

# **INHERENTLY SAFE REACTORS DESIGNS**

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## **1. INTRODUCTION**

The electrical power production industry worldwide is faced with a need for new generating capacity. In the USA, the need is for an increase of 1/3 of the existing capacity. The utilities in the USA are currently building gas turbine and combined cycle plants to meet peaking and intermediate needs. Of the forecast need of 150 GWe for the period 2000-2007, 65 Gwe are for peaking and intermediate capacity, and 85 Gwe are needed for uncommitted baseload capacity.

A new generation of nuclear power plants possessing passive inherently safe safety features is replacing the older designs, and are being constructed or in the conceptual design stage. The inherent safety features are a response to the understanding that human factors significantly contributed to the Three Mile Island and the Chernobyl reactor accidents. These plants would contribute to the reduction of the environmental costs of emitting nitrogen oxides (NO<sub>x</sub>), sulphur dioxide (SO<sub>2</sub>), and carbon dioxide (CO<sub>2</sub>) from fossil fuels. As these emissions become economically monetized, nuclear electricity is expected to offer both an environmental and a cost advantages.

The safety improvements in the new generation of plants are quantified by the Core Damage Frequency (CDF), resulting from Probabilistic Risk Analyses (PRAs). The design goals for CDFs for the new generation of nuclear power plants are set at  $10^{-7}$  per reactor.year. This is compared to the value of  $10^{-4}$  for existing power plants. For existing plants the frequency of 1/10,000 translates into one severe core accident per 200 years for a world with 500 reactors ( $500 \times 200 = 10,000$ ). For the new generation of reactors, even with a world with more than double the number of reactors as today (434 plants as of the year 2001) at 1,000 plants, the frequency of 1/10,000,000 translates into one severe core accident in 10,000 years ( $1,000 \times 10,000 = 10,000,000$ ).

The new design also satisfy the Nuclear Regulatory Commission's (NRC), Severe Accident criteria, which require that these new plant possess the capability to protect the public from radiation releases, even in the improbable situation of a severe accident.

## **2. ADVANCED LIGHT WATER REACTOR (ALWR)**

The USA Department of Energy (DOE) has launched cost share programs with the USA electric utilities, and vendors to develop advanced LWR reactor designs. On the basis that reactor safety is a global concern, government agencies and companies from about 20 other countries participate in the program. The development of a new generation of reactors has moved in two directions:

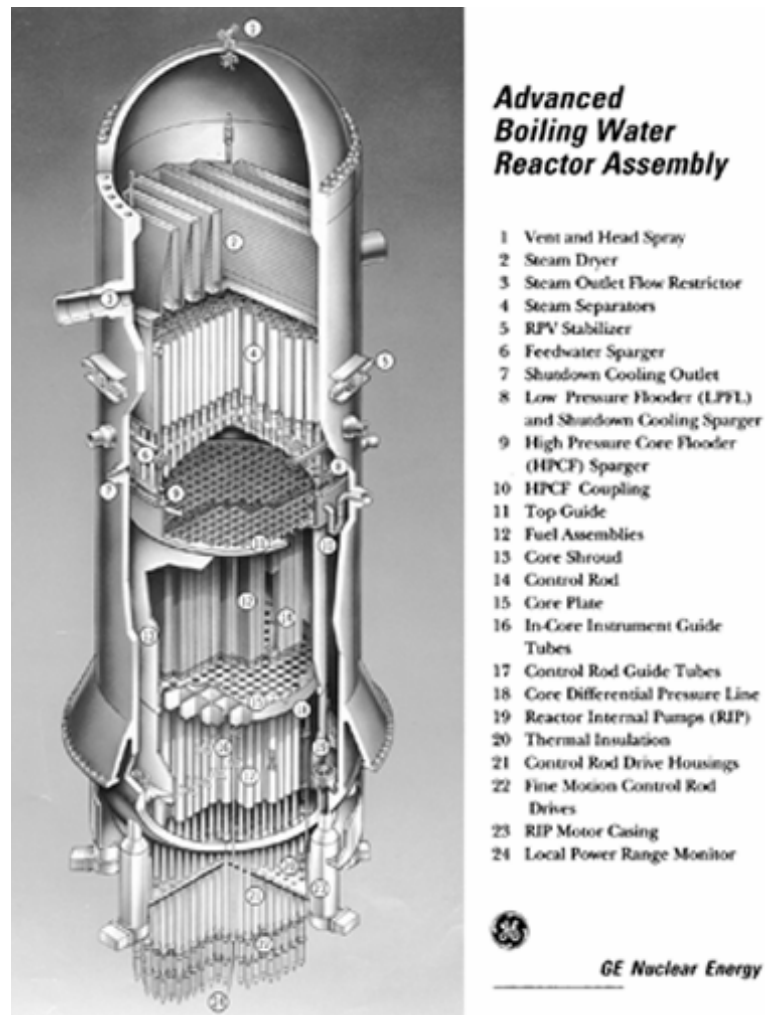
1. Evolutionary plant designs.
2. Passive plant designs

### 3. EVOLUTIONARY PLANT DESIGNS

In these designs, evolutionary changes in the design are incorporated in the sense that the designs started from existing plant designs and built into them improved safety margins. Modern safety systems, advanced instrumentation and controls, and simplified operations and maintenance are incorporated in these designs.

Two large designs have been developed with an electrical capacity of 1350 Mwe and have received final design approval from the NRC.

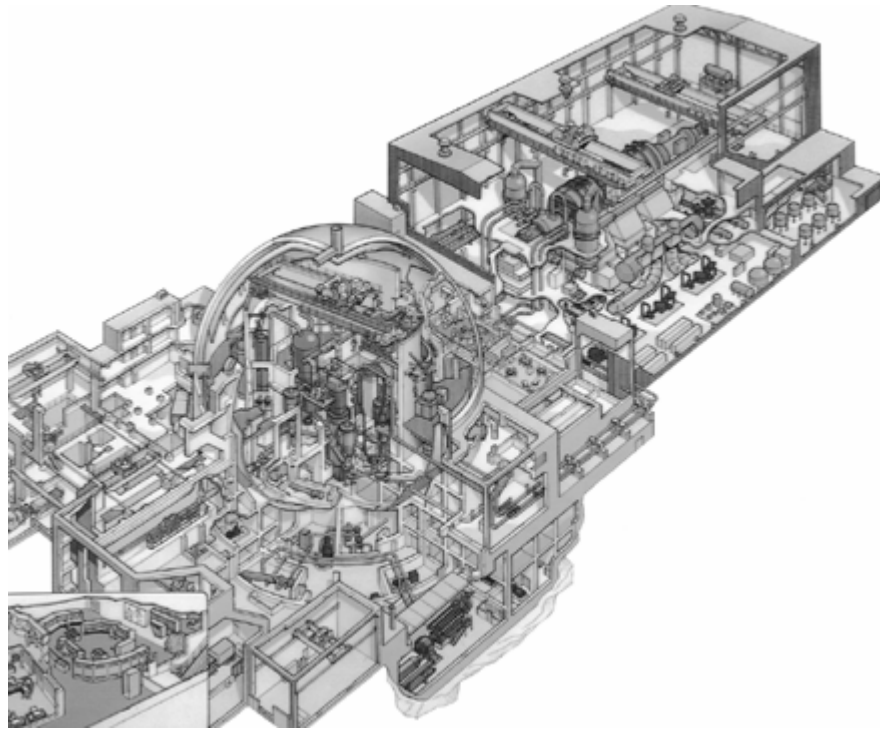
The first one shown in Fig. 1 is the Advanced Boiling Water Reactor ABWR designed and built by the General Electric Company. Plants are being in Japan following this concept.



**Fig 1 The ABWR pressure vessel design.**

The second concept that has received final design certification from the NRC is the ABB-CE System 80+ which evolved from the Combustion Engineering (CE) Pressurized Water Reactor System 80 plants, shown in Fig. 2.

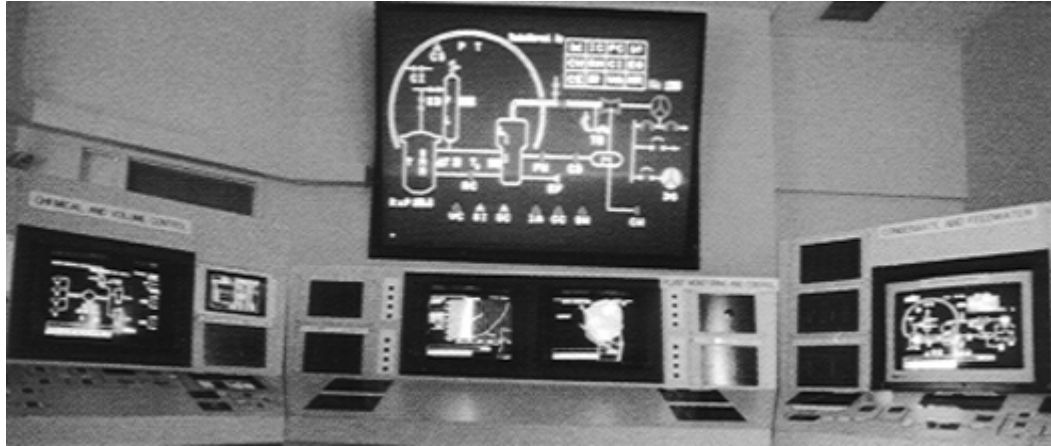
The System 80+ plus plants are being built in Korea, and are the basis of the first standardized system used in the three units at the Palo Verde Nuclear Generating plant in Arizona, shown in Fig 3.



**Fig 2 The plant layout of the ABB-CE System 80+ evolutionary design.**



**Fig 3. The three standardized System 80+ units at Palo Verde Nuclear Station in Arizona, USA.**



**Fig 4 The Control Room of the System 80+ features human factors engineering features.**

### **3.1 THE ADVANCED BOILING WATER REACTOR (ABWR)**

The ABWR reactor pressure vessel is 21 meters high and 7.1 meters in diameter, and is designed for a 60 years lifetime. The vessel is mostly made of a single forging including the 4 vessel rings from the core beltline to the bottom head.

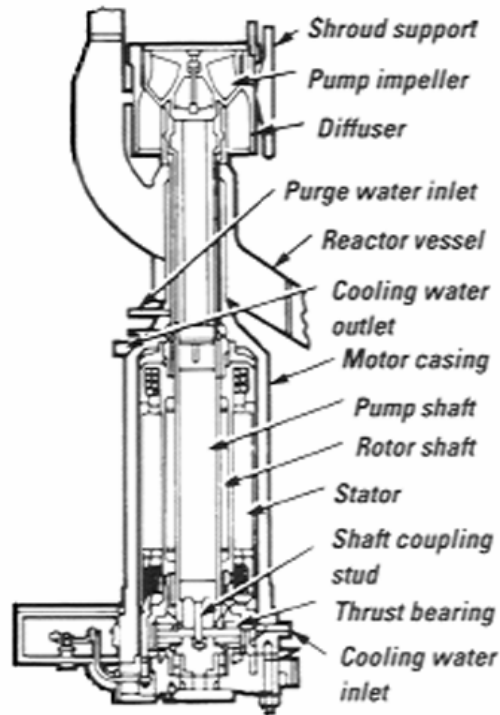
The external recirculation loops in older designs using jet pumps, and subject to leakage failures have been eliminated. The 10 canned rotor Reactor Internal Pumps (RIPs) shown in Fig. 5 replaced them. As a result the vessel has no nozzles greater than 2 inches in diameter anywhere below the top of the core. Over 50 percent of the welds and all the piping and pipe supports in the primary system have been eliminated. This eliminates the largest source of occupational exposure in the BWR earlier designs. These pumps are improved versions of those used in Europe with significant operational experience. The RIP pumps are continuously purged with clean water to keep the crud from settling into them from the vessel, reducing the radiation levels around them. With them compact containment designs are possible, greater excess flow capacity, absence of nozzles below the core and higher reliability are possible.

Selection of materials in the ABWR eliminates cobalt from the design, since cobalt could activate in the crud into Co60, a strong gamma emitter. The condensers are made of titanium metal and valve seat use cobalt free alloys. The steel used in the primary system is low carbon steel alloys, which are nuclear grade materials. These alloys mitigate a source of worry in earlier designs concerning Intergranular Stress Corrosion Cracking (IGSCC).

Fine Motion Control Drives (FMCRDs) are being used. They perform with an electric stepping motor moving the drive in 0.75 inch increments. The control rods can be scrammed into the reactor core using the hydraulic system, but as a backup, the stepping motor can be used to scaram them. Earlier Locking Piston Drives had 3 inches increments. A clean water purge is provided for the FMCRDs to keep radiation levels to low values.

The ABWR Control and Instrumentation (C&I) system has four separate divisions of system logic and control, including four separate redundant multiplexing networks. These systems are made of digital and fiber optic technologies. Microprocessors in each system receive incoming sensor information and generate output control signals. The controllers are fault tolerant continually generating signals to simulate input data and compare them against the

desired outcome. Controllers for both sensors and equipment are on cards. If the controller detects a problem, the malfunctioning cards can be replaced. Multiplexing and fiber optics have eliminated 0.4 million meters of cable and 3780 m<sup>3</sup> of cable trays compared with the older designs. The four redundant divisions in the control system use a 2 out of 4 logic. On line repair of one of the systems while the others are still functional is possible. Solid state technology allows the control system to be enclosed in two cabinets, compared with 18 cabinets for the older design.



**Fig 5 Reactor Internal Pump for the ABWR.**

The control room is designed according to the wishes of plant operators who wanted to operate the plant from a single console with touch screen panels and Cathode Ray Displays (CRTs) displays, and not have to run around the control room looking for side panels. A system, its subsystems, and components can be called in a series of exploding displays. Touching the screen then can operate these. The entire system can be operated by a system master command. For instance, the Reactor Heat Removal (RHR) system has six operating modes, each of which the operator with a single touch, can invoke, and the computer will configure the system by aligning valves or turning on pumps.

None of the major facilities are shared with other foreseeing the trend toward standardization. The plant is designed to fit any site in the world from the perspective of seismic potential.

The containment building is a Reinforced Concrete Containment Vessel (RCCV) with a leak tight steel lining. The reactor building surrounds the containment, and doubles as a secondary containment. A negative pressure differential is maintained in the reactor building and directs any radioactive release to a gas treatment system.

Large modules are prefabricated in a factory and assembled on site. The entire control room, weighing 400 tonnes, the steel lining of the RCCV, and the turbine generator pedestal are examples of these prefabricated modules.

Three independent and redundant safety systems are available in the ABWR. The systems are mechanically and electrically separated. Each division has redundant sources of AC power and its own dedicated emergency diesel generator. Each division is located in a different quadrant of the reactor building, and they are separated with firewalls. Fires, floods or loss of power if affecting one division, will not affect the others. Each division has a low and a high-pressure system. Each system has its own dedicated heat exchanger to control core cooling and remove the decay heat.

One of the high-pressure safety systems are designed to keep the core covered at all times coolant injection systems, the Reactor Core Isolation (RCIC) system is powered by reactor steam. This provides a source of cooling in case of the Station Blackout (SB) accident, where it is assumed that both the local and outside sources of electrical power are not available to control and operate the plant.

The safety systems have the capability of keeping the core covered at all times. In the event of a Loss Of Coolant Accident (LOCA), operator action is not required for 72 hours, since the plant response has been fully automated. The operational transients leading to a plant shutdown have been reduced to a frequency of once per year.

### **3.2 THE SYSTEM 80+ PRESSURIZED WATER REACTOR**

The system 80 design is characterized by being the first standardized system built in the USA. It obtained a Final Design Approval (FDA) by the NRC in 1994. In the past, nuclear power plants have been designed, built and constructed on a plant by plant custom basis. This has led to costly delays, and in some cases project abandonment, because safety license could not be obtained without a lengthy review after plant completion.

With standardization of the design:

1. Plant design is pre-approved , eliminating licensing delays during and after construction.
2. Standardization allows for plant completion in about half the time for a custom designed plant.
3. Final costs can be significantly cut through lower interest charges paid during the construction period.

An example of a standardized station is the three units at Palo Verde, arizona, USA, shown in Figs. 2 and 3. Several units are under construction in Korea and Taiwan.

Another unique feature of the System 80+ design is the incorporation of an advanced control room design designated as the Nuplex 80+ Advanced Control Complex. As shown in Fig. 4, a 6x8 feet overview display lets all persons in the control room keep track of key status indicators simultaneously, minimizing the risk of human error.

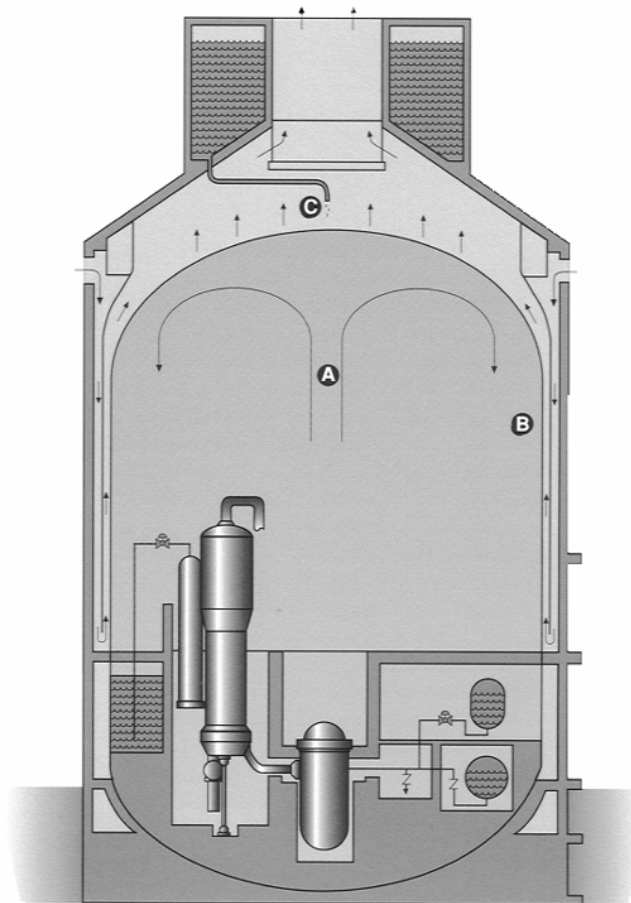
## **4. PASSIVE PLANT DESIGNS**

In these designs, the emphasis is on passive safety features that would negate the need for active safety systems which are prone to human error as occurred in the Three Mile Island and the Chernobyl accidents.

Two plant designs stand out in this category: The AP600 design by the Westinghouse company, and the Simplified Boiling Water Reactor by the General Electric Company.

#### 4.1 THE AP600 ADVANCED PASSIVE PRESSURIZED WATER REACTOR

A cooperative program between the Electric Power Research Institute (EPRI) and the US DOE, resulted in this concept. The Advanced Reactor Corporation (ARC), representing 16 US utilities chose it as the lead passive plant design for a next generation of nuclear power plants.



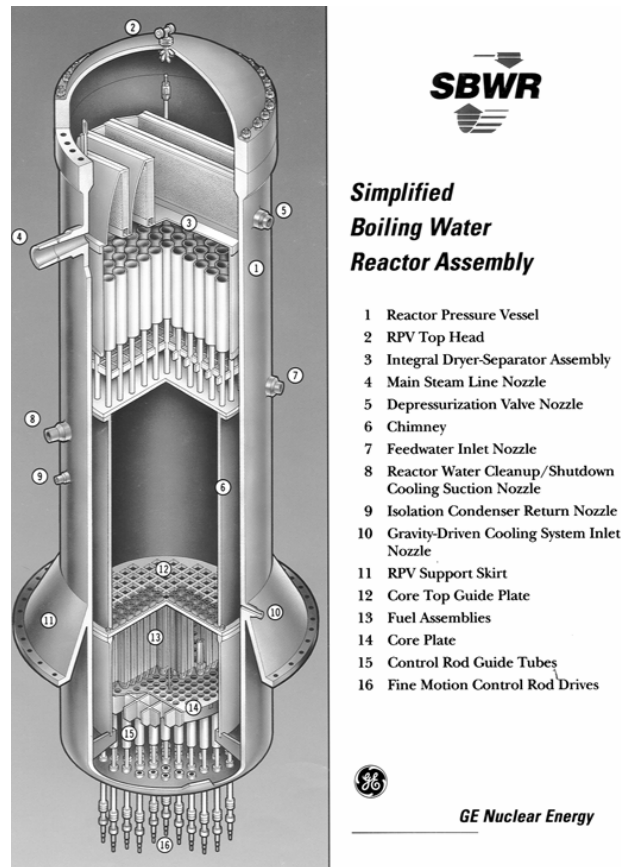
**Fig 6 The Passive Containment Cooling in the AP600 Pressurized Water Reactor Concept.**

The AP600 emphasizes a simplified plant design and safety by greatly reducing the number of small components that can fail during plant operation. Compared with a typical plant of the same size, it is designed to have 50 percent fewer valves, 35 percent fewer pumps, 80 percent fewer heating, ventilating and cooling units, 45 percent seismic building volume, and 70 percent less cables.

Gravity and natural circulation are to cool the reactor core and transport heat through the containment vessel to the atmosphere. In case of a coolant release, internal condensation transfers heat from the flashed steam to the steel containment, as shown in Fig. 6. That steel structure would initially be cooled with gravity fed water from tanks on top of the containment.

In addition, the steel containment is continuously cooled by natural circulation of air between the containment and the surrounding biological concrete shield structure. It is recognized that in existing PWR containment structures, a coolant leakage in the containment, without adequate cooling, would eventually breach the containment. The leakage would occur at the weakest points in the structure at the piping and instrumentation ducts seals.

The concrete containment is not meant to contain any coolant releases, but is provided both as a biological radiation shield, and as protection from the outside elements, like tornado or hurricane driven missiles.



**Fig 7. The Simplified Boiling Water Reactor (SBWR) Assembly.**

The 1.7 inch thick steel walls of the containment vessel enclose a larger space than in conventional PWR designs, and can thus withstand a long buildup of pressure inside the containment.

The reactor is placed at the bottom of the containment so that gravity fed water continually covers its core from emergency flooding tanks, in addition to the nitrogen pressurized accumulator tanks.

Automatic depressurization valves on top of the pressurizer open above a setpoint venting steam into a quenching tank that condenses it into water.

Instead of injecting water into the hot leg or cold leg or both like in current designs, Emergency water injection is directly into the core through a safety injection nozzle.

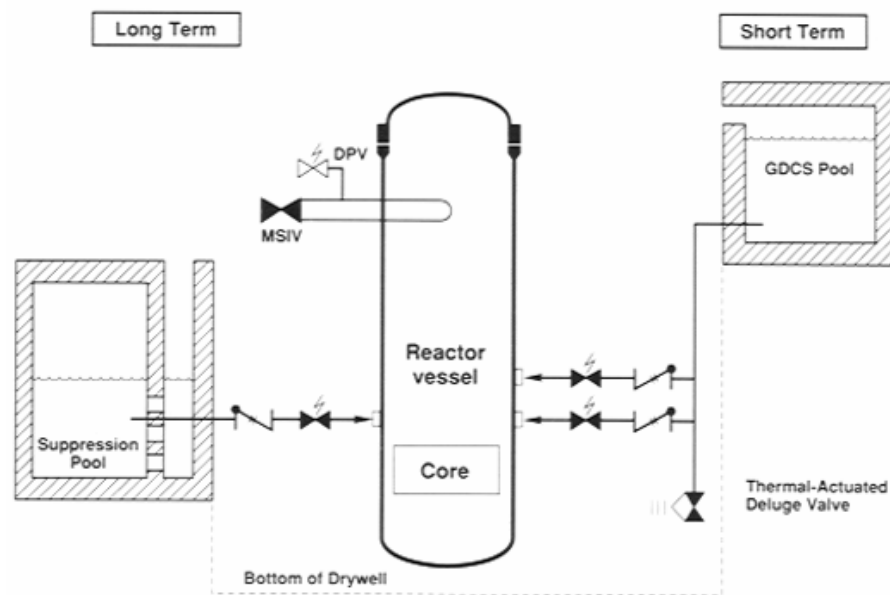


High inertia canned motor pumps improve safety and reliability. These pumps are closely coupled to the steam generator to avoid small LOCA core uncover.

#### 4.2 THE SBWR, SIMPLIFIED BOILING WATER REACTOR

In this 600 Mwe design, there exists complete dependence on natural circulation for cooling the core. The recirculation pumps have been eliminated as shown in Fig. 7.

Natural circulation cooling has been used in earlier BWRs such as the Dresden and Humboldt Bay plants. A 60 Mwe plant in the Netherlands, Doodeward used natural circulation over a 25 years lifetime at a capacity factor of 84 percent. In fact, most large BWRs operate with natural circulation up to 50 percent of their rated power or about 500-600 Mwe. To enhance natural circulation in the SBWR, the pressure drop through the core has been reduced from 25 psi to 5 psi. This reduction was achieved by shortening the SBWR fuel bundles from 3.7 meters to 2.7 meters. In addition the core's chimney, which is the annular region above the top of the core has been lengthened from 3 to 6 meters, enhancing the natural circulation process. The vessel of the SBWR is made forged rings with a diameter of 6 metres, and a height of 24 metres.



**Fig 8 Passive Safety Features of the SBWR System.**

The most attractive feature of the SBWR is its passive safety system shown in Fig. 8. In the case of Loss of Coolant Accident (LOCA), the water level in the reactor's core would drop to a level at which the safety system is initiated. A Depressurization Valve (DPV), ahead of the Main Steam Isolation Valve (MSIV) to the turbine, opens and the reactor's core is rapidly depressurized. Upon reaching a pressure of 30 psig, water from a Gravity Driven Cooling System (GDCS) pool, flows into the core. There are 3 independent pools of this type situated 12 meters above the core to provide enough head of water to overcome the reactor's pressure.

Isolation Condensers (ICs), situated in water pools on top of the reactor building replenish the water in the GDCS pools. These ICs are essentially heat exchangers, and were used in earlier BWRs. Steam in the drywell portion of the containment structure is diverted by the pressure in the drywell into the IC where it is condensed and returned to the GDCS pool and, then to the reactor core.

The operation of the ICs leads to a cooling of the containment. Heat is first transferred from the IC to the surrounding IC water pool. As the temperature of the pool rises, boiling ensues, and steam is released. A vent releases the steam to the atmosphere, making the atmosphere the ultimate heat sink. This approach to cooling the containment is designated as the Passive Containment Cooling System (PCCS). This approach eliminates the need for safety grade core cooling, for heat removal pumps, and for the supporting diesel generator units. The water available in the pools can support heat removal for up to 72 hours without any operator's intervention. The IC pool is outside the containment structure so that any escaping steam does not contain any radioactivity.

For long term cooling, water is also available from the pressure suppression pool. Thus dependence is on Isolation Condenser Pool, the Isolation Condenser pool, and the pressure suppression pool as sources of cooling water, as shown in Fig. 9. This provides three redundant passive cooling systems, each one being capable to independently mitigate the consequences of a LOCA.

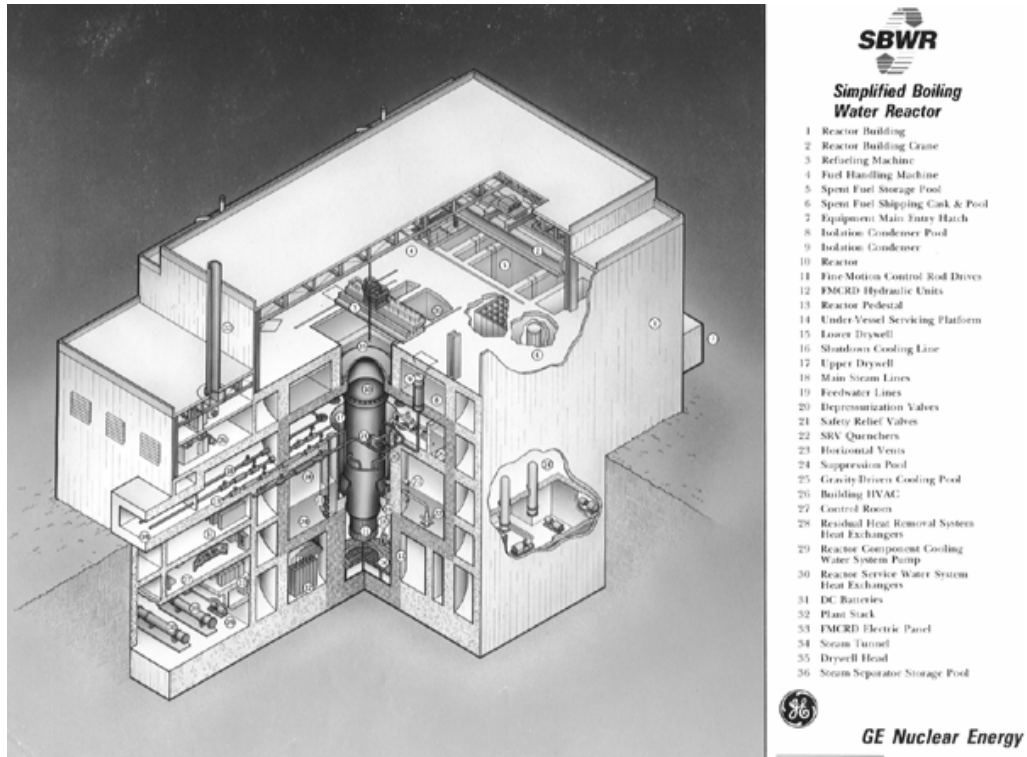
#### **4.3 THE MHTGR, MODULAR HIGH TEMPERATURE GAS COOLED REACTOR**

This reactor design comprises a 100 MWe graphite core, gas cooled reactor. Its small size and low power density achieve inherent safety. For the MHTGR the power density is 3 [W/cc], compared with PWR at 100 [W/cc]. The graphite core offers a high thermal inertia capable of absorbing a great amount of heat under accident conditions. The core is cooled with an inert gas: Helium.

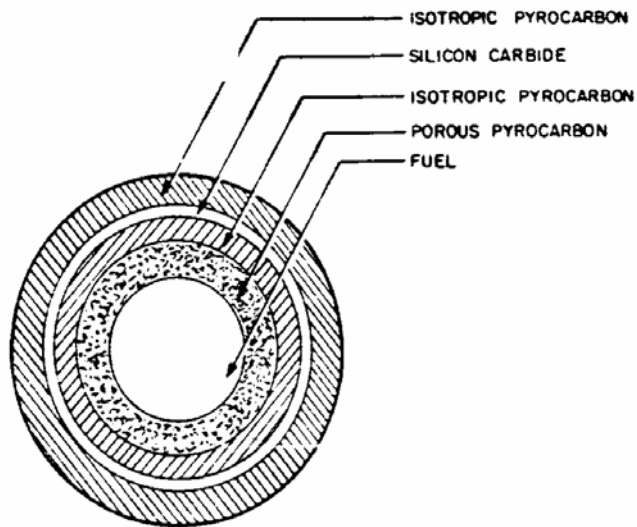
The design possesses a high negative temperature coefficient of reactivity, which would terminate the accident after a modest temperature rise without a radioactive release from its encapsulated fuel particles. The fuel particles themselves act as miniature pressure vessels containing the fission products. Figure 10 shows a multilayered TRISO pyrolytic graphite and silicon carbide coated MHTGR fuel particle.

As shown in Fig.11, the core of the MHTGR is limited in diameter, permitting the decay heat to be conducted and radiated to the environment without overheating the fuel to the point where the fission products would be released. A steel vessel now replaces the previous prestressed concrete pressure vessel of previous HTGR designs. The radiative cooling property is here gained without the active intervention of the operators.

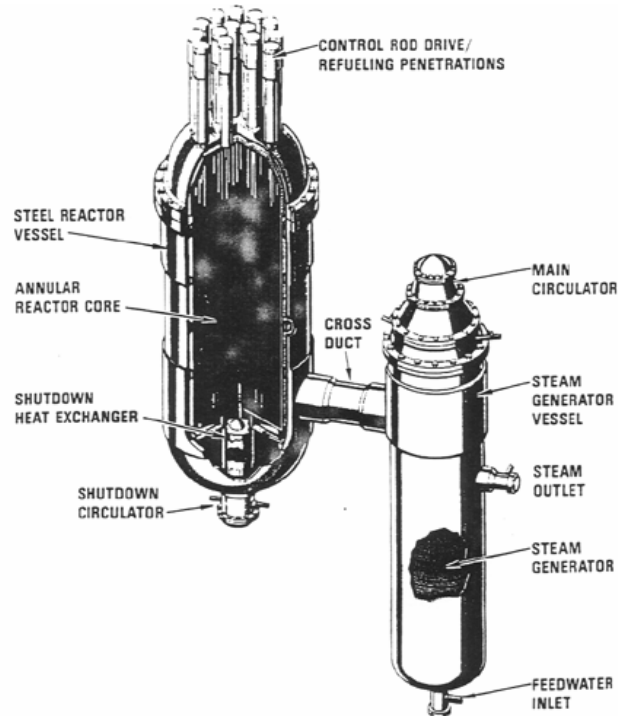
Many of the safety features of the MHTGR have been demonstrated in Germany on a 15 Mwe reactor: the Arbeitsgemeinschaft Versuchs Reactor (AVR) which was started in 1968.



**Fig. 9: Plant Layout for the SBWR showing the three independent passive cooling water pools.**



**Fig. 10: Pyrolytic carbon and Silicon carbide coated TRISO particle for the MHTGR.**



**Fig. 11 Pressure Vessel and Steam Generator for the Modular High Temperature Gas cooled Reactor (MHTGR).**

The MHTGR offers some perceived advantages compared to the PWR concept as outlined in Table 1. Chief among them are operation at high temperature resulting in higher overall thermal efficiencies, and the ability to produce process heat for industrial applications such as high temperature electrolysis of water to produce hydrogen as a future fuel.

#### **4.4 THE PIUS, PROCESS INHERENT ULTIMATE SAFETY REACTOR**

The PIUS concept was conceived with the following objectives:

1. Alleviate the public concern by relying on laws of nature, particularly natural convection, rather than the failure prone equipment and human intervention in the operation of a nuclear power plant.
2. Improve the safety margin to operate under more adverse conditions than exist in current PWRs, such as in third world countries.
3. Ease the reactor siting restrictions away from population centers, and make possible the combined production of electricity and heat for densely populated areas.
4. Build economical nuclear power plants in smaller unit plant sizes, in an environment where the economies of scale favor large size units.
5. Simplify the design to create an intrinsically safe water power reactor which eliminates the need for add on engineered safety systems and the associated quality assurance measures.

It is recognized that as long as the core of a reactor remains unharmed, no incidents in a light water reactor will result in the release of radioactive matter with major environmental impact.

Accordingly the design philosophy here is to protect the core against melting or overheating under all credible circumstances.

**Table 1: Comparison of the main features of 860 MWe MHTGR and PWR.**

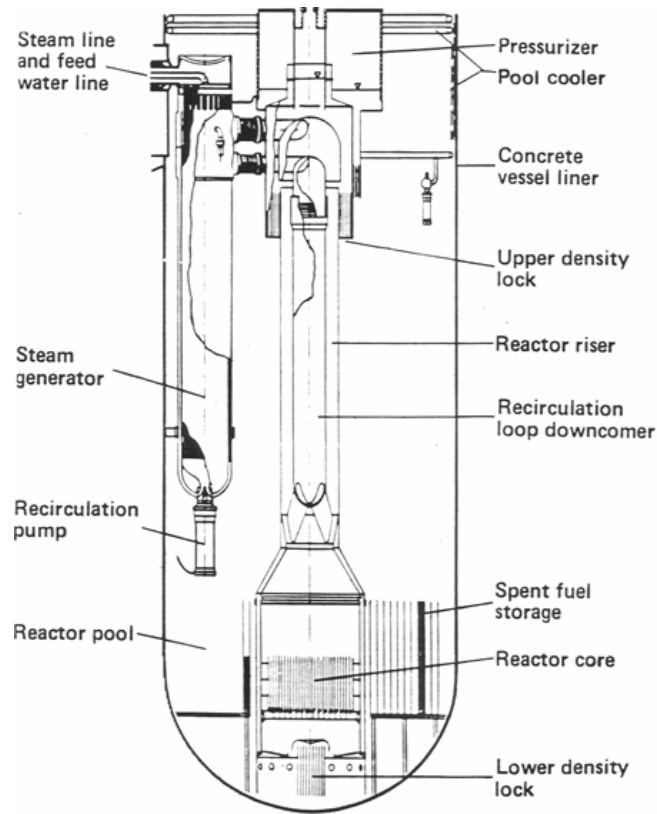
<b>Operating Characteristic</b>	<b>MHTGR</b>	<b>PWR</b>
<b>Steam temperature (degrees F)</b>	1,000	540
<b>Net plant thermal efficiency (percent)</b>	38.5	32
<b>Reactor core power density (Watts/cc)</b>	7.1	100
<b>Makeup cooling water required with wet cooling towers with forced draft (gallons/min)</b>	14,400	19,900
<b>Fuel required per billion Watts of electricity during 30 years of plant operation (short tons of U<sub>3</sub>O<sub>8</sub>)</b>		
PWR: Low Enrichment Uranium, MHTGR: Medium enrichment, once through fuel cycle.	4,510	6,000
PWR: Low enriched Uranium with Pu recycle, MHTGR: High enrichment Uranium with U <sup>233</sup> recycle.	2,280	4,100
<b>Radioactive wastes</b>		
Liquid (Curies/year)	134	310
Solid (m <sup>3</sup> /year)	61	180
Gaseous (Curies/year)	450	450

The core will remain unharmed when two design criteria are met:

1. The core shall remain submerged in water at all times.
2. The power level of the submerged core shall not rise to a level where the cooling capability of the submerging water is exceeded. Dryout of the core must be avoided.

The designers of the PIUS concept adopted the design principle that the process of heat extraction from the core shall be such that fulfillment of the two design criteria will be ensured on the basis of the laws of thermodynamics alone. Following any credible incidents, with the primary system intact, or subject to foreseeable damage. Following this design principle, makes the plant independent from mechanical or electrical components and systems, defeating Murphy's law, where if something can go wrong, it will. The plant becomes insensitive to human error by the operators. It becomes immune to destructive human intervention.

The concept was developed in Sweden by ASEA/ATM. The design principle depends on immersing the core, as shown in Fig. 11, in a large pool of cold borated water. The pool of water is contained at full reactor pressure within a large Prestressed Concrete Pressure Vessel (PCR.V). The steam generators and the primary cooling system are also located in the pool. All piping and instruments penetrations are at the top of the PCR.V. Such a configuration makes impossible for any type of leak to lead to the uncovering of the reactor core, and it remains submerged under water under all credible conditions.



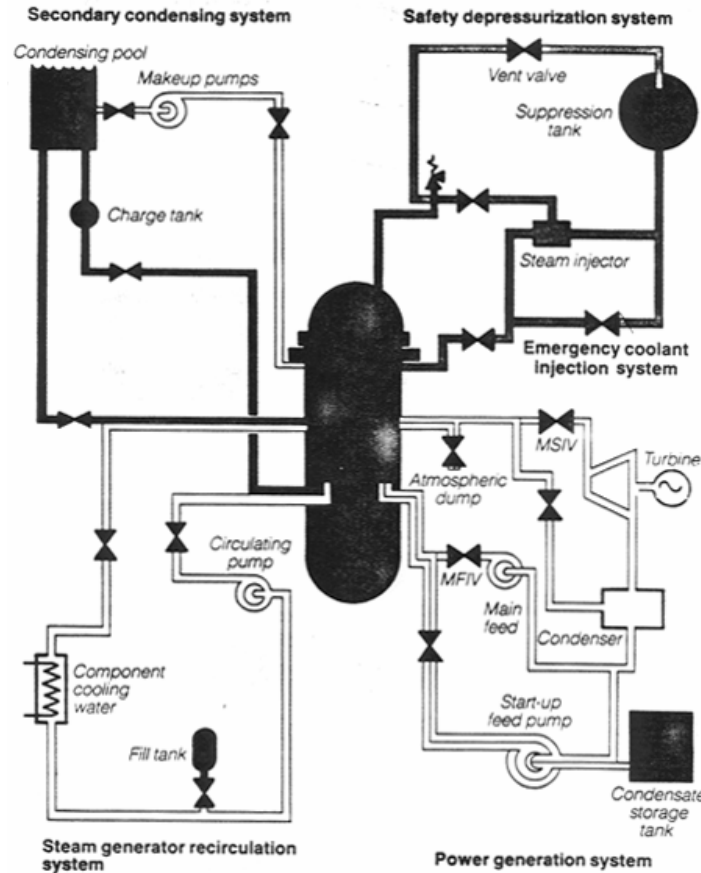
**Fig. 12: The primary system of a 500 MWe PIUS plant.**

Under normal operation, the primary coolant water is pumped through the core to the steam generators where steam is produced and sent to the electrical generators and turbines. A stagnant connection of the primary circuit to the reactor pool is provided in terms of the lower and upper density locks. The interface there is pressure balanced so that the two water circuits do not mix under normal operational conditions. If the normal flow of coolant in the primary circuit is interrupted for any reason, such as a loss of power to the circulating pumps, borated water from the pool would naturally enter the primary system. The boron in pool's water is a strong absorber for neutrons, and would shut down the chain reaction and the power generation in the core. In addition, the water ingress would cool the reactor from the remaining decay heat by establishing natural circulation of the water through the core and the pool. The water in the pool is sufficient to keep the core cooled for at least a week, in the absence of other sources of water. Over this time, the decay heat would have declined to low levels, and makeup water could then be added.

The design of the core is a typical PWR 16x 16 square fuel assembly without channels. The fuel rods have a larger diameter than normal PWRs at 0.5 in in diameter. This reflects the reduced specific power of the fuel at 23 kW/ kg U.

The core height is reduced to about 6.5 feet. This, combined with the reduced specific power, decreases drastically the core pressure drop and permits the use of a pressure balance system.

Each fuel assembly can accommodate 4 shutdown neutron absorber rods. These rods are more of a licensing requirement for early plants, but may not be necessary for later plants since shutdown can be achieved by boron injection. Each fuel assembly has 4 rod positions for gadolinia neutron absorbers which are replaced each year. For longer fuel cycles the number can be increased.



**Fig. 13 The Safe Integral Reactor (SIR) flow diagram.**

The power level control is designed to be by boron content and coolant temperature, without the need for the control rods. Boron dilution makes possible a 2 percent per minute load increase. The gadolinia shim the coolant temperature coefficient of reactivity is kept strongly negative, a desirable neutronic safety feature. This negative coefficient in combination with once through steam generation makes possible excellent load following properties. Fast load steps of up to 25 percent of full power are possible without boron dilution. The system can withstand grid disturbances such as short circuits of 250 ms duration. The plant startup from cold conditions is about the same as that for standard light water reactors.

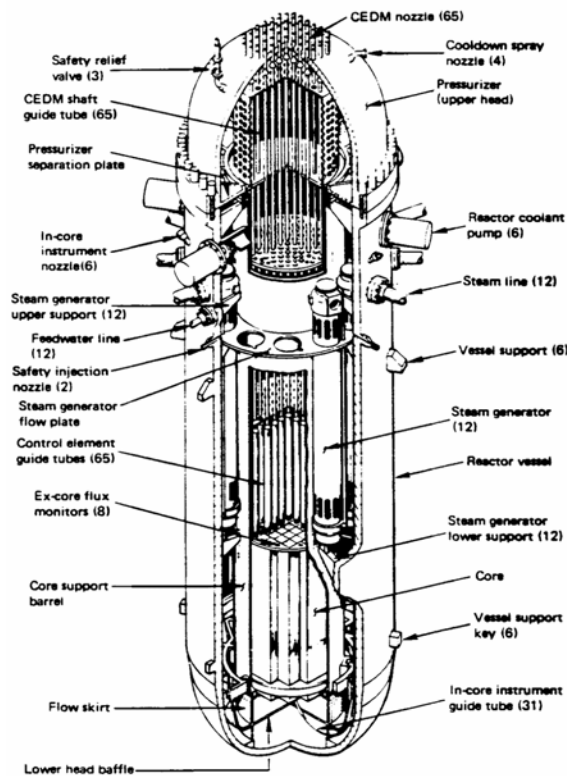
Let us consider the response of the plant to a loss of auxiliary power accident, or a station blackout during days. The following events would occur:

The recirculation pumps are starved of electric power and they trip, or stop functioning. Because of the drop in pressure, connection is established through the lower and upper locks between the

primary circuit and the coolant pool. The boron injection from the pool water into the core shuts the reactor immediately. Natural circulation is established naturally leading to core cooling even though the circulating pumps are not operational. Hot water collects in the upper part of the pool and is cooled by natural circulation to an external atmospheric pool. Thus any short-term damage to the core is averted. For the longer term, three scenarios can be envisioned:

1. Water is refilled in the external pool within about three days at 4 m<sup>3</sup>/hr, by using off site vehicles and personnel. This process can last indefinitely if needed at all.
2. The plant is left alone. In this case the depressurization valves open when the atmospheric pool has boiled dry. The pool water boils off through the depressurization valves. The core starts to be uncovered after about 20 days. Boric acid congestion of the core occurs after one week, unless more volume of water is made available under the core.
3. The plant is left alone. The depressurization valves remain closed. The internal water pool heats up and boils off through the relief valves. The core starts getting uncovered within 12 days. A gain of a sufficient period of time for adding any needed water.

The protection using passive measures is effective against conceivable accidents caused by equipment malfunction and operator errors, as well as against unforeseen events such as earthquakes, sabotage or military attack.



**Fig. 14: The Integral Safe Reactor pressure vessel containing the steam generators.**

#### **4.4 THE SIR, SAFE INTEGRAL REACTOR**



Combustion Engineering, now a unit of Switzerland's ABB Combustion Engineering Nuclear Systems, developed the Safe Integral Reactor (SIR) concept, jointly with Rolls Royce and Associates, Ltd., Stone and Webster Engineering Corp., and the UK Atomic Energy Authority.

The basic design philosophy of the SIR is adopted from naval reactor designs where the heat exchangers and the associated piping are enclosed together with the reactor core within the same reactor vessel. In this way as shown in Fig. 13, the possibility of coolant leakage from the piping is eliminated, since any leakage would be contained within the reactor vessel's space as shown in Fig. 14. This eliminates the traditional reactor coolant loop primary system piping associated with more conventional PWRs. The reactor core, the pressuriser, and the steam generators are contained in a single reactor pressure vessel. The reactor coolant pumps are mounted on the side of the reactor vessel.

Integral type coolant circuits have been used in the Advanced Gas cooled Reactors (AGRs) in the UK, and in the type of reactor commonly used for ship propulsion. In this type of coolant circuit, the primary circulating coolant is contained within the vessel, eliminating the need to circulate the primary coolant through the connecting pipe work to the steam generator, where coolant leakage is most likely to occur in the loop design approach to the primary circuits.

A safety depressurization system is provided to divert steam to a suppression tank. This tank also feeds an emergency coolant injection system. The condensate storage tank as well as a secondary condensing pool are sources of extra emergency coolant.

The SIR reactor is 325 Mwe. Its size is limited by the practical construction and transportation requirements of the pressure vessel. It contains a pressure-suppression containment and 12 cylindrical once-through steam generators, only 11 of which are needed to reach full power.

The safety systems are primarily passive relying on natural circulation and a large heat capacity rather than active AC power and equipment.

Reactor control is maintained by the use of control rods and burnable poisons, with the traditional PWR boron shim being eliminated for the sake of simplicity and corrosion protection.

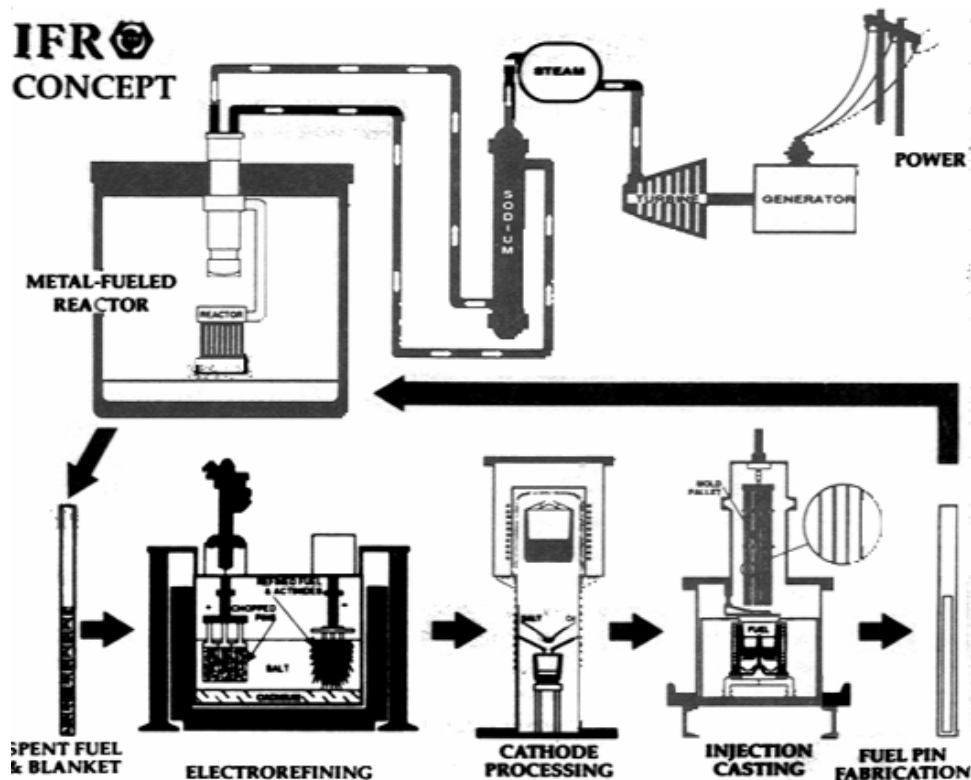
## **5. THE INTEGRAL FAST REACTOR (IFR) DESIGN**

The Integral Fast Reactor (IFR) concept builds on the experience acquired from the successful operation of the experimental Breeder Reactor (EBR II), a 40 MW research reactor operated over 25 years by Argonne National Laboratory (ANL). It was sited at the Idaho National Engineering Laboratory (INEL) near Idaho Falls, Idaho in the USA. The experience of EBR II is the basis of other international effort in fast reactors including The Phenix and the Super Phenix in France. The concept also builds on the experience gained in the Fast Flux Test Facility (FFTF) at Hanford in the state of Washington, where the mixed oxide fuel (MOX) as a mixture of  $UO_2$  and  $PuO_2$  was tested.

The IFR offers the advantage of fast reactors in their ability of breeding new fissile fuel, as the old fissile fuel is being consumed. This extends the available supplies of nuclear fissile fuels practically indefinitely.

It offers a large degree of inherent safety from two perspectives:

1. The core of the reactor is immersed in a large pool of sodium liquid metal possessing a large thermal inertia, and capable of absorbing the heat generated by the fuel under any credible accident condition.
2. The coolant is operated at atmospheric pressure and is not pressurized like in the gas or water cooled designs. In the case of sudden depressurization of a pressurized coolant it is lost to the system. In the case of water as a coolant, it flashes into steam and is lost. This cannot happen in the case of the IFR since the coolant is operated at atmospheric pressure. If the coolant pumps fail, the reactor naturally shuts itself off, without the need for human intervention. If the secondary steam system shuts off, the reactor shuts itself off even without any control rod movements or actions on the part of the reactor operators.



**Fig 14: Flow Diagram of the Integral Fast Reactor Concept.**

One technical difficulty remains in that the liquid metal is chemically reactive with air and water. Thus, the coolant must be covered with an inert gas atmosphere, and double walled heat exchangers tubing must be used to avoid contact between the liquid metal and water.

The IFR program devised an ingenious way of using metallic fuels effectively. The use of metallic fuels has been replaced by the use of ceramics as metal oxides to avoid a swelling problem and consequent rupturing of the cladding in earlier metallic fuels. The IFR uses a method for electrorefining the fuel onsite solving two problems at once. It is quicker and cheaper than in traditional fuel reprocessing, and the produced new fuel could be immediately loaded into the reactor core.

The reprocessed metallic uranium, plutonium and zirconium fuel is spiked with a small percentage of fission products, making it useless for weapons manufacture and cannot be diverted to non-civilian activities. Thus the fuel is thought to offer nonproliferation characteristics and would be suitable for global deployment.

Letting the fuel pellets float freely within the cladding has solved the swelling problem in metallic fuels. This allows them to swell freely. The noble gas fission gases such as Xenon and Krypton are evacuated from the fuel and migrate to a containment area in the head of the reactor. This also solves any problem of cladding rupture that would otherwise result from the pressure buildup caused by these gases. For enhanced heat transfer a little sodium is added to the fuel pellets.

The ventilated fuel offers good burnup, compared with the previous metallic fuel designs. It is designed to have its center point temperature below the boiling point of the sodium coolant, due to its good thermal conductivity. As a result, if an accident occurs, the coolant would not boil like in the case of a light reactor system.

The IFR seems to be the best of any possible breeder design with the lowest doubling time and the highest breeding ratio. The product of fuel reprocessing does not contain any actinides such as uranium or plutonium, which are all recycled back to be burned in the core and produce energy. Accordingly, the fission products will be after about 200 years at the same level of radiological activity as the ores the uranium was initially mined from. The design, shown in Fig. 14, in this way solves the problem of disposing of the long-lived actinides in the waste from light water reactors.

The fuel electrorefining process uses high temperature with the uranium and plutonium moved from an anode to a cathode leaving the waste product behind. These impurities collect in the anode compartment or remain dissolved in the electrolytic bath of molten salts. Dissolving the metallic fuel pins in a liquid cadmium bath at 900 degrees F, starts the process.

The IFR has been the USA's contribution to programs on fast reactors in the industrialized nations with limited fossil fuel supplies such as Japan, France and the UK.

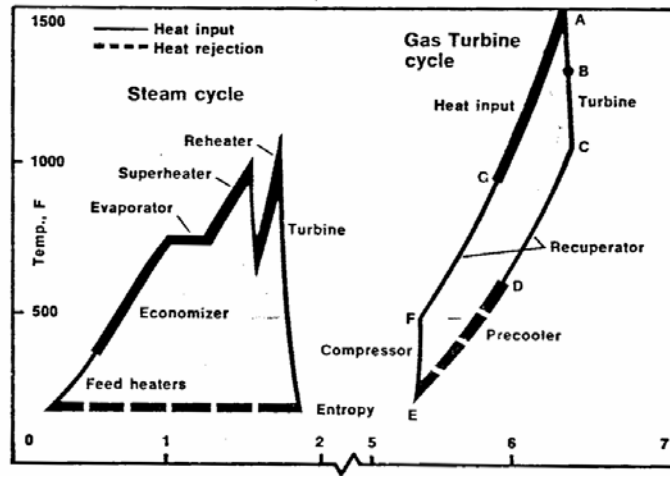
## **6. BRAYTON CYCLE CONCEPTS**

### **6.1 INTRODUCTION**

Gas cooled reactors using graphite as a moderator material offer a high degree of safety due to the large thermal inertia of the graphite. In addition, the fuel particles are enclosed in layers of pyrolytic graphite and silicon carbide that prevent radioactive releases even at high temperatures.

Past designs used the Steam or Joule cycle as shown in Fig. 15. The newer designs take advantage of new developments in turbine technology such as magnetic bearing, and use the Brayton or Gas Turbine cycle shown in Fig. 15. With the use of Brayton gas turbine cycle, the helium coolant is enclosed in a single circuit moving from the compressor to the turbine. The possibility of its depressurization or leakage is minimized, and it is not reactive with graphite like steam would be. The designs can operate at higher temperatures and offer a high value of the thermal efficiency around 40 percent, compared with the 30 percent value for light water reactors. The high temperatures offer the possibility of process heat generation and use in

industrial processes such as high temperature water electrolysis for the production of Hydrogen for future non fossil transportation fuel supplies.



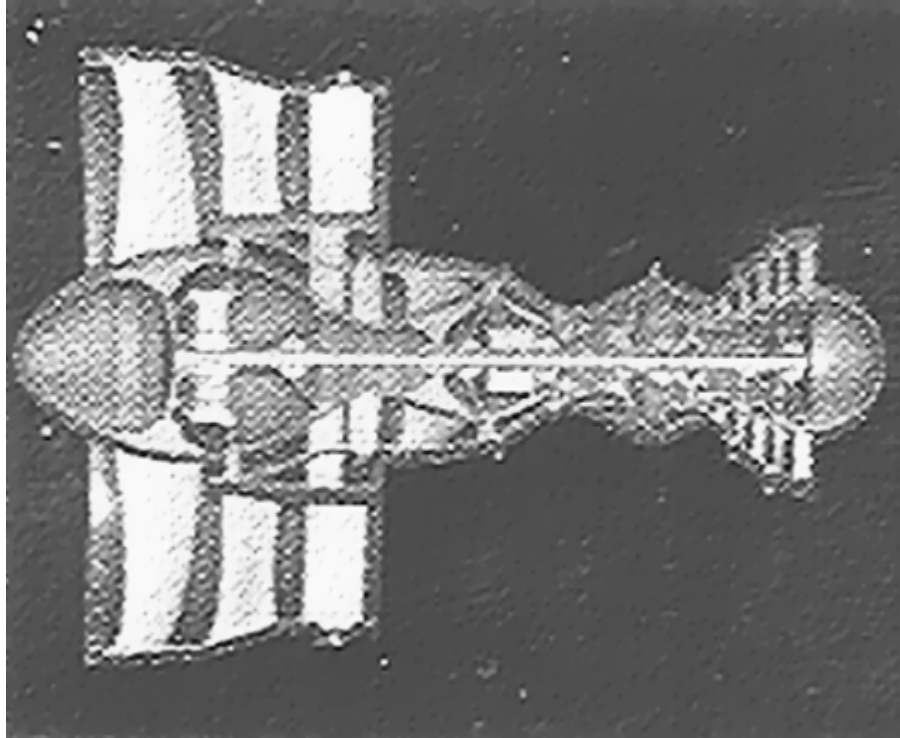
**Fig. 15: Comparison of the Brayton cycle to the Steam Cycle on a Temperature Entropy Diagram.**

**Table 2: Gas cooled Reactors Experience.**

Concept	Power (MWe)	Characteristics
AVR	15	Experimental Pebble Bed Reactor, operated for 21 years in Germany.
THTR	300	Demonstration pebble bed reactor with steam turbine, operated for 5 years in Germany.
FSV (Fort Saint-Vrain)	330	Operated in the USA for 14 years.
HTR	30	Reached criticality in 1998 in Japan.
HTR-10	10	Reached criticality in 1999 in China.
VG-400, VGM, VGM-P		Russian gas cooled reactor designs.
HTR-MODUL	80	German pebble bed reactor licensed for Siemens/Interatom in 1987.
HTR-100	100	German pebble bed reactor design by HRB/BBC.
GT-MHR	300	Helium Turbine design Russian-USA-French-Japan design.
PBMR	110	Pebble bed South African design, to be licensed in the USA by Exelon Corp.
PROTEUS	-	Critical test facility in Switzerland.

These designs build on a wealth of operational experience in the development of the gas-cooled reactor concept as shown in Table 2.

The Brayton cycles designs build on new experience from the space and aeronautical field that did not exist a few years ago, such as the use of magnetic bearings as shown in Fig. 16. The advantage of magnetic bearings is the elimination of the dependence on liquid oil for lubrication.



**Fig. 16: Use of magnetic bearings in turbine design in advanced propulsion technology.**

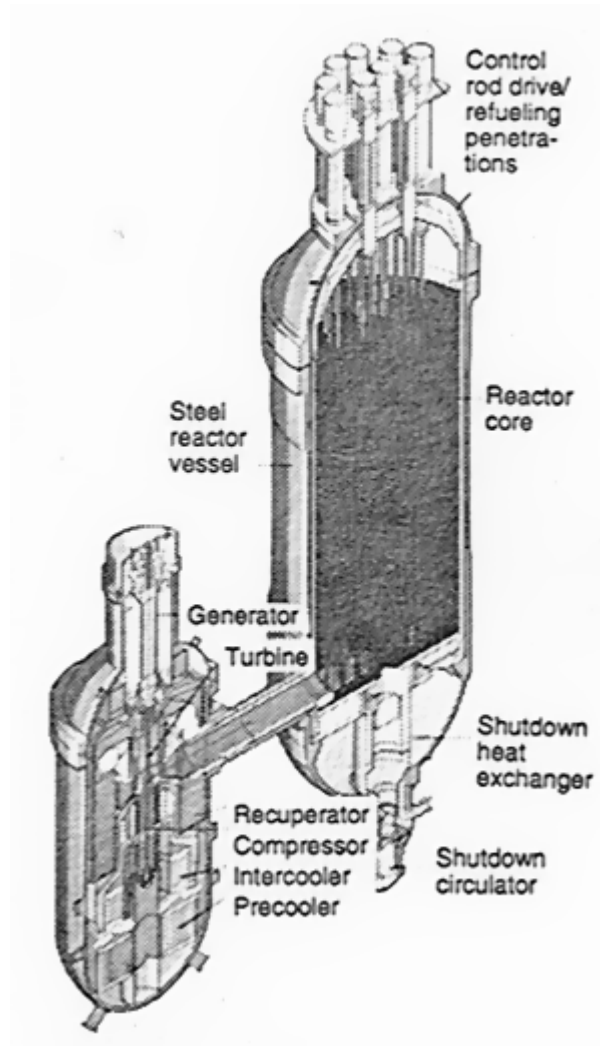
## **6.2 THE GAS TURBINE MODULAR HELIUM REACTOR: GT-MHR**

Is based on an agreement between General Atomics in the USA and Russia. This concept pioneers the development of a next-generation modular nuclear reactor using the direct-cycle gas turbine, rather than the steam cycle. This approach operates at higher temperatures than light water reactors, increasing the overall thermal efficiency of the plant by 50 percent. This in turn reduces the cost of producing electricity to the range of 3 cents per kilowatt-hour, compared with the cost of 5 cents per kilowatt-hour in existing designs.

The ability of handling higher temperatures is possible because of the use of ceramics such as silicon carbide. Helium as a coolant replaces water, which even though a splendid coolant, is corrosive. This limits the use of water to a temperature of 700 degrees C. Helium, in contrast, is inert and noncorrosive. It has no thermal limit, is in the gaseous form whether at room temperature or 3,000 degrees C, which is a tremendous advantage.

Every time a coolant is passed through a heat exchanger before it can do its work, significant losses in efficiency occur. As shown in Fig. 17, the turbine is directly driven by the helium gas as it comes out of the reactor core. Since no heat exchange is needed, this improves the overall efficiency of the plant.

The 300 MWe units will be able to burn either uranium fuel, or fuel made from weapons grade plutonium, thus converting weapons grade material into electricity. If plutonium is used, up to 95 percent of it is used during power production. A mixture of Uranium and Thorium can also be used.



**Fig. 17: The entire Gas Turbine Modular Helium Reactor power plant is contained in two pressure vessels, enclosed into a concrete containment structure below ground.**

This kind of development project is also advocated as a way for western nations to turn around the economic depression and cultural pessimism that are eroding scientific and technological capabilities threatening their futures. In addition it would help the former nations of the Soviet Union and Third world countries out of their current economic devastation. It has been even advocated as a contribution for preventing atomic scientists in these countries from getting involved in nuclear weapons projects in proliferation prone countries. Along this line, the U.S. has pledged \$ 1.6 billion in terms of cleaning up problems in Russia's nuclear and weapons programs.

The new technology of magnetic bearings virtually eliminates friction. It has also great properties in the dynamic dampening of rotating shafts; a technology that did not exist a few years ago. The gas turbines builds on new turbine technology developed for jet engines such as the Boeing 747. High-strength high-temperature steel vessels are used.

New technology has also become available in the recuperators from the fossil fuel power production field. Today's recuperators are five times smaller than the designs of a few years ago. Highly efficient plate-fin heat exchange equipment is used.

The design consists of two pressure vessels, both located underground in a concrete containment structure. The first vessel houses the reactor system. The second vessel houses the power conversion system including the gas turbine, two compressors, and a generator.

The helium gas in the core is heated to 1,562 degrees F. The heated helium flows to the power generator unit generating electricity at an overall thermal efficiency of 48 percent, compared with the value of 28-30 percent in water cooled reactors.

The reactor possesses passive inherently safe features in that it can shut itself down and cool itself down in the case of an emergency. Its cooling towers are one-sixth the size of those of conventional power plants, which reduces the overall cost. The cooling towers can be air-cooled rather than water cooled, which suggests that the plants can be located where water resources are scarce.

Compared with light water reactors, a reduction of 50 percent in the amounts of nuclear waste and thermal discharge is claimed.

The control and monitoring system are based using digital programmable equipment. Panels based on traditional equipment are used for controlling the safety systems.

Operation at high temperatures open the possibility for process heat applications such as the extraction and production of chemical products and mineral fertilizers, coal gas saturation, production of synthetic natural gas from coal, and ferrous and nonferrous metallurgy, as well as district heating. The expected completion date for the 4 modules first Russian design is 2005.

### **6.3 THE PEBBLE BED MODULAR REACTOR: PBMR**

The PBMR is being pursued jointly by the Exelon corporation in the USA and South Africa's state owned utility Eskom. This concept is a high temperature helium cooled reactor with unit sizes of 110 MWe, These small sizes can be factory built before assembly at a site. The pressure vessel of the PBMR is shown in Fig. 18.

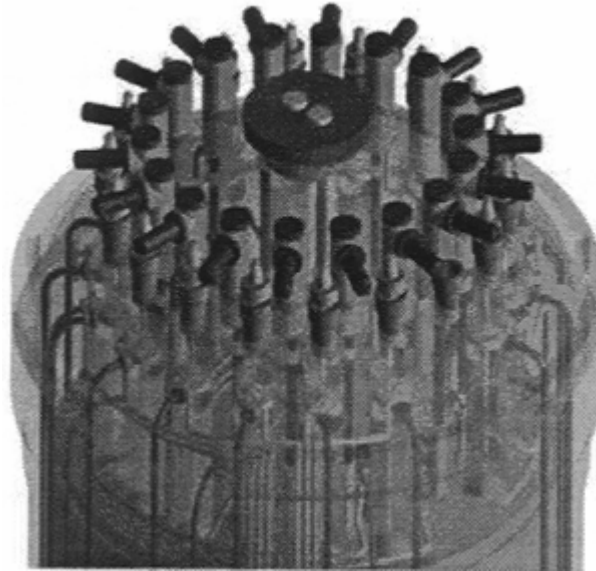
This design is the dependence on fuel in the form of pebbles 6 cm in diameter. About 400,000 of these pebbles lie within a graphite lined vessel that is 20 m high and 20 m in diameter as shown in Fig. 19. Each pebble contains about 15,000 fuel particles where the fuel is enclosed in layers of pyrolytic graphite and silicon carbide.

The use of graphite as fuel element cladding, moderator, core structural material, and reflector, provides the reactor with a high degree of thermal inertia. A core melt situation would be practically unlikely, since a large difference exists between the normal average operating temperature of 1,095 degrees C, and the maximum tolerable fuel temperature of 2,800 degrees C.

Helium at a temperature of about 500 degrees C is pumped in at the top of the reactor, and withdrawn after sufficient burnup from the bottom of the reactor. The coolant gas extracts heat from the fuel pebbles at a temperature of 900 degrees C. The gas is diverted to three

turbines. The first two turbines drive compressors, and the third drives an electrical generator from which electrical power is produced.

Upon exit from the compressors or generator, the gas is at 530 degrees C. It passes through recuperators where it loses excess energy and leaves at 140 degrees C. A water cooler takes it further to about 30 degrees C. The gas is then repressurized in a turbo-compressor. It then moves back to the regenerator heat-exchanger, where it picks up the residual energy before being fed to the reactor.



**Fig. 18: The Reactor Pressure Vessel of the Pebble Bed Modular Reactor.**

Refueling is done online, eliminating refueling outages. The PBMR would shut down every few years for maintenance of other mechanical parts of the plant. The staff would be constantly taking pebbles out of the bottom, checking their burnup, eliminating any leakers, and then reloading them back from the top, or adding fresh pebbles to replace the discarded ones..

The spent fuel pebbles are passed pneumatically to large storage tanks at the base of the plant. This storage space can hold all spent fuel throughout the plant's life. These tanks can hold the fuel for 40 to 50 years after shutdown. About 2.5 million walls are normally used over the 40 years design life of a typical reactor. The silicon carbide coating on the fuel particles can isolate the fission products, at least in theory for a million years. For permanent storage, these pebbles are easier to store than fuel rods from PWRs.

The PBMR is expected to have an overall thermal efficiency of about 40-42 percent. It operates at a low power density of less than  $4.5 \text{ MW/m}^3$ , compared with a value of 100 for a PWR.

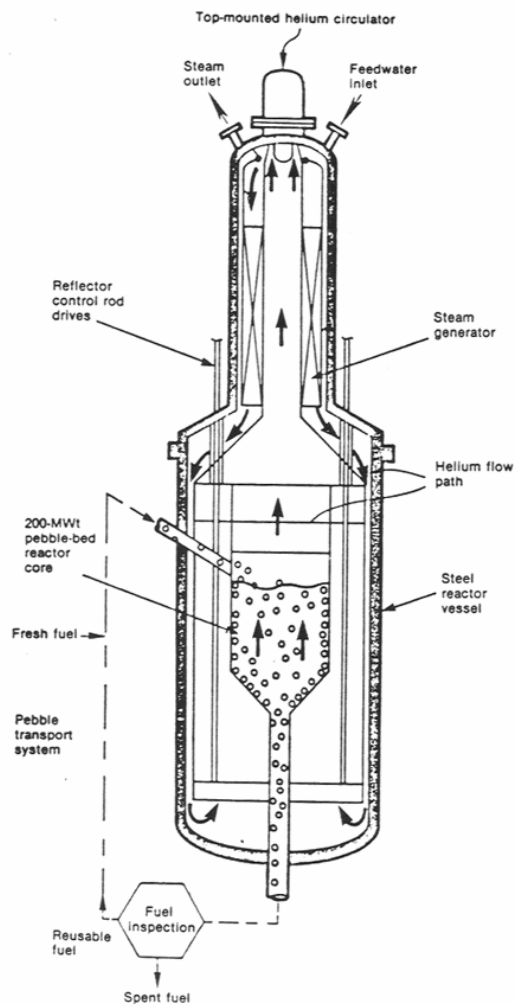
The economic advantage of the PBMR is that it can allow a utility to make a decision on investing about 120 million dollars rather than the 2-3 billion dollars for other power plants. In addition, the construction time can be reduced to the 18-36 months level as opposed to the 5 or more years for light water reactors. The operating costs of the concept because of the staffing



characteristics and the lower fuel costs. It can satisfy the information technology needs which are increasing the load demand in the USA at 4-4 1/2 percent rate per year, for instance around Chicago.

It possesses a high degree of inherent safety. The worst case scenario would produce temperatures below fuel damage temperatures.

The design provides a containment structure provide for regulatory needs only the reason is that the type of accidents envisioned would take hours to days to develop, as compared to minutes in PWRs. In the USA a Nuclear Regulatory Commission (NRC) policy statement designated as SECY 93-092, provides guidance in this regard. It concludes that conventional containment is not needed for such a design. Nevertheless, the PBMR design allows for the release of the helium coolant in the case of a loss of coolant accident. The containment structure is there to intercept any fission products release within days of the initiation of any accident.



**Fig. 19: Conceptual Design of Pebble Bed Gas Cooled Reactor with a steam cycle rather than a gas cooled cycle..**

The design would require a smaller Emergency Planning Zone (EPZ). The control rods are used only for compensating for the initial heat up and for achieving full cold shutdown. For temperature control, the helium pressure is lowered or raised. To decrease the power level, the temperature is decreased by increasing the helium pressure and vice versa.

In a bridging to a helium economy in the future, high temperature systems can be used to dissociate water into oxygen and hydrogen on a global scale. This would satisfy in a nonpolluting manner the needs of both industrialized and developed nations. Expectations are for the doubling of electricity demand worldwide by 2020.

## **7. DISCUSSION**

The core meltdown frequencies for two conventional reactor designs, the Surry PWR and the Peach Bottom BWR were estimated by the WASH-1400 reactor safety study to be  $6 \times 10^{-5}$  and  $3 \times 10^{-5}$  [accidents/reactor.year], respectively. The uncertainty in these estimates is believed to be a factor of 5-10 either way. On this basis, a core melt frequency of  $10^{-4}$  cannot be ruled out.

Studies before the Three-Mile Island (TMI) accident suggested that these frequencies were as high as  $10^{-3}$ . After the improvements mandated by TMI, it is suggested that present day reactors have frequencies 1.5-3 times higher than the WASH-1400 study. This puts them back at about  $10^{-4}$ .

For an evolutionary PWR design built by Westinghouse and Mitsubishi, and for the evolutionary ABWR built by General Electric, the frequencies are estimated at the level of  $1.1 \times 10^{-6}$  and  $5 \times 10^{-6}$ , respectively. Thus the evolutionary designs can be thought as having a core melt frequency around  $10^{-5}$ .

The USA's Nuclear Regulatory Commission (NRC) has promulgated the value of  $10^{-4}$  as a safety goal. Thus it is seen that the evolutionary designs are an order of magnitude lower than the NRC goal. With a world with about 500 reactors, a core meltdown frequency of  $10^{-4}$ , translates into:

$$500 \text{ reactors} \times 10^{-4} \text{ [accident/reactor.year]} = 1/20 \text{ [accident/year]},$$

or 1 accident each 20 years. With the figure associated with the evolutionary designs, at  $10^{-5}$ , this figure becomes 1 accident per 200 years. It does appear that the latest figure is one that could be more acceptable to the public, suggesting any new nuclear power plant construction cannot follow the older standard designs any more, and the latter should probably be retired in favor of the new evolutionary designs.

If the world opts for more nuclear power plants construction, as a remedy for greenhouse emissions, and a movement towards a hydrogen based energy economy, a world with a 1,000 nuclear power plants, can be envisioned. If we desire in such a world for the core meltdown frequency not to exceed 1 accident per 200 years, this would mandate a core melt frequency design goal of less than  $5 \times 10^{-6}$ . In this situation, to account for the statistical error involved at a factor of ten either way, a move to the passive designs which would provide a core melt frequency of less than  $5 \times 10^{-7}$ . This appears to be the clear alternative for the world to benefit from nuclear electrical production.

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