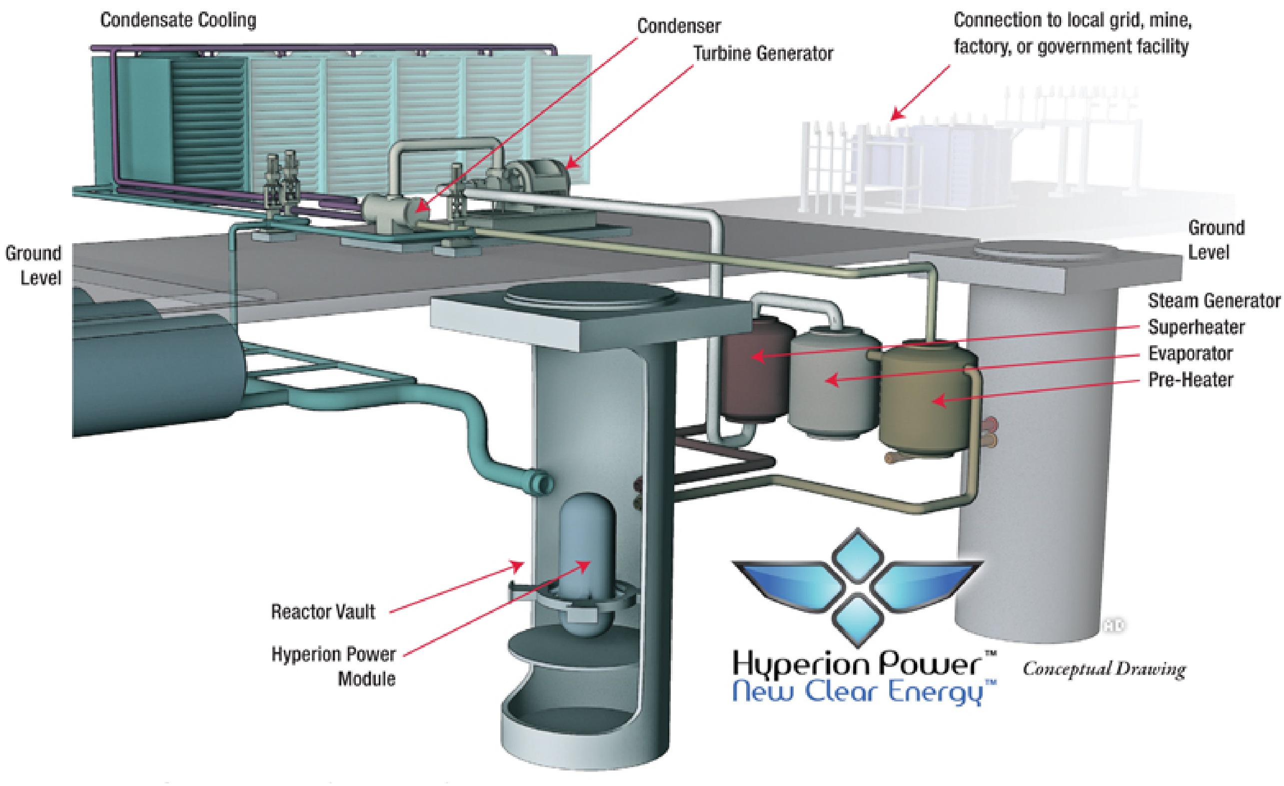


Figure 3.10.2. Nulcear plant layout

Hyperion produces only 25 MWe, just the size to provide electricity for about 20,000 average American sized homes or its industrial equivalent. Joined together, modules can produce more consistent energy for larger projects.

Absolute containment of all gases and other contaminants is ensured by multiple gas-tight chambers, in the unlikely event that a single chamber fails. Further, the module will be buried in the ground during its operational life. This will protect the module from almost all conceivable threats, natural or man-made, and make tampering extremely difficult. Additionally, active area security will be provided by the operator.

Hyperion technology could provide a 30% reduction in capital costs compared to conventional reactors (from 2,000 to 1,400 $ per kW). It's possible to reduce also operating costs also referring to conventional power plants.

The company expects to produce 4000 units in 3 years, providing 100GW of power, about 20% of America's total energy usage.

About Hyperion's project, Pete Knollmeyer, vice president for strategic planning at Savannah River Nuclear Solutions, said: "The design and licensing processes will take several years each and construction could take an additional three to four years."

2.11 SVBR 100 (LMFBR, sodium cooled)

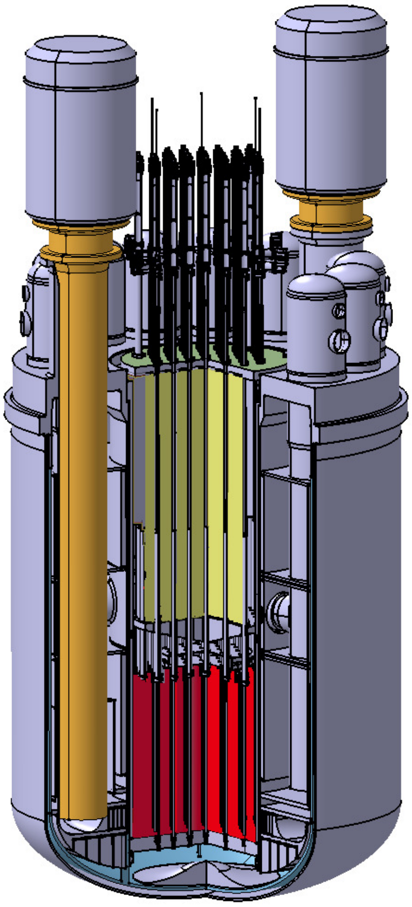


Figure 3.11.1 Schematic view of the SVBR-100 core

	SVBR 100
Power	265-280MWt 100MWe
Core Size	4.5m Diam; 7.6m Height
Weight	270tons
Core Inlet temperature	280-320°C
Core Outlet Temperature	440-482°C
Coolant	Liquid-metal (Sodium)
Average power density	160 kW/dm^3
Coolant volume	8m^3

	SVBR 100
Fuel	18 exagonal, U-Pu
Enrichment	<20%
Steam Capacity	460-580 t/h
Steam Pressure	4.7-9.5 MPa
Steam Temperature	400°C
Refueling	8 years
Reactor Life	60years
Commercialization	About 2020

The SVBR-100 is a small fast-breeder reactor with a heavy metal coolant. It’s inspired by nuclear submarines propulsion systems.

Its modular design of the SVBR-100 makes possible a factory fabrication process, to start a large scale production, enhancing quality levels and ensuring better control on the production, with lower production costs.

SVBR-100’s coolant is a lead-bismuth eutectic loaded into the reactor at the factory. After a first test, the coolant is allowed to “freeze” in the core, so that the module can be shipped with its coolant load inside. This module’s transportation can be made via railroad flat car.

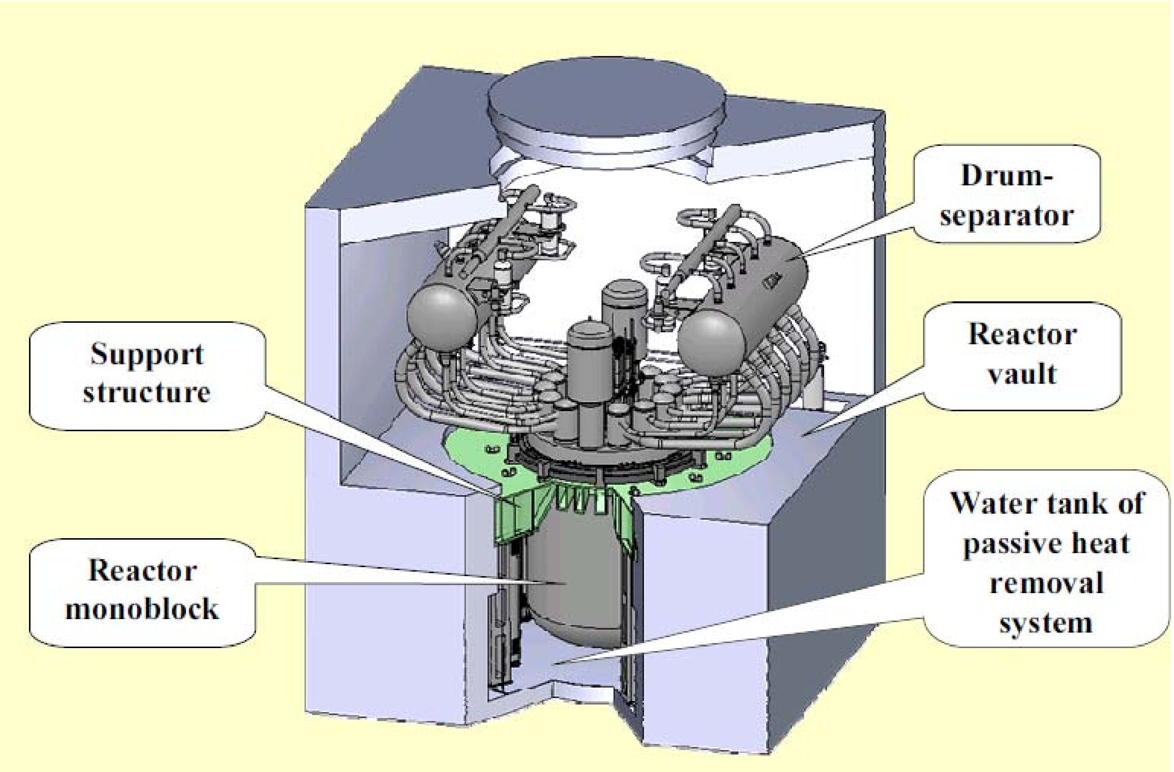


Figure 3.11.2 Schematic view of the primary circuit

280MW is a very small thermal capacity compared to standard commercial reactors. To satisfy ah higher request for energy it is possible to install together different modules of SVBR-100 to reach the needed output.

SVBR-100 can use a bootstrapping or breed/burn approach to enhance the energy extracted from uranium and make this energy 100 times greater. Inside fast breeder reactors, fertile isotope U-238 is progressively converted to fissile Pu-239 which then is burned in the fission process. U-238 is added from stocks of un-enriched uranium as the process goes on.

It has been developed also an advanced fuel cycle that, ultimately, produces wastes that need to be stocked for less than 550 years, instead of the current pressurized water reactors that generate plutonium wastes requiring storage times of between 100,000 to 500,000 years.

An high proportion of the shorter lived radioactive waste products are extracted during scheduled reactor fuel reprocessing. To contrast proliferation a part of the dangerous nuclear actinides are intentionally left in the reprocessed fuel to make it impossible to construct bombs from the plutonium component of the fuel. This fuel, after being reprocessed, is returned for further burning in other reactors.

Steam turbines installed in LNG carriers were rated at 32,400 horsepower. In the near future diesel fuelled engines will probably be expensive to operate, the SVBR-100 reactor could be readily used as propulsion for large ships at relatively low costs per nautical mile. The first reactor propelling Russian Alfa submarines was a 155 MWt lead-bismuth reactor that developing 40,000 horsepower. So it is clear that this technology can be also directed towards propulsion other than the only energy production.

2.12 4S (LMFBR, sodium cooled)

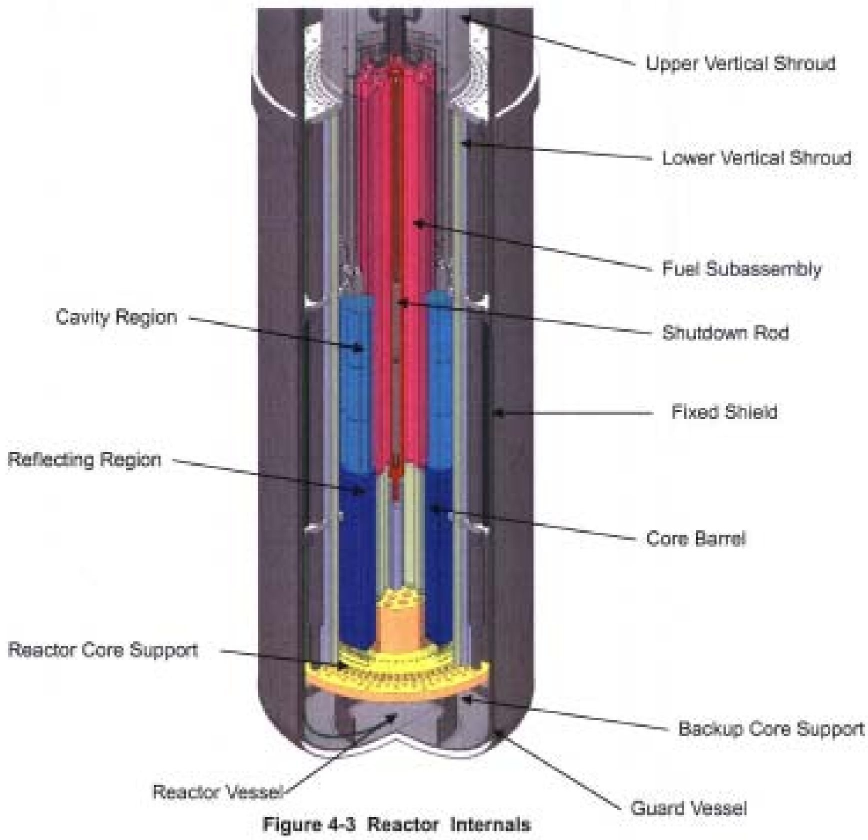


Figure 3.12.1 4S core scheme

	4S
Power	30-135MWt 10-50MWe
Core Size	2.5m Diam; 25mm Thick
Core Type	Pool
Core In-Outlet T	355-510°C
Coolant	Liquid-metal (Sodium)
Primary Pressure	<0.1MPa
Fuel	Metal fuel U-Zr alloy
N° Fuel Assemblies	18 (Hexagonal)
Maximum Fuel Cladding Temperature	650°C

	4S
Fissile Inventory	1.69 tons
Enrichment	<20%
Secondary Coolant Inlet Temperature	310°C
Secondary Coolant Outlet Temperature	485°C
Steam Temperature Inlet	210°C
Steam Temperature Outlet	453°C
Steam Pressure	10.5MPa
Refueling	30 years
Core Lifetime	30 years

The 30 years refueling time is one of the most significant features of the 4S reactor design. Whereas most current reactor designs require refueling every 18-30 months, the 30-year lifetime of the reactor makes the 4S ideal for remote areas where it might be too dangerous to store nuclear fuel for the necessary periodic refueling.

Unlike conventional reactor designs, relying on thermal neutrons, the 4S reactor relies on fast neutrons in order to sustain a fission chain reaction.

Fast neutrons, having such high energies, leak from the core, affecting neutron economy, instead of being absorbed by the nuclear fuel. To stop these neutrons from escaping the reactor core, a reflector made of liquid sodium is used to redirect the neutrons back into the core enhancing neutron economy.

Hitachi submitted documents to the Nuclear Regulatory Commission showing the kind of fuel that would be used in the 4S reactor: metallic alloy made of 10 % zirconium and 90% uranium. The uranium will be enriched to between 17%-19% with uranium-235.

1.1 millimeter layer of HT9 steel surrounds the entire fuel pin, representing the cladding. To enhance heat transfer, a thin layer of sodium is placed around the fuel. The entire fuel pin is 5 meters long, with the fuel occupying 2.5 meters. The 2.5 meters of empty space in the upper part is necessary do store gases released during the fission process. This large volume is necessary because of the 30-year expected lifetime of the fuel pin.

The heat generated inside the reactor passes to secondary sodium thanks to the IHX located at the upper region in the reactor vessel. One EM pump unit makes sodium circulate in this secondary loop. Then the heat is transferred to the steam system through heat transfer tubes in the SG.

The SG is provided with heat transfer tubes with double wall. Between the inner and outer tube, wire meshes are inserted, filled with helium, working as a detection system for a one side tube failure.

4S safety concepts emphasize on simplicity achieved by strong reliability of passive, largely used, and inherent safety features as the greater part of the defense in depth strategy. The principal goal of the 4S safety concept is to get rid of the evacuation as an emergency measure. Confinement of radioactive material, prevention and mitigation are the keyword of 4S security philosophy during accidents scenarios. During a severe accidents 4S systems can grant a high security level preventing loss of coolant, loss of flow, avoiding transient overpower and the water-sodium reaction.

For heat removal from a shutdown reactor, two independent passive systems are provided, which are Reactor Vessel Auxiliary Cooling System and Intermediate Reactor Auxiliary Cooling System.

The RVACS is completely passive, removing shutdown heat from the vessel surfaces using air moved by natural circulation. No vane, valves or damper are present in the flow path; so, the RVACS is always in operation.

Secondary sodium is used in the IRACS to remove shutdown heat.

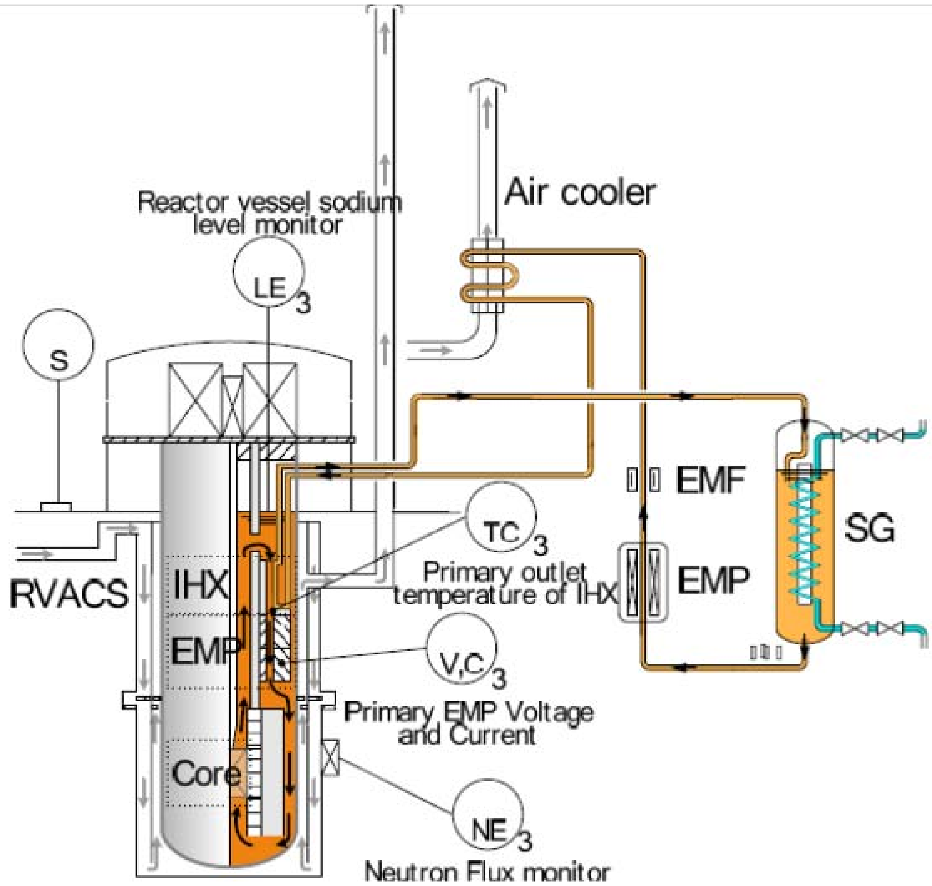


Figure 3.12.2 4S Safety systems scheme

The latest developments at NRC suggest that currently the biggest obstacle for 4S reactor is legal and related to requirements rather than technical or safety related. The NRC is aware of this issue, and set up a number of public reviews to provide a new pattern that would make it financially feasible to build and operate SMRs.

2.13 Candle (LMFBR, sodium cooled)

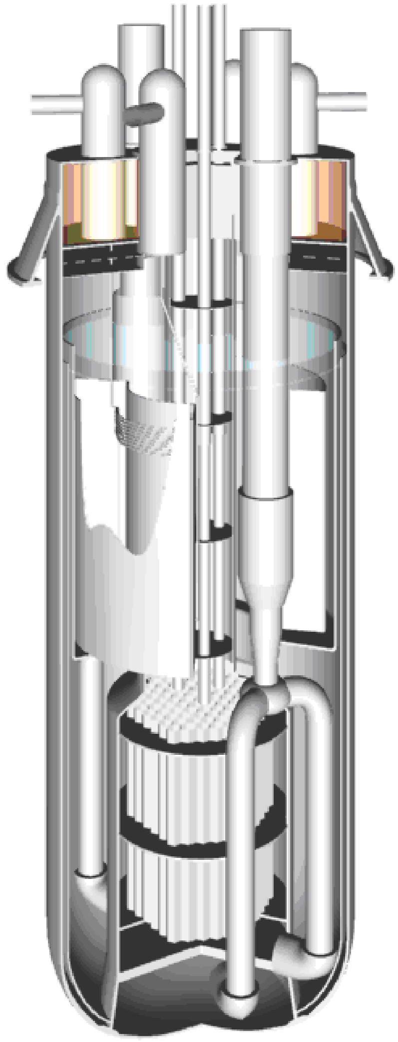


Figure 3.13.1 Candle reactor core

	Candle
Power	30-135MWt 10-50MWe
Core Size	2.5m Diam; 25mm Thick
Core Type	Pool
Core In-Outlet T	355-510°C
Coolant	Liquid-metal (Sodium)
Primary Pressure	<0.1MPa
Fuel	Metal fuel U-Zr alloy
N° Fuel Assemblies	18 (Hexagonal)

	Candle
Maximum Fuel Cladding Temperature	650°C
Fissile Inventory	1.69tons
Enrichment	<20%
Secondary Coolant T	310/485
Steam Temperature	210/453°C
Steam Pressure	10.5MPa
Refueling	30 years
Core Lifetime	30 years

Technological Institute of Tokyo is studying the possibility of creating lead-bismuth reactors that use CANDLE burnup conception. CANDLE is a Constant Axial shape of Neutron flux, nuclide number densities and power shape During Life of Energy producing reactor.

CANDLE conception provides constant distribution of isotope concentrations, neutron fluxes and power shape during the reactor life, and they shift with the constant speed in axial direction. There is no need to take care of its compensation during the burnup process because of the fact that reactivity excess doesn’t change during core life.

CANDLE active zone is divided in three zones in axial direction: burned out subzone, burning zone (where the main part of energy is produced), zone of fresh fuel.

Fuel (enriched uranium or plutonium) should be loaded only once at he beginning in the CANDLE system, and it will be needed only to create the burning zone. About fresh fuel zone, it will contain natural or depleted uranium or thorium.

During reactor life, while fissile materials is consumed in the burning zone, on the other hand side is accumulated in the fresh fuel zone. After the exploitation of the first reactor burning zone it can be used to load a second reactor.

CANDLE most clear advantage is called super deep burnup (>40%), and it’s a Japanese conception and project. Due to this such a high efficiency coefficient, using natural uranium, spent fuel proceeding and closing of the nuclear cycle can be not so advisable. This eliminates one of the potential threats to the non-proliferation system.

Previous CANDLE conception was projected to be used for big fast and heat reactors; this time Japanese specialists pay a lot of attention to the lead-bismuth reactors to be used in plant of smaller size. But it’s not so easy to realize the CANDLE strategy in such circumstances, because of the big radial neutron leakage. However excellent reflector characteristics of lead-bismuth have played an important role in this particular aspect of the project, precisely, enhancing neutron economy inside the reactor.

2.14 HTR-PM (Gas cooled)

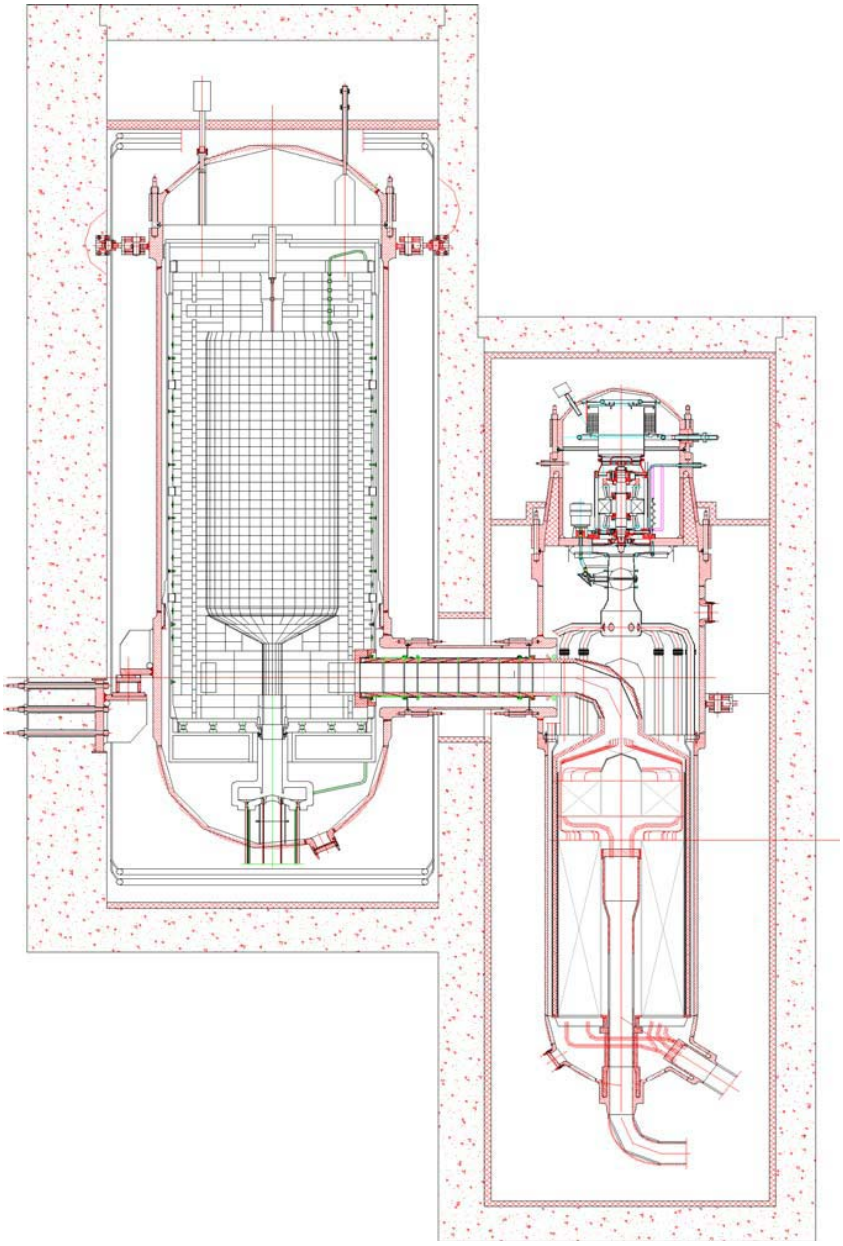


Figure 3.14.1 HTR-PM reactor

	HTR-PM
Power	458MWt 195MWe
Core Size	6.7m Diam; 24m Height;
	146-250mm Thick
Core In-Outlet T	250-750°C

HTR-PM	
Coolant	Helium 7.0MPa
Primary Coolant Flow	96kg/s
Fuel	UO2
Enrichment	9.08%
Steam Flow Rate	99.4kg/s
Max Steam Temperature	570°C
Min Steam Temperature	205°C
Cooling Water Tempersature	16°C
Steam Pressure	13.24MPa
Condenser Water Recirulation Rate	7700kg/s
Refueling	2 years
Plant Lifetime	40years

The HTR-PM power plants are designed to be a series of commercial plants, starting from a demonstration plant. The investment for the HTR-PM project will come from the market, i.e. from the future utility company. Other financing sources are needed for the development of some new technology in the stage of demonstration plant, for example financial support from government, and the Chinese central government agrees to support the technology development activities. HTR-PM project can be divided into four categories: technical design, marketing, project, and organization. About technical design, the main work is to find and optimize an HTR-PM standard design based on the enveloping or reference site conditions.

Helium is used as coolant in HTGR reactors with graphite as moderator as well as structural material. A single-zone core design was adopted, in which the spherical fuel elements are placed. Active reactor core has 3.0 meters of diameter and effective height of 11.0 meters generating an effective core volume of 77.8 m^3. 420,000 spherical fuel elements are contained during standard operations inside the core.

The reflectors include top, side and bottom graphite reflectors. 30 graphite blocks compose reflector in the circumferential and corresponding numbers of channels are designed for reactor shutdown systems and for helium flow. To facilitate the pebble flow, the bottom reflector has the shape of a cone. The hot helium of different temperatures is mixed in the bottom area of the reflector and then directed to the hot gas duct where the hot helium goes to the Steam generator.

The primary helium flows at 7.0MPa with a flow rate of is about 99kg/s from the bottom to the top of the core. Once reached the top reflector level it reverses the flow direction and flow into the pebble bed in a downward flow pattern. In this passages helium reaches an average temperature of

750ºC and then flows to the Steam generator. A connecting vessel links the reactor core and steam generator vessels. Inside the connecting vessel, the hot gas duct is designed.

Now China is searching for new energy sources to sustain its rapid growth; nuclear energy is one of the most suitable. Nowadays a viable and safe electricity supply is the main object for nuclear energy, but, in the future, hydrogen production and desalination may become important. Standard PWRs are the choice for the Chinese nuclear power plants, but MHTGR is very attractive because of its safety, its high efficiency and its possibility for direct hydrogen production.

Figure 3.14.2 Spherical fuel elements

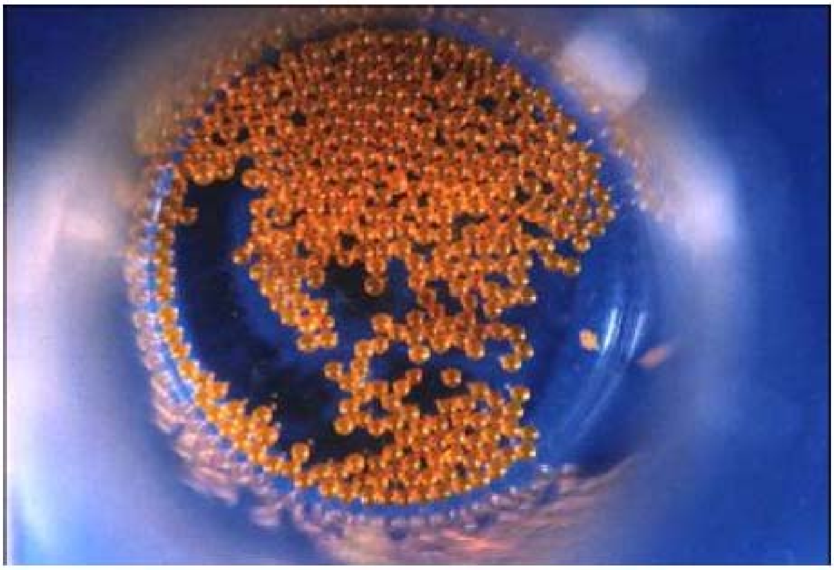


Figure 3.14.3 Fuel uranium kernel

So, the HTR-PM can meet market requirement in short term and in long term. A roadmap for the demonstration plant is set up, while a long-term one of the HTR-PM is in the preliminary stage.

One branch of HTR research is the reactor itself, standard design, experimental verification, safety review. Another line of project is for the nuclear fuel plant. The long-term development is about development of new technologies, including helium turbine, hydrogen production technology, gas-cooled fast reactor technology and very high-temperature gas-cooled reactor technology.

A development team, completely representing the industrial Chinese background, is set up to enhance the HTR-PM project, including research, design, construction, manufacturing and operation.

Another aspect of this team is the institution of a future architecture and engineering (AE) company for the HTR-PM project, responsible for the engineering design, components supply and all issues and problems concerning the construction.

A joint venture company will be created to develop this, among a Chinese electricity company, namely China Huaneng Group, and a nuclear industry company, the Chinese Nuclear Engineering and Construction Corporation (CNECC), Tsinghua University and other local investors located near the final site of the HTR-PM power plant.

2.15 Flexblue (PWR underwater)

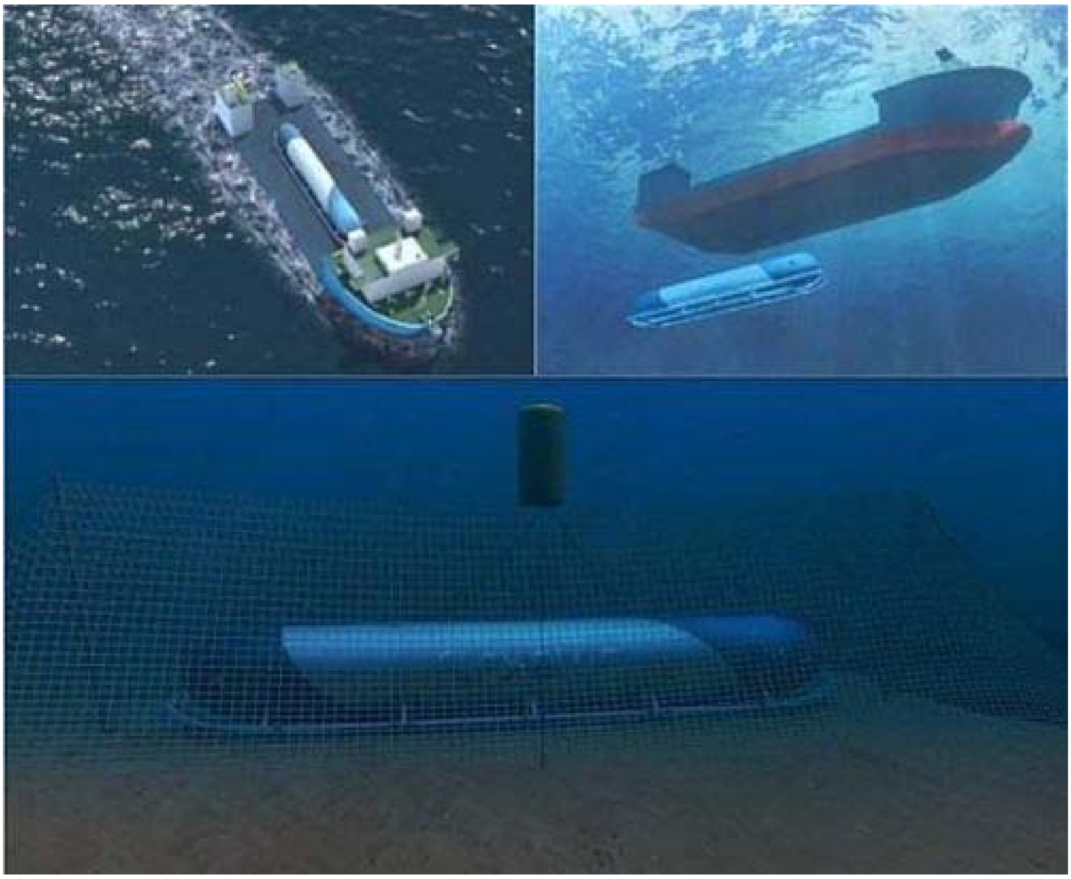


Figure 3.15.1 Deposition of Flexblue reactor

	Flexblue
Power	50-250MWe
Size	12-15m Diam; 100m Length
Fuel	UO2 enriched
First prototype	2013
Weight	12000tons

Flexblue is a small subsea nuclear power plant with an output rating of 50 to 250 MWe.

The power plant is composed of a nuclear reactor, PWR, and a steam turbine-alternator. Submarine power cables will transport electricity from the Flexblue plant to the coast.

Flexblue has to be located on an extremely stable seafloor at a depth of 60 to 100 metres a few kilometres off shore. Ballast tanks will be used to raise or deposit the plant during installation and maintenance and refueling. The reactor will be transported on site using boats, similar to the common cargo barge.

A Flexblue plant is designed to meet the electricity needs of regions with a population of 100,000 to 1,000,000, living standards and the of local industries. A standard, on-land nuclear power plant, produces about 1,000 megawatts, a huge amount of power that can serve large cities or areas with a massive presence of industries. But there is demand, for smaller, much cheaper reactors that could serve areas where infrastructure is not as advanced, or where the grid is not able to sustain large injection of power. The reactor can be placed underwater without having to build extensive support structures.

The Flexblue is composed by a cylindrical hull of around 100 metres in length and 12 to 15 metres in diameter for a total weight of 12,000 tonnes. Each hull and power plant could be moved using a purpose-built vessel. Reactor control will be made a remote control, because of its undersea conception. Each reactor will have a on board control room to manage critical operation or startup; it will be accessible anytime using small submersible.

Flexblue is based on proven technologies: DCNS has 40 years’ experience in nuclear engineering and know how in submarine design and production, even related to nuclear propulsion systems, that inspired this reactor design. Flexblue’s main goals are: performance, reliability, safety, durability and environmental protection.

Figure 3.15.2 Flexblue module underwater

EU civil protection regulation stated that the EPR must resist air attacks. Flexblue won’t be suffering air attacks, so it has been decided to make it resistant to the impact of an explosive torpedo as this seems the most realistic threat. To enhance safety it’s been decided to protect these installation with at least a military ship as patrol in the interested area.

It must be stressed that operational costs are really low compared to standard nuclear power plant. It has been estimated a cost of some hundred million Euros, instead of 5 billion of the EPR.

This reactor design joins technology from actual land reactors to those developed for nuclear-powered submarines. A first prototype could be presented in 2013, and, if successful, commercial production would start in 2016.

2.16 SM-MSR (MSR, epithermal)

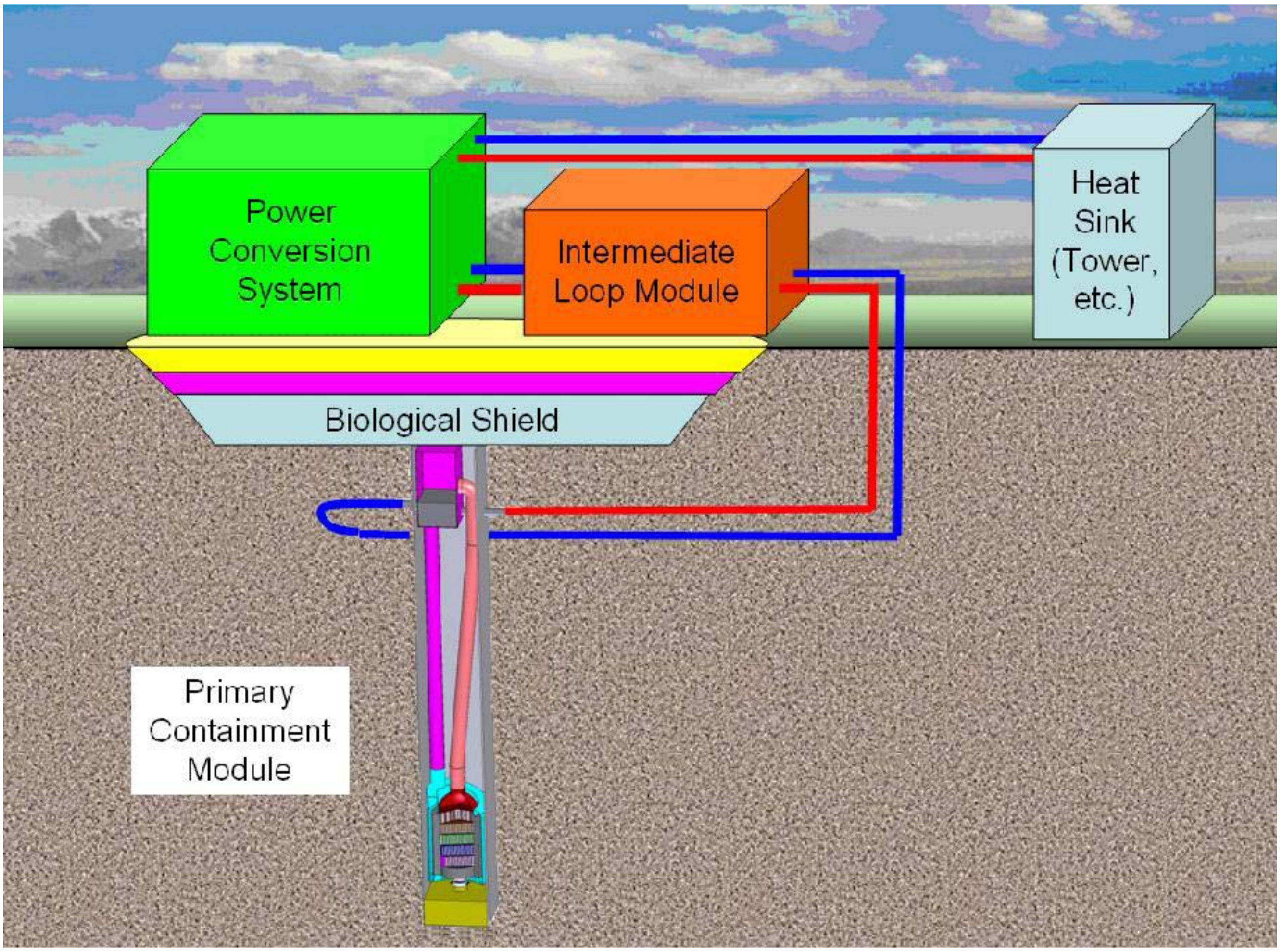


Figure 3.16.1 Schematic plant layout

	SM-MSR
Power	240MWt 100MWe
Core Size	2.35m Diam; 4.8m Height; 2cm thick
Efficiency	42.7
Primary footprint	23.8m^2
Fuel	2LiF-BeF2UF4
Fuel-coolant flow rate	538kg/s
Compressor/turbine inlet	300-960K
Helium Cycle	81kg/s, TM=538°C,Tm=205°C
Compression ratio	5.52

	SM-MSR
Weight	410tons
Design life cycle	30 years
Refueling	online

SM-MSR is a modular molten salt reactor. The primary containment is placed below grade to take advantage of earth for shielding and intruder prevention. It contains the core, made of a circular cylinder of graphite that moderates neutron, till epithermal energies. There are also vertical lengths of the hot and cold leg piping contained in the primary module.

An intermediate loop heat exchanger ensures the connection between primary and secondary circuit.

This module is 25 meters tall to allow sufficient height to support natural circulation of the primary coolant, in order to enhance life cycle, avoiding potential circulating pump failures. At the top of containment is located an intermediate heat exchanger which provides heat transfer as well as: salt addition, thermal expansion of the primary fluid, fission product gas removal and addition of fuel to the liquid fuel salt.

There's also a large biological shield which allows the region of space immediately above the core to be a 40 hour per week occupancy zone, as well as providing mitigation against human intrusion into the system. In order to isolate environment from radioactive fuel in form of salt, a secondary loop is provided.

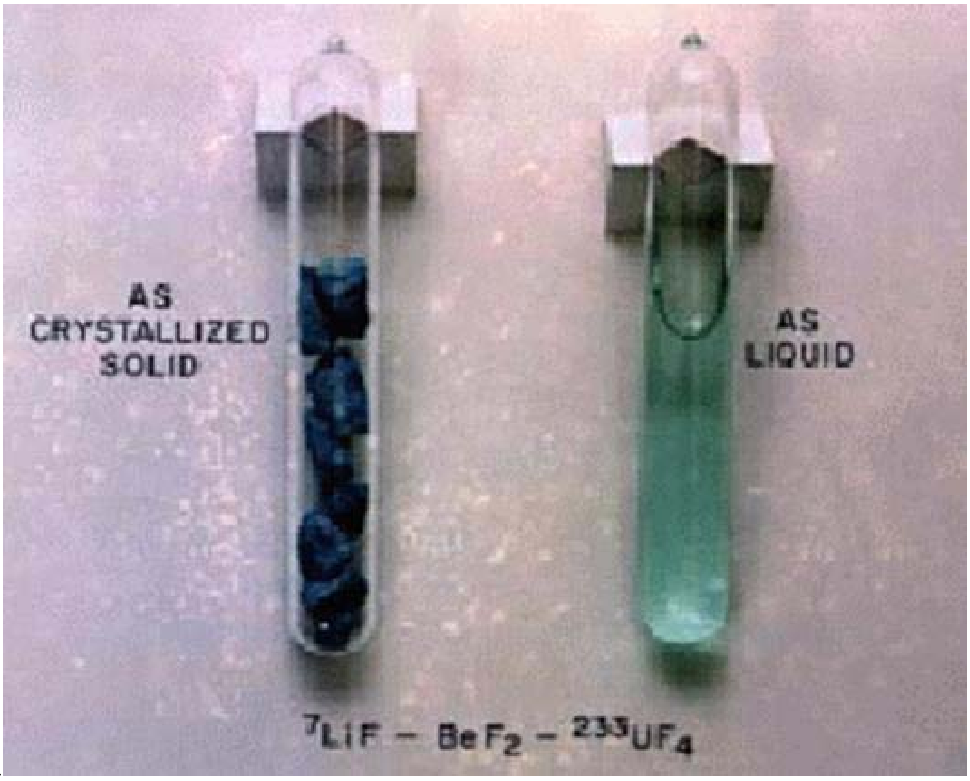


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Figure 3.16.2 Frozen and Liquid Fluoride Salts of Lithium, Beryllium, and Uranium

To transfer thermal energy to the power system a circulating pump is needed, and heat is transferred to helium. The heat exchanger is made by three nearly equal sized sections extracting heat from the secondary salt and transfer it to the helium circulating in the power conversion system.

A Brayton power cycle ensures the conversion o venery, granting the electrical output. This cycle is designed with double re-heat process to maximize efficiency. In addiction there is a heat sink mechanism which will be site specific.

This kind of reactor in designed for district heating and water desalinization as a part f the GNEP mission under which it will be deployed; the heat sink system will incorporate these features as needed.

The primary system is composed by the primary containment structure, the moderator chamber, the primary loop piping and components, the fission-product gas collection system, the core safety dump tanks, the intermediate heat exchanger, the upper biological shield, and components for fuel salt addition and removal. Obviously there are also some detection system components for monitoring as well as other auxiliary systems necessary to operations of the SMMSR.

The primary containment structure is a parallelepiped with dimensions of 4.88 square meters of base and by 25 meters height. The structure is made of a carbon-steel alloy with a Hastelloy-N plated interior surface. Hastelloy-N alloy was invented at Oak Ridge National Labs and it is a nickel-base alloy that that can be used as a container for molten fluoride salts. It is resistant to oxidation due to hot fluoride salts (until 1150K) and air. So a great corrosion resistance is reached thanks to this plating especially in case of fuel or secondary salt leakage. The alloy and plating are 1 centimeter thick, but the necessary structural rigidity is reached using a skeleton of beams and supports.

The lower parts are provisioned with an emergency decay heat removal system in case it's necessary to use the emergency core dump tank. The structure has a removable upper shield near which there is one of the human ingress/egress point.

Human contact with the salt is inhibited due to fuel activation after the system has been operated at full power for a short time. This feature supports the GNEP objective, enhancing proliferation resistance in a passive way. On the other hand side, the primary system must be designed permitting emergency operation to be performed without needing direct access to the primary components. The SMMSR is designed to achieve that goal.

One of the operational most important feature is that reactor is semi-autonomous, so that no operator action is needed during standard and transient operations. Natural circulation simplifies operation as well, enhancing long-term operational stability. No control devices such as rods, soluble poisons, or moveable reflectors are needed.

The reactor regulates power and temperature autonomously thanks to the properties of liquid fuels. There's no need for pressurization systems, as the fuel remains liquid at temperatures up to 1700K and atmospheric pressure. A passive freeze valve is designed to put the system into a fail-safe state upon initiation of an excessive over-power or over-temperature accident.

The size-limits of the system are based on rules regulating transportation over land using a heavy transporter. The containment structures outer dimension are designed to match the special permit parameters of transportation.