Appendix A: Overview of Reactor Technologies

Background

Although a small number of commercial ‘fast’ reactors which utilize fast neutrons to maintain a critical nuclear reaction have been operated in Russia and France, all of the world’s currently operating commercial nuclear power plants are thermal plants which utilize slow or ‘thermal’ neutrons to maintain a critical nuclear reaction. The principal components of a thermal reactor are fissile material (fuel), a moderator to slow the neutrons to thermal speeds, a reactor coolant system to remove the heat produced by fission, and hardware to house the fuel, moderator, and reactor coolant. A very large variety of moderators and coolant combinations have been tried. However, the combined economics and safety considerations have resulted in very few of these combinations being commercialized. These include the Pressurized Water Reactor (PWR), the Boiling Water Reactor (BWR), the CANDU reactor, the MAGNOX and AGR gas cooled reactors, and the HTGR. Although MAGNOX and AGR plants continue operation in the UK, the UK organizations no longer support development of the technology and these plants are not commercially available. The principal characteristics of the other technologies are discussed in the following subsections.

Commercial nuclear power plants world-wide have been constructed and operated for the generation of electricity utilizing a steam cycle. Limited process applications of nuclear power have been implemented (e.g., the Bruce Energy Center in Ontario is no longer supplied with nuclear generated steam, and district heating systems in Switzerland). These applications have utilized a small fraction of the reactor output, and have operated at low pressure.

For electricity generation, the steam from the turbine is discharged to a vacuum condenser. The vacuum in the condenser is maintained as low as technically feasible in order to maximize electricity output. The condensers are cooled by water. This water is supplied by and returned to a lake, river, or ocean if such bodies of water of sufficient capacity are available, or by a partially closed circuit cooling water system utilizing either natural draft or forced draft cooling towers that reject heat to the atmosphere. The amount of heat rejected to the environment ranges from approximately 66% of the reactor output for water cooled reactors to 58% for the High Temperature Gas Reactor.

All figures utilized in the following sections were obtained from public domain literature.

Pressurized Water Reactor (PWR) NPPs

PWRs utilize light/ordinary water as the coolant and moderator. The fuel and coolant moderator are housed in a large pressure vessel (see Figure A-1). The reactor coolant pumps circulate the reactor coolant through the reactor pressure vessel to remove the heat of fission and through the Steam Generators (vertical U-tube heat exchangers) to generate steam on the secondary side which subsequently powers a turbine or other industrial application. The reactor coolant temperature at the reactor vessel outlet remains below the saturation temperature. Since the reactor coolant does not pass through the turbine, this is referred to as an indirect cycle.

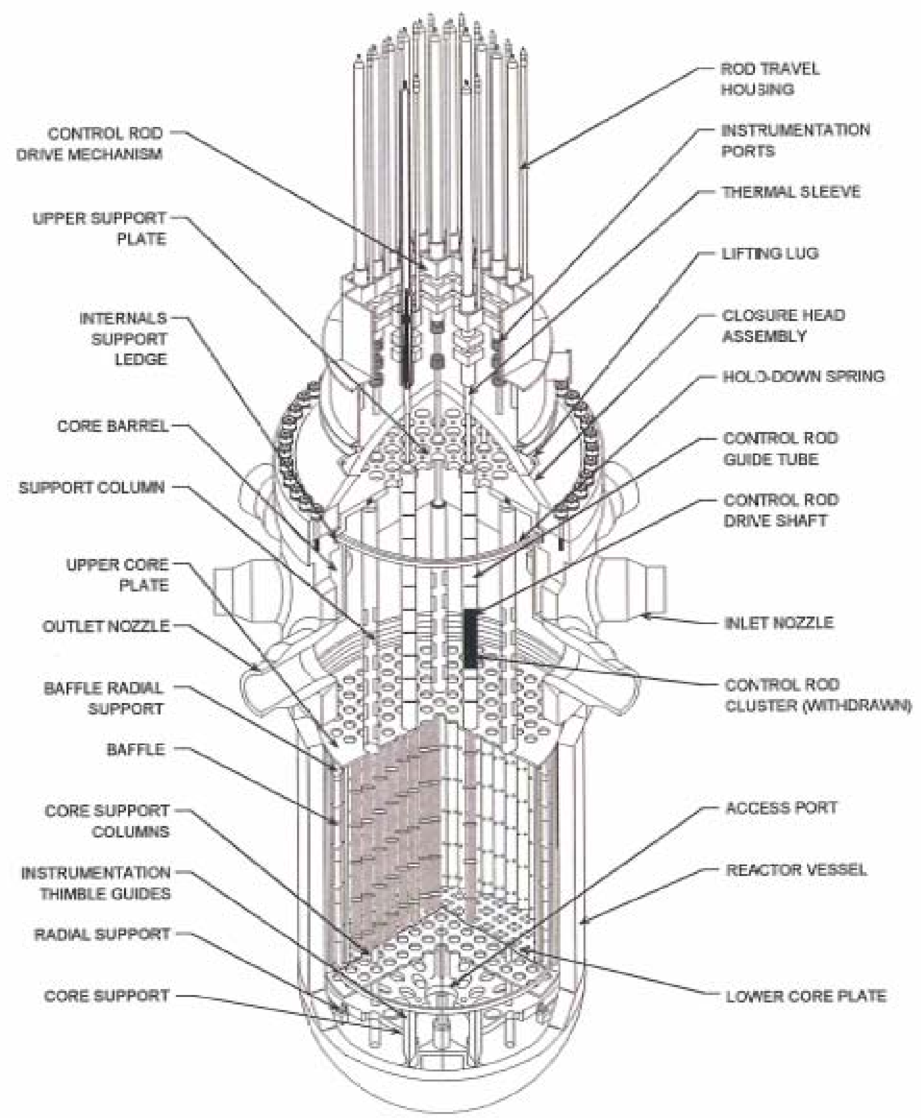
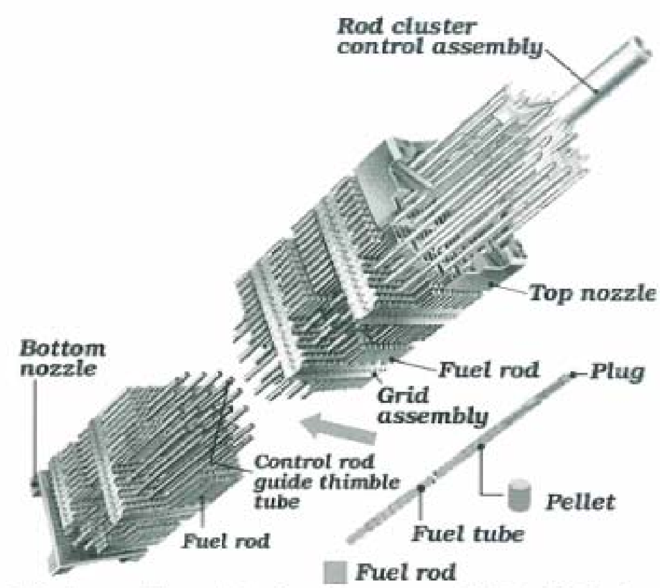


Figure A-1, PWR Pressure Vessel Configuration

Light water as the moderator has the advantage of a short thermalization distance (the distance it takes to slow fast neutrons to thermal speeds) which results in small reactor size, low cost, and low activation. The disadvantage of a light water coolant and moderator is that light water is a significant neutron absorber, requiring the enrichment (increase in fissile content) of the fuel. PWRs are refueled off-power on cycles that vary between 18 months and two (2) years.

The vertical PWR fuel assemblies are approximately four (4) meters long, and consist of square arrays of fuel rods. The fuel rods consist of a tubular zirconium alloy fuel sheath, which is closed at each end and filled with uranium dioxide fuel pellets. The U_{235} content of the fuel is in the range of 3.5% to 4%. Natural uranium contains approximately 0.7% U_{235}. A typical PWR fuel assembly is shown in Figure A-2. The control rods, arranged in clusters, penetrate guide tubes that occupy some of the fuel element locations in the lattice.



A fuel assembly consists of a square array of 179 to 264 fuel ro
and 121 to 193 fuel assemblies are loaded into an individual re
After the nuclear fuel is used at nuclear power plants, it can be
as recycled fuel through chemical processing at a reprocessing

Basic specification of PWR fuel assembly

Type	14×14		15×15	17×17
	10ft	12ft	12ft	12ft
Cross-section size (mm square)	197		214	214
Fuel assembly length (mm)	3,473	4,057	4,057	4,058

Figure A-2, PWR Fuel Assembly

Traditionally, PWRs except those designed by Combustion Engineering, utilize from two (2) to four (4) 'loops' in the reactor coolant system, with each loop containing a reactor coolant pump and a Steam Generator. The capacity of each loop is between 300 MWe and 400 MWe. The minimum number of two (2) loops is determined by safety considerations. A typical two (2) loop PWR reactor coolant system is shown in Figure A-3.

Combustion Engineering has standardized on a two (2) loop reactor coolant system configuration, increasing the Steam Generator size as required to match the reactor power. The largest CE reactors in service (at Palo Verde) have one Steam Generators and two (2) reactor coolant pumps in each of two (2) loops. Westinghouse has adopted this general configuration for the AP1000.

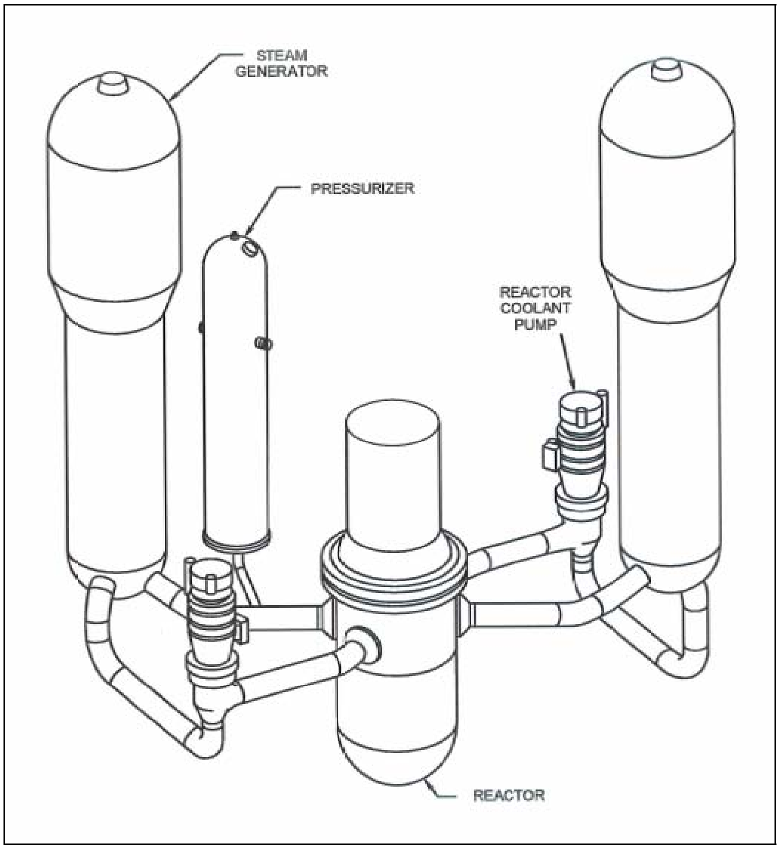


Figure A-3, Typical 2-Loop PWR Reactor Coolant System Arrangement

A typical PWR configuration is presented in Figure A-4. The PWR reactor and reactor coolant system is housed within a robust containment system that prevents the release of radioactivity into the environment in the event of a reactor coolant system failure (Loss of Coolant Accident), and protects the reactor and reactor coolant system from external events (e.g., aircraft crash).

Most modern PWR containment structures consist of vertical post tensioned concrete cylinders with hemispherical domes that are steel lined, and employ a separate reinforced concrete structure surrounding the containment that provides protection from external events. The AP1000 has a cylindrical steel containment structure that is housed within a reinforced concrete protective structure (shield building). The AP1000 incorporates passive post Loss of Coolant Accident (LOCA) and containment cooling.

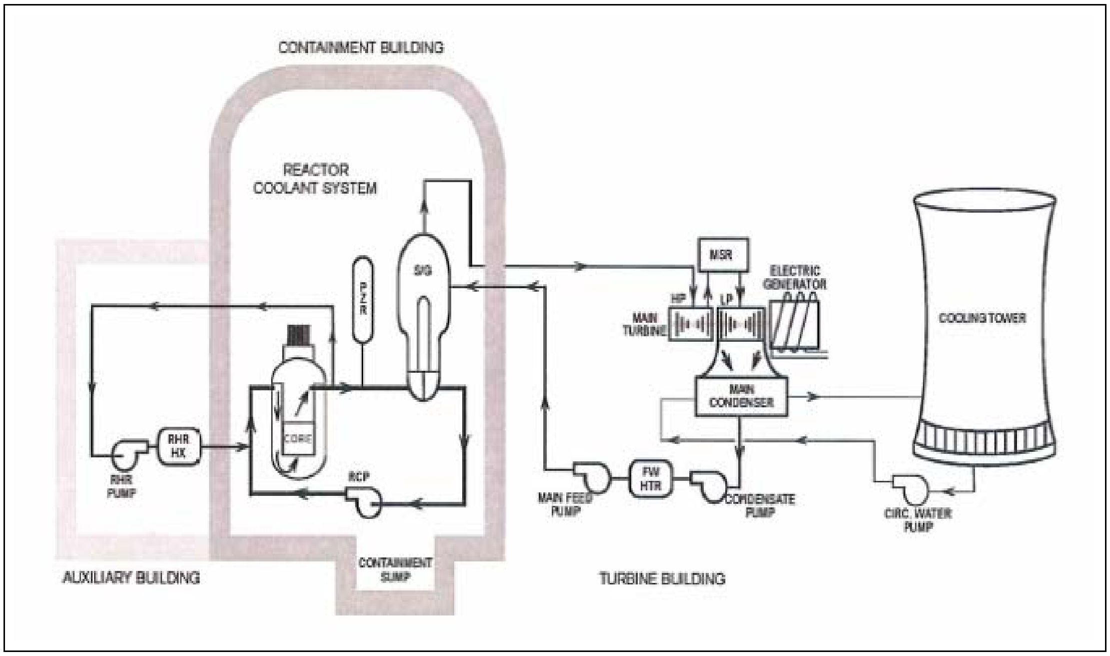


Figure A-4, Typical PWR Nuclear Power Plant Configuration

The configuration in which the reactor coolant in the pressure vessel serves as the moderator results in PWR reactors having a negative void reactivity coefficient (i.e., reactor power decreases when reactor coolant is lost). This situation occurs largely because the coolant and moderator are both lost simultaneously. PWRs have a negative temperature reactivity coefficient (i.e., power increases with dropping temperature).

CANDU Reactor NPPs

The CANDU reactor is a variation of the indirect cycle PWR concept, in which the fuel and coolant within the reactor assembly are housed in a large number of horizontal ‘pressure tubes’ instead of a large pressure vessel, as illustrated in Figure A-5. The pressure tubes are approximately 103 mm inside diameter and have an in-core length of 6 meters. The CANDU 6E has 380 pressure tubes. CANDU fuel bundles are approximately 500 mm long and 100 mm in diameter, and consist of 28, 37 or 41 fuel elements. Each fuel element is contained in a tubular zirconium alloy fuel sheath which is closed at each end and filled with uranium dioxide fuel pellets. All operating CANDU plants utilize natural uranium fuel. There are 12 fuel bundles in each fuel channel. The heavy water reactor coolant is separated from the low pressure heavy water moderator that surrounds the pressure tubes. The reactor coolant pumps circulate the reactor coolant through the pressure tubes to remove the heat of fission, and through Steam Generators (as is the case with PWRs) to generate steam that subsequently drives a turbine. Boiling in the pressure tubes which results in a quality of up to

4% is permitted in the CANDU 6 and CANDU 6E. The general configuration of a CANDU nuclear power plant is the same as for a PWR, as shown in Figure A-4.

The advantage of a heavy water coolant and moderator is neutron economy. Since heavy water does not absorb a significant number of neutrons, CANDU reactors can operate on natural uranium and other low fissile content fuels. The principal disadvantages of heavy water coolant and moderator are the high cost of heavy water (greater than $300/kg), a relatively long thermalization distance (about 10 times that of PWRs) resulting in large reactor size, moderator and coolant activation (tritium is produced), and the complexity of on-power refueling.

The configuration in which the reactor heavy water coolant is separated from the heavy water moderator results in traditional CANDU reactors having a strong positive void reactivity coefficient (i.e., reactor power increases when reactor coolant is lost). This situation occurs largely because the moderator is not lost simultaneously with the coolant. CANDU reactors have a near zero reactivity temperature coefficient.

Current CANDU 6 plants have a post-tensioned, reinforced concrete epoxy lined containment that utilizes a passive light water dousing system to limit peak pressure in the event of a loss of coolant accident. The CANDU 6E adopts the conventional PWR approach, utilizing a steel lined, post-tensioned concrete containment without dousing.

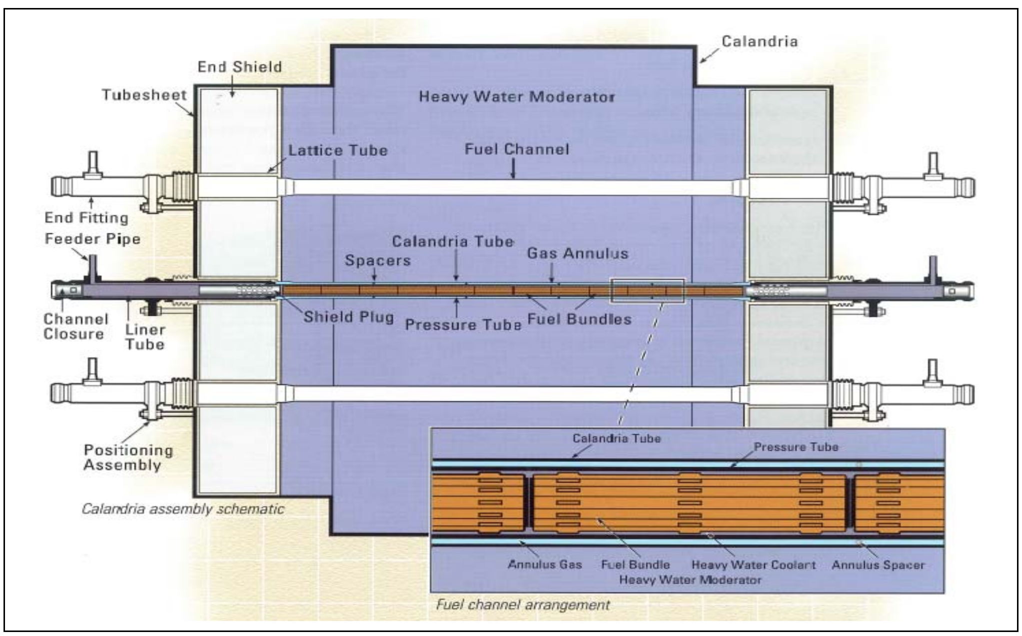


Figure A-5, CANDU / ACR Reactor Configuration

ACR-1000 Nuclear Power Plant

The ACR-1000 is a variation of the basic CANDU design, utilizing a horizontal pressure tube reactor configuration as illustrated in Figure A-5. A major difference from traditional CANDU reactors is the use of light water coolant instead of heavy water. Since light water has a much higher neutron absorption cross-section than heavy water, the ACR-1000 (similar to PWRs) requires enriched fuel. In addition, the neutron absorption in the light water coolant results in a very high and unacceptable positive void reactivity coefficient (i.e., reactor power increases rapidly on loss of coolant) if utilized in a traditional CANDU reactor configuration. Hence, the ACR-1000 introduces a number of design changes that reduce the void reactivity coefficient to near zero. These include reducing the fuel channel spacing and increasing the annulus gap between the fuel channel pressure tube and Calandria tube (both of which reduce the heavy water moderator volume), and adding dysprosium (a neutron absorber or poison) to the fuel. The latter increases the amount of fuel enrichment required.

The ACR-1000 fuel bundle with 43 fuel elements has an external configuration similar to the CANDU 6E 41 element fuel bundle. However, the uranium oxide fuel pellets are enriched to approximately 2.7% U_{235}, and dysprosium (neutron absorber) is included in the central element.

The traditional CANDU disadvantages of tritium generation in the moderator and the need for on-power refueling are retained by the ACR-1000. However, the reactor size is reduced relative to conventional CANDU plants of the same output as a result of reduced spacing of the fuel channels, which reduces the volume of heavy water moderator required. The advantage of light water coolant is that the cost of heavy water coolant is avoided, as is the cost of the facilities required to maintain the heavy water coolant.

Boiling Water Reactor (BWR) NPPs

As is the case with PWRs, BWRs utilize a common, light water coolant and moderator and house the fuel and the common coolant/moderator within a large pressure vessel. However, BWRs allow boiling in the reactor vessel, with qualities in the range of 20% at the core outlet. Steam separators are incorporated into the top section of the pressure vessel to separate the steam from the water fraction. A typical BWR pressure vessel is shown in Figure A-6. The BWR 6, shown in the illustration utilized external recirculation pumps to return water from the steam separators to the bottom of the reactor core. The external pumps were replaced by canned recirculation pumps that penetrate the lower head of the pressure vessel on the ABWR, and recirculation pumps are eliminated on the ESBWR which utilizes natural circulation.

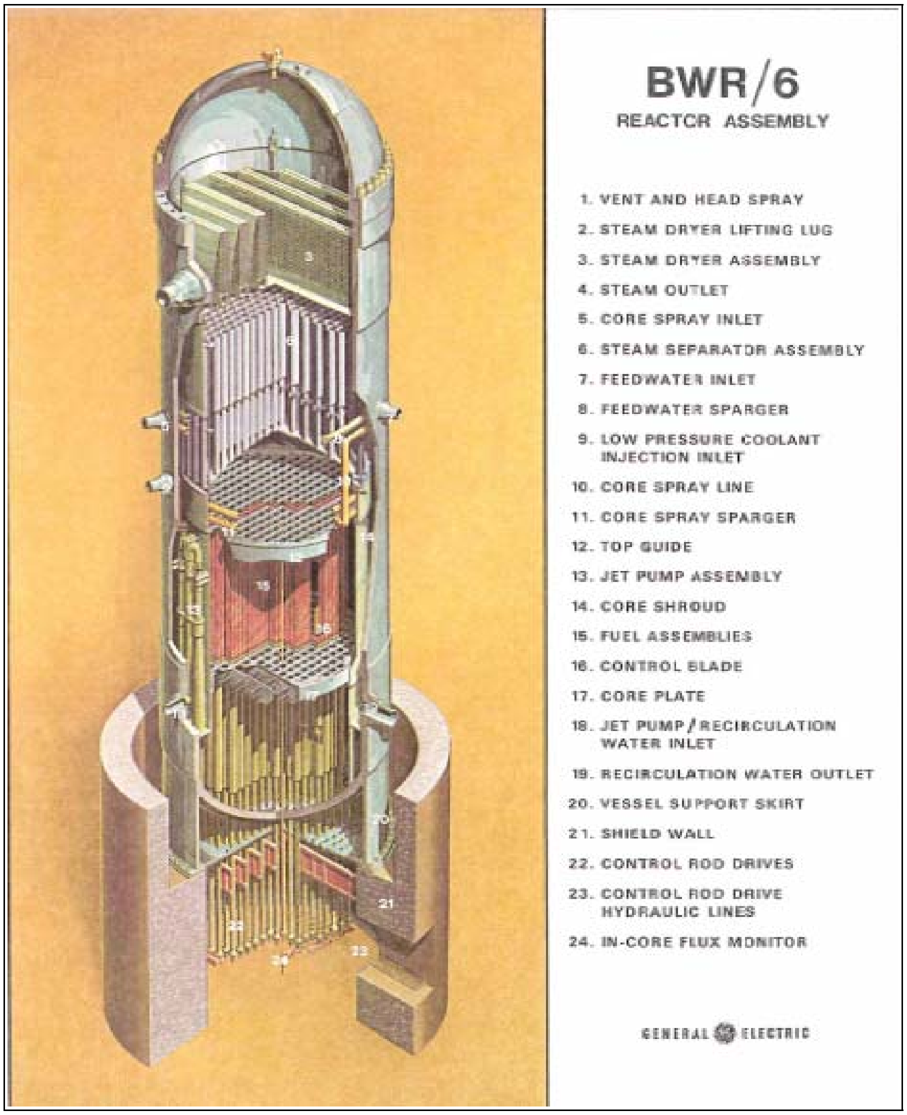


Figure A-6, Typical BWR Pressure Vessel

A typical BWR fuel assembly is shown in Figure A-7. The fuel elements consist of a tubular zirconium alloy fuel sheath which is closed at each end and filled with uranium dioxide fuel pellets. The U_{235} content of the fuel is in the range of 3.7%. Natural uranium contains about 0.7% U_{235}. The fuel assemblies are housed within channels of square cross-section that assure flow stability in the core. The shut-off rods have a cruciform shape, and are inserted between the fuel assemblies utilizing hydraulic drives that are located below the reactor pressure vessel.

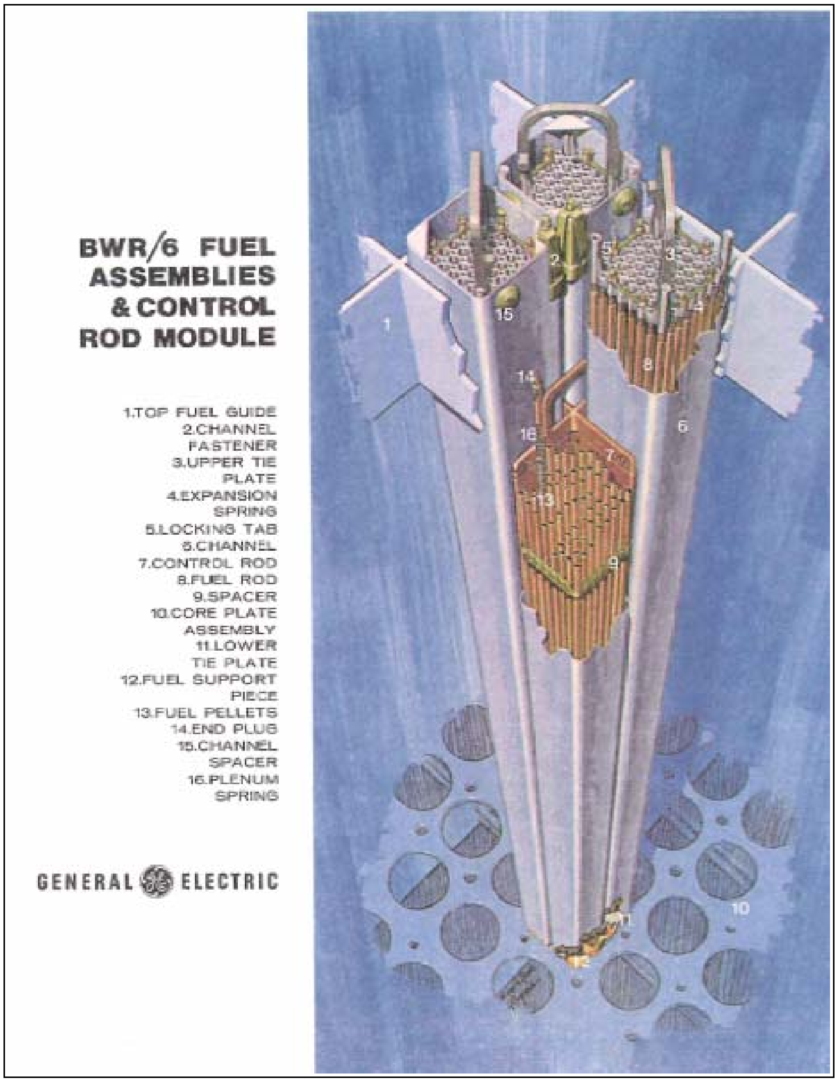


Figure A-7, Typical BWR Fuel Assembly

The steam produced in the reactor core, after its separation by the steam separators, passes through driers and is directed to the turbine which drives the generator. After being condensed by the vacuum condenser, the steam condensate is returned to the reactor. This is referred to as a direct cycle. The general arrangement of a BWR nuclear power plant is illustrated in Figure A-8.

The BWR reactor and is housed within a robust containment system to prevent the release of activity to the environment in the event of a reactor coolant system failure. Fast acting valves at the containment boundary close in the event of a steam pipe failure outside of the containment. BWR containments, which have evolved in design over the years, are typically steel and of small size relative to the containments of PWRs. This is facilitated by the absence of Steam Generators. To prevent over pressurization of the containment in the event of a loss of coolant accident the escaping water/steam from the break is vented directly to a suppression pool located within the containment. Steel BWR containments are provided with a robust concrete surrounding structure to protect the containment and the reactor systems from external events. The ESBWR incorporates passive post Loss of Coolant Accident (LOCA) and containment cooling.

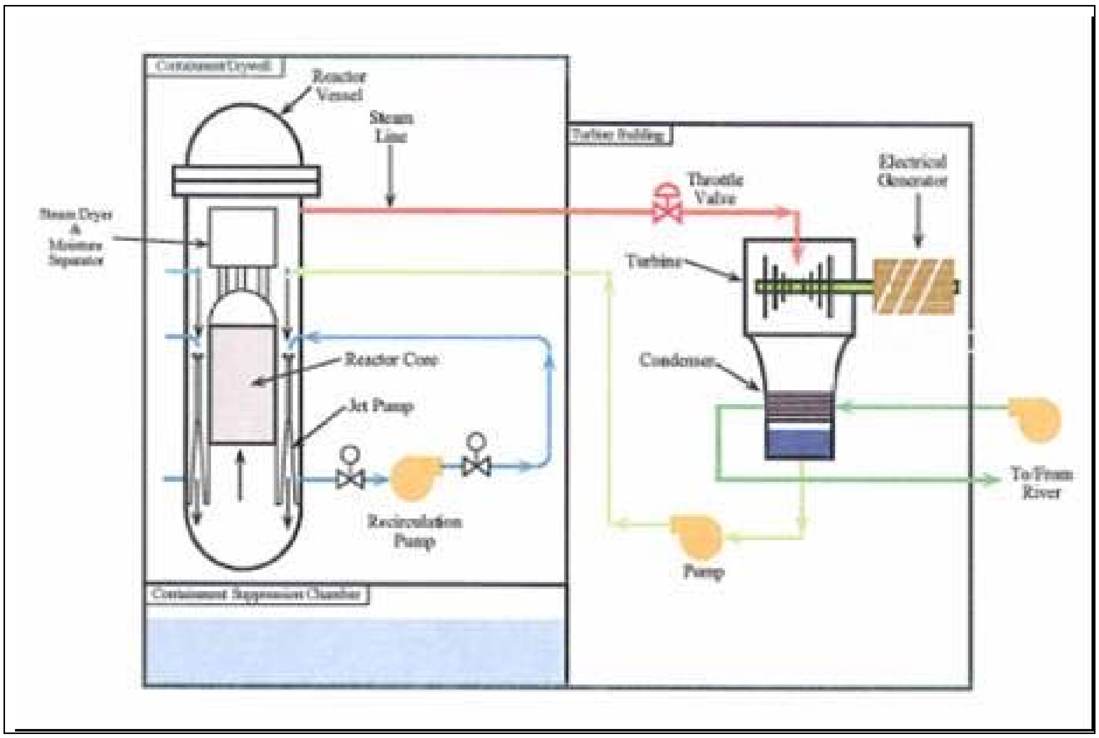


Figure A-8, BWR Nuclear Power Plant Configuration

The configuration in which the reactor coolant in the pressure vessel serve as the moderator results in BWR reactors having a negative void reactivity coefficient (i.e., reactor power decreases when reactor coolant is lost). This situation occurs largely because the coolant and moderator are lost simultaneously. BWRs have a negative temperature reactivity coefficient (i.e., power increases with dropping temperature).

The principal advantage of a BWR relative to a PWR is the elimination of Steam Generators. The disadvantages include a much taller pressure vessel, and radioactive steam lines and turbine during plant operation that results in a requirement for extensive shielding. The principal source of radioactivity in the steam is N_{16} which has a short half life (a few seconds), so this activity is not a problem during reactor shutdown. Historically, BWRs had lower capacity factors than PWRs. However, during the last three (3) year reporting period for US reactors, BWRs have equaled the PWR performance.

High Temperature Gas Reactor (HTGR) NPPs

Early development of the High Temperature Gas Reactor (HTGR) was undertaken during the OECD Dragon project, which began in 1959 and involved the participation of thirteen nations. The Dragon project resulted in the development of the first BISO (two coatings) and TRISO (three coatings) particle fuels, and in the world's first prismatic type HTGR (Dragon) that was constructed at the Winfrith Atomic Establishment in England. Subsequent development of the HTGR was centered in Germany and the USA from the late 1960s through the 1980s, with demonstration and commercial units being constructed in both countries.

Both the German and US modular HTGR concepts developed in the 1980s and those currently under development take advantage of the TRISO fuel particle (see Figure A-9). The fuel particles are still referred to as TRISO, although a fourth coating has been added. They have an outside diameter of less than one mm, and consist of a uranium, plutonium oxide, or oxycarbide kernel with four coatings. A porous, pyrolytic carbon inner layer accommodates the fission gases and the fission product recoil. The high density inner pyrocarbon layer protects the kernel during the application of the silicon carbide layer, and serves both as a barrier to fission product gases and a secondary structural element for internal pressure. The high density outer pyrocarbon layer protects the silicon carbide layer during fuel element pressing, and also serves both as a barrier to fission product gases and a secondary structural element for internal pressure. The silicon carbide coating serves as the primary barrier against the diffusion of metallic fission products, and as the primary structural element for internal pressure retention. The radionuclide retaining capability of the TRISO particle is maintained up to very high temperatures, with 1600°C typically used as a design limit under accident conditions.

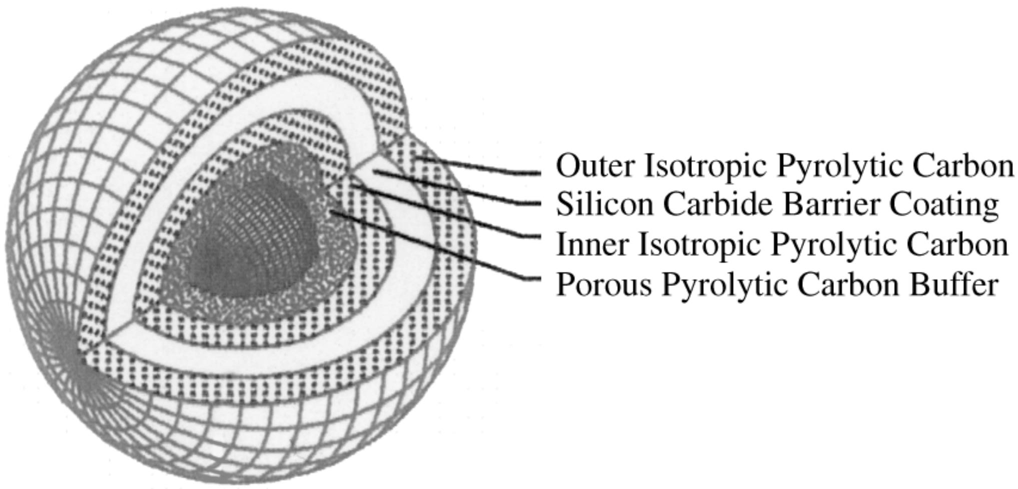


Figure A-9, TRISO Fuel Particle

Inherent shutdown is primarily achieved in modular HTGRs by the strong negative reactivity temperature coefficient of the graphite moderator. Passive fuel cooling is provided following postulated accident conditions by the conduction and radiation of heat from the high temperature capability TRISO fuel to the pressure vessel surroundings. This requirement, in conjunction with fuel and pressure vessel temperature limits, constrains the size of prismatic core modular HTGRs to approximately 600 MW_{th} and of pebble bed HTGRs to about 500 MWth, and also limits the coolant core outlet temperature to 850°C to 950°C.

Thousands of TRISO particles are housed within a graphite matrix to serve as fuel for the HTGR. In the Pebble Bed Modular Reactor (PBMR), the TRISO particles are contained within billiard ball size pebbles, while in the GA PRISMATIC CORE design the TRISO particles are contained within small cylindrical compacts (see Figure A-10) that are located in columns within the prismatic graphite fuel blocks (see Figure A-11).

The PBMR is refueled on-power. New fuel pebbles are introduced at the top of the reactor core, while depleted fuel pebbles are removed from the bottom of the core. In order to maintain uniform reactivity over the length of the core, fuel pebbles are recirculated by returning non-depleted pebbles from the bottom of the core to the top of the core. Each pebble makes approximately eight passes through the core. In contrast, the GA prismatic design is refueled off-power. Typically, only 1/3 of the prismatic fuel blocks are exchanged during a refueling outage. Refueling outages typically take place at two year intervals. Since all nuclear power plants must shut down periodically for inspection and maintenance, it is not anticipated that on-power fueling capability will significantly increase PBMR plant availability. However, it does complicate plant design and operation.

Figure A-10, TRISO Particles & Compacts (GA)

Figure A-11, TRISO Particles & Compacts (GA)

Both the PBMR and the GA-HTGR utilize a steel pressure vessel to house the reactor core, and provide the reactor coolant pressure boundary. Due to the use of graphite moderator, HTGR pressure vessels are relatively large in comparison with PWR pressure vessels for a reactor of the same capacity. Figure A-12 shows a typical HTGR core cross section. The dark annular portion is the active/fueled part of the core. In the PBMR, this consists of fuel pebbles, while in the GA design it consists of prismatic fuel blocks. The central graphite column facilitates increased reactor size by minimizing the distance from the innermost TRISO fuel particle to the pressure vessel wall, and acts as a heat sink during the early stage of a loss of coolant accident.

The HTGRs have a zero void reactivity coefficient as the helium coolant does not absorb neutrons.

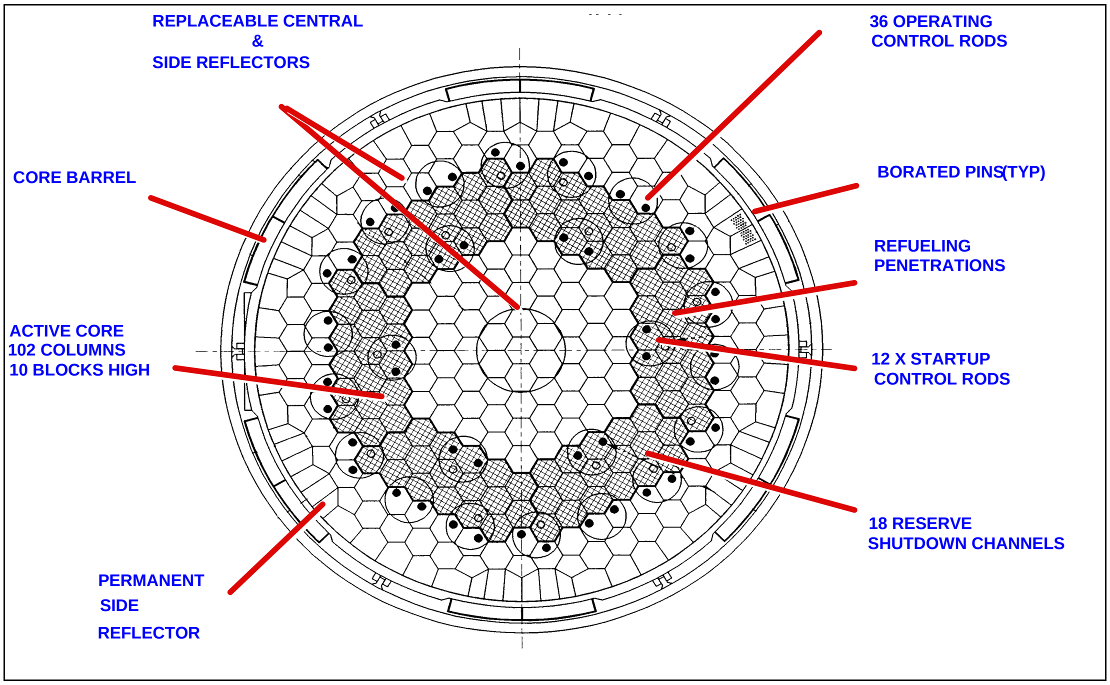


Figure A-12, HTGR Core Arrangement

Current HTGR designs offer the safety advantages of inherent shutdown based on the negative temperature reactivity coefficient (no shutdown systems are required for ultimate safety) and the inherent ability to reject decay heat to the environment without the need for active systems.

The primary containment system for fission products is provided by the TRISO fuel particle coatings. However, civil structures are required to protect the reactor from external events such as aircraft crashes. The GA direct cycle HTGR building arrangement is shown in Figure A-13. A Steam Generator replaces the power conversion system for steam production in the reference configuration.

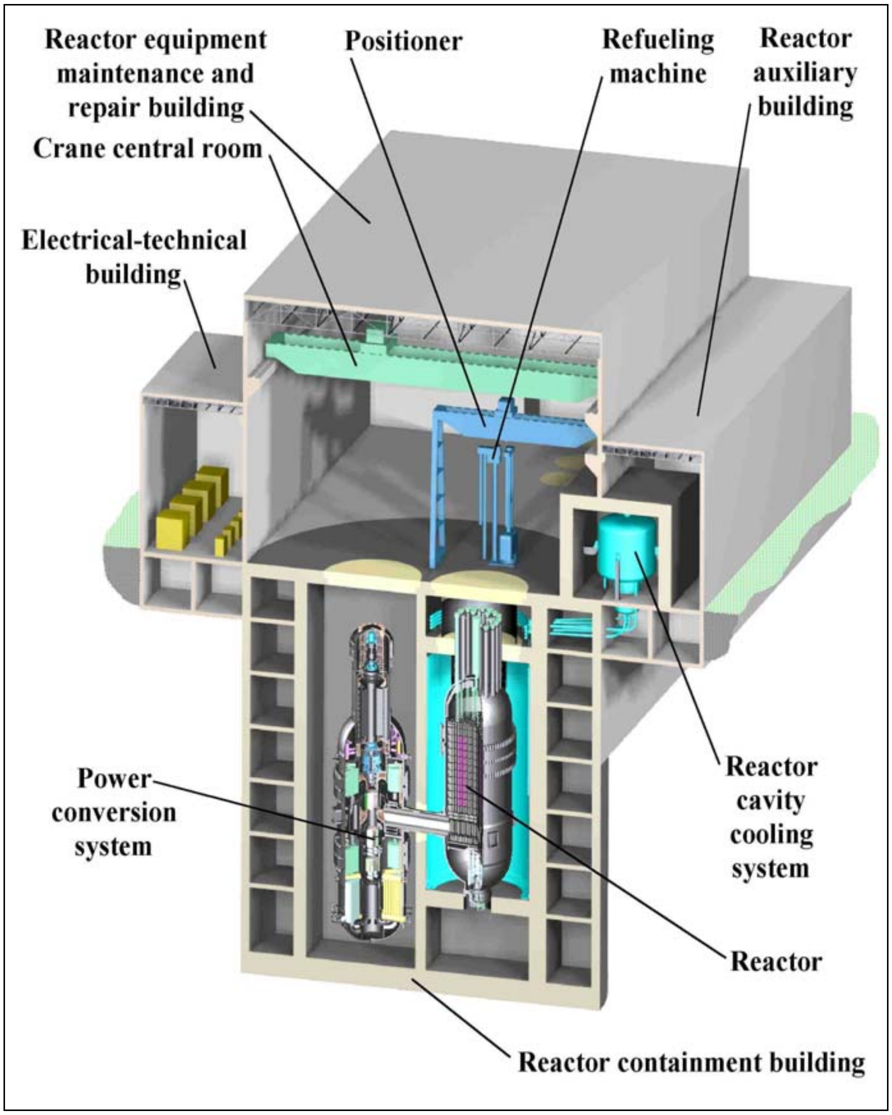


Figure A-13, GA Direct Cycle HTGR Layout

HTGRs operate at high temperatures, with core outlet temperatures ranging from 850 °C to 950 °C. This facilitates high pressure steam generation (above 17 MPa) with substantial superheat that is suitable for serving a range of process heat applications, and for electricity generation utilizing a direct cycle. The latter is the focus of current efforts at GA and PBMR. High temperature process heat applications are the focus of HTGR design efforts in China and Japan. Both Japan and China operate HTGR research reactors, and China has a commercial HTGR under construction with a projected in-service date of 2010.

HTGR development by GA and PBMR is focused on direct cycle designs in which the helium coolant from the reactor is passed through a turbine to drive a generator. As shown in Figure A-14, this configuration has the potential of achieving thermodynamic efficiencies that are approximately 50% higher than those available with water cooled reactors. A typical direct cycle configuration is shown in Figure A-15.

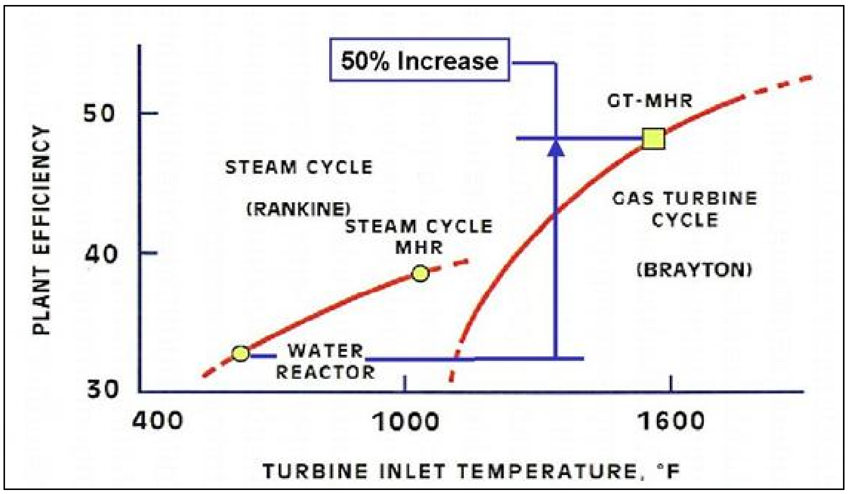


Figure A-14, Relative NPP Thermodynamic Efficiencies

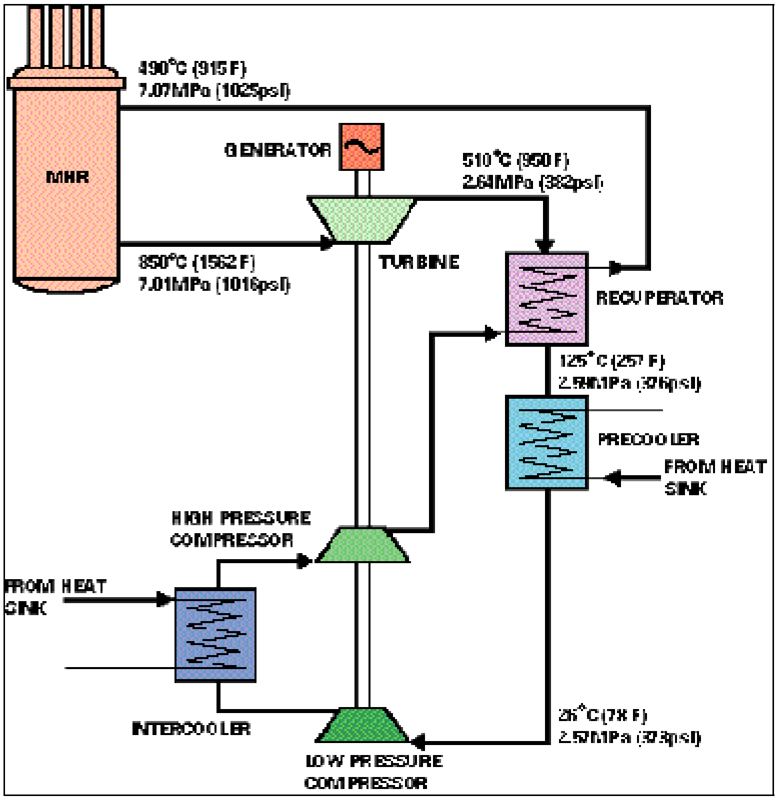


Figure A-15, Typical Direct Cycle HTGR Arrangement

Table A-1, HTGR Plants Constructed & Operated

Feature	Dragon	Peach Bottom	AVR	Fort St. Vrain	THTR	HTTR	HTR-10
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MWt/MWe)	20/ -	115/40	46/15	842/330	750/300	30/ -	10/ -
Fuel Elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He Temp (In/Out°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He Press (Bar)	20	22.5	11	48	40	40	20
Pwr Density (MW/m^3)	14	8.3	2.3	6.3	6	2.5	2
Fuel Coating	TRISO	BISO	BISO	TRISO	BISO	TRISO	TRISO
Fuel Kernel	Carbide	Carbide	Oxide	Carbide	Oxide	Oxide	Oxide
Fuel Enrichment	LEU/ HEU	HEU	HEU	HEU	HEU	LEU	LEU
Reactor Vessel	Steel	Steel	Steel	PCRV	PCRV	Steel	Steel
Operation Years	1965-1975	1967-1974	1968-1988	1979-1989	1985-1989	1998-	1998-

Appendix B: ABWR Technical Summary

Introduction

The Advanced Boiling Water Reactor (ABWR) developed by General Electric was the first of the next generation advanced light water reactor (ALWR) plants to be constructed. Four units are now in operation; two in Japan and two in Taiwan. The information in this Appendix was provided by General Electric.

The general arrangement of the ABWR is illustrated in Figure B-1.

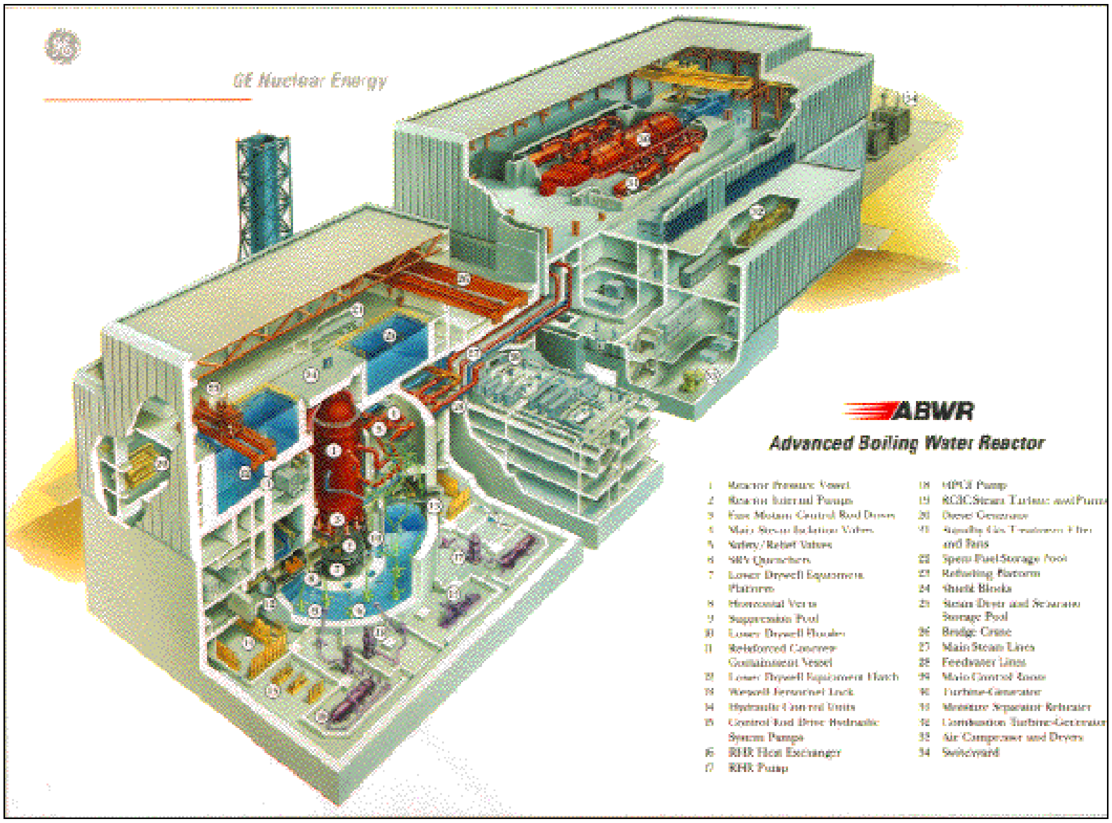


Figure B-1, ABWR Station General Arrangement

ABWR Technical Data

a) POWER

1) Net electrical output: 1356 MWe
2) Gross thermal power: 3926 MWt

b) REACTOR CORE

1) Active height: 3.71 m
2) Active diameter: 5.16 m
3) Number of fuel elements: 872
4) Average power rating: 196 W/cm
5) Average core power density: 50.6 W/litre

c) FUEL ASSEMBLIES

1) Fuel material: UO2, UO2-Gd2O3
2) Average reload enrichment: 3.2 %
3) Number of rods per assembly: 62
4) Fuel rod diameter: 12.3 mm
5) Cladding material: Zircaloy 2
6) Cladding thickness: 0.86 mm
7) Fuel discharge burnup: 32,000 MWd/t
8) (equilibrium) reference case

d) CONTROL SYSTEM

1) Number of control rods: 205
2) Form of control rods: Cruciform

3) Neutron absorber: B_4C

4) Control rod drive motion: Electric, fine motion

5) Hydraulic: scram

6) Burnable poison: Gd_2O_3

e) PRIMARY COOLANT SYSTEM

1) Type: internal recirculation, pump system

2) Operating pressure: 73.1 kg/cm^2

3) Feedwater inlet temperature: 215.5 °C

4) Steam outlet temperature: 287.4 °C

5) Number of recirculation pumps: 10

6) Recirculation mass flow:52,200 t/hr

7) 100% rated

f) REACTOR PRESSURE VESSEL

1) Internal height: 21 m

2) Internal diameter: 7.1 m

3) Wall thickness minimum: 174 mm

4) Materials: Low alloy steel/stainless steel cladding

g) CONTAINMENT

1) Type: Reinforced concrete containment vessel (steel lined)

2) Design pressure: 3.16 kg/cm^2

3) Height: 36.1 m

4) Inside diameter (maximum): 29 m

h) TURBINE

1) Number: 1
2) Maximum rating (@ 722 mm Hg): 1381 MWe
3) Speed: 1500 rpm/1800 rpm
4) Turbine inlet pressure: 69.2 kg/cm^2
5) Turbine inlet temperature: 283.7 C

ABWR Reactor Pressure Vessel

The ABWR reactor pressure vessel (see Figure B-2) is 21 meters high and 7.1 meters in diameter, and is designed for an operating life of 60 years.

Much of the vessel, including the four (4) vessel rings from the core beltline to the bottom head, is made from a single forging. The vessel has no nozzles greater than 2 inches in diameter anywhere below the top of the core, as the external recirculation loops have been eliminated. Because of these two (2) features, over 50% of the welds and all of the piping and pipe supports in the primary system have been eliminated, and along with it the greatest source of occupational exposure in the BWR.

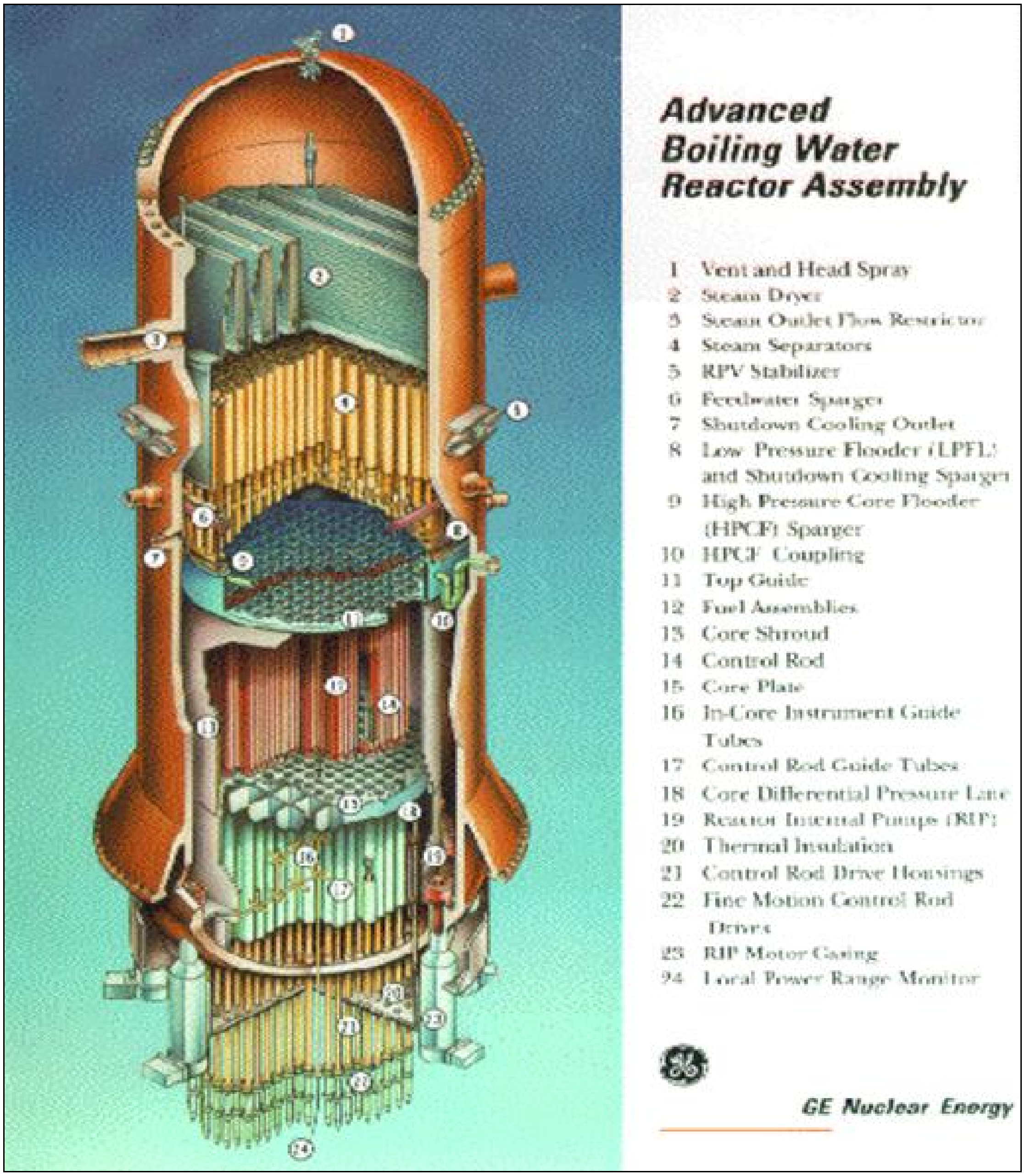


Figure B-2, ABWR Assembly

Materials & Water Chemistry

By utilizing their 30 years of experience in operating BWR reactors, GE has taken special care in selecting the appropriate material. The reactor coolant chemistry specifications have also been refined to better assure component and fuel reliability. Cobalt has been eliminated from the design. The steel used in the primary system is made of nuclear grade materials (low carbon alloys), which are resistant to intergranular stress corrosion cracking.

External Recirculation System Eliminated

One of the unique features of the ABWR is its external recirculation system elimination. The external recirculation pumps and piping have been replaced by ten internal recirculation pumps mounted to the bottom head. These Reactor Internal Pumps (RIPs) are improved versions of those used in Europe for which there are over 1000 pump years of operating experience. The reliability and durability of these pumps has proven to be so high that only two (2) will be removed for servicing during an outage. The RIP motors are continuously purged with clean water to keep crud in the vessel from settling, such that radiation levels surrounding the pumps are vastly reduced.

Fine Motion Control Rod Drives

Fine Motion Control Rod Drives (FMCRD) are being introduced in the ABWR. Day-to-day operation is performed with an electric stepping motor which moves the drive in 0.75" increments, compared to the Locking Piston Drive which had 3" increments (hence the name ‘fine motion’). The control rods are scrammed hydraulically but can also be scrammed by the electric motor as a backup. The FMCRD are so reliable, it is not necessary to inspect all of them during the lifetime of the plant. Therefore, only three (3) drives will be removed for inspection during outages, which represents a huge time saving. Typically, 30 Locking Piston Drives are removed every outage. The FMCRDs have continuous clean water purging to keep the radiation levels very low.

Digital Control & Instrumentation Systems

The Control and Instrumentation (C&I) systems use state of the art digital and fiber optic technologies. The ABWR has four (4) separate divisions of safety system logic and control, including four (4) separate, redundant multiplexing networks to provide absolute assurance of plant safety. Each system includes microprocessors to process incoming sensor information, and to generate outgoing control signals, local and remote multiplexing units for data transmission, and a network of fiber optic cables. The controllers are fault tolerant, as they continually generate signals to simulate input data and compare the results against the expected outputs. Controllers for both sensors and equipment are located on cards which are remotely distributed. Should a problem be detected by the controller, a signal will be sent to the Main Control Room. Within minutes, the malfunctioning card can be replaced with a spare.

Multiplexing & Fiber Optics

Multiplexing and fiber optics have dramatically reduced the amount of cabling in the plant. This has another benefit, as it shortens the critical path for the construction schedule by one month.

Control Room Design

The entire plant can be controlled from one (1) console. The panels in the centre control non-safety systems in the Nuclear Island. The panel on the left controls the safety systems, and those on the right control the balance of plant. The CRTs allow the operator to call up any system or its subsystems and components just by touching the screen. It is possible to operate an entire system by means of a system master command.

Plant Layout

ABWR is designed to envelop the site conditions, which covers almost all of the available nuclear sites in the world, including the sites with high seismic potential.

The reactor and turbine building are arranged ‘in-line’, and none of the major facilities are shared with the other units. The containment is a reinforced concrete containment vessel (RCCV) with a leak tight steel lining. The containment is surrounded by the reactor building, which doubles as a secondary containment. A negative pressure is maintained in the reactor building to direct any radioactive release from the containment to a gas treatment system. The reactor building and the containment are integrated to improve the seismic response of the building without an additional increase in the size and load bearing capability of the walls.

Construction of the plant will make use of large modules which are prefabricated in the factory and assembled on site. A 1000 ton crawler crane can lift the modules and place them vertically into the plant. Use of RCCV, modular construction and other construction techniques serve to reduce the construction time from 66 to 50 months.

Particular attention was paid to designing the plant for ease of maintenance. Monorails are available to remove equipment to a conveniently located service room through an equipment hatch.

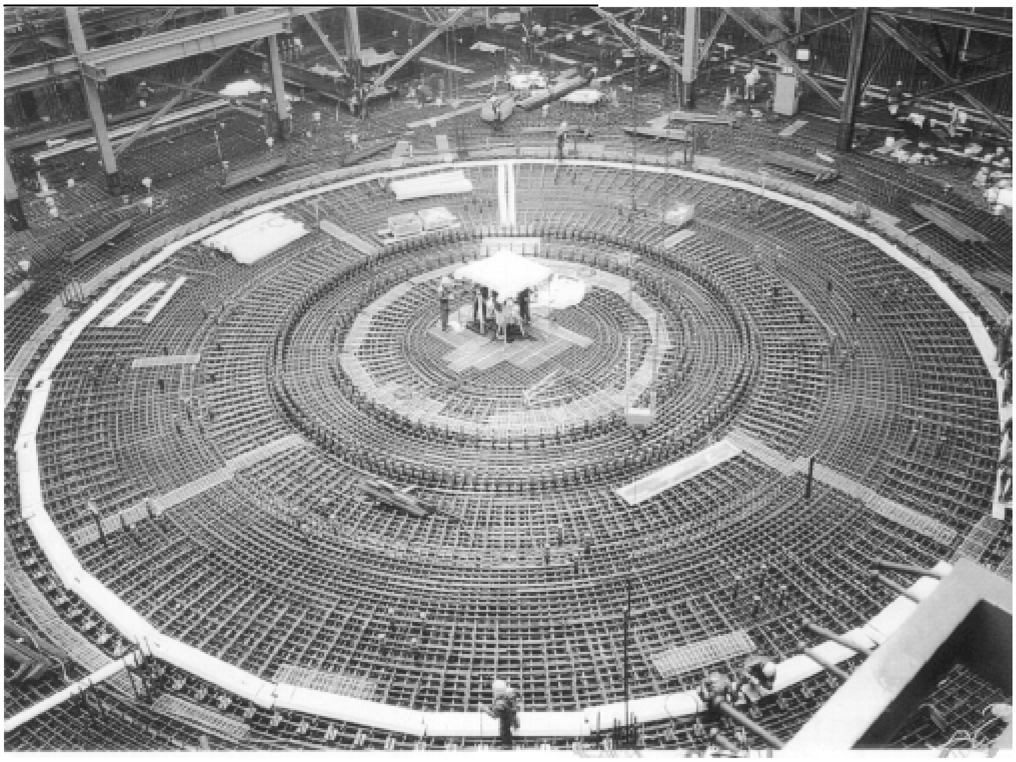
Removal of the Reactor Internal Pumps and FMCRDs for servicing has been automated. Handling devices, which in the case of the FMCRD are operated remotely from outside the containment to engage and remove the equipment. The pump or driver is laid on a transport device and removed through the equipment hatch. Just outside the hatch are the dedicated service rooms. One room is for the RIPs and the other is for the FMCRDs, where the equipment can be decontaminated and serviced within a shielded environment. The entire operation is completed efficiently and with virtually no radiation exposure to personnel.

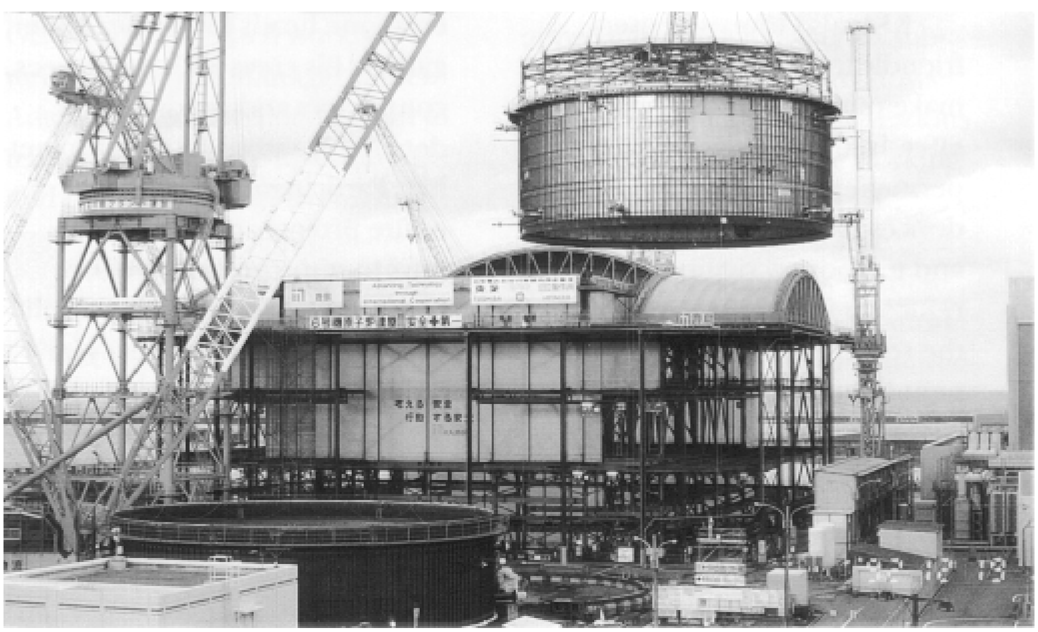
Simplified Active Safety Systems

Another unique feature of the ABWR is its simplified active safety system. The ABWR has three (3) completely independent and redundant divisions of safety systems. The systems are mechanically separated and have no cross connections as in earlier BWRs. They are electronically separated so that each division has access to redundant sources of AC power, and for added safety, its own dedicated emergency diesel generator. Divisions are physically separated. Each division is located in a different quadrant of the reactor building, and separated by firewalls. A fire, flood or loss of power which disables one (1) division has no effect on the capability of other safety systems. Finally, each division contains both a high and low pressure system, and each system has its own dedicated heat exchanger to control core cooling and remove decay heat. One of the high pressure systems, the Reactor Core Isolation Cooling (RCIC) system, is powered by reactor steam and provides the diverse protection needed should there be a station blackout.

The safety systems have the capability to keep the core covered at all times. Because of this capability and the generous thermal margins built into the fuel designs, the frequency of transients which will lead to a scram and therefore a plant shutdown have been greatly reduced (to less than one per year). In the event of a Loss of Coolant Accident, plant response has been fully automated so that operator action is not required for 72 hours, which is the same capability as passive plants.

The following images show an ABWR under construction in Japan.





Appendix C: ACR-1000 Technical Summary

An overview and technical summary of the ACR-1000 reactor design is provided in the following pages.

ACR-1000® Technical Summary

AECL

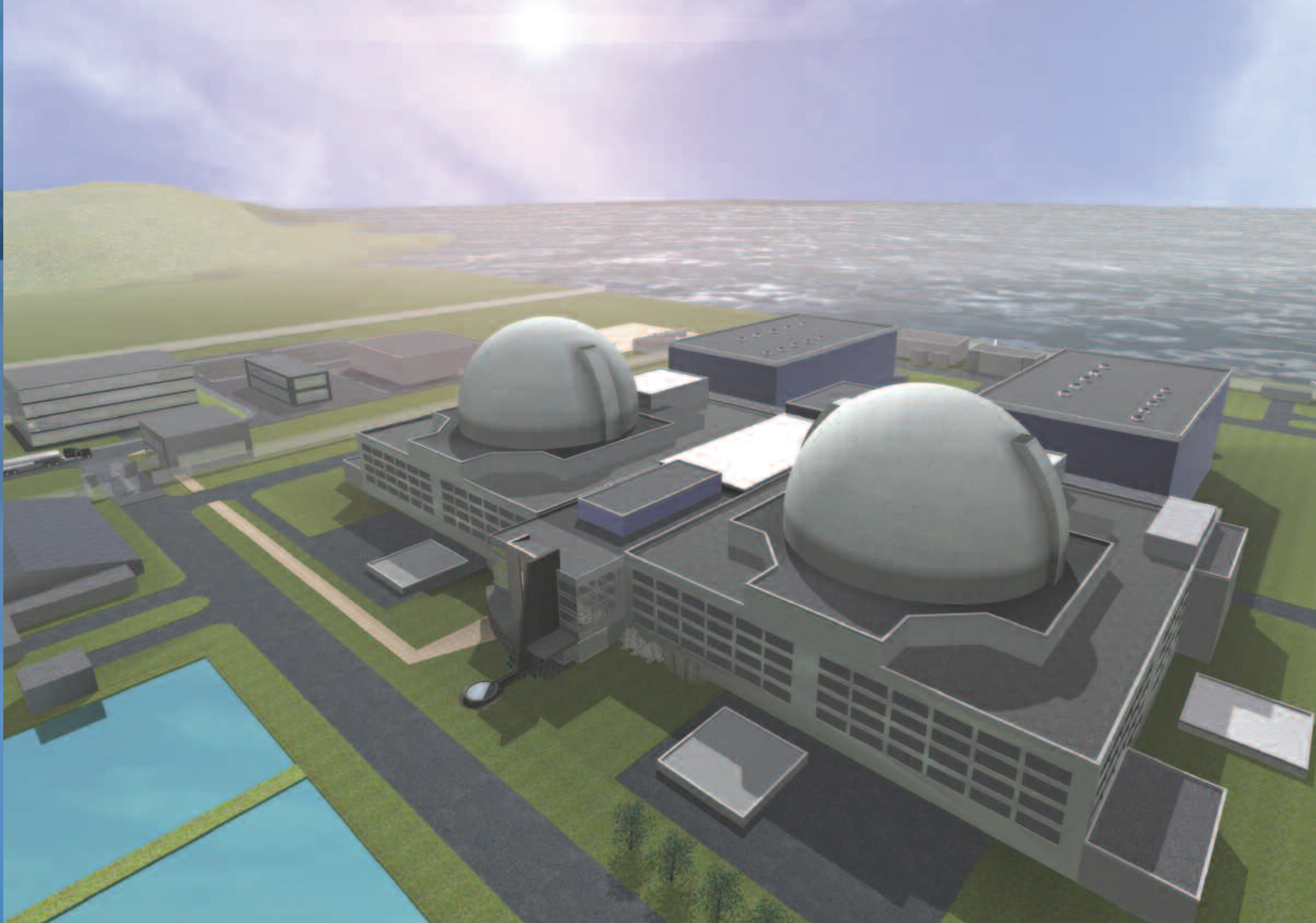


Figure S-1 Pictorial View of Two-Unit ACR-1000 Plant

Summary

The ACR-1000®* is an evolutionary, Gen III+**, 1200 MWe class pressure tube reactor, designed to meet industry and public expectations for safe, reliable, environmentally friendly, low-cost nuclear power generation.

The reactor core consists of fuel and light-water coolant in pressure tubes with a heavy water moderator. Derived from the well-established CANDU*** line of reactors, the ACR-1000 was developed from valuable project-based experience in the design, construction and operation of CANDU plants for utilities around the globe.

The ACR-1000 retains basic CANDU design features such as: modular, horizontal fuel channel core, low-temperature heavy water moderator, water-filled vault, two diverse shutdown systems, on-power fuelling and an accessible reactor building for on-power maintenance.

To achieve outstanding safety, operation, performance and economics, the ACR-1000 incorporates a specific set of innovative features and state-of-the-art technologies.

Enhanced Safety

- A small, negative coolant void reactivity offers a good balance of nuclear protection between loss-of-coolant accidents and fast cool-down accidents
- Enhanced prevention and mitigating measures for severe accident management, based on insight gained from Probabilistic Safety Analysis (PSA) during the design process
- A strengthened calandria tube providing additional assurance that it will contain a pressure tube failure
- New and improved passive designs for emergency core cooling (ECC), moderator cooling, reactor vault cooling and containment cooling. Design simplifications include sharing of long-term emergency cooling and shutdown safety functions
- Reduced operator decision-making and action workload through state-of-the-art automation and human/machine interface

Improved operation, performance and economics

- Reduction in heavy water inventory by approximately 60% over traditional CANDU reactors, cutting capital costs and improving environmental performance and occupational safety
- Ability to burn alternate fuels such as mixed oxides (MOX) and thorium
- Less refuelling and lower spent fuel volume per MWh, through use of low enriched uranium (LEU) in a CANFLEX®****-ACR fuel bundle, as a result of increased fuel burn-up
- Simplified reactor control resulting from reduced pressure tube lattice pitch and use of LEU fuel for a highly stable, more compact core. Further simplification achieved with mechanical zonal control rods and eliminating the liquid zone control system
- Improved on-power maintenance and testing, additional redundancy in actuating signals for trip channels, reduced risk of spurious trips and overall increased reliability, through use of quadrant-based layout for safety and heat sink systems
- Enhanced power manoeuvring ability due to a lower xenon load after shutdown than in traditional CANDU plants
- Higher overall thermal cycle efficiency, resulting from increased coolant and steam supply pressure and temperature

This document provides a brief description of the main features of an ACR-1000 two-unit plant, including overall plant design, major systems and their key components, and the plans to complete construction of an ACR-1000 within 42 months for the first unit of the n^{th} integrated two-unit plant. AECL experience and services in support of regulatory approvals, operations and final decommissioning are also described.

* ACR-1000® (Advanced CANDU Reactor®) is a registered trademark of Atomic Energy of Canada Limited (AECL).

** Gen III+ is the classification given to nuclear technologies by an international team, including Canada, that is collaborating on the research to develop the next generation, Gen IV reactors. ACR-1000 is one of the technologies that are considered as a generation III+ design.

*** CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

**** CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI).

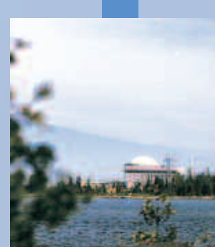
CANDU®: The Evolution

ZEEP
research reactor
10 Watts
Criticality: 1945

NPD
CANDU
demonstration reactor
24 MWe
In-service: 1962

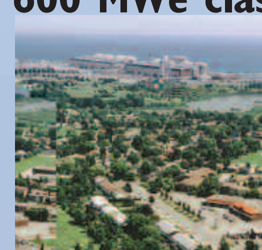
NRU
research reactor
200 MW
Criticality: 1957

NRX
research reactor
42 MW
Criticality: 1947



Douglas Point
CANDU commercial
prototype
220 MWe
In-service: 1968

CANDU
600 MWe class



Pickering A
4 units, 542 MWe
In-service: 1971-73

Pickering B
4 units, 540 MWe
In-service: 1983-86

All figures for operating commercial units indicate gross output.
Source: Nuclear Engineering International (NEI)

CANDU 6® 700 MWe class

Pt. Lepreau 680 MWe
In-service: 1983

Embalse 648 MWe
In-service: 1984

Wolsong Unit 1 679 MWe
In-service: 1983

Wolsong Unit 2 715 MWe
In-service: 1997

Wolsong Unit 3 715 MWe
In-service: 1998

Wolsong Unit 4 715 MWe
In-service: 1999

Gentilly 2 675 MWe
In-service: 1983

Cernavoda Unit 1 708 MWe
In-service: 1996

Cernavoda Unit 2 708 MWe
Projected in-service date: 2007

CANDU 900 MWe class

Bruce A 4 units 900 MWe
In-service: 1977-79

Bruce B 4 units 915 MWe
In-service: 1984-87

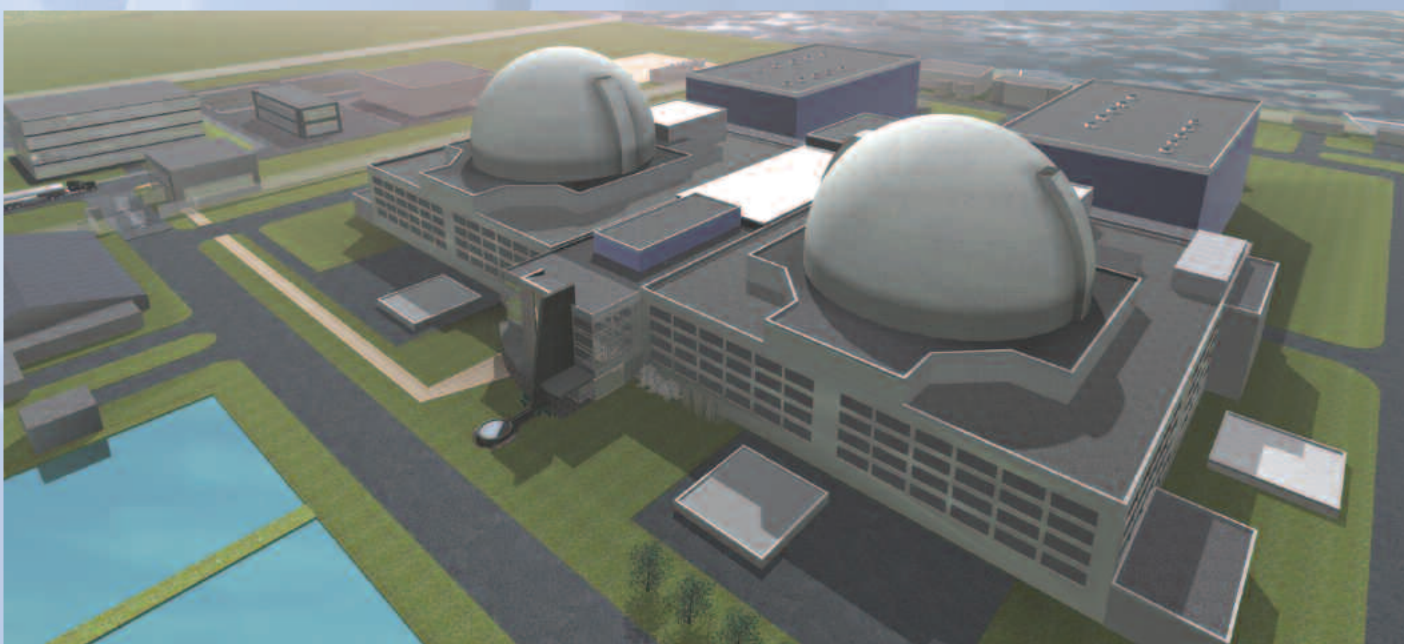
Darlington 4 units 935 MWe
In-service: 1990-93

Figure S-2 Evolution of ACR-1000

Qinshan Phase III Unit 1 728 MWe
In-service: 2002

Qinshan Phase III Unit 2 728 MWe
In-service: 2003

ACR-1000® 1200 MWe class



Artist's impression of a 2-unit ACR-1000 Nuclear Power Plant: 1200 MWe class Gen III+

AECL

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1. An Introduction to ACR-1000 Evolution

1.1 The ACR-1000

The ACR-1000 is built to meet industry and public expectations for safe, reliable, environmentally friendly, low-cost nuclear power generation. It has been developed by AECL from experience and feedback gained in the design, construction and operation of CANDU plants operated by ten utilities around the world.

With a 60-year design life, the ACR-1000 is a light-water-cooled, heavy-water-moderated pressure-tube reactor derived from the well-established CANDU line. It retains basic CANDU design features while incorporating a specific set of innovative features and state-of-the-art technologies to ensure its safety, operation, performance and economics are second to none.

Enhanced safety features include a core design with a small negative coolant void reactivity, larger thermal margins due to the use of CANFLEX®**** fuel, and design improvements based on insights gained from Probabilistic Safety Analysis (PSA) performed during the design process.

The latest design tools (CADDS) linking material management, documentation, safety analysis and project execution databases are used to ensure that accurate and complete configuration management can be readily maintained by the plant Owner.

1.2 Design Features

The ACR-1000 benefits from the proven principles and characteristics of CANDU design and the extensive knowledge base of CANDU technology gained over many decades of operation.

Proven CANDU strengths

- Modular, horizontal fuel channel core
- Separate low-temperature and pressure moderator
- Reactor vault filled with light water surrounding the core
- On-power refuelling
- Two independent passively driven, safety shutdown systems
- Reactor building access for on-power maintenance

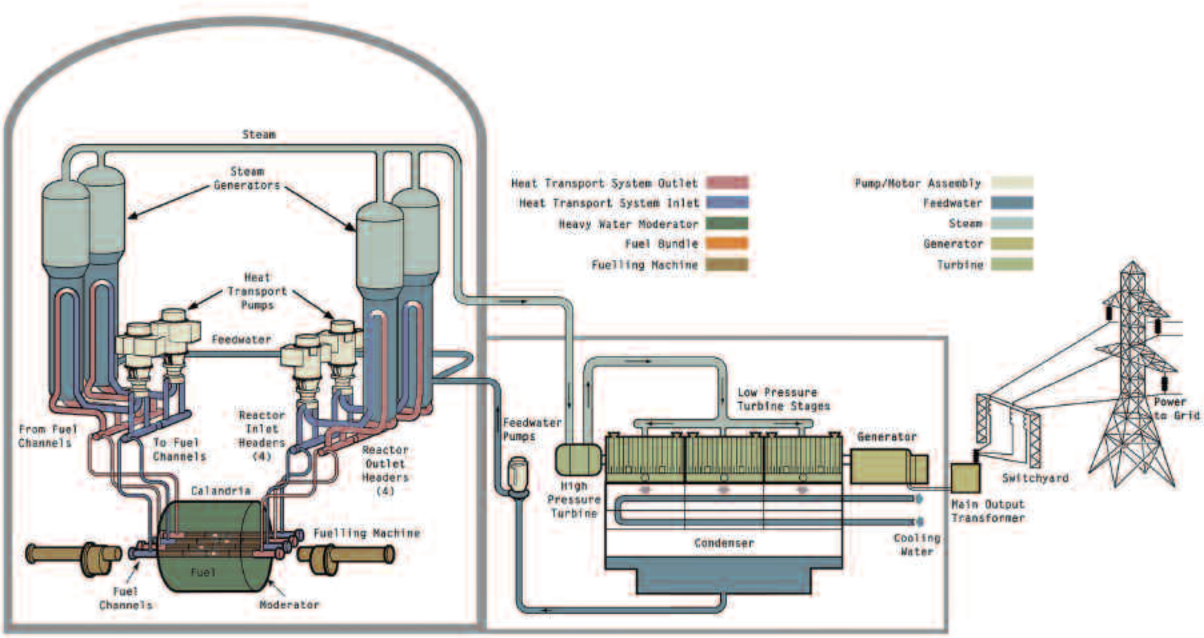


Figure 1-1 Overall ACR-1000 Plant Flow Diagram

ACR innovations

- A more compact core design, which reduces heavy water inventory and results in lower costs and reduced emissions
- Use of light water as reactor coolant, resulting in reduction of systems for heavy water coolant cleanup and recovery and simplification of containment atmosphere cleanup systems
- Improved fuel burn-up through the use of low enriched uranium (LEU) fuel, contained in advanced CANFLEX®-ACR fuel bundles
- Efficient means for burning other fuel types such as mixed oxides (MOX) and thorium fuels
- Increased fuel safety margins
- Improved plant thermal efficiency through use of higher pressures and higher temperatures in the coolant and steam supply systems
- Enhanced accident resistance and core damage prevention features
- Improved performance through use of SMART CANDU™ advanced operational and maintenance information systems and provision of designed-in maintenance features such as lifting devices, platforms and laydown areas
- Approximately 60% reduction in spent fuel quantities compared to current operating CANDU plants

Significant design simplifications

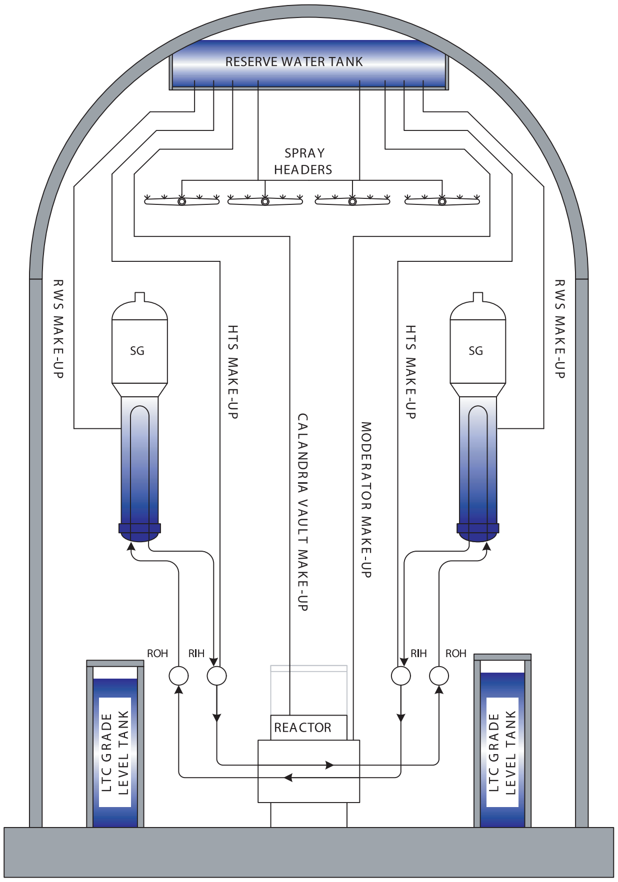
- Steel-lined containment designed for all design basis events
- Sharing of long-term emergency cooling and shutdown cooling safety functions
- Use of light water coolant enabling a simplified Emergency Coolant Injection (ECI) system, which replaces large motor-operated, safety-qualified injection valves with passive check valves
- Reduced inspections through selection of feeder materials for increasing resistance to flow-assisted corrosion (FAC) and robust fuel channel design margins
- Reduced on-line and start-up time with computerized testing of major safety systems and automatic calibration of in-core detector control signals
- Fuelling machine simplification with electric drives eliminating complex pneumatic systems. This accelerates the on-line fuelling process, reduces maintenance and speeds pressure tube in-service inspection
- Faster movement of personnel, without risk of airborne contamination spread, through use of ventilation systems that allow main airlock doors to be open during an outage
- Maintenance-based design providing required space allocation, reduction in temporary scaffolds and hoists, and provision for built-in electrical, water and air supplies for on-power and normal shutdown maintenance
- Reduction in number of sensors due to permitted sharing between systems

These technical improvements, along with advancements in project engineering, manufacturing, and construction, result in significantly reduced capital cost and construction schedule, while enhancing the inherent safety of the ACR-1000 design.

1.3 Passive Safety Features

The ACR-1000 design includes a number of "passive" safety features, some of which are design improvements over the already robust systems in existing CANDU plants. Examples of optimized features include:

- Two independent passively driven shutdown systems, each of which is capable of safely shutting down the reactor
- Increased safety margins with negative reactivity coefficients
- Passive emergency coolant injection operation
- Cool, low-pressure moderator serving as a passive heat sink for decay heat from fuel channels in severe accident situations
- Large concrete reactor vault, surrounding the core in the calandria vessel and containing a large volume of light water to further slow down or arrest severe core damage progression by providing a second, passive, core heat sink
- Elevated reserve water tank (RWT) in upper level of the containment building to deliver passive make-up cooling water by gravity to heat transport system, steam generators, moderator and the calandria vault. This delays progression of severe accidents and provides even more time for mitigating actions by the operator
- Passive, robust, seismically-qualified containment consisting of:
  - Thickened pre-stressed concrete structure designed to withstand aircraft crashes
  - Leak-tight inner steel liner to reduce potential leakages
  - Passive spray system from elevated reserve water tank to reduce reactor building pressures in the event of a severe accident
  - Passive Hydrogen Recombiner



ONLY ONE LOOP OF HTS IS SHOWN

Figure 1-2 Reserve Water System

2. Plant Design

2.1 Layout: Inherently Safer and Faster to Build

Designed for efficient operation, increased safety and easier and faster maintenance, the plant is laid out to provide separation by distance, elevations and the use of barriers for safety-related structures, systems and components. Each corner of the reactor auxiliary building houses redundant safety equipment in a four-quadrant configuration.

Security and physical protection have been addressed to ensure that the response to potential common and abnormal events, such as fires, aircraft crashes and malevolent acts meets latest criteria.

The plant layout is also designed to achieve the shortest practical construction schedule while supporting easier maintenance practices. Buildings are arranged to minimize interferences during construction, with allowance for on-site fabrication of module assemblies. Through the use of open-top construction, provisions exist for flexible equipment installation sequences.

The footprint of the two-unit plant has been minimized with the adoption of common areas for the main control room, service and maintenance buildings. A single-unit plant can be adapted from the two-unit layout with no significant changes to the reference design. The plant is designed for an exclusion zone of 500 metre radius. The size of the power block for a 2-unit ACR-1000 station is 48,700 m^2* (actual area).

** Power block consists of 2 reactor buildings, 2 reactor auxiliary buildings, 2 turbine buildings, 1 service building, 1 main control building, 1 maintenance building, 1 crane hall, 2 secondary control buildings and four diesel generator buildings.*

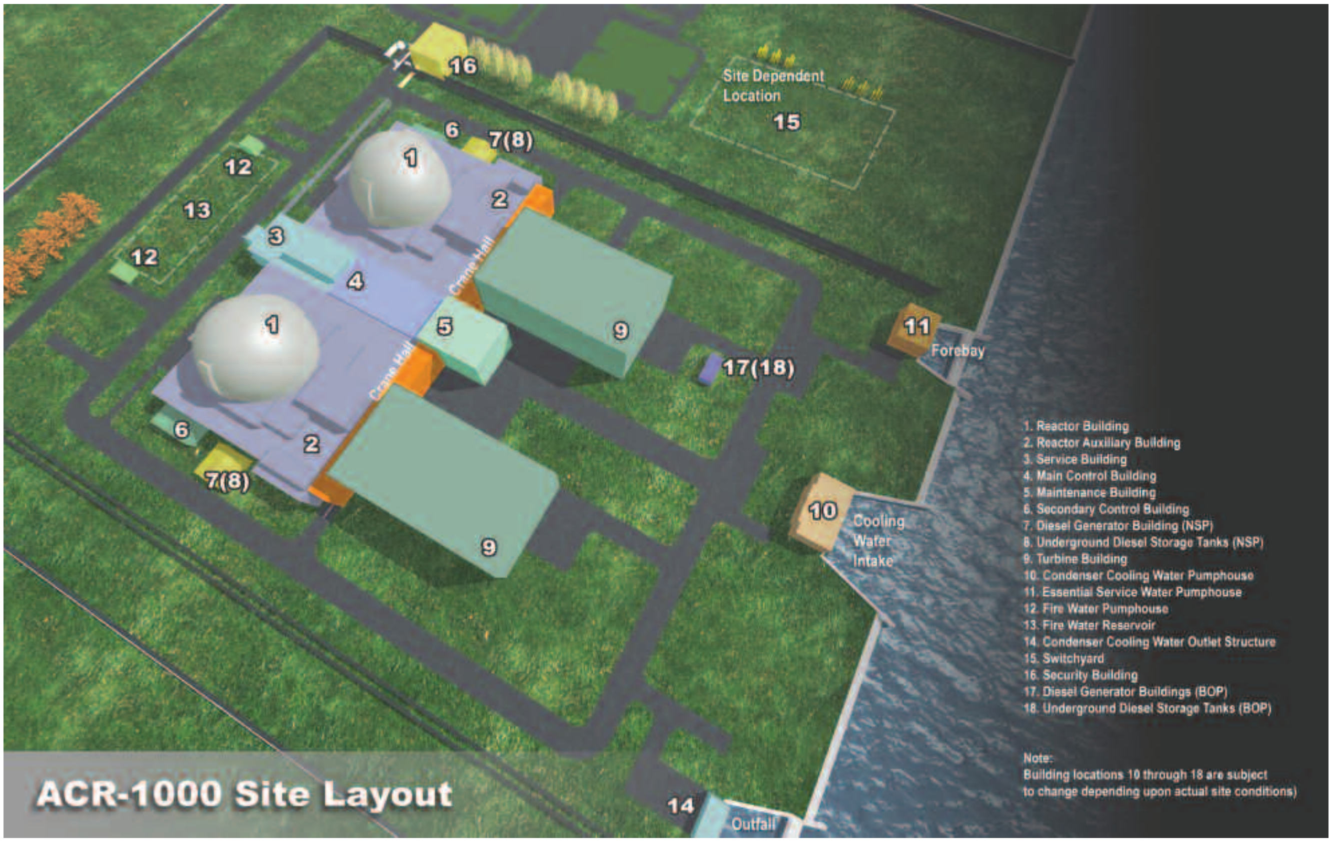


Figure 2-1 Two-Unit Plant Layout of Major Structures

Major buildings and structures of two-unit plant

- Reactor Buildings (2)
- Reactor Auxiliary Buildings (2)
- Turbine Buildings (2)
- Main Control Building
- Secondary Control Buildings (2)
- Maintenance and Service Buildings
- Condenser Cooling Water Pumphouse
- Essential Service Water Pumphouse
- Main Switchyard

Reactor Building

Strengthened over previous CANDU designs, the pre-stressed concrete reactor building is seismically-qualified and tornado-proofed. The concrete outer wall has an inner steel liner that will achieve significantly reduced leak rates in the event of an accident.

An isolation system ensures "button-up" in case of accidents.

The entire structure, including concrete internal structures, is supported on a reinforced concrete base slab to ensure a fully enclosed boundary for environmental protection and biological shielding.

During reactor operation, internal shielding permits personnel access to an environment that is temperature-controlled for personal comfort. Airlocks are designed as routine entry/exit doors.

Containment structure perimeter walls are separate from internal structures, so as to eliminate any interdependence and to provide flexibility in construction.

The reactor building is the principal component of the containment system.

	CANDU 6	ACR-1000
Containment Structure		
Type	Pre-stressed concrete / epoxy liner	Pre-stressed concrete / steel liner
RB inside diameter	41.4 m	56.5 m
RB containment wall thickness	1.07 m	1.8 m
Building height (base slab to top of dome)	51.2 m	74.0 m

Reactor Auxiliary Building

The reactor auxiliary building is a multi-level, reinforced concrete and steel structure that is seismically-qualified and tornado protected. It surrounds the reactor building and accommodates the umbilicals that run between the principal structures, the electrical systems, and the spent fuel bay and associated fuel-handling facilities. It also houses the long-term cooling (LTC) pumps and heat exchangers, the spent fuel bay cooling and purification system pumps and heat exchangers, the essential cooling water pumps, heat exchangers and valve stations, and the essential service water valve stations. Safety and isolation valves for the main steam lines are housed in a seismically-qualified concrete structure on top of the building.

Turbine Building

The turbine building is located to one side of the reactor auxiliary building, so that turbine shaft alignment is perpendicular to the reactor building. This is also an optimum location for access to the main control room, the piping and cable tray runs to and from the reactor auxiliary building, and the condenser cooling water ducts to and from the main pumphouse. Access routes are provided between the turbine building and the reactor auxiliary building.

The turbine building houses the turbine generator and its auxiliary systems: condenser, condensate and feedwater systems, the building heating plant, and any compressed gas required for the balance of plant (BOP). Blow-out panels in the walls and roof serve to relieve internal pressure in the event of a steam-line break.

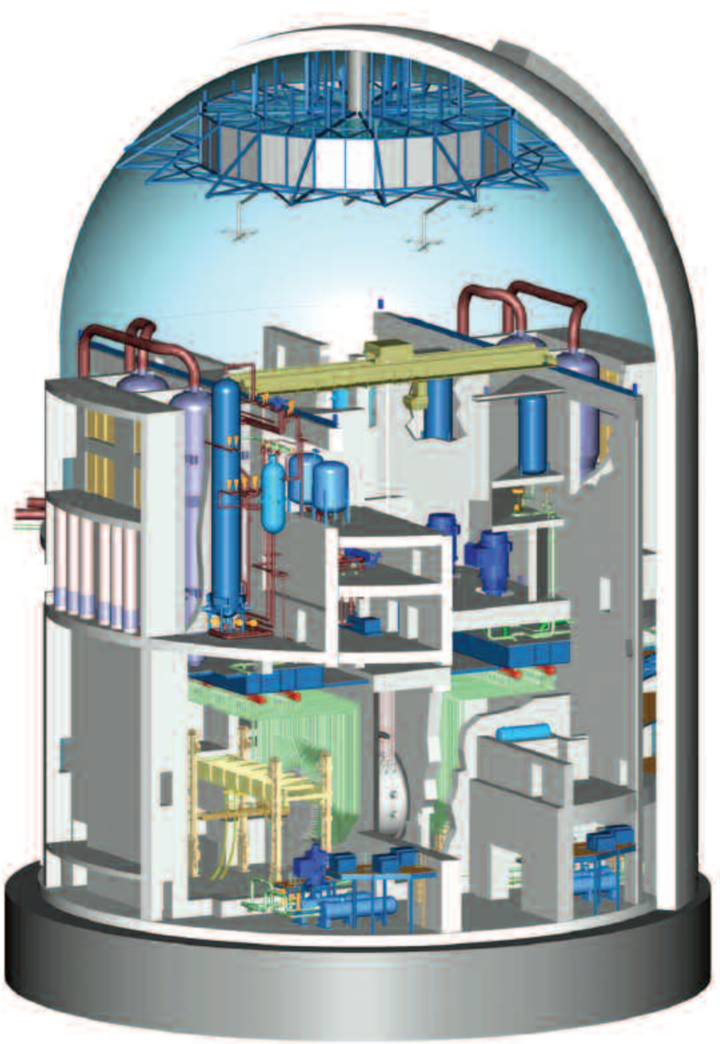


Figure 2-2 Reactor Building

Main Control Building

Seismically-qualified and tornado-protected, the main control building is a multi-level structure located between the two units. It has a superstructure of steel and reinforced concrete and reinforced concrete substructure. It contains the main control room (MCR) and associated control and electrical equipment for the two units. Each side of the MCR has dedicated panels, computers, displays and operator consoles with separation of cabling and equipment for each unit.

Secondary Control Building

Each unit has a completely separate secondary control building (SCB) with sufficient control and monitoring equipment to shut down the unit, initiate required cooling and ensure a safe, maintained shutdown state should the MCR become uninhabitable or non-functional. The SCB is located so that the MCR and SCB cannot be simultaneously rendered inoperable due to any design basis event. SCB human-system interface components are similar to those in the MCR so as to minimize human error should the operator relocate from one area to the other.

Maintenance and Service Buildings

The seismically-qualified maintenance and service buildings are located between the two-unit ACR-1000 plant. They house all conventional plant services, including radioactive waste handling facilities, heavy water management systems, common services, central stores, central active/non-active maintenance shops, and change rooms for staff. They are multi-level structures with a reinforced concrete substructure and braced steel-frame superstructure.

Condenser Cooling Water Pumphouse

The condenser cooling water (CCW) pumphouse has a reinforced concrete substructure and braced steel-frame superstructure. It contains the CCW pumps, plant water system pumps, screen wash pumps, trash racks, screens, and chlorination equipment, if required. Together with related intake and outfall structures, the pumphouse serves the two-unit ACR-1000 plant, housing separate CCW and plant water systems with adequate separation for each unit. Sites with limited cooling water availability can use cooling towers instead of the conventional CCW system.

Essential Service Water (ESW) Pumphouse

The essential service water (ESW) pumphouse contains the ESW pumps. It has a reinforced concrete substructure, braced steel-frame superstructure and is seismically and tornado-qualified.

Main Switchyard

The switchyard is designed to allow flexible operation for power output, switching and maintenance. A breaker-and-a-half design with single voltage is proposed for high reliability. Each ACR-1000 unit will have at least four bays of power inputs/outputs from the main transformers and grid system, with options to add more as future plant and grid requirements may dictate.

2.2 Plant Siting

2.2.1 Unit Output

Each unit of the ACR-1000 two-unit integrated plant design has a nominal gross electrical output of 1165 MWe. Output can be optimized by adjusting the turbine/condenser design to suit any site cooling water conditions.

2.2.2 Adaptation to Site Requirements

The ACR-1000 can accommodate a wide range of geotechnical and meteorological data and conditions. Some of these flexible design features include:

- Cooling water systems for all nuclear steam requirements, saltwater or freshwater. Conventional cooling towers can also be used
- Cooling water temperatures from typical cold to typical warm sites. A generic set of reference conditions has been developed for potential ACR-1000 sites
- Tornado protection as required. The design basis tornado (DBT) is defined by a maximum wind speed of 483 km/h. DBT for the ACR-1000 is selected to satisfy tornado design requirements for North American sites and potential sites overseas

- Plant exclusion zone capability of only 500 m radius. All unauthorized persons are restricted from this zone. Larger zones may be selected where desired
- Design basis earthquake (DBE) protection of up to 0.3 g acceleration. This is the maximum ground motion of potentially severe quakes, with low probability of being exceeded during the life of the plant

2.3 Nuclear Systems

ACR-1000 nuclear systems are located in the reactor building and the reactor auxiliary building. These buildings are robust and shielded where necessary to ensure all radioactive substances are always secure. Systems include:

- Heat transport system with light water coolant in a two-loop, figure-eight configuration with four steam generators, four heat transport pumps, four reactor outlet headers and four reactor inlet headers. This configuration is standard on all CANDU 6 reactors and the larger 935 MWe Darlington Nuclear Generating Station (NGS) CANDU design
- Heavy water moderator system
- Reactor assembly consisting of calandria vessel installed in concrete vault
- Fuel handling system consisting of two fuelling machine heads, each mounted on a fuelling machine bridge and supported by columns, located at each end of the reactor. The fuelling machines have been simplified to enhance maintainability and accelerate pressure tube in-service inspection
- Two independent shutdown systems, emergency core cooling (ECC) system, containment system and associated safety support systems

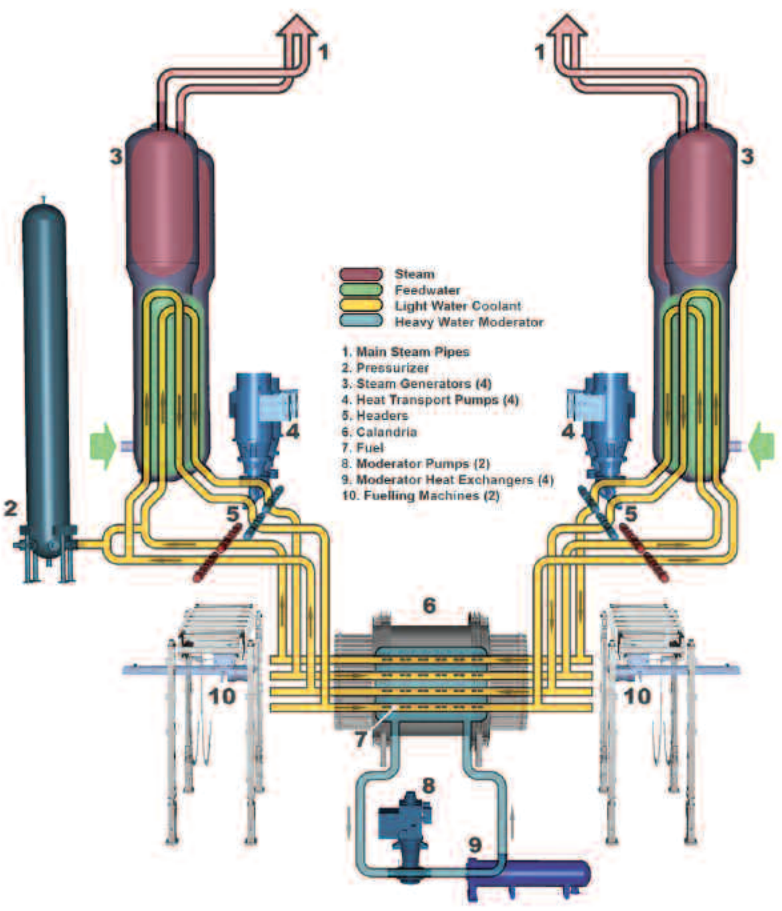


Figure 2-3 Nuclear Systems Schematic

2.4 Heat Transport System and Auxiliary Systems

The ACR-1000 heat transport system (HTS) circulates pressurized light water coolant through the reactor fuel channels to remove heat produced by nuclear fission in the core. The use of light water coolant is a design simplification allowing for the reduction of systems for cleanup and recovery. It also simplifies containment atmosphere cleanup systems.

The ACR-1000 HTS consists of 520 reactor fuel channels with associated corrosion-resistant stainless steel feeders, four inlet headers, four outlet headers and interconnecting piping. The system includes four steam generators and four electrically-driven heat transport pumps in a two-loop, figure-eight configuration. Headers, steam generators and pumps are all located above the reactor.

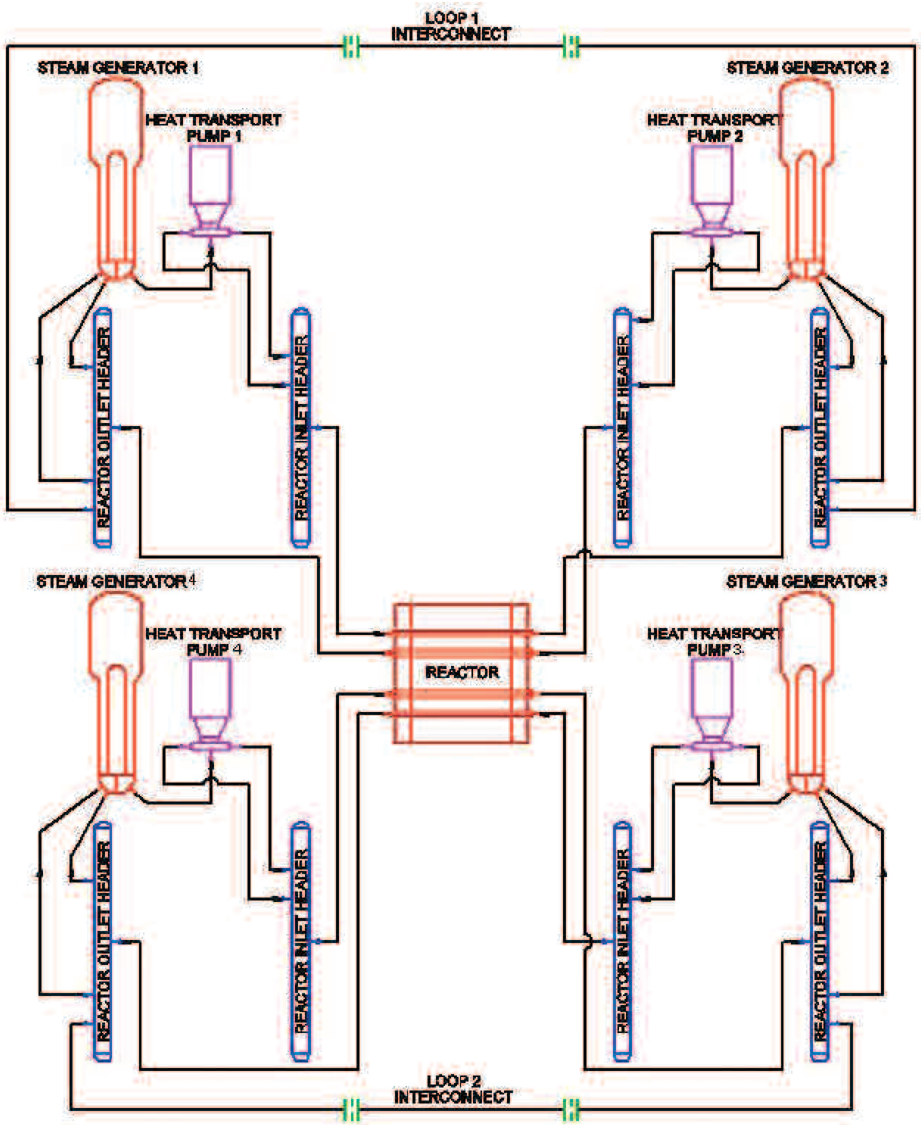


Figure 2-4 Heat Transport System Flow Diagram

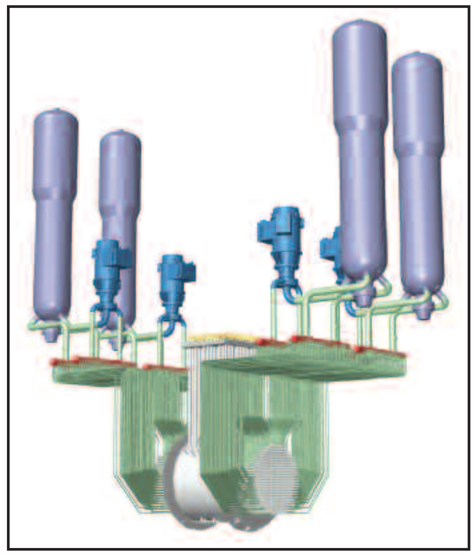


Figure 2-5 3D View of Heat Transport System Layout

Figure 2-6 3D View of Heat Transport System in Reactor Building

Table 2-1 Heat Transport System Design Data

	CANDU 6	Darlington	ACR-1000
Reactor outlet header pressure [MPa (g)]	9.9	9.9	11.1
Reactor outlet header temperature [°C]	310	310	319
Reactor inlet header pressure [MPa (g)]	11.2	11.3	12.5
Reactor inlet header temperature [°C]	260	267	275
Single channel flow (maximum) [kg/s]	28	27.4	28

Pressure and Inventory Control System

The ACR-1000 heat transport pressure and inventory control system consists of pressurizer, pumps, feed and bleed valves and a coolant storage tank. This system provides:

- Pressure and inventory control for each heat transport system loop
- Overpressure protection
- Controlled degassing flow

Light water in the pressurizer is heated electrically to pressurize the vapour space above the liquid. The volume of the vapour space is designed to cushion pressure transients, without allowing excessively high or low pressures in the heat transport system.

The pressurizer also accommodates change in reactor coolant volume from zero power to full power. This permits reactor power to be increased or decreased rapidly, without imposing severe demand on the coolant feed and bleed components of the system.

When the reactor is at power, pressure is controlled by the pressurizer; heat is added with the electric heaters to increase pressure, and removed by spraying cold water via the reactor inlet headers to reduce pressure. The coolant inventory is adjusted by the feed-and-bleed circuit. Pressure can also be controlled by the feed-and-bleed circuit with the pressurizer isolated at low reactor power and when the reactor is shut down. The feed-and-bleed circuit is designed to accommodate the changes in coolant volume that take place during heat-up and cool-down.

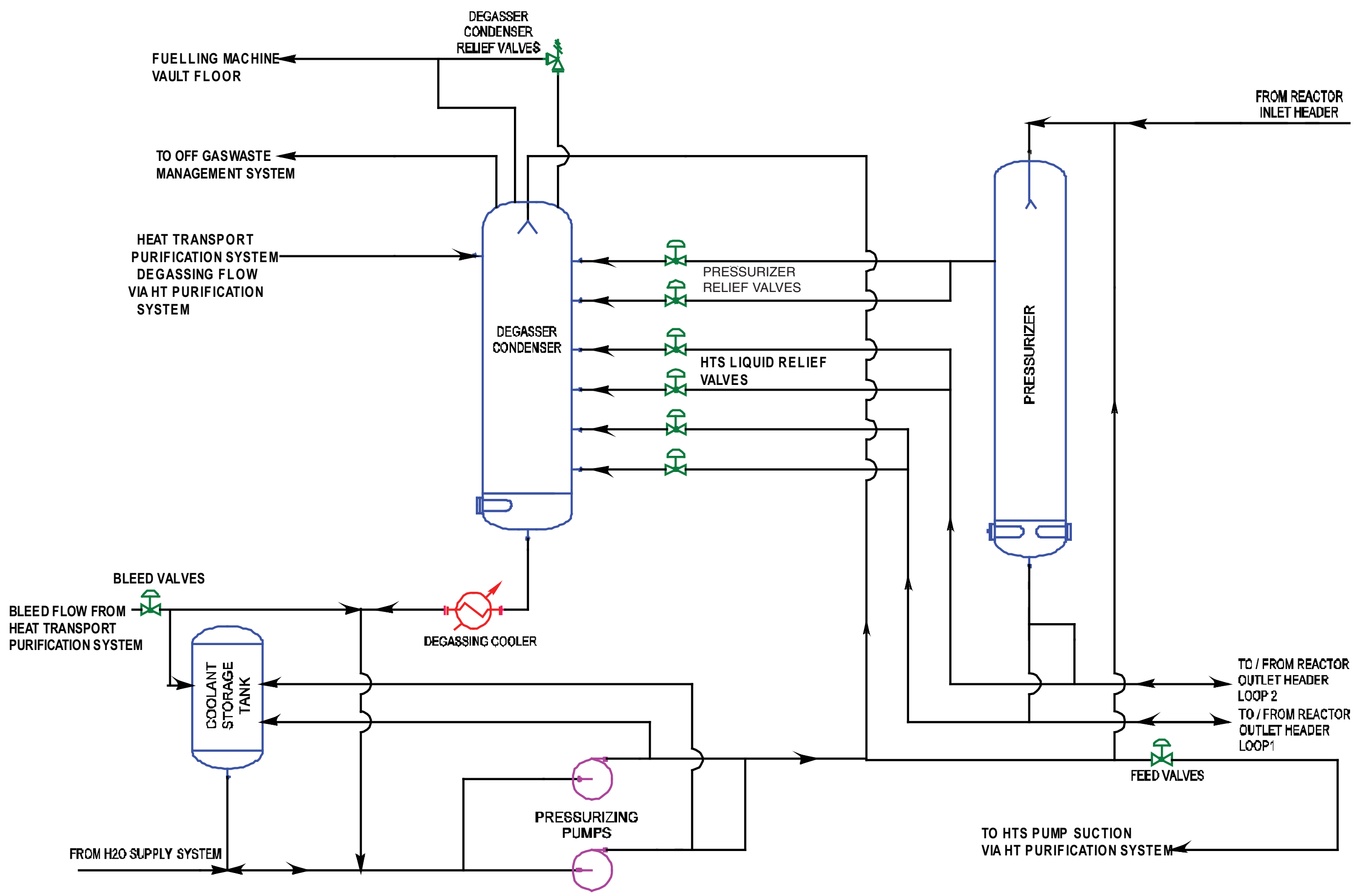


Figure 2-7 Pressure and Inventory Control Flow Diagram

2.4.1 Heat Transport Pumps

The ACR-1000 heat transport pumps are an enhanced, larger version of the double-discharge design used in the CANDU 6 and Darlington reactors.

The ACR-1000 retains the CANDU mechanical multi-seal design, which allows for easy replacement. Backup seal cooling extends pump survivability, even during accident conditions, if service water is lost.

Table 2-2 Heat Transport Pump Data

	CANDU 6	Darlington	ACR-1000
Number	4	4	4
Rated flow [L/s]	2228	3240	4300
Motor rating [MWe]	6.7	9.6	10.0

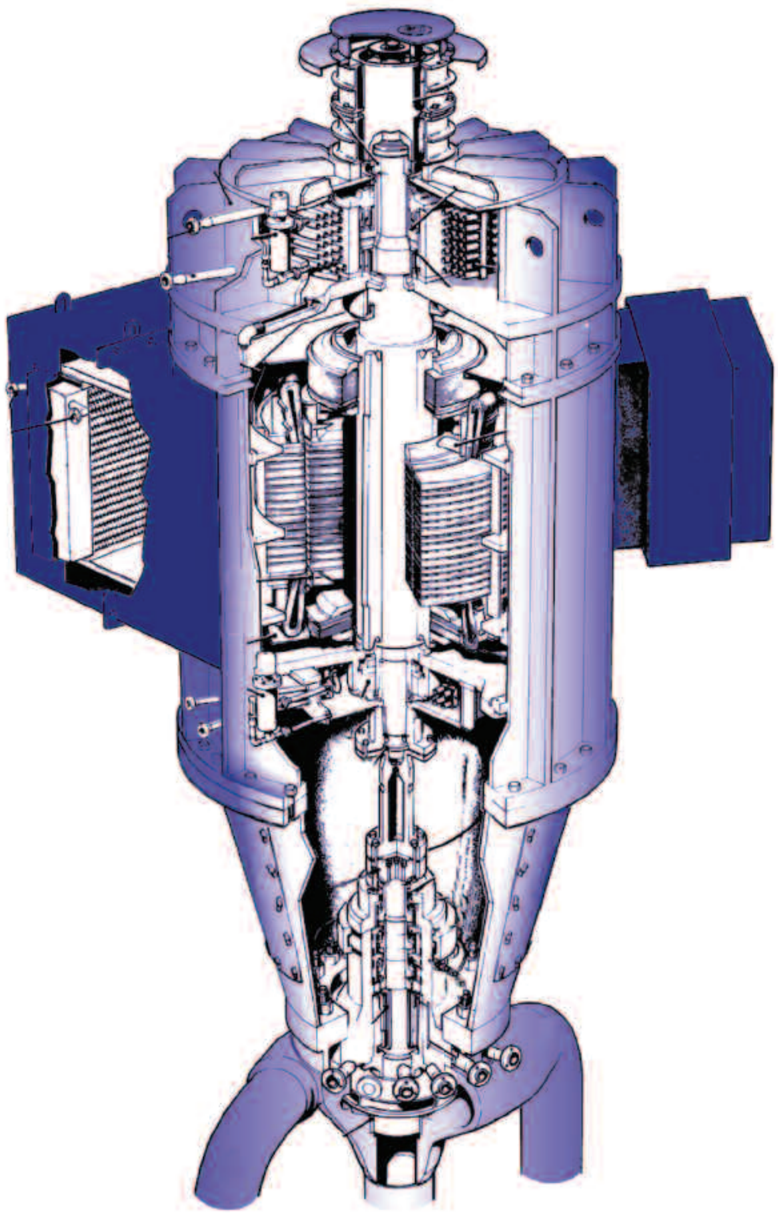


Figure 2-8 Heat Transport System Pump

2.4.2 Steam Generators

The ACR-1000 steam generators are similar to the CANDU 6 and Darlington designs, except for the larger physical size. For the ACR-1000, steam generator tubing diameter is increased to take advantage of the change to light water coolant.

ACR-1000 tubing is made of Incoloy-800, a material with proven operating performance and service at CANDU 6 and Darlington stations. Steam wetness at the steam nozzle has been reduced to 0.1%, based on latest steam separator technology, leading to improved turbine cycle economics.

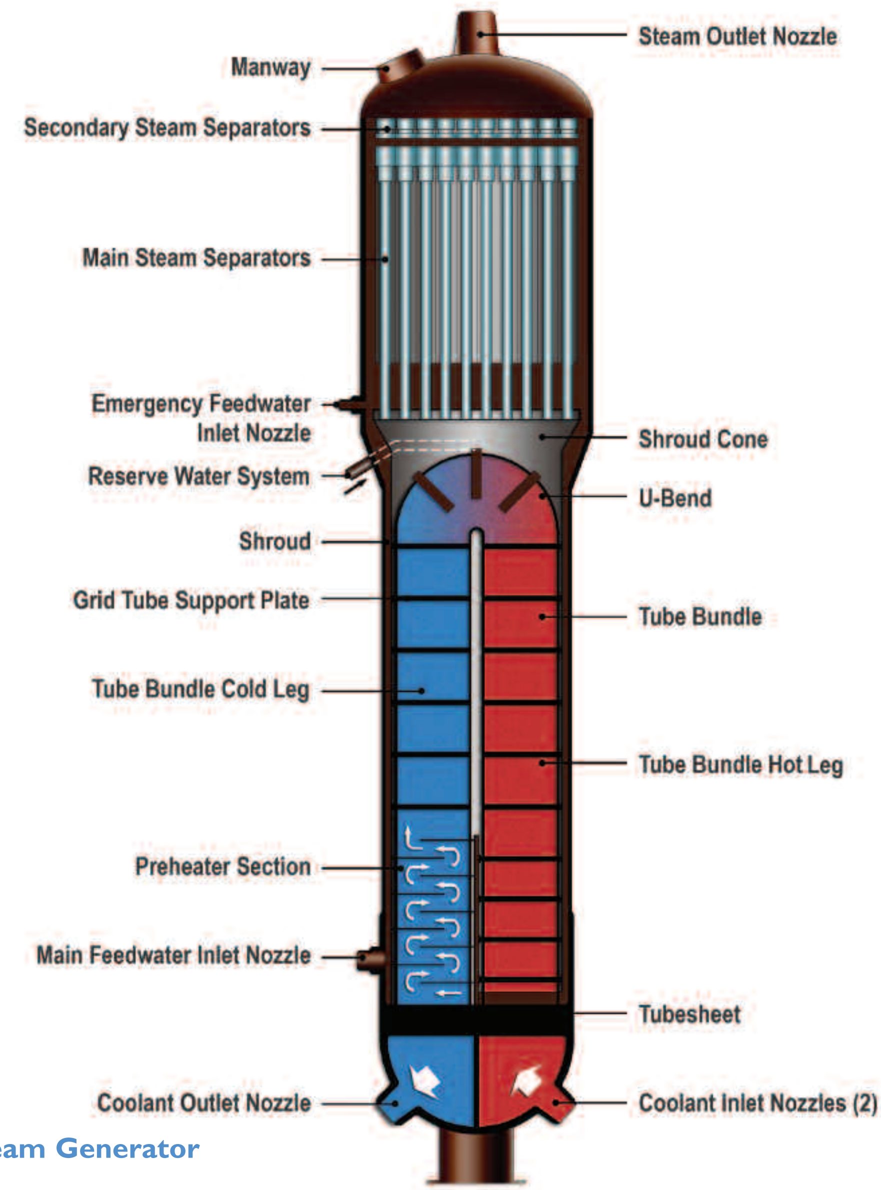


Figure 2-9 Steam Generator

Table 2-3 Steam Generator Design Data

Steam Generators	**CANDU 6**	**Darlington**	**ACR-1000**
Number	4	4	4
Type	Vertical U-tube / integral pre-heater	Vertical U-tube / integral pre-heater	Vertical U-tube / integral pre-heater
Nominal tube diameter [mm]	15.9 (5/8")	15.9 (5/8")	17.5 (11/16")
Steam temperature (nominal) [°C]	260	265	275.5
Steam quality	0.9975	0.9975	0.999
Steam pressure [MPa (g)]	4.6	5.0	5.9

2.5 Moderator System

The ACR-1000 moderator is a low-pressure, low-temperature system that is fully independent of the heat transport system. It consists of pumps and heat exchangers that circulate heavy water moderator (D_2O) through the calandria vessel and remove heat generated within the moderator during reactor operation. Heavy water acts as both moderator and reflector for the neutron flux in the core.

Inlet and outlet nozzles are located at the top of the calandria vessel to prevent inadvertent draining and are oriented to ensure uniform moderator temperature distribution inside the calandria.

The ACR-1000 moderator system also fulfills a safety function that is unique to ACR/CANDU. It serves as a backup heat sink in the event of loss of fuel cooling via the heat transport system, thereby mitigating core damage consequences. Heat exchangers are provided with seismically-qualified cooling water and standby power.

Another safety improvement in the ACR-1000 is the connection to the reserve water tank. It provides additional passive gravity-fed inventory to the calandria vessel, extends core cooling and delays severe accident event progression.

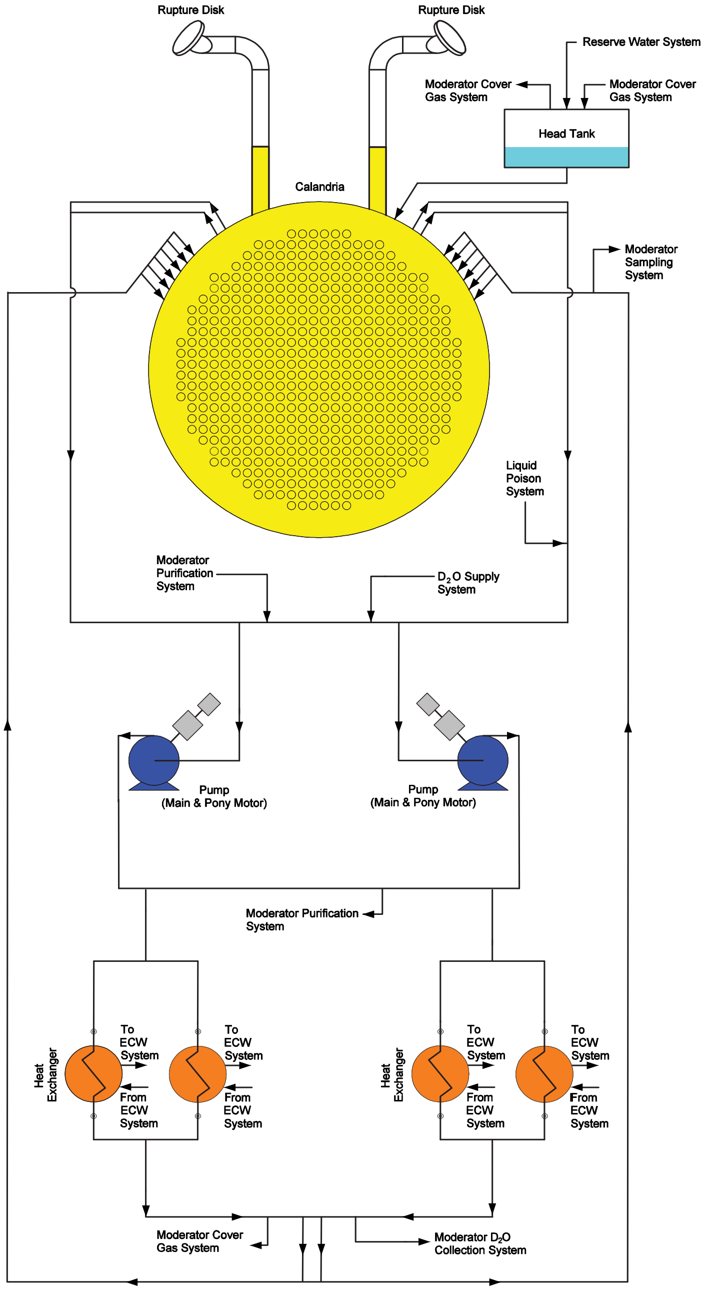


Figure 2-10 Moderator System Flow Diagram

Table 2–4 Heavy Water Inventory Design Data

	CANDU 6	Darlington	ACR-1000
Moderator System [Mg D_2O]	265	312	250
Heat Transport System [Mg D_2O]	192	280	0
Total [Mg D_2O]	457	592	250

2.6 Reactor Assembly

The ACR-1000 reactor assembly consists of the horizontal, cylindrical, low-pressure calandria and end-shield assembly. This enclosed assembly contains the heavy water moderator and the 520 fuel channel assemblies. The reactor is supported within a concrete, light-water-filled calandria vault. Fuel is enclosed in the fuel channels that pass through the end shields. Each fuel channel permits access for on-line fuelling operation while the reactor is at power.

The ability to replace fuel as required for maintaining power means minimal "excess" reactivity in the core at all times, an inherent safety feature. This feature also contributes to operational flexibility for improved outage planning, since fixed cycle times are not required and prompt removal of defect bundles can be accomplished without shutdown.

2.6.1 Reactor Core Characteristics

The ACR-1000 reactor core offers the following distinctive advantages:

- Compact size due to smaller fuel channel lattice pitch than CANDU, resulting in reduced heavy water requirements
- Use of light water as coolant
- Negative coolant void reactivity
- Simplified reactor control through negative feedback in reactor power
- Flattened axial and radial profiles to optimize channel thermal power output

The physical size of the ACR-1000 core, while producing greater power output, is similar to that of the CANDU 6.

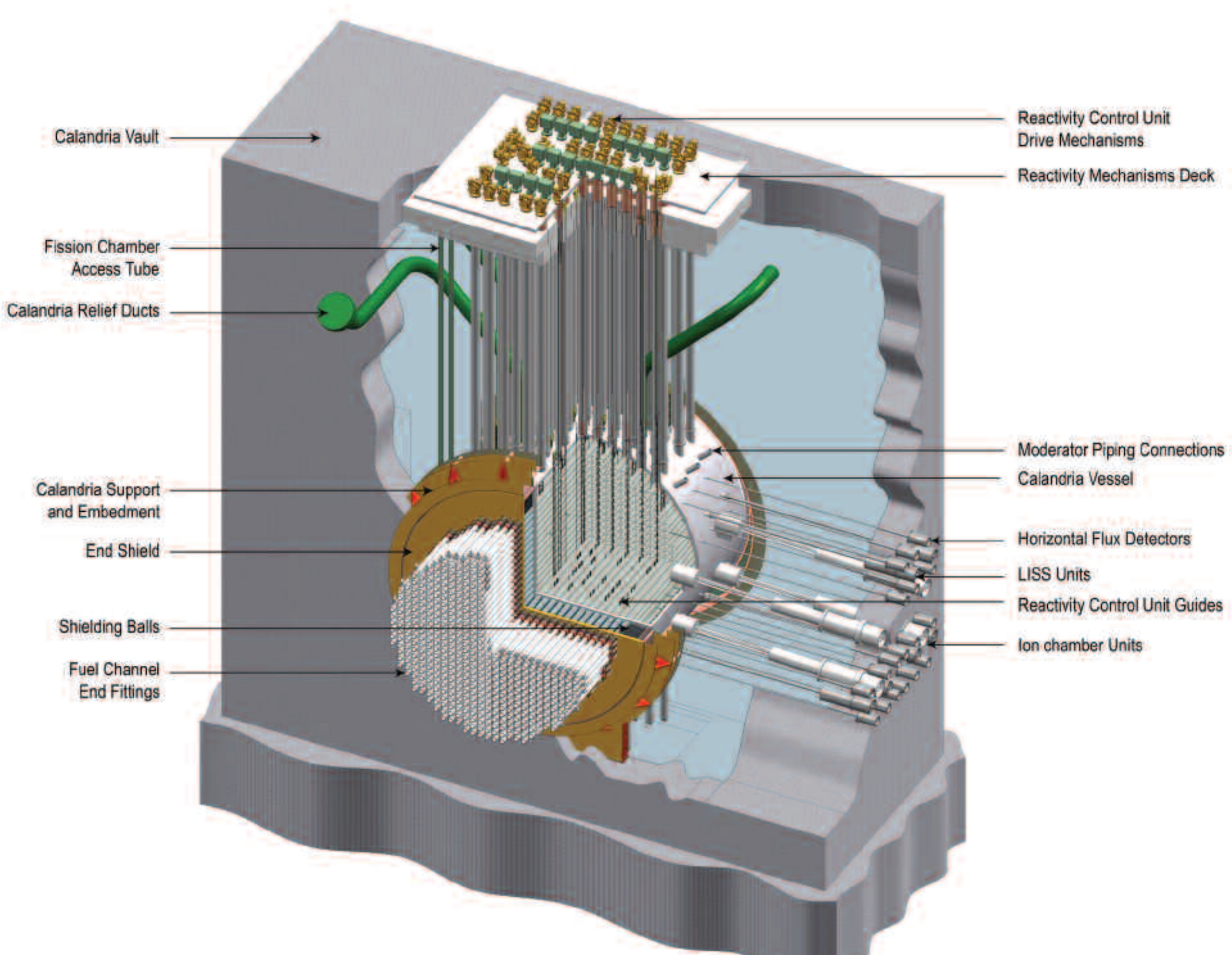


Figure 2-11 Reactor Assembly

2.6.2 Reactor Control

The neutronic coupling in the compact ACR-1000 core and negative power coefficient ensure core stability. All harmonic modes, including the first axial mode, are stable at all power levels under nominal operating conditions. Stable reactor physics characteristics allow simpler control mechanism design.

Mechanical zonal control units provide primary control in the ACR-1000. Each zone control assembly consists of two independently movable segments. On-power refuelling and zone-control actions provide day-to-day reactivity control. The reactor regulating system also includes control absorber units, physically similar to the mechanical shutoff rods that can be used to reduce power if larger reductions are required.

Table 2- 5 Reactor Core Design Data

	CANDU 6	Darlington	ACR-1000
Reactor			
Output [MWth]	2064	2657	3187
Coolant	Pressurized D_2O	Pressurized D_2O	Pressurized Light Water
Moderator	D_2O	D_2O	D_2O
Calandria diameter [m]	7.6	8.5	7.5
Fuel channel	Horizontal Zr 2.5wt%Nb alloy pressure tubes with modified 403 SS end-fittings	Horizontal Zr 2.5wt%Nb alloy pressure tubes with modified 403 SS end-fittings	Horizontal Zr 2.5wt%Nb alloy pressure tubes with modified 403 SS end-fittings
Fuel channels	380	480	520
Lattice pitch (mm)	286	286	240
Pressure tube wall thickness (mm)	4	4	6.5

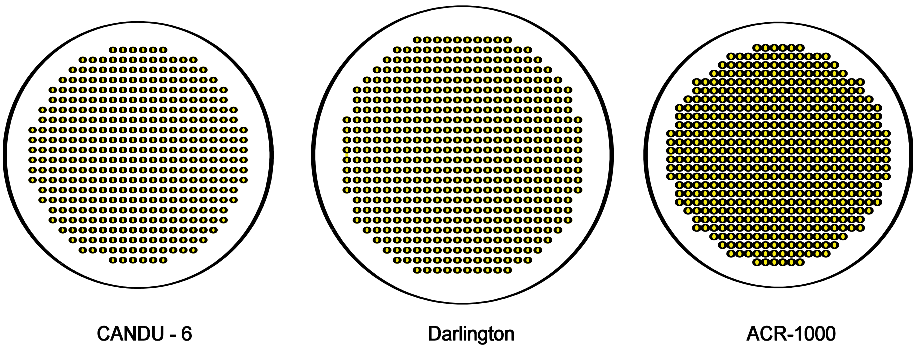


Figure 2-12 Comparison of Core Sizes

2.6.3 Fuel Channel Assembly

The ACR-1000 fuel channel assembly consists of a zirconium-niobium (Zr-2.5%Nb) pressure tube, centred in a zircaloy calandria tube. The pressure tube is roll-expanded into stainless steel end fittings at each end.

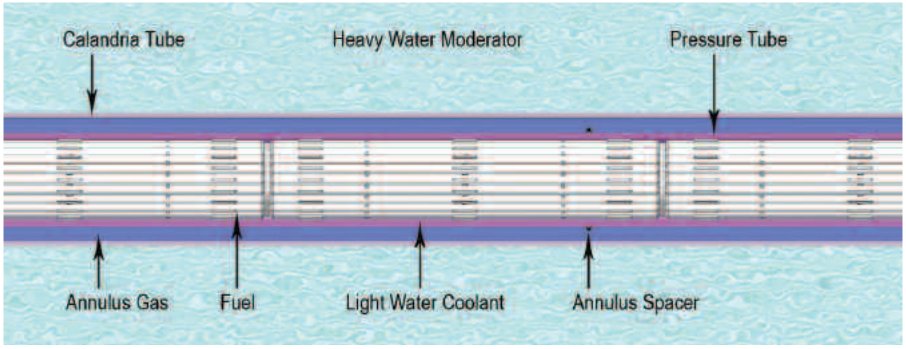


Figure 2-13 Fuel Channel

Each pressure tube is thermally insulated from the low-temperature moderator by the annulus gas between the pressure tube and the calandria tube. Fixed spacers, positioned along the length of the pressure tube, maintain annular space and prevent contact between the two tubes. Each end-fitting holds a liner tube, a fuel support plug and a channel closure. Reactor coolant flows through adjacent fuel channels in opposite directions. The ACR-1000 calandria tube has been thickened compared to the CANDU design to ensure it can withstand a pressure tube rupture.

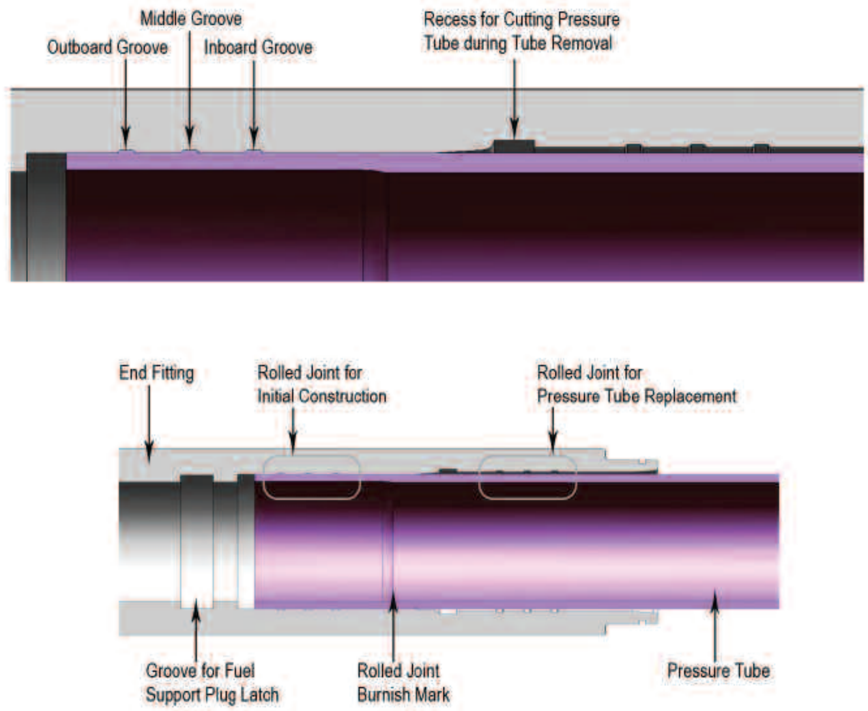


Figure 2-14 Fuel Channel Grooves

The ACR-1000 is designed for 60 years of reactor operation with provision for mid-life refurbishment, including replacement of fuel channels. Special design features, such as additional rolled joint grooves, are provided in the end-fittings to facilitate pressure tube replacement.

2.7 Fuel Handling Systems

The ACR-1000 fuel handling systems consist of:

- New fuel handling and storage system
- Fuelling machines and their supports
- Spent fuel handling and storage

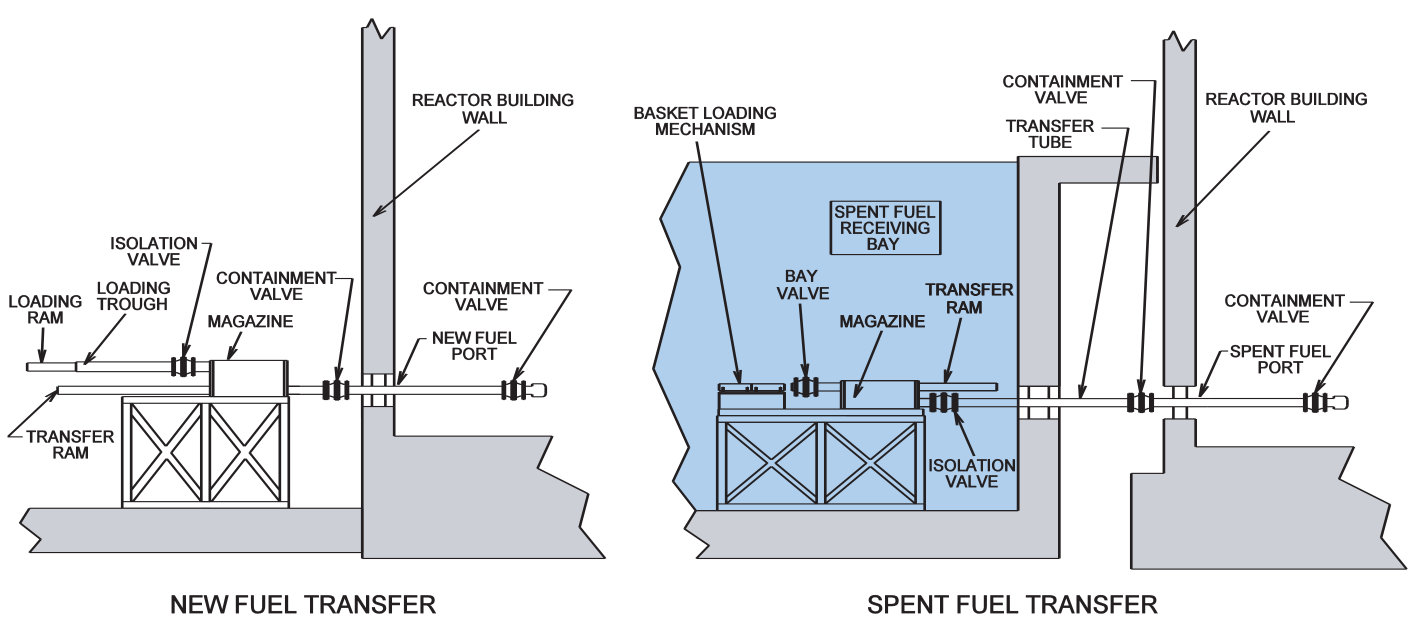


Figure 2-15 New Fuel and Spent Fuel Transfer Mechanisms

The new ACR-1000 fuel handling and storage system includes the storage of the new low enriched uranium (LEU) fuel and the supply of the two fuelling machines to maintain full-power operation. The need for operator access to the reactor building is minimized with all new fuel storage, inspection and fuelling machine loading being performed from an accessible area in the reactor auxiliary building.

Evolved from the CANDU 6 design, the simplified ACR-1000 fuel handling machines incorporate significant advances. Key design improvements include replacing water and oil hydraulic drives with electric drives, a larger capacity magazine and a mechanical ram with absolute resolvers for position feedback. Further design simplifications include change to light water operation, with heavy water eliminated from the fuel handling systems. These changes, along with built-in redundancy, will result in improved system performance, extended in-service periods and reduced maintenance requirements, including accelerated de-fuelling for pressure tube in-service inspection.

Two fuelling machines are located on opposite sides of the reactor and mounted on bridges supported by columns. The normal refuelling operation is an eight-bundle shift, in the direction of coolant flow, in which spent bundles are removed from the outlet end of a fuel channel, while fresh bundles are inserted at the inlet end.

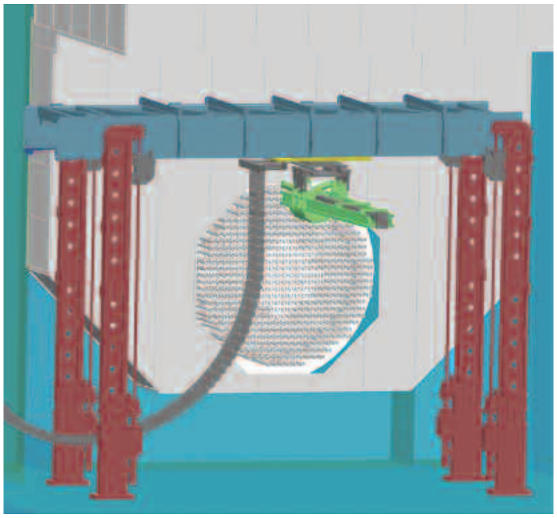


Figure 2-16 Fuelling Machine and Carriage

The ACR-1000 transfer and storage system handles spent fuel from the time it is discharged from the fuelling machine to the time it is moved to the storage bay in the reactor auxiliary building.

Once spent fuel is discharged, the transfer system uses recirculating water, which also cools the fuel, to push it through a pipe to receiving bays. The system then unloads the fuel from its magazine and moves it in baskets to the storage bay through a shielded tunnel. In the storage bay, spent fuel baskets are stacked in frames with capacity for at least 10 years operation. A storage bay bridge and handling tools permit manipulation of spent fuel and containers. Baskets are also suitable for direct transfer to dry fuel storage, which can be provided at Owner request—for an additional 50 or beyond.

The entire fuelling and spent fuel unloading process is automated and carried out from the station control room.

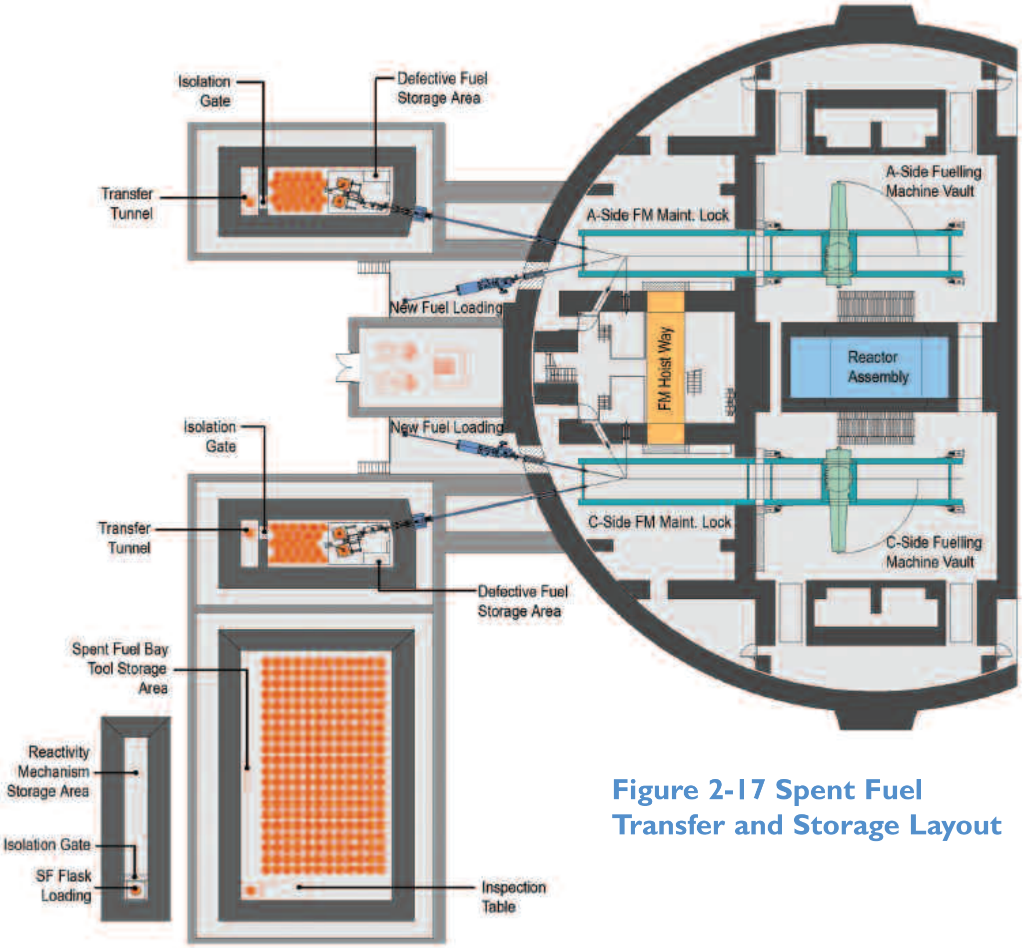


Figure 2-17 Spent Fuel Transfer and Storage Layout

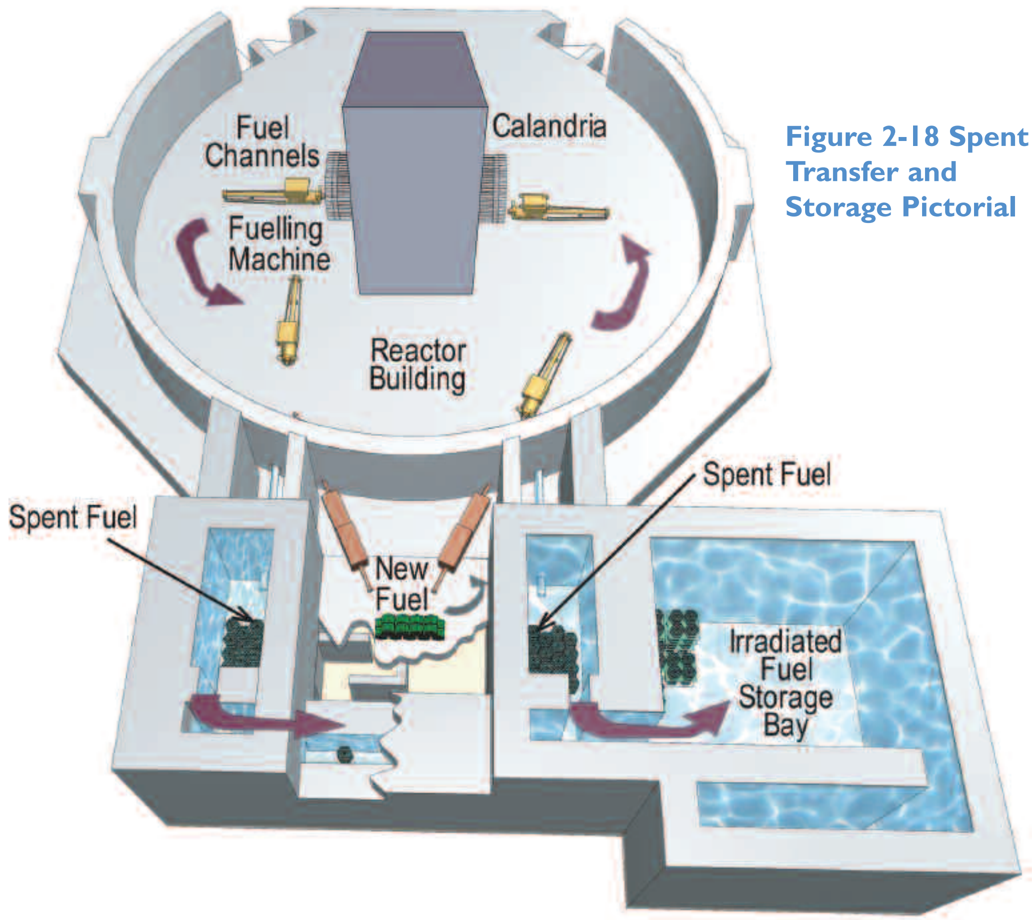


Figure 2-18 Spent Fuel Transfer and Storage Pictorial

	CANDU 6	Darlington	ACR-1000
Fuel	Natural UO_2	Natural UO_2	Low enriched UO_2
Fuel burn-up [MWd/te U]	7,500	7,791	20,000
Fuel bundle assembly	37 element	37 element	43-element CANFLEX®-ACR
Bundles per fuel channel	12	13	12

2.8 Fuel

The ACR-1000 uses the 43-element CANFLEX®-ACR fuel bundle design.

The centre element contains neutron absorbers, while the remaining elements contain U-235 enriched UO_2 pellets. A burnable absorber is used in some of the elements that contain enriched pellets to optimize the power rating of the fuel. The neutron absorbers of the centre element are used for management of coolant void reactivity. A very thin layer of CANLUB covers the inside surface of the fuel cladding to enhance fuel performance.

The ACR inherent feature for operating with neutron absorbers makes it ideally suited to burn other fuel types such as mixed oxides (MOX) and thorium.

Figure 2-19 CANFLEX®-ACR Fuel Bundle

2.9 Safety Systems

ACR-1000 safety systems are designed to mitigate the consequences of plant process failures, ensuring reactor shutdown, removal of decay heat and prevention of radioactive releases.

Design follows the traditional CANDU practice of providing:

- Shutdown System 1, (SDS1)
- Shutdown System 2, (SDS2)
- Emergency Core Cooling (ECC) System
- Containment System
- Emergency Feedwater System

SDS1, SDS2, the ECC and containment systems meet high reliability requirements that have been established during system design and verified by reliability analysis.

Safety support systems are also provided to ensure reliable electrical power, cooling water and instrument air supplies to the safety systems. Eight nuclear steam plant (NSP) standby generators are provided for the two units. Four NSP standby generators are "pre-assigned" to specific distribution buses in the respective unit. Two additional BOP standby generators provide backup to the NSP for postulated station blackout events.

Safety systems and their support services are designed to perform their safety functions with a high degree of reliability. This is achieved through the use of redundancy, diversity, separation, testability, the application of appropriate quality assurance standards, and the use of stringent technical specifications, including seismic and environmental qualification for accident conditions.

2.9.1 Shutdown Systems

The ACR-1000 incorporates two passive, fast-acting, fully capable, diverse and separate shutdown systems, which are physically and functionally independent of each other.

SDS1 consists of mechanical shutoff rods that drop by gravity into the core when a trip signal de-energizes the clutches that hold the shutoff rods out of the core. The design of the shutoff rods is based on the proven

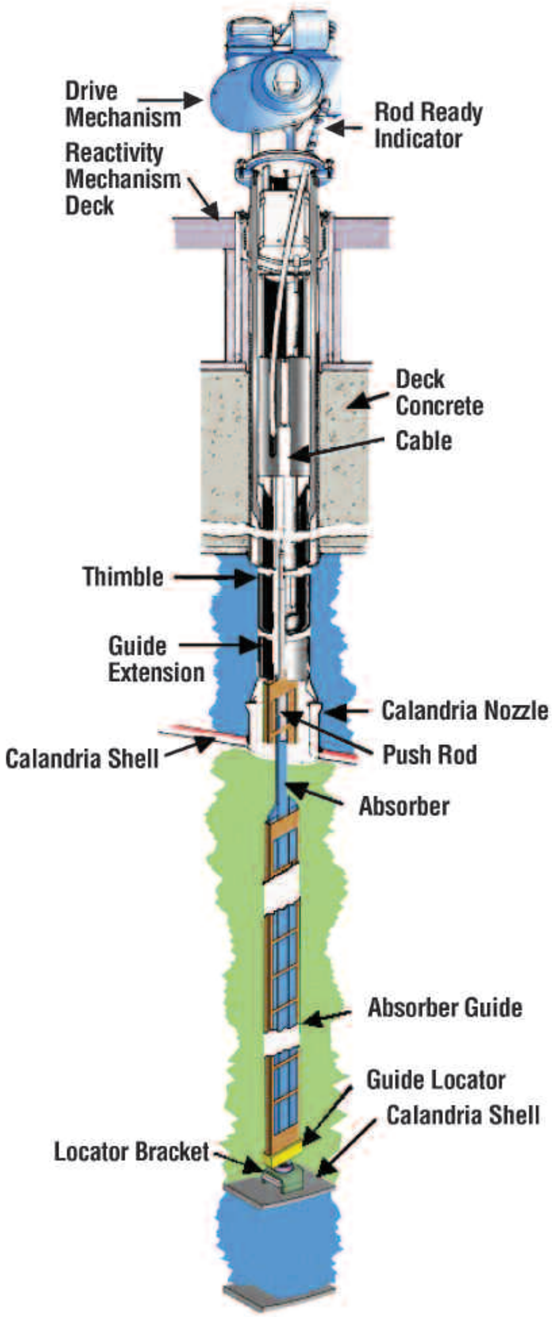


Figure 2-20 SDS1 Shutoff Rods

CANDU 6 design. The in-core portion of the shutoff rods has been designed to accommodate the smaller ACR-1000 core lattice pitch.

SDS2 injects a concentrated solution of gadolinium nitrate into the low-pressure moderator to quickly render the core sub-critical. The gadolinium nitrate solution is dispersed uniformly with pressurized gas, maximizing shutdown effectiveness.

The reactor can be put into a guaranteed shutdown state (GSS) using a rod-based system. Design simplifications have been provided to achieve this.

2.9.2 Emergency Core Cooling (ECC) System

The ACR-1000 emergency core cooling (ECC) system consists of two subsystems:

- *Passive emergency coolant injection (ECI) system:* The ECI system has accumulator tanks that will supply high-pressure water to the HTS and refill the fuel channels in the short term after a loss of coolant accident (LOCA)

  During normal operation, the ECI system is poised to detect any LOCA that results in a depletion of HTS inventory to such an extent that make-up by normal means is not assured. When the HTS pressure drops below the pressure of the ECI accumulator tanks, water is injected into the heat transport system.

  Valves on the ECI interconnect lines between the reactor outlet headers (ROH) open upon detection of a LOCA to assist in establishing a sustainable cooling flow path.

  In addition, core makeup tanks (CMTs) provide passive makeup to the intact HTS loop following a LOCA and prevent voiding for secondary side depressurization events.

- *Long-term cooling (LTC) system:* The LTC system provides long-term recirculation and recovery. It is used for cooling of the reactor after postulated transients, including LOCA, and during maintenance.

  LTC is initiated automatically when HTS is sufficiently depressurized, at which time the LTC system begins operation in long-term recovery mode.

2.9.3 Containment System

The ACR-1000 containment system forms a continuous, pressure-retaining envelope around the reactor core and the heat transport system. This prevents releases of radioactive material to the external environment.

The containment boundary consists of a steel-lined, pre-stressed concrete reactor building, access airlocks and a containment isolation system. The containment design ensures a low leakage rate. Hydrogen control is provided in the reactor building by passive autocatalytic recombiners and igniters to limit the hydrogen content to below deflagration limit within the containment, following a core damage accident.

Finally, the provision of a spray system connected to the elevated reserve water tank (RWT) will reduce reactor building pressures, if required, in the event of severe accidents.

Heat removal from the containment atmosphere is also normally provided by the operation of local air coolers, which are suitably located in various compartments of the reactor building, to reduce pressure and further reduce leakage over a longer period following an event.

2.9.4 Emergency Feedwater (EFW) System

The emergency heat removal function is accomplished by the EFW system. The system provides an independent supply of feedwater to the steam generators to remove decay and sensible heat to cool down the reactor following a total loss of the main and emergency feedwater systems.

The emergency feedwater system consists of emergency feedwater pumps driven by normal Class IV power and backed up by standby Class III electrical power. These pumps provide emergency feedwater to the steam generators at a rate sufficient to remove decay heat from the reactor core following a design basis event. Emergency feedwater is supplied from the reserve feedwater tank. All the components and valves of the system are seismically-qualified and are located in the seismically-qualified reactor building and reactor auxiliary building.

2.10 Essential Service Water Systems

The ACR-1000 adopts a four-division concept for essential service water systems. All divisions are physically separate, redundant and equipment in each is identical. Systems are sized to ensure that, under accident conditions, two divisions are capable of handling plant safety shutdown heat loads.

2.11 Balance of Plant (BOP)

The balance of plant (BOP) comprises the turbine building, steam turbine, generator, condenser, and the feedwater heating system with associated auxiliary and electrical equipment. The BOP also includes the water treatment facility, auxiliary steam facilities, condenser cooling water pumphouse and/or cooling towers, and associated equipment to provide all conventional services to the plant.

Turbine Generator	CANDU 6	Darlington	ACR-1000
Steam Turbine Type	Hitachi impulse-type, tandem-compound	Tandem-compound	Impulse-type tandem-compound
Steam Turbine Composition	One double-flow high-pressure cylinder	One double-flow high-pressure cylinder	One double-flow high-pressure cylinder
Net to turbine (MWth)	2060	2650	3180
Gross/Net electrical output* (nominal) [MWe]	728/666	935/881	1165/1085
Turbine Generator Efficiency**	35.3%	35.3%	~36.6%
Steam temperature at main stop valve [°C]	258	263	273
Final feedwater temperature [°C]	187	177	217
Condenser Vacuum [kPa (a)]	4.9	4.2	4.9

CANDU 6 data quoted is based on the Qinshan Phase III CANDU 6 design.
* Approximate values: electrical output is dependent on site conditions.
** Motor-driven feedwater pump, CANDU 6 and ACR-1000 outputs are based on reference cooling water temperature of 18.8°C. Darlington output is based on reference cooling water temperature of 11°C.

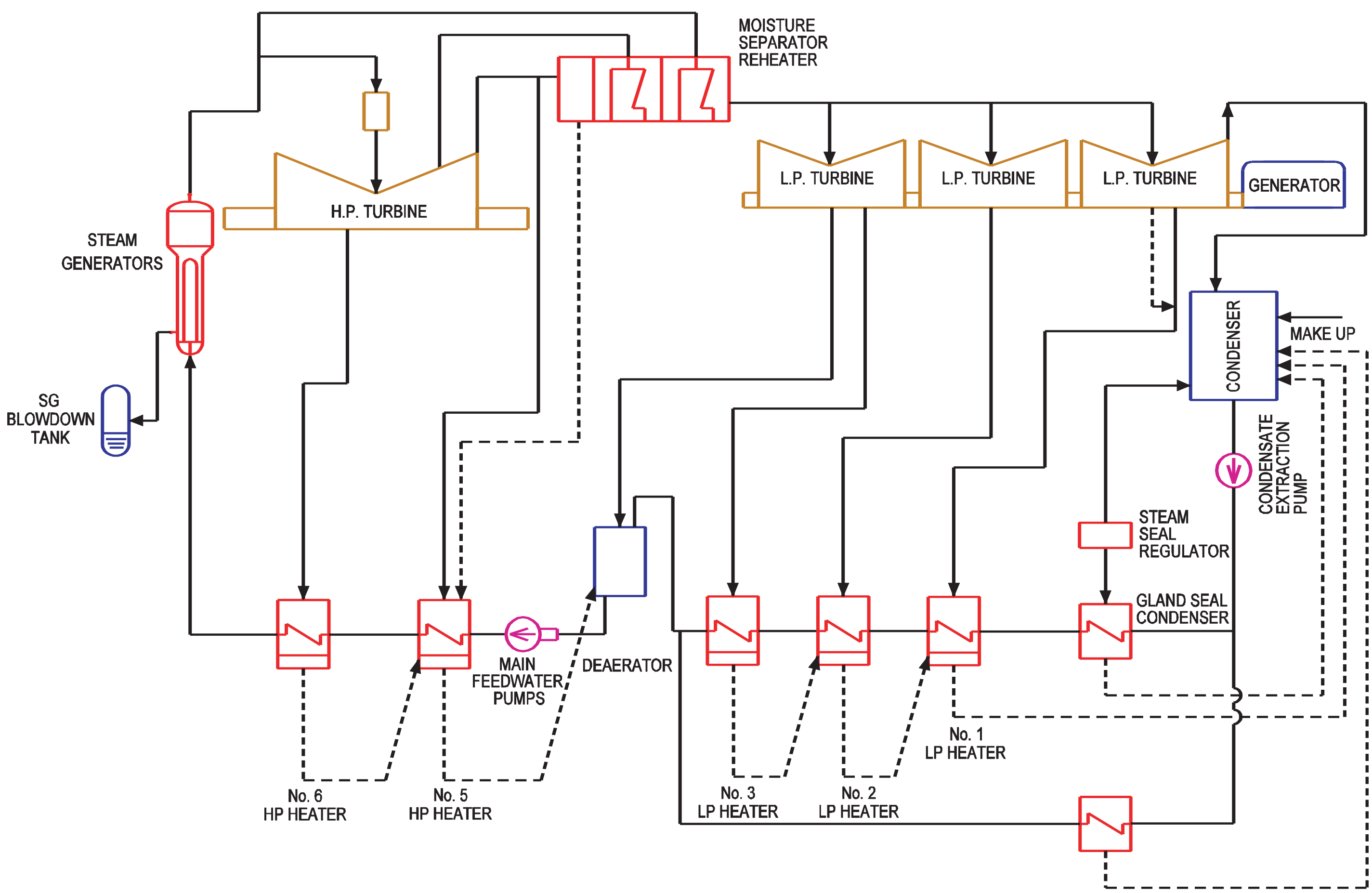


Figure 2-21 Turbine Generator and Auxiliaries Flow Diagram

2.11.1 Turbine Generator and Auxiliaries

The turbine generator system and the condensate and feedwater systems are based on conventional designs. They meet the design requirements specified by the NSP designer to assure the performance and integrity of the nuclear steam plant. These include requirements for materials (i.e., titanium condenser tubes, absence of copper alloys in the feed train), chemistry control, feed train reliability, feedwater inventory and turbine bypass capability.

In the event of station blackouts, the reactors are designed to stay at power for the duration of the event with the turbine generators disconnected from the grid. In this mode of operation, power is only supplied to internal auxiliaries as needed for the safe operation of the plant.

The BOP is capable of daily and weekly power manoeuvring to as low as 50%.

Figure 2-22 Qinshan Low-Pressure Turbine Rotor

CANDU plants have operated successfully using North American, European and Japanese turbine generators with fresh water and seawater condenser cooling water.

2.11.2 Steam and Feedwater Systems

The ACR-1000 main steam system supplies the steam from the steam generators in the reactor building to the turbine through the steam balance header. The feedwater system takes hot, pressurized feedwater from the feedwater train in the turbine building and discharges it into the pre-heater section of the steam generators. The system maintains the required steam generator level by controlling feedwater flow.

The condenser steam discharge valves (CSDVs) are designed to discharge up to 100% of steam flow directly to the condenser. This feature provides for operational flexibility in support of load following operation in conjunction with overall reactor control. It also provides a backup safety function for fuel cooling, via steam generator cooling, by making use of the large inventory in the condenser.

The safety functions of overpressure protection and cooling of the steam generator secondary side are provided by main steam safety valves (MSSVs). In addition, main steam isolation valves (MSIVs) can be used to prevent releases from containment in the event of steam generator tube leaks to the secondary side.

2.11.3 BOP Services

Conventional plant services include potable water supply, heating, ventilation, air conditioning, chlorination (if required), fire protection, compressed gases and electric power systems.

Service Water Systems

The balance of plant (BOP) water systems provide cooling water, de-mineralized water and domestic water to plant users. The systems consist of the condenser cooling water (CCW), plant water system, water treatment facility and chlorination systems.

Heating, Ventilation and Cooling Systems

Heating, ventilation, air conditioning and chilled water (from the chilled water system) are supplied to plant buildings to ensure a suitable environment for personnel and equipment during winter and summer. The building heating plant provides the steam and hot water demands of the entire plant. Steam extracted from the turbine is used as the normal building heating source. Dedicated, separate ventilation systems are provided for the main control building and secondary control building.

Fire Protection System

Water supply for the main fire protection system comes from a fresh water source via a buried pipe circuit. The main system provides fire protection for the entire station (i.e., both NSP and BOP).

The fire protection system also includes standpipe and fire hose systems, portable fire extinguishers for fire suppression, and a fire detection and alarm system covering all plant buildings and areas.

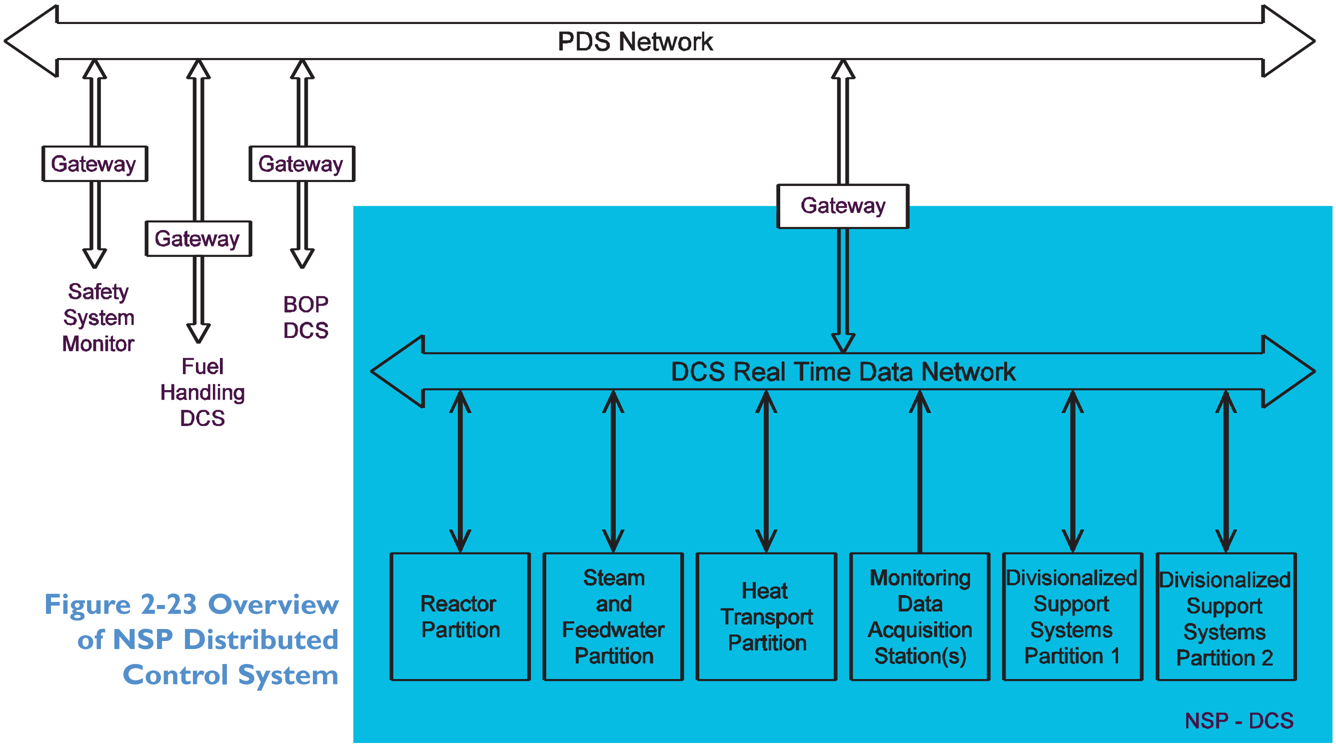


Figure 2-23 Overview of NSP Distributed Control System

Fire-resistant barriers are provided for mitigation purposes, where necessary, to isolate and localize fire hazards and to prevent spread of fire to other equipment and areas. The four-quadrant layout in the reactor auxiliary building provides maximum separation of redundant safety equipment for added fire protection.

2.12 Instrumentation and Control

The ACR-1000 unit control and monitoring systems apply modern distributed control, display and network communication technologies. Safety system logic and control are based on four-channel architecture to provide fault tolerance protection and to minimize spurious reactor trips. This results in enhanced monitoring capability and contributes to lower operating and capital costs due to:

- Reduction in the number of instrumentation and control components, leading to improved reliability and reduced maintenance and construction costs
- Design simplification through permitted sharing of systems, enabling the reduction in the number of sensors
- Increased automation, thus reducing frequency of operator error
- Improved information and data communications systems that provide detailed information on unit operational state, enabling early detection and diagnosis of faults and improving timely preventive equipment maintenance, thereby reducing unplanned plant outages

Most control functions are performed by a state-of-the art distributed control system (DCS) that uses small, programmable digital controller modules in place of a single central computer. The controllers communicate with one another by means of data highways, which use reliable, high-security data transmission methods. Manual control commands to be executed by the DCS are entered by the operators via the plant display system.

Control Centre

The ACR-1000 plant control centre enables operating staff to monitor, control and effectively operate the units in both normal and abnormal modes.

A computerized plant display system (PDS) is used for all plant control and monitoring. Integrated computer technology is used throughout the controls, displays, panels and consoles. These link operating procedures, testing requirements and configuration management to achieve high plant performance and enhanced operator effectiveness.

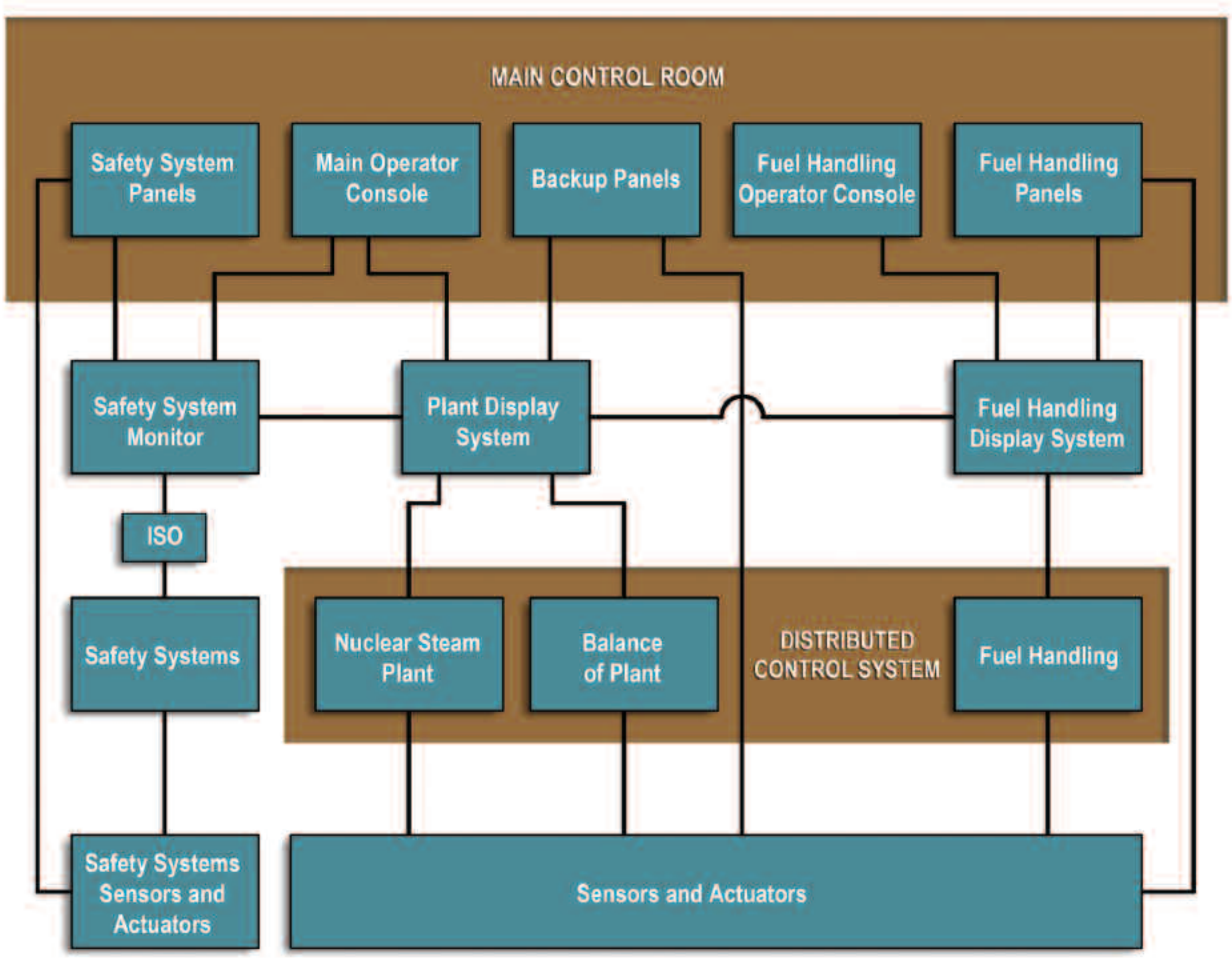
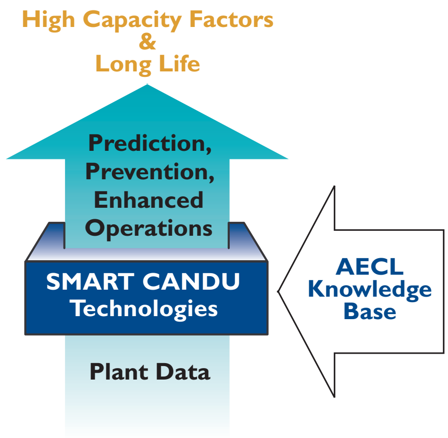


Figure 2-24 Plant Control and Monitoring Systems

The control centre information system includes an advanced alarm annunciation capability, based on the CANDU annunciation message list system (CAMLS) implemented on the Qinshan units. It conveys up-to-date unit information through fault and status displays. The control centre information system also includes an alarm interrogation application that allows operations staff to view fault and status display and to interrogate alarm history from any of the control centre panels or console workstations. The control centre information system includes on-line procedures for operator support.

Each unit has a completely separate secondary control building (SCB) to control and monitor equipment required to shut down the unit, initiate the required fuel cooling, and monitor equipment and plant state to ensure the unit remains in a safe shutdown should the main control room (MCR) become unavailable.

The ACR-1000 will also provide an integrated package of software tools and work processes aimed at plant performance optimization throughout its life cycle. SMART CANDU technologies use the AECL knowledge base and plant data to predict, prevent and enhance operations. The SMART CANDU suite of tools includes ChemAND and other superior engineering tools.



CAMLS
Intelligent Annunciation Message List System that assists operators in coping with events such as blackouts.
ChemAND
Health monitor for plant chemistry. Predicts future performance of components, determines maintenance requirements and optimal operating conditions.
ThermAND
Health monitor for heat transfer systems and components. Ensures optimal margins and maximum power output.
MIMC
Maintenance Information Management Control system that links health monitor data to the plant work management system.

Figure 2-25 SMART CANDU

2.13 Electrical Power System

The electrical power system consists of connections to the off-site grid, main turbine generator, associated main output system, on-site standby diesel generators, battery power supplies, uninterruptible power supplies (UPS) and the distribution equipment. Essential standby generators, batteries, UPS and the equipment distributing power from these sources are seismically and environmentally-qualified. This equipment is provided in a four-bus configuration, which improves reliability, allows for on-power maintenance and minimizes potential for spurious trips.

The electrical distribution system (EDS) supplies electrical power to all process and instrumentation and control loads within the unit. The EDS is divided into four classes of power based on availability: Class I is delivered from batteries, Class II from UPS, Class III from standby generators and Class IV from the main generator or grid.

In a two-unit ACR-1000 plant, each unit has a dedicated electrical distribution system with inter-unit ties only in the Class III distribution system. Four seismically-qualified, essential standby generators are provided for each unit. Two additional standby generators are provided to support station operation, including 'blackouts.'

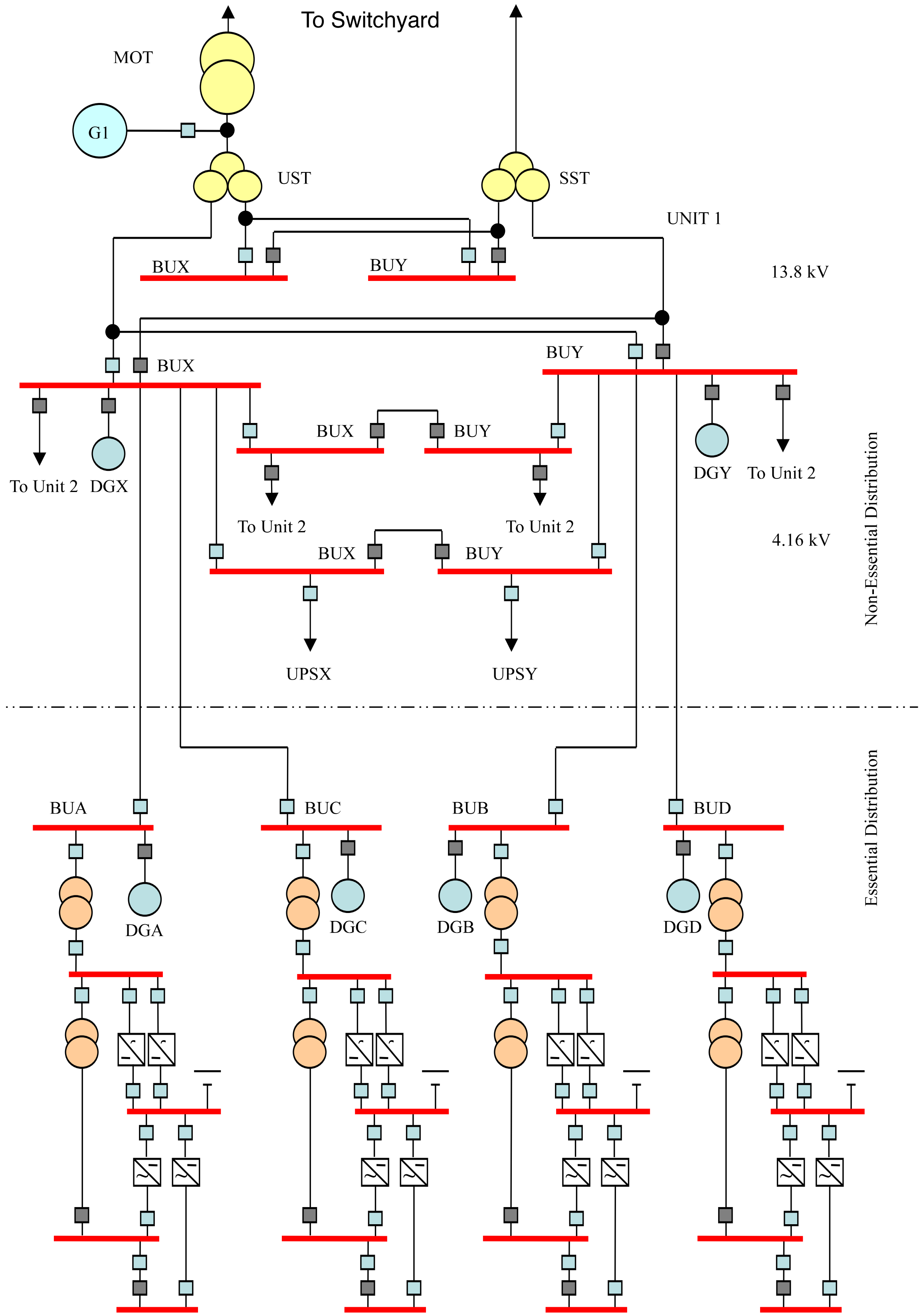


Figure 2-26 Unitized Electrical Power System

3. Nuclear Safety and Licensing

3.1 Safety Design

Nuclear safety requires that the radioactive products from the nuclear fission process be contained, both within the plant systems for worker protection and outside the plant structure to protect the public. This is achieved at all times by:

- Controlling the reactor power, and if necessary, shutting the reactor down
- Removing reactor heat, including decay heat following shutdown, in order to prevent heat up of fuel
- Containing radioactive products that are normally produced and contained within the fuel
- Monitoring the plant to ensure that the above functions are being carried out, and if not, ensuring that mitigating actions are being taken

These nuclear safety functions are carried out to a high degree of reliability by applying the following principles:

- The use of high-quality components and installations
- Maximizing the use of inherent safety features of the ACR-1000
- Implementing multiple defence-in-depth barriers for prevention of radioactive release
- Providing enhanced features to mitigate and reduce consequences of design basis events and severe accidents

The implementation of these safety measures is provided by safety systems, safety support systems, systems important to safety and robust buildings and structures that meet high standards for diversity, reliability and protection against common-mode events such as seismic occurrences, fires, flooding and unauthorized acts.

3.2 Defence-in-Depth

The ACR-1000 is based on the CANDU principle of defence-in-depth by providing the following multiple, diverse barriers for accident prevention and mitigation of consequences:

- High-quality process systems to accommodate plant transients and to minimize the likelihood of accidents
- Reliable safety systems for reactor shutdown, emergency core cooling, containment, and emergency heat removal (emergency feedwater)
- Reliable safety support systems to provide services to the safety systems and other mitigating systems
- Backup systems for heat sinks and essential controls
- Passive heat sinks to increase resistance against both design basis events and severe accidents

The ACR-1000 has at least seven barriers:

1) Fuel sheath which contains the radioactive material
2) Heat transport system, including pressure tubes
3) Calandria tubes designed to withstand a pressure tube rupture
4) Cool, low-pressure moderator
5) Cool, low-pressure reactor vault
6) Reserve water system
7) Steel-lined, concrete containment structure

The design of the safety systems follow the design principles of separation, diversity and reliability. High degrees of redundancy within systems are provided to ensure the safety functions can be carried out, even when systems or components are impaired. Protection against seismic, flooding and fire events is also provided, ensuring highly reliable and effective mitigation of postulated events, including severe accidents.

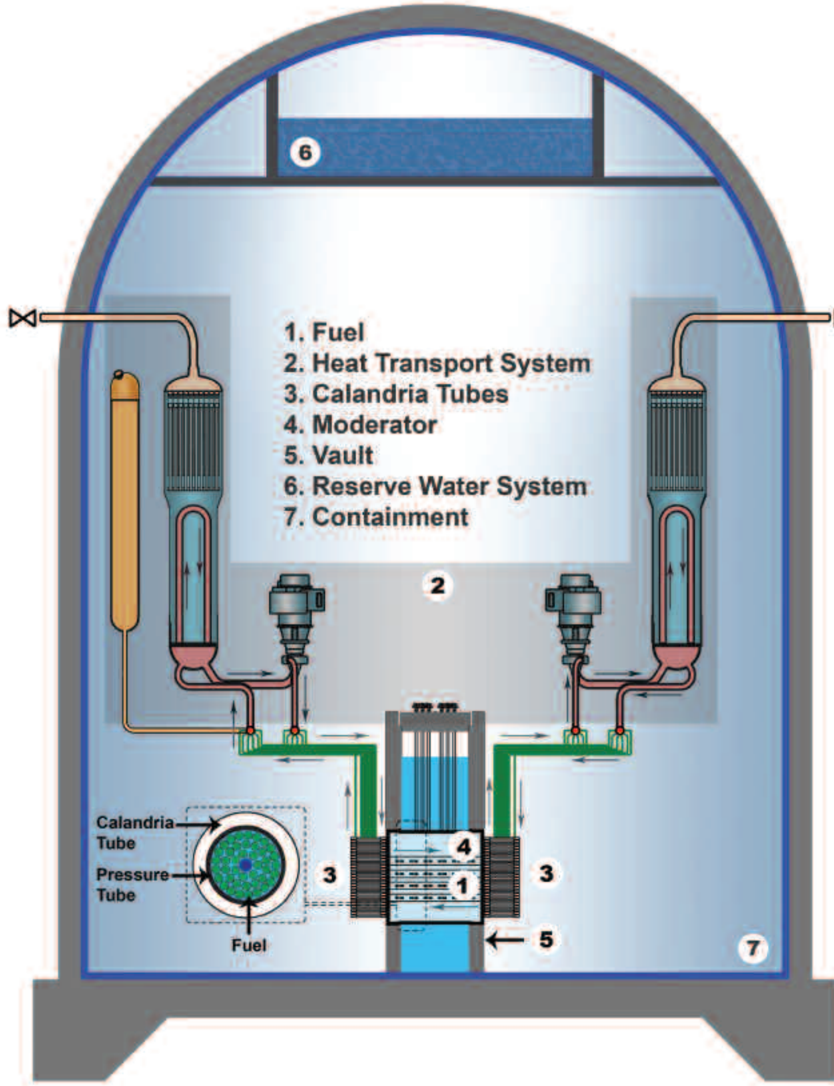


Figure 3-1 Barriers for Prevention of Releases

3.3 Inherent Safety Features

The ACR-1000 maintains the traditional CANDU inherent safety characteristics:

- Heavy water moderator, which is very efficient in slowing down neutrons, resulting in a fission process which is more than an order of magnitude slower than LWRs. Reactor control and shutdown are inherently easier to perform
- On-power refuelling, which reduces the 'excess' reactivity as required. Reactor characteristics are constant and no additional measures, such as boron addition to the coolant (and its radioactive removal), are needed
- Natural circulation capability in the reactor coolant system, which can cope with transients due to loss of forced flow
- Reactivity control devices. These are in the low-pressure moderator, do not penetrate the reactor coolant pressure boundary and therefore cannot be ejected
- Moderator backup heat sink, which maintains core coolability for loss-of-coolant accidents, even when combined with the unavailability of emergency core cooling
- Negative power reactivity coefficient, which makes reactor power more stable and easier to control
- Small negative full-core void reactivity offering a good balance of nuclear protection between loss-of-coolant accidents and fast cool-down accidents
- Very flat and stable flux across the core minimizing demand on the reactor control system
- Larger safety and operating margins due to the use of CANFLEX-ACR fuel, with lower element rating and higher critical heat flux limits

3.4 Severe Accidents

A severe accident is one in which the fuel is not cooled within the heat transport system. The ACR/CANDU design principle is to prevent severe accidents and to mitigate severe accident events, in addition to minimizing their consequences. This is achieved by providing a number of design measures:

- Normal heat removal systems
- Heat removal systems using emergency feedwater system
- Passive emergency feedwater supply from reserve water system
- Emergency core cooling
- Passive emergency heat transport system make-up from reserve water system
- Heat removal using moderator systems
- Passive thermal capacity of moderator
- Passive emergency moderator heat sink make-up from reserve water system
- Heat removal by reactor vault water
- Passive thermal capacity of reactor vault water
- Passive emergency reactor vault heat sink make-up from reserve water system
- Passive containment cooling via spray
- Severe accident management monitoring capabilities

Severe accident management, in addition to providing multiple mechanisms for fuel cooling and barriers to release, also includes mitigating measures within containment. In addition to the robust, concrete outer and inner steel liners, which by themselves can withstand the largest pipe breaks, containment is also provided by:

- Passive, hydrogen recombiners and igniters that will limit the hydrogen content to below the deflagration limit
- A spray system to reduce the build-up of containment pressure and reduce leakages
- Highly reliable local air coolers that can be used for containment heat removal

PSA studies estimate that the summed frequency of internal initiating events leading to reactor core damage during at-power operation is only 3.4×10^{-7} for the ACR-700 and is expected to be better for the ACR-1000. This exceeds EPRI requirements by approximately two orders of magnitude and is comparable to latest LWR designs. This marginal value is comprised of probabilities of seven dominant initiating events, all of which are relatively small.

3.5 Licensing Basis

The ACR-1000 builds on the successful CANDU track record of accommodating regulatory requirements of offshore jurisdictions in various host countries (China, South Korea, Romania, Argentina) while retaining the standard nuclear platform.

The ACR-1000 is designed to meet regulatory requirements in Canada and other countries:

- The ACR-1000 is an evolutionary, enhanced design based on current regulations. Future licensability in Canada and abroad will be based on this experience
- ACR-1000 design meets the requirements of applicable IAEA Safety Series documents for nuclear power reactors
- The design meets the Canadian and international requirements for nuclear plant siting
- International codes and standards, as they apply to the ACR-1000 design, have been incorporated. ACR-1000 has benefited from the extensive review of US NRC requirements—both its written regulations and via dialogue

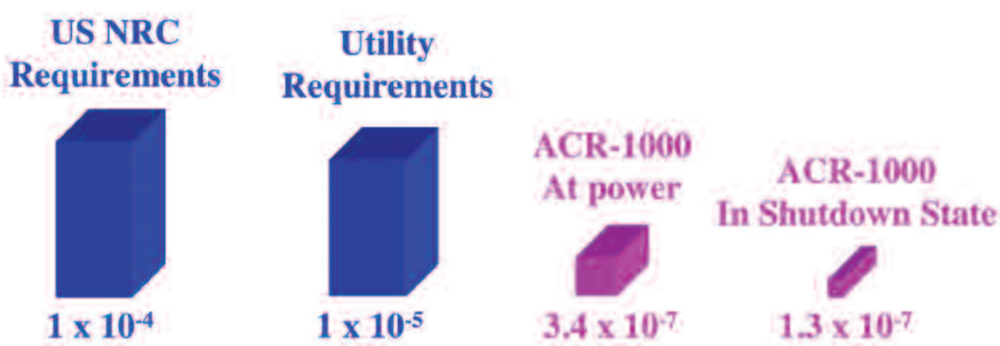


Figure 3-2 Core Damage Frequencies per Year

4. ACR-1000 Deployment

The feedback gained from AECL's past construction projects, associated with improvements and optimization of key project elements, results in an optimum 42-month schedule (n[th] Unit) from first Containment Concrete to fuel load. Deployment of the ACR-1000 requires the coordination and timely delivery of key project elements, including: licensing programs, environmental assessments, design engineering, procurement, construction and commissioning start-up programs.

Design Engineering: Prior to a project contract, a series of activities are executed to ensure design readiness and a seamless transition to the procurement and construction phases. Preliminary design and research and development programs are executed in parallel with the environmental assessment and licensing programs, ensuring continuous improvement and plant configuration is maintained. The final design program ensures plant reliability, equipment and component maintainability and constructability requirements are maximized to the fullest extent.

Licensing: The ACR-1000 builds on the successful CANDU track record of accommodating requirements of offshore jurisdictions in various host countries while retaining the standard nuclear platform. Licensing programs are executed and coordinated with the engineering design programs and environmental assessment, and are structured in a manner to support regulatory process requirements.

Configuration Management: The ACR-1000 makes use of the latest computer technology for managing the complete plant configuration from design to construction and finally, turnover to the Owner. State-of-the-art electronic drafting tools are integrated with material management, wiring and device design, and other technology applications.

Project Management: The ACR-1000 project management structure provides fully integrated project management solutions. Performance management programs are executed from project concept, through a project readiness mode, and finally project closeout. The project management framework consists of three key elements:

- Total project execution planning
- Critical decision framework to control each phase of the project
- Comprehensive risk management program

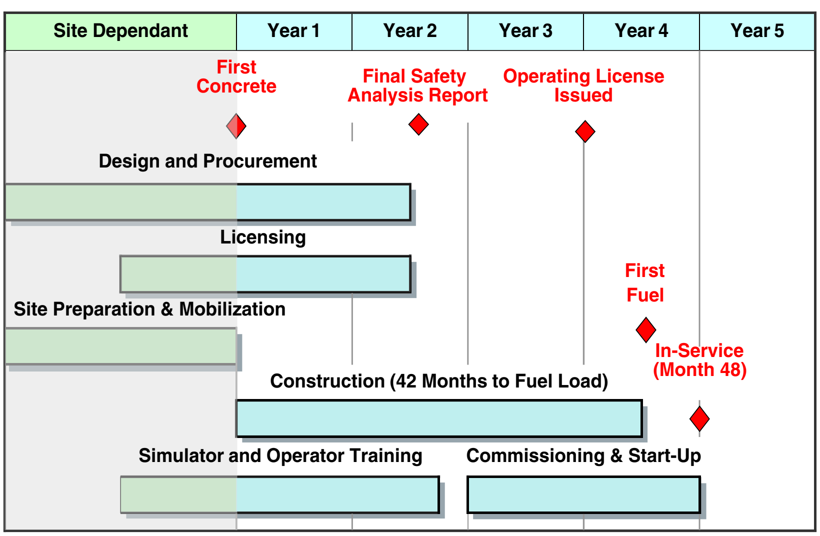


Figure 4-1 42-Month Deployment Schedule (Nominal)

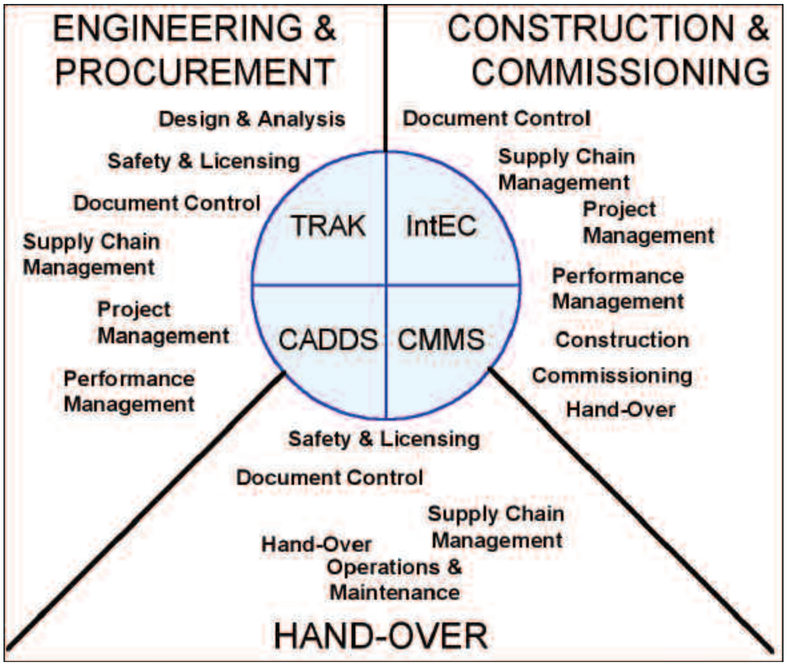


Figure 4-2 Design Engineering Applications

The open-top/vertical installation construction method enables an improved logic that reduces costs while reducing the schedule risk. The internal structure of the reactor building is initially built as vertical walls without floors. Major modules, including the floors, are then installed in parallel.

Commissioning: The commissioning and plant start-up programs for the ACR-1000 are being developed with input received from design staff and plant operations staff. Identification of key design parameters that require confirmation to meet overall system objectives are reviewed to ensure commissioning plans can be produced to check those identified parameters. In addition, acceptance criteria will be developed between the designer and experienced commissioning technical staff.

Procurement: Standardized procurement and supply processes are implemented to support time, cost and performance benefits to the project, including benefits such as efficiency through variety control (standardization), economy in manufacturing and servicing, and avoidance of repetitive effort in producing new specifications and processes for each procurement.

Construction Programs: Constructability programs are implemented to ensure simplification, maximized concurrent construction, increased construction productivity, minimized construction rework, decreased construction equipment costs, minimized unscheduled activities, and reduced capital costs and construction risk.

Construction Strategy: The main elements of the ACR-1000 construction strategy are:

- Open-top construction method using a very-heavy-lift crane
- Concurrent construction
- Modularization and prefabrication
- Use of advanced technologies to minimize interferences.

Test programs will be defined as part of the overall plan, including:

- Preoperational tests
- Fuel loading, initial criticality, and low power tests
- Power tests
- Test run and performance tests

Figure 4-3 Module Lift Using VHL Crane

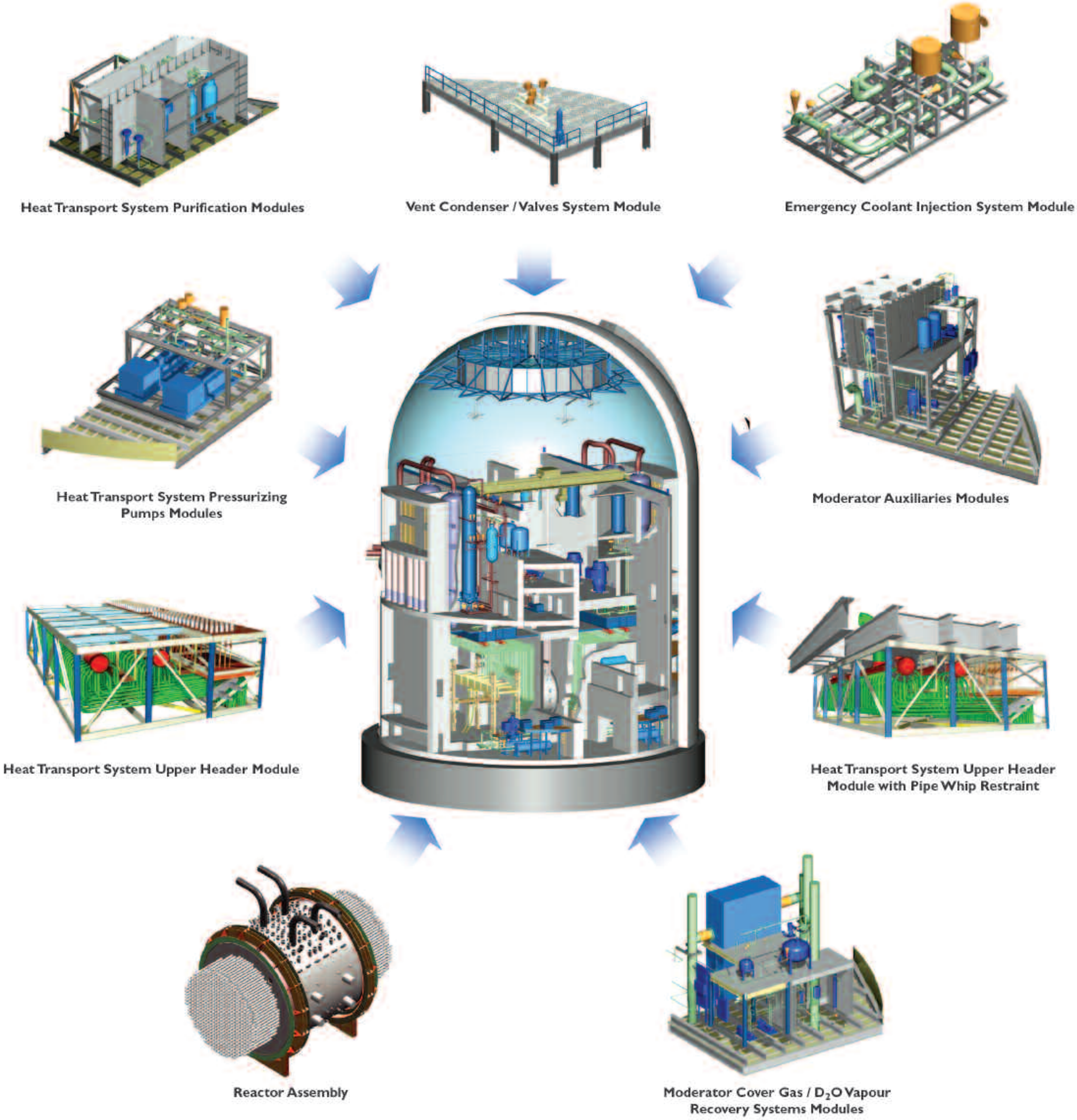


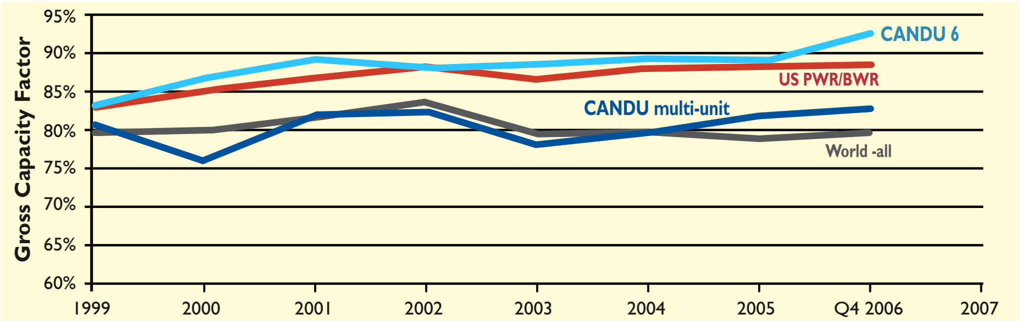
Figure 4-4 Typical Reactor Building Modules

5. Operation and Maintenance

5.1 Consistently Better Performance

The lifetime capacity factor for the ACR-1000 is expected to be greater than 90% over the operating life of 60 years. The year-to-year expected capacity factor is 95%. These expectations are based on the proven track record of CANDU 6s, which have collectively surpassed the U.S. PWR/BWR Gross Capacity Factor (GCF) with a combined average of 92.4% in 2006. These results are consistently better than LWRs around the world.
The ACR-1000 has made a number of improvements to achieve these incremental performance targets.

CANDU 6/PHWR Performance Trends (1999 - 2006)
Reference: CANDU Owners Group Newsletter



COGnizant Volume 12, Issue 6, 2007, 2006 U.S. and world data based on Q4 results (courtesy of NEI)
The graph is for comparison of trends only

Figure 5-1 Comparison of Gross Capacity Factors

5.2 Enhanced Performance Features

Incorporation of feedback from operating reactors (both CANDU and other designs) is an integral feature of the design process. Various new features and maintenance improvement opportunities have been incorporated to enhance operating performance throughout station life.

Major enhancements include:

- Use of improved material and plant chemistry specifications, based on operating experience from CANDU plants. For example, life-limiting components such as HTS feeders and headers have been replaced with stainless steel to limit the effect of feeder corrosion
- Implementation of advanced computer control and interaction systems for monitoring, display, diagnostics and annunciation. These include ergonomic operator consoles, touch displays, large colored screens, smart communications for improved operator awareness and plant status through modern human-factors engineering
- Providing integrated SMART CANDU modules for annunciation, on-line monitoring of systems and components, and providing a predictive maintenance capability

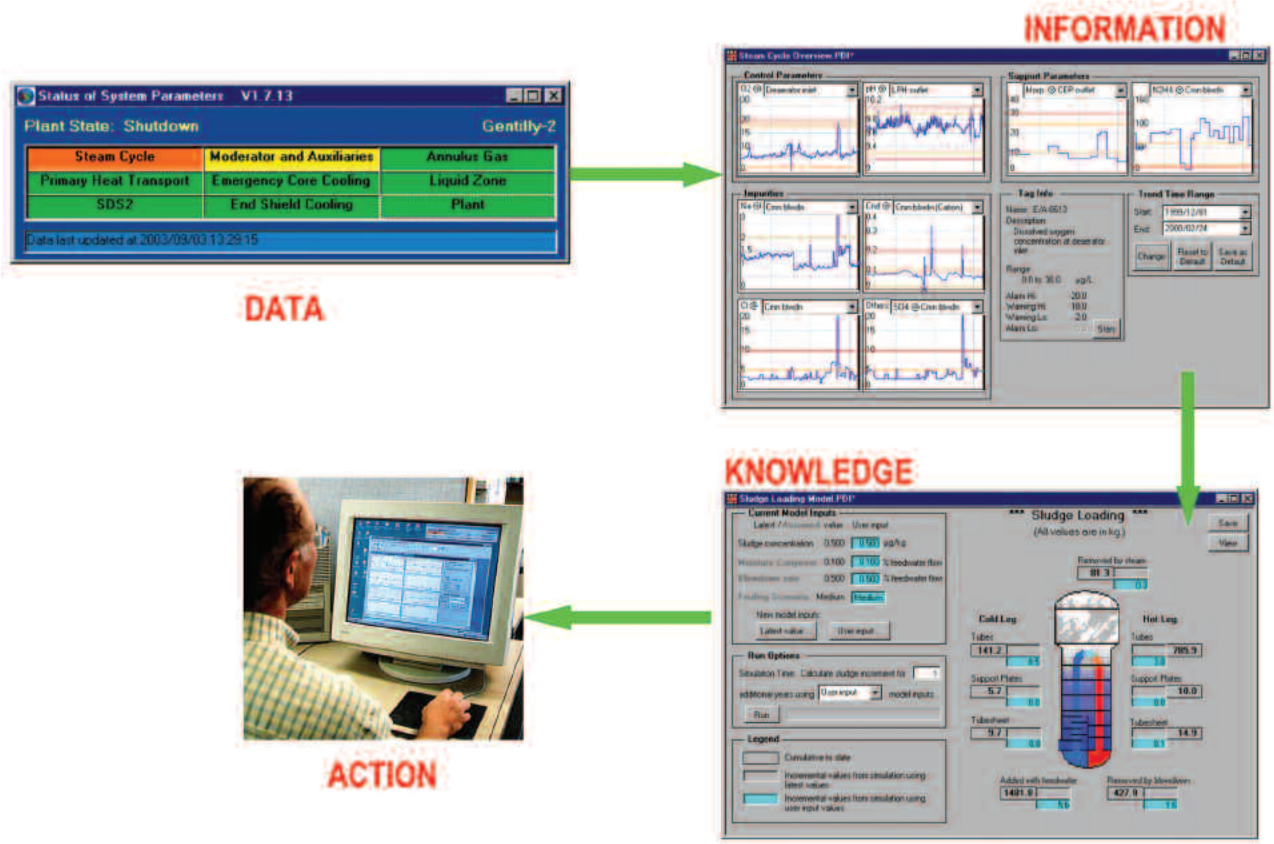


Figure 5-2 ChemAND – Performance Monitor for Plant Chemistry

- Enhancing power maneuvering capability:
  - Load-following the grid provides up to 2.5% power variation, while operating at 97.5%
  - Daily load-cycling capability includes rapid load reduction from steady state 100% power operation to 75%, and periodic load reduction from 100% to 60% and as low as 50% when required (e.g. , weekends)
  - Use of LEU fuel and light water coolant has resulted in a lower xenon load following reactor power reduction compared to CANDU. This simplifies reactor operation and makes the ACR-1000 inherently more responsive
- Ensuring station blackout capability for return to full power on restoration of electrical grid. The ACR-1000 has the capability to continue operation of house load without a grid connection, enabling a rapid return to full power upon reconnection

5.3 Enhanced Maintenance Features

The lifetime capacity factor of a plant is impacted by the number and duration of maintenance outages. The traditional 'annual' outage of up to one month for currently operating CANDU plants has been improved to a 'major' outage of only 21 days every three years for the ACR-1000. A number of enhancements to achieve these objectives have been incorporated.

- A maintenance-based design strategy has been implemented. The program incorporates lessons learned and ensures maintainability of systems and components. It will define the improvements made to maintenance programs for earlier designs. The new program is based on the SMART CANDU technology. It will identify and take mitigating actions, if required, to ensure plant states are diagnosed and maintained within their design performance limits. This will lead to improved preventive maintenance and reduced forced outages at a rate of less than five days/year. Only the best available equipment for critical components will be used

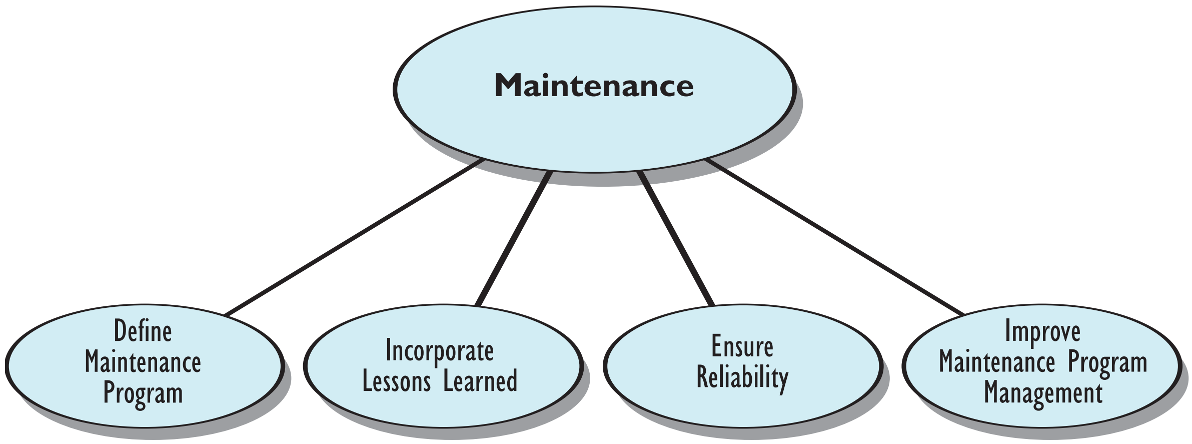


Figure 5-3 Maintenance Basis

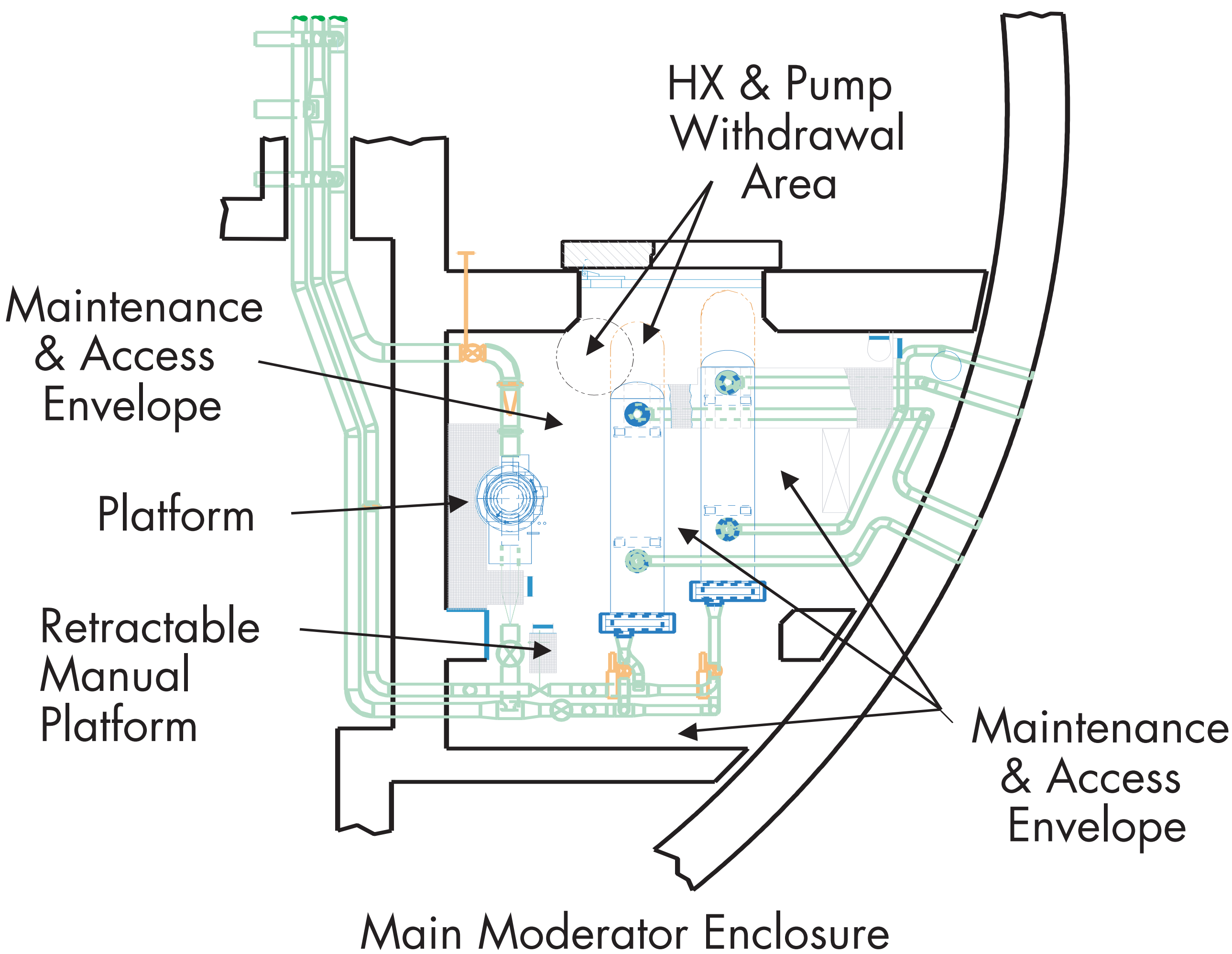


Figure 5-4 Typical System Equipment Module

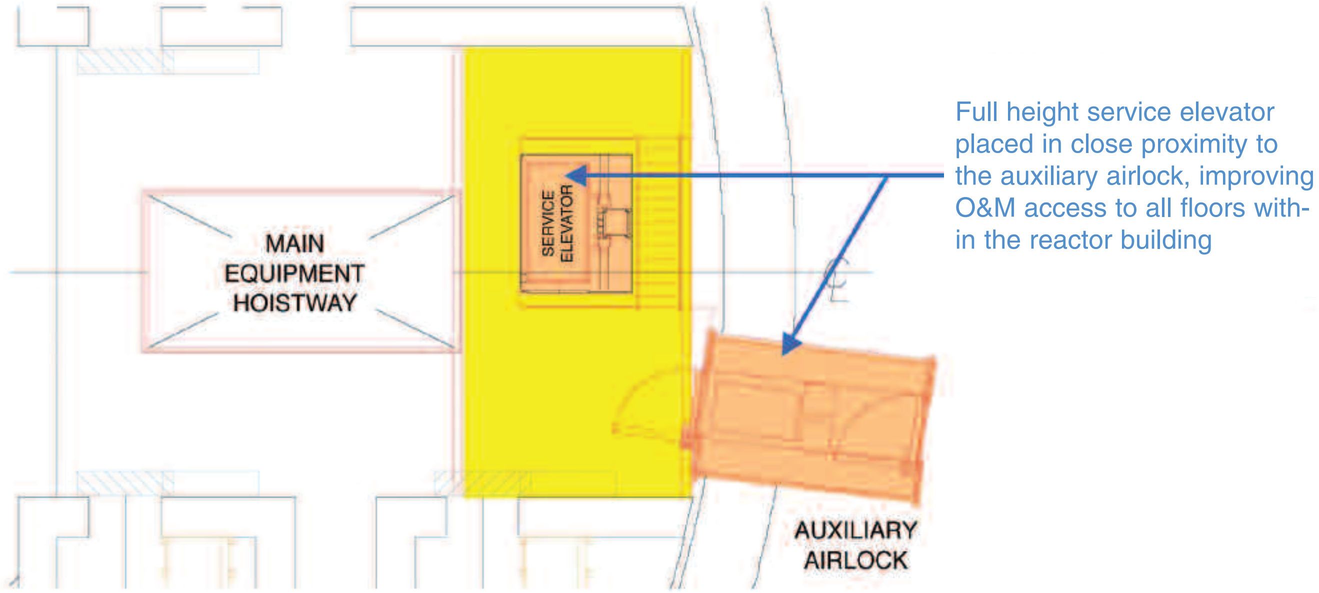


Figure 5-5 Service Elevator

- Plant layout has been improved by providing generous space, laydown areas, good lighting, and use of permanent walkways and platforms to minimize need for temporary scaffolding. Provision for electrical, water and air supplies are built-in for on-power and normal shutdown maintenance
- Effective use of on-power reactor building accessibility and on-power maintenance of four-division design safety systems will minimize the amount of maintenance that must be performed during shutdown
- Computerized testing of major safety systems and automatic calibration of in-core detector control signals reduce both on-line testing and start-up testing time
- More durable materials and robust design margins simplify fuel channel inspections
- Shielding in radiologically-controlled areas has been increased. This feature, along with reduced tritium releases due to use of light water coolant, will result in enhanced radiological protection to further reduce worker exposure and occupational dose. Dose to an individual station staff member is expected to be less than 50 mSv in any single year
- The design for planned outages every three years is accomplished by selection of equipment and system design. It is based on probabilistic safety evaluations using three-year outage intervals

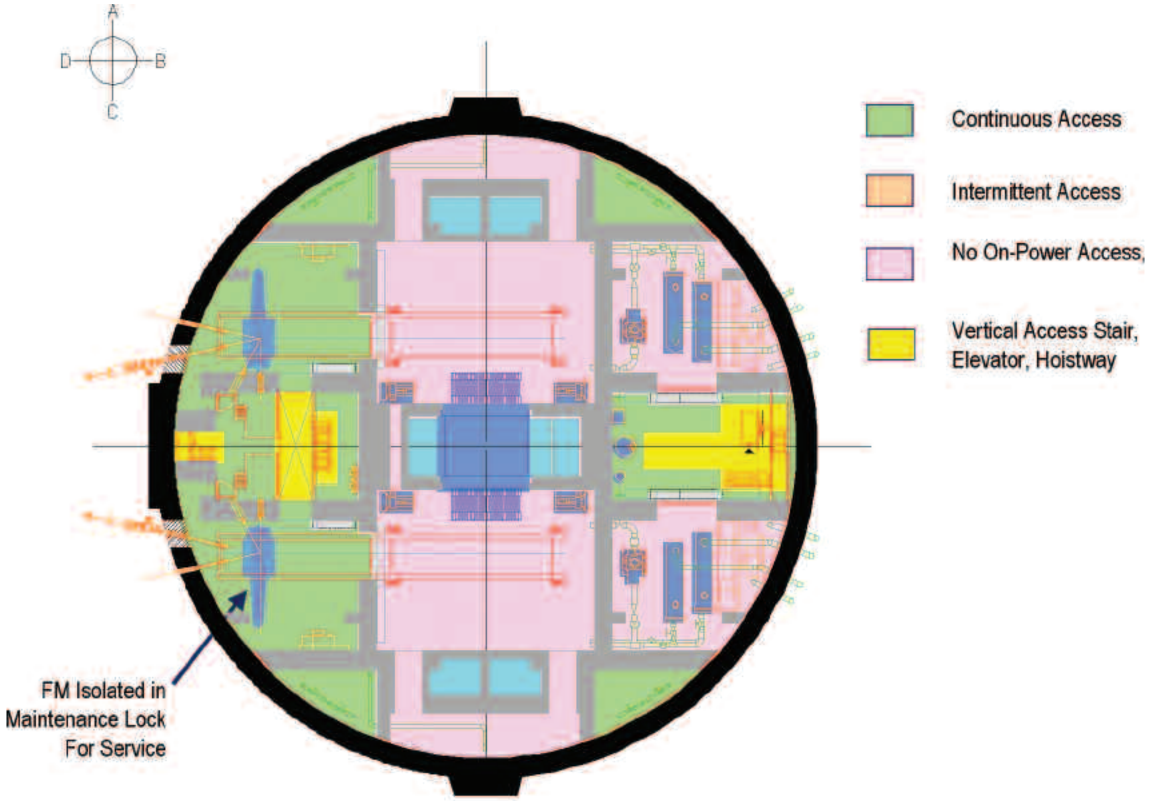


Figure 5-6 Accessible Areas in the Reactor Building – Level 100 m

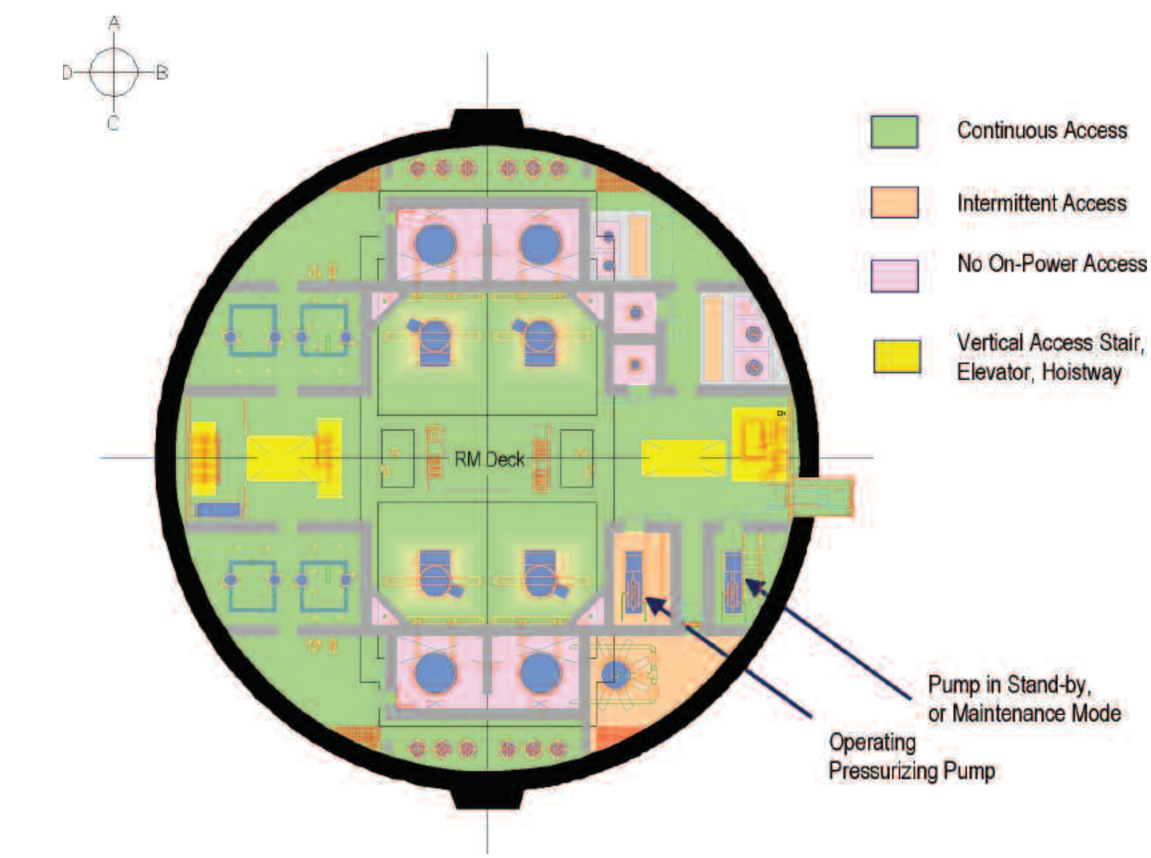


Figure 5-7 Accessible Areas in the Reactor Building – Level 125.4 m

The plant layouts above show the accessible areas in the plant, enhanced for ease of operation and maintenance.

6. Radioactive Waste Management

The waste management systems for the ACR-1000 will minimize the radiological exposure to operating staff and the public. Exposures for workers from the plant are monitored and controlled to ensure they are within the limits recommended by the International Commission on Radiological Protection. The systems for the ACR-1000 have been proven over many years at other CANDU sites. They provide for the collection, transfer and storage of all radioactive gases, liquid and solid, including spent fuel and wastes generated within the plant:

- Gaseous radioactive waste gases, vapours or airborne particulates are monitored and filtered. Active gases are treated by the off-gas management system (OGMS) with an absorber bed. Any tritium releases from isolated moderator areas are collected by a vapour recovery system and stored on site
- Liquid radioactive wastes are stored in concrete tanks located in the maintenance building. Any liquid requiring removal of radioactivity, including spills, is treated using cartridge filters and ion-exchange resins
- Solid radioactive wastes can be classified by five main groups: spent fuel, spent ion-exchange resins, spent filter cartridges, compactable and non-compactable solids. Each type is processed and moved, using specially designed transporting devices, if necessary. After processing, wastes are collected and prepared for on-site storage by the utility or for transport off-site

AECL has developed the MACSTOR®**** (Modular Air-Cooled Storage) system for safe, above-ground storage of spent fuel. MACSTOR has been developed from more than 30 years of experience.

*****MACSTOR® is a registered trademark of Atomic Energy of Canada Limited (AECL).*

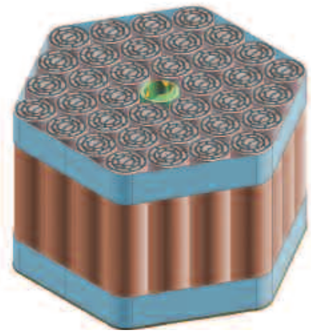


Figure 6-1 Spent Fuel Storage Basket

MACSTOR saves up to one-third of the space required for comparable systems, requires less manpower, has low operating and construction costs, and permits easy fuel retrieval.

With highly efficient heat-rejection and shielding capabilities, it is constructed using multiple barriers to provide radiation shielding for operators and the public, while being appropriately qualified and equipped with monitoring facilities.

Figure 6-2 MACSTOR Fuel Transfer

Figure 6-3 AECL's MACSTOR System

7. Decommissioning

AECL, through its membership in the OECD/NEA co-operative programme on decommissioning, has adopted a three-stage decommissioning strategy:

1) Placement of the station into a static state. This dormancy state is a modified IAEA Stage I concept such that:
    - Buildings around the reactor building are decommissioned for alternate use
    - The reactor building is isolated and sealed
    - The plant is monitored to ensure its dormant state

2) IAEA Stage II dormancy period, assumed to be 40 years or more, depending on the Owner's plans

3) IAEA Stage III final decommissioning to unrestricted use of the land

As an evolution of CANDU 6, the enhanced ACR-1000 design features a number of systems that have been simplified and/or optimized; some have also been eliminated. Thus, the amount of materials to be decommissioned is less than CANDU 6. Some examples are:

- Reduction of heavy water by elimination or downsizing of heavy-water-related systems
- Reactor core size reduction
- Consideration of alternative structural material yielding less cumulative radioactivity at end of life
- Civil structure size reductions

ACR-1000 design features that assist in maintenance and inspection during the lifetime of the reactor will also facilitate decommissioning. For example, the division of the reactor building into separate compartments, with proper isolation and shielding, allows the segregation of contaminated from non-contaminated systems, facilitating efficient dismantling, removal and disposal.

AECL has decommissioned three prototype Nuclear Power Plants and one research reactor to a static state. It has decommissioned at least one facility to IAEA's Stage III. AECL has also participated in decommissioning plans of facilities in Japan, the U.S. and elsewhere.

AECL has all the experience and facilities required to support Owner decommissioning plans.

8. Conclusion

Evolution

Capitalizing on the proven features of CANDU technology, AECL has designed the evolutionary ACR-1000 to be cost-competitive with all forms of energy, including other nuclear technologies, while achieving higher safety and performance standards consistent with customer expectations.

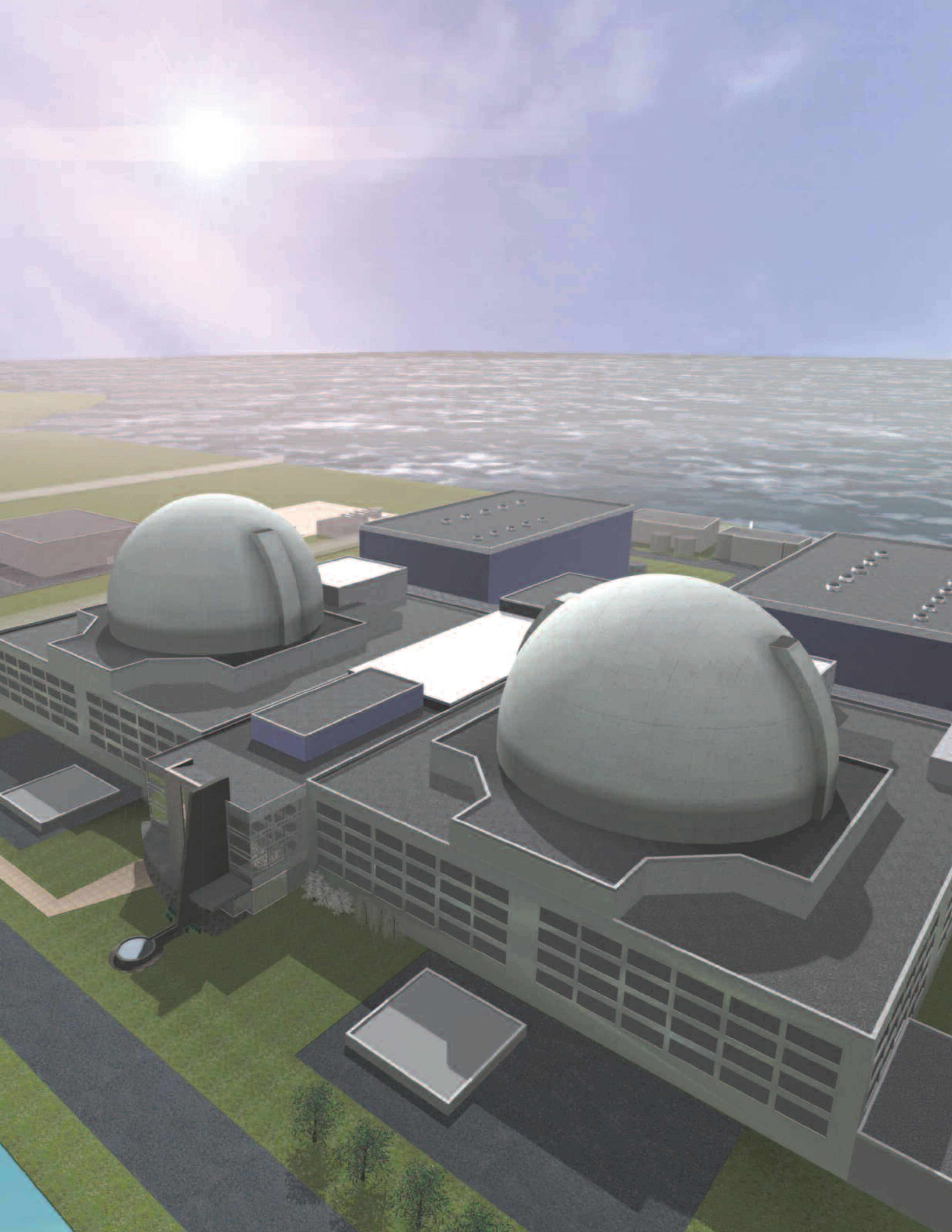
Proven CANDU Features

- Heavy water moderator and horizontal fuel channel design
- Series of parallel pressure tubes—rather than a single pressure vessel—allowing simpler manufacturing and reduced cost
- Two independent, passive, fast-acting safety shutdown systems and a unique inherent emergency cooling capability
- On-power fuelling for flexible outage planning and minimal 'excess' reactivity burden
- Multiple heat removal systems to prevent and mitigate severe accidents

ACR-1000 Innovations

- Extended fuel life through use of low enriched uranium fuel
- Reduced heavy water inventory by approximately 60% of traditional CANDU reactors, by use of light water coolant and reduced lattice
- Compact, highly stable reactor core design
- Reduced spent fuel volume
- Improved thermal efficiency through optimized, higher-pressure steam turbines
- Modular, prefabricated structures and systems
- Advanced construction techniques
- Quadrant-based safety and heat sink system layout design for improved on-power maintenance and testing, additional redundancy in actuating signals for trip channels, reduced risk of spurious trips and overall increased reliability
- Enhanced safety design including addition of reserve water system for passive accident mitigation
- Improved power manoeuvrability with lower inherent xenon load after shutdown than traditional CANDU
- Improved design for maintainability and operability
- Design validated by exhaustive proof-testing
- Comprehensive Risk Management Program

The ACR-1000 meets customer expectations for safe, reliable and economically competitive power production. It is the preferred choice… based on a wealth of experience, technical excellence and innovations in engineering.



Company Profile

AECL is an integrated nuclear technology, products and services company. Our 4,000 employees are dedicated to delivering leading edge nuclear services, R&D support, design and engineering, construction management, specialized technology, waste management and decommissioning in support of CANDU® reactor products and nuclear products from other vendors, worldwide.

AECL delivers power through partnership.

AECL

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Appendix D: AP1000 Technical Summary

An overview and technical summary of the AP1000 reactor design is provided in the following pages.

Ready to Meet Tomorrow's Power Generation Requirements Today

AP1000™

Simple

Safe

Innovative

Westinghouse

You can be sure ...
if it's Westinghouse

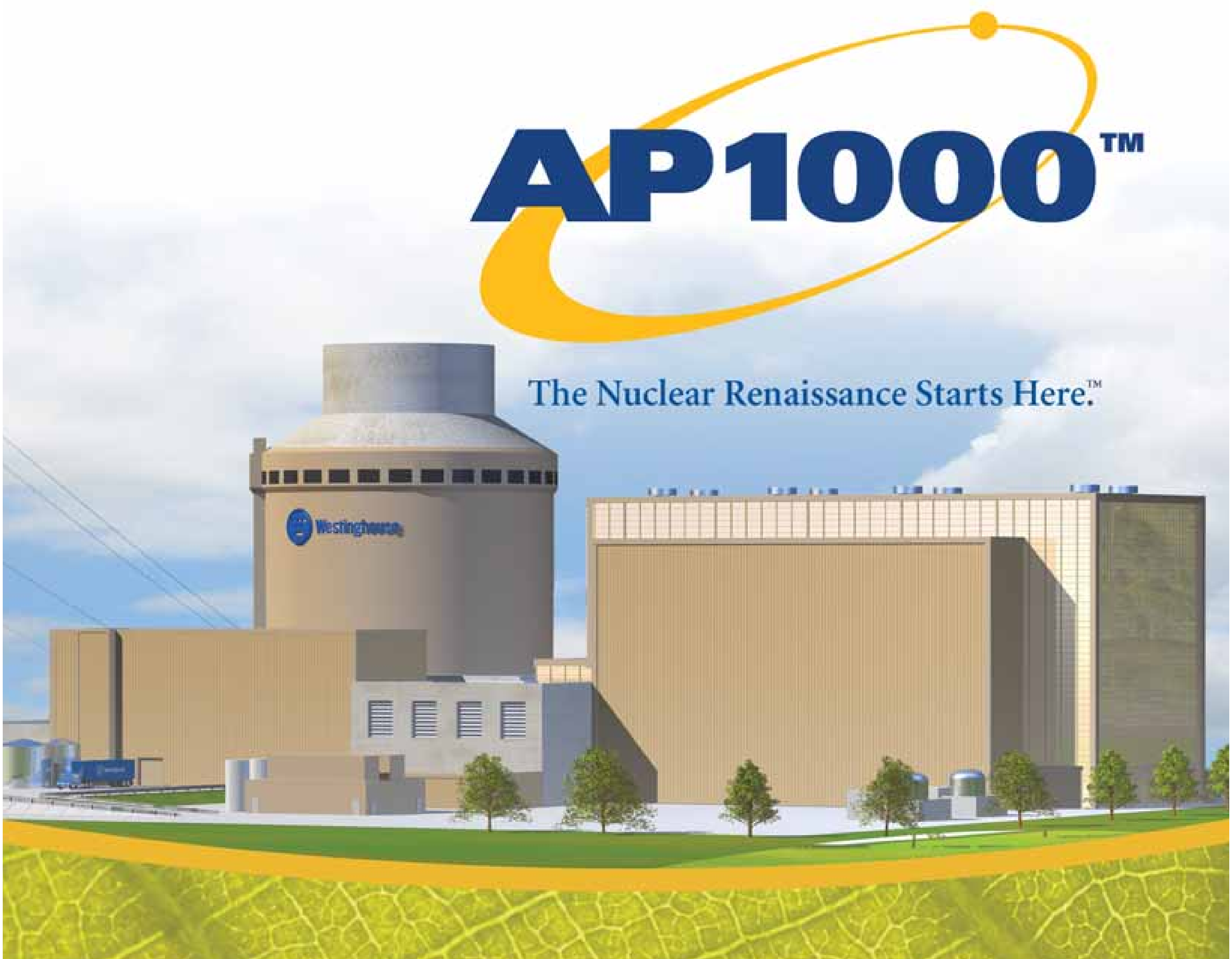
Nuclear Power
- The Environmentally Clean Option

As the world's population rises, so does its reliance on electricity. Likewise, energy demands are soaring as new technologies and expanded development create additional energy needs. This trend will only continue as nations grow and developing countries emerge.

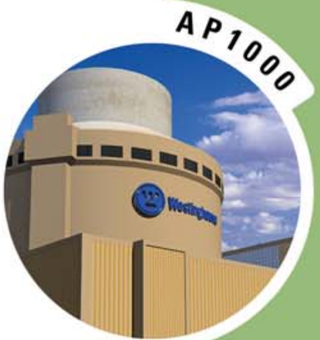
For the most part, fossil fuels have powered whole nations and economies. But as fossil fuels dwindle and as the effects of pollution and global warming increase, it's time to look for better solutions to the world's energy needs.

Continued reliance on fossil fuels for the vast majority of our energy needs is simply not realistic. Viewing the situation in a worldwide context magnifies the problem. With an additional two billion people expected to need energy by 2020, fossil fuels cannot adequately satisfy the demand without further harming the environment. Likewise, renewable energy sources are still in their infancy, as well as being an unrealistic means to provide baseload generation.

It's time to realize a generation of power that is safe, plentiful, economical and clean. It's time for a new generation of nuclear power.



Westinghouse AP1000



Featuring proven technology and innovative passive safety systems, the Westinghouse AP1000™ pressurized water reactor can achieve competitive generation costs in the current electricity market without emitting greenhouse gases and further harming the environment.

Westinghouse Electric Company, the pioneer in nuclear energy, once again sets a new industry standard with the AP1000. The AP1000 is the safest and most economical nuclear power plant available in the worldwide commercial marketplace, and is the only Generation III+ reactor to receive Design Certification from the U.S. Nuclear Regulatory Commission (NRC).

The established design of the AP1000 offers three distinct advantages over other designs:

- Unequaled safety
- Economic competitiveness
- Improved and more efficient operations

Based on nearly 20 years of research and development, the AP1000 builds and improves upon the established technology of major components used in current Westinghouse-designed plants. Components such as steam generators, digital instrumentation and controls, fuel, pressurizers, and reactor vessels are currently in use around the world and have years of proven, reliable operating experience.

Historically, Westinghouse plant designs and technology have forged the cutting edge of nuclear plant technology around the world. Today, nearly 50 percent of the world's 440 nuclear plants are based on Westinghouse technology. Westinghouse continues to be the nuclear industry's global leader.

(Generation III+ is the Department of Energy's nomenclature for Generation III Advanced Light Water Reactors with improved economics and safety.)

Westinghouse

You can be sure...
if it's Westinghouse

AP1000™ at a Glance

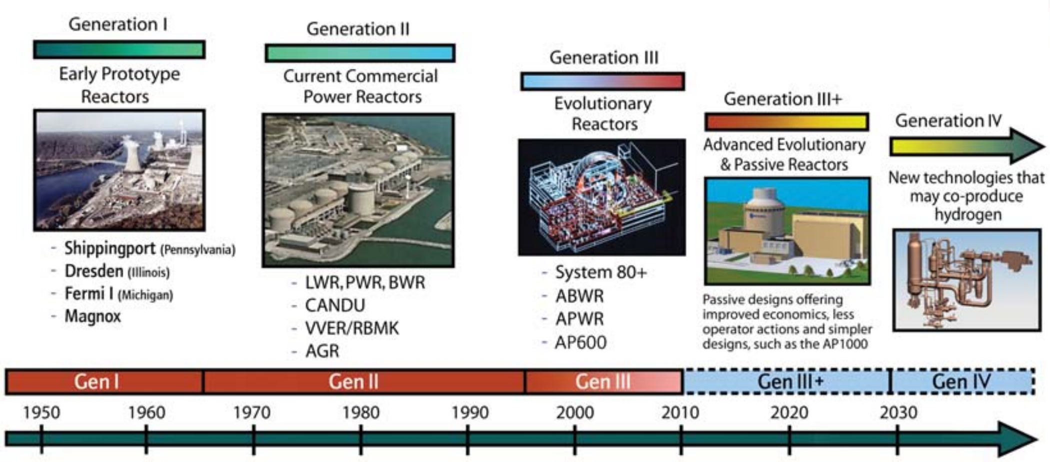
The AP1000™ is a two-loop pressurized water reactor (PWR) that uses a simplified, innovative and effective approach to safety. With a gross power rating of 3415 megawatt thermal (MWt) and a nominal net electrical output of 1117 megawatt electric (MWe), the AP1000, with a 157-fuel-assembly core, is ideal for new baseload generation. The standardized reactor design complies with the Advanced Light Water Reactor Utility Requirements Document (URD). Additionally, the AP1000 received Final Design Approval from the U.S. NRC in September 2004, and Design Certification in December 2005. The AP1000 is the first and only Generation III+ reactor to receive such certification from the NRC.

Simplified Plant Design

Simplification was a major design objective of the AP1000. Simplifications in overall safety systems, normal operating systems, the control room, construction techniques, and instrumentation and control systems provide a plant that is easier and less expensive to build, operate, and maintain. Plant simplifications yield fewer components, cable, and seismic building volume, all of which contribute to considerable savings in capital investment, and lower operation and maintenance costs. At the same time, the safety margins for AP1000 have been increased dramatically over currently operating plants.

The Technology

The AP1000 is comprised of components that incorporate many design improvements distilled from 50 years of successful operating nuclear power plant experience. The reactor vessel and internals, steam generator, fuel, and pressurizer designs are improved versions of those found in currently operating Westinghouse-designed PWRs. The reactor coolant pumps are canned motor pumps, the type used in many other industrial applications where reliability and long life are paramount requirements.



Licensed Passive Safety Systems

The unique feature of the AP1000 is its use of natural forces - natural circulation, gravity, convection and compressed gas - to operate in the highly unlikely event of an accident, rather than relying on operator actions and ac power. Even with no operator action and a complete loss of all on-site and off-site ac power, the AP1000 will safely shut down and remain cool.

Because natural forces are well understood and have worked as intended in large-scale testing, no demonstration plant is required. The Westinghouse advanced passive reactor design underwent the most thorough pre-construction licensing review ever conducted by the U.S. NRC.

Large Safety Margins

The AP1000 meets the U.S. NRC deterministic-safety and probabilistic-risk criteria with large margins. The safety analysis is documented in the AP1000 Design Control Document (DCD) and Probabilistic Risk Assessment (PRA). Results of the PRA show a very low core damage frequency (CDF) that is 1/100 of the CDF of currently operating plants and 1/20 of the CDF deemed acceptable in the URD for new, advanced reactor designs. It follows that the AP1000 also improves upon the probability of large release goals for advanced reactor designs in the event of a severe accident scenario to retain the molten core within the reactor vessel.

Ready for Implementation

Having received Design Certification, the AP1000 has the highest degree of design completion of any Generation III+ plant design. Demonstrating confidence in the AP1000 plant design and its readiness for implementation, several U.S. utilities have selected the AP1000 design in their applications to the U.S. NRC for combined construction and operating licenses (COL). Additionally China is building four AP1000s with the first unit scheduled to be online by 2013.

Plant Simplifications Yield Fewer Components, Cable and Building Volume

Component	1000 MWe Reference Plant	AP1000
Pumps	280	180
Safety class valves	2,800	1,400
Safety class piping, ft	110,000	19,000
Cable, million ft	9.1	1.2
Seismic building volume, million ft3	12.7	5.6

Operational Technology Incorporated into the AP1000

Component	Prior Use of Technology
Reactor vessel and internals	Doel 4, Tihange 3
CRDMs	Westinghouse plants worldwide
Fuel	South Texas 1&2, Doel 4, Tihange 3
Large Model F steam generators	ANO-2, San Onofre, Waterford, Palo Verde
Canned motor reactor coolant pumps	Fossil boilers and other industrial applications (inverted canned motor pumps)
Pressurizer	70 Westinghouse plants worldwide

Unequaled Safety

The AP1000™ pressurized water reactor is based on a simple concept: in the event of a design-basis accident, such as a main coolant pipe break, the plant is designed to achieve and maintain safe shutdown condition without operator action, and without the need for ac power or pumps. Rather than relying on active components, such as diesel generators and pumps, the AP1000 relies on natural forces - gravity, natural circulation, and compressed gases - to keep the core and the containment from overheating.

The AP1000 provides multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core-damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up. Defense-in-depth is integral to the AP1000 design, with a multitude of individual plant features including the selection of appropriate materials; quality assurance during design and construction; well-trained operators; and an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits. In addition to these protections, the following features contribute to defense-in-depth of the AP1000:

- *Non-safety Systems.* The non safety-related systems respond to the day-to-day plant transients, or fluctuations in plant conditions. For events that could lead to overheating of the core, these highly reliable non-safety systems actuate automatically to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems.

- *Passive Safety-Related Systems.* The AP1000 safety-related passive systems and equipment are sufficient to automatically establish and maintain core cooling and containment integrity indefinitely following design-basis events, assuming the most limiting single failure, with no operator action, and no on-site or off-site ac power sources. An additional level of defense is provided through diverse mitigation functions that are included within the passive safety-related systems.

- *In-vessel Retention of Core Damage.* The AP1000 is designed to drain the high capacity in-containment refueling water storage tank (IRWST) water into the reactor cavity in the event that the core has overheated. This provides cooling on the outside of the reactor vessel preventing reactor vessel failure and subsequent spilling of molten core debris into the containment. Retention of debris in the vessel significantly reduces uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena such as the interaction of molten core material with concrete.

- *Fission Product Release.* Fuel cladding provides the first barrier to the release of radiation in the highly unlikely event of an accident. The reactor coolant pressure boundary, in particular the reactor pressure vessel and the reactor coolant piping, provide independent barriers to prevent the release of radiation. Furthermore, in conjunction with the surrounding shield building, the steel containment vessel provides additional protection by establishing a third barrier and by providing natural convection air currents to cool the steel containment. The natural convection cooling can be enhanced with evaporative cooling by allowing water to drain from a large tank located at the top of the shield building on to the steel containment.

AP1000 exceeds safety goals

- Safety performance optimized and verified by probabilistic risk assessment (PRA)
- Calculated core damage frequency that exceed NRC and Utility Requirements Document (URD) goals

	NRC Req	Current Plants	URD Req	AP1000
CDF	1×10^{-4}	5×10^{-5}	1×10^{-5}	5×10^{-7}

CDF – core damage frequency, events per year

Non Safety-related Active Systems for Defense-in-Depth

Many of the active safety-related systems in existing and evolutionary PWR designs are retained in the AP1000 but are designated as non safety-related.

The AP1000 active non safety-related systems support normal operation and are also the first line of defense in the event of transients or plant upsets. Although these systems are not credited in the safety analysis evaluation, they provide additional defense-in-depth by adding a layer of redundancy and diversity. In addition to contributing to the very low CDF, the non safety-related, active systems require fewer in-service inspections, less testing and maintenance, and are not included in the simplified technical specifications. For defense-in-depth, most planned maintenance for these non-safety systems can be performed while the plant is operating.

Examples of non safety-related systems that provide defense-in-depth capabilities for the AP1000 design include the chemical and volume control system, normal residual heat removal system, and the startup (auxillary) feedwater system. These systems utilize non-safety support systems such as the standby diesel generators, the component cooling water system, and the service water system. The AP1000 also includes other active non safety-related systems, such as the heating, ventilation and air-conditioning (HVAC) systems, which remove heat from the instrumentation and control (I&C) cabinet rooms and the main control room. These are, in simpler form in the AP1000, familiar systems that are used in current PWRs as safety systems. In the AP1000, these HVAC systems are simplified non-safety first line of defense, which are backed up by the ultimate defense, the passive safety-grade systems.

This defense-in-depth class of systems includes the containment hydrogen control system, which consists of the hydrogen monitoring system, passive autocatalytic hydrogen recombiners, and hydrogen igniters (powered by batteries).

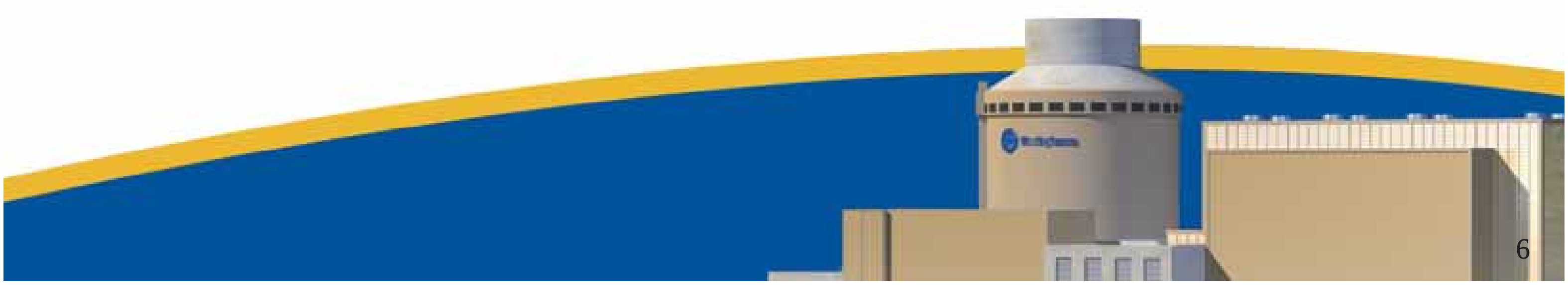
Probabilistic Risk Assessment (PRA)

From a letter dated July 20, 2004, from the Chairman of the Advisory Committee on Reactor Safeguards to the Chairman of the U.S. NRC on its Reactor Safeguards Report about the safety aspects of the Westinghouse Electric Company Application for Certification of the AP1000 Passive Plant Design:

"The AP1000 Design Certification application included a PRA in accordance with regulatory requirements. This PRA was done well and rigorous methods were used. We found that this PRA was acceptable for certification purposes. The mean estimates of the risk metrics are:

CDF (Core Damage Frequency)	$5 X 10^{-7}$ per year
LRF (Large Release Frequency	$6 X 10^{-8}$ per year

“These risk metrics are well within the agency's expectations for advanced plants. The fact that the PRA was an integral part of the design process was significant to achieving this estimated low risk."



Passive Safety Systems

A major safety advantage of passive plants versus current or evolutionary light water reactors(LWRs) is that long-term accident mitigation is maintained without operator action or reliance on off-site or on-site ac power.

The AP1000 uses extensively analyzed and tested passive safety systems to improve the safety of the plant. The Advisory Council on Reactor Safeguards (ACRS) and the U.S. NRC have scrutinized these systems and ruled that they meet the U.S. NRC single-failure criteria, and other safety criteria such as Three Mile Island lessons learned, and generic safety issues.

The AP1000 passive safety systems require no operator actions to mitigate design-basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas to achieve their safety function. No pumps, fans, diesels, chillers, or other active machinery are used, except for a few simple valves that automatically align and actuate the passive safety systems. To provide high reliability, these valves are designed to move to their safeguard positions upon loss of power or upon receipt of a safeguards actuation signal. Only a single move is required for each valve, which are powered by multiple, reliable Class 1E dc power batteries. The passive safety systems do not require the large network of active safety support systems (ac power, diesels, HVAC, pumped cooling water) that are needed in typical nuclear plants. As a result, in the case of the AP1000, those active support systems no longer must be safety class, and they are either simplified or eliminated. With less safety-grade equipment, the seismic Category 1 building volumes needed to house safety-grade equipment are greatly reduced. In fact, most of the safety equipment can now be located within containment, resulting in fewer containment penetrations.

The AP1000 passive safety systems include:

- Passive core cooling system (PXS)
- Containment isolation
- Passive containment cooling system (PCS)
- Main control room emergency habitability system

Passive Core Cooling System

The AP1000 passive core cooling system (PXS) performs two major functions:

- Safety injection and reactor coolant makeup from the following sources:
  - Core makeup tanks (CMTs)
  - Accumulators
  - In-containment refueling water storage tank (IRWST)
  - In-containment passive long-term recirculation

- Passive residual heat removal (PRHR), utilizing:
  - Passive residual heat removal heat exchanger (PRHR HX)
  - IRWST

Safety injection sources are connected directly to two nozzles dedicated for this purpose on the reactor vessel. These connections, which have been used before on two-loop plants, reduce the possibility of spilling part of the injection flow in a large break loss-of-coolant accident.

High Pressure Safety Injection with CMTs

Core makeup tanks (CMTs) are called upon following transients where the normal makeup system is inadequate or is unavailable. Two core makeup tanks (CMTs) filled with borated water in two parallel trains are designed to function at any reactor coolant system (RCS) pressure using only gravity, and the temperature and height differences from the reactor coolant system cold leg as the motivating forces. These tanks are designed for full RCS pressure and are located above the RCS loop piping. If the water level or pressure in the pressurizer reaches a set low level, the reactor, as well as the reactor coolant pumps, are tripped and the CMT discharge isolation valves open automatically. The water from the CMTs recirculates then flows by gravity through the reactor vessel.

Medium Pressure Safety Injection with Accumulators

As with current pressurized water reactors, accumulators are required for large loss-of-coolant accidents (LOCAs) to meet the immediate need for higher initial makeup flows to refill the reactor vessel lower plenum and downcomer following RCS blowdown. The accumulators are pressurized to 700 psig with nitrogen gas. The pressure differential between the pressurized accumulators and the dropping RCS pressure ultimately forces open check valves that normally isolate the accumulators from the RCS. Two accumulators in two parallel trains are sized to respond to the complete severance of the largest RCS pipe by rapidly refilling the vessel downcomer and lower plenum. The accumulators continue delivery to supplement the CMTs in maintaining water coverage of the core.

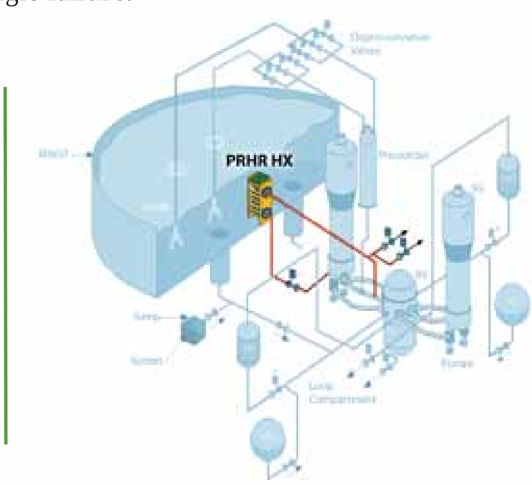
Low Pressure Reactor Coolant Makeup from the IRWST

Long-term injection water is supplied by gravity from the large IRWST, which is located inside the containment at a height above the RCS loops. This tank is at atmospheric pressure and, as a result, the RCS must be depressurized before injection can occur. The AP1000 automatically controls depressurization of the RCS to reduce its pressure to near atmospheric pressure, at which point the gravity head in the IRWST is sufficient to overcome the small RCS pressure and the pressure loss in the injection lines to provide IRWST water to the reactor.

Passive Residual Heat Removal

The AP1000 has a passive residual heat removal subsystem that protects the plant against transients that upset the normal heat removal from the primary system by the steam generator feedwater and steam systems. The passive RHR subsystem satisfies the U.S. NRC safety criteria for loss of feedwater, feedwater-line breaks, and steam-line breaks with a single failure.

The system includes the passive RHR heat exchanger consisting of a 100-percent capacity bank of tubes located within the IRWST. This heat exchanger is connected to the reactor coolant system in a natural circulation loop. The loop is isolated from the RCS by valves that are normally closed, but will open if power is lost or upon other signals from the instrumentation and control protection system. The difference in temperature and the elevation difference between the hot inlet water and the cold outlet water of the heat exchanger drives the natural circulation loop. If the reactor coolant pumps are running, the passive RHR heat exchange flow will be increased.



The IRWST is the heat sink for the passive RHR heat exchanger. The IRWST water volume is sufficient to absorb decay heat for about two hours before the water starts to boil. After that, the steam from the boiling IRWST condenses on the steel containment vessel walls and then drains back into the IRWST by specially designed gutters.

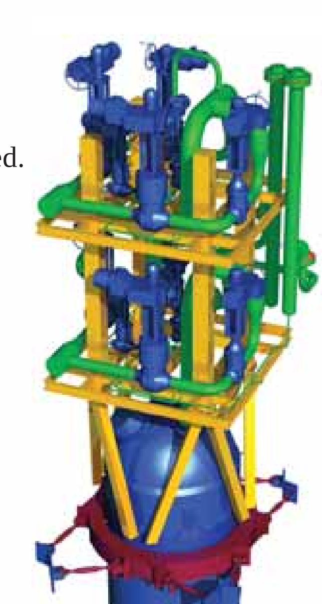
Automatic Depressurization System

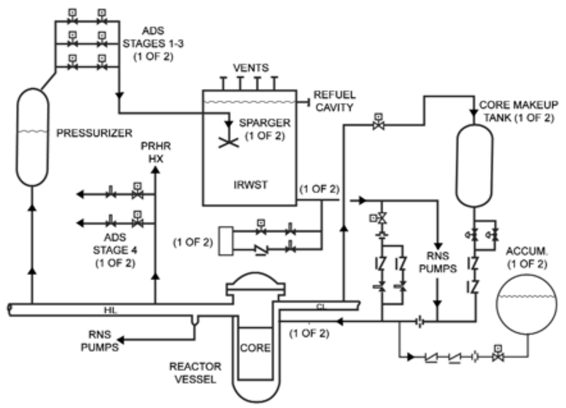
The automatic depressurization system (ADS) depressurizes the reactor coolant system (RCS) and enables lower pressure safety injection water to enter the reactor vessel and the core. It is activated by a level set-point in the core makeup tank (CMT). The ADS is comprised of three stages of motor-operated valves (MOVs) located above the pressurizer, and a fourth stage connected to the RCS hot legs and controlled by a squib valve, which opens by the actuation of an explosive charge. The first three stages of MOVs are arranged in six parallel sets (two normally closed valves in series). These MOV valves are activated on two-out-of-four actuation signals. The fourth stage of this system consists of four large valves, in two pairs, that open off the hot legs, reducing the pressure to atmospheric, allowing gravity injection from the IRWST. This eventually evolves into a long-term cooling mode with containment sump recirculation. The ADS valves are arranged to open in a prescribed sequence determined by the core makeup tank (CMT) level and a sequence timer.

The automatic RCS depressurization feature meets the following criteria:

- The reliability (redundancy and diversity) of the ADS valves and controls satisfies the single-failure criterion as well as the failure tolerance called for by the low core-damage frequency goals.

- The design provides for both real demands (i.e., RCS leaks and failure of the CVS makeup pumps) and spurious instrumentation signals. The probability of significant flooding of the containment due to the use of the ADS is less than once in 600 years. The design is such that for small-break loss-of-coolant accident(LOCA) up to 8 inches (20.32 cm) in diameter, the core remains covered.





Increased Safety Margins

Design Basis Accident	Typical Plant	AP1000
Loss Flow Margin to DNBR Limit	~ 10-14%	~ 16 %
Feedline Break, Subcooled Margin	> 0°F	~ 140°F
Steam generator tube rupture	Operator actions required in 10 minutes	Operator actions not required
Small loss-of-coolant accident (LOCA)	3" LOCA, core uncovers, PCT <1500°F	<8" LOCA, No core uncovery
Large LOCA peak clad temperature with uncertainty	1700-2000°F *	<1600°F *
Anticipated transient without reactor trip, pressure (% core life)	3200 psig (90%)	2800 psig (100%)

* Based on ASTRUM analysis. AP1000 was licensed with a very conservative "bounding" best-estimate large break LOCA analysis

Containment isolation

Containment isolation is provided to prevent or limit the escape of fission products that may result from postulated accidents. In the event of an accident, the containment isolation provisions are designed so that fluid lines penetrating the containment boundary are isolated. The containment isolation system consists of the piping, valves and actuators that isolate the containment.

Containment isolation is improved in the AP1000 because:

- The number of normally open penetrations is reduced by 50 percent, thanks to the simpler passive safety systems
- Penetrations that are normally open and at risk are fail safe - they fail in the closed position
- There is no recirculation of irradiated water outside of containment for design-basis accidents
- The steel containment is a high integrity (steel) pressure vessel, rather than a concrete vessel

The function of the AP1000 passive containment cooling system (PCS) is to prevent the containment vessel from overheating and exceeding the design pressure, which could result in a breach of the containment and the loss of the final barrier to radioactive release.

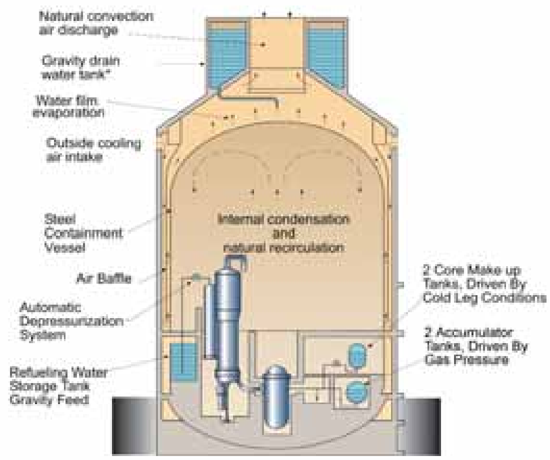
Passive Containment Cooling System (PCS)

The PCS consists of the following components:

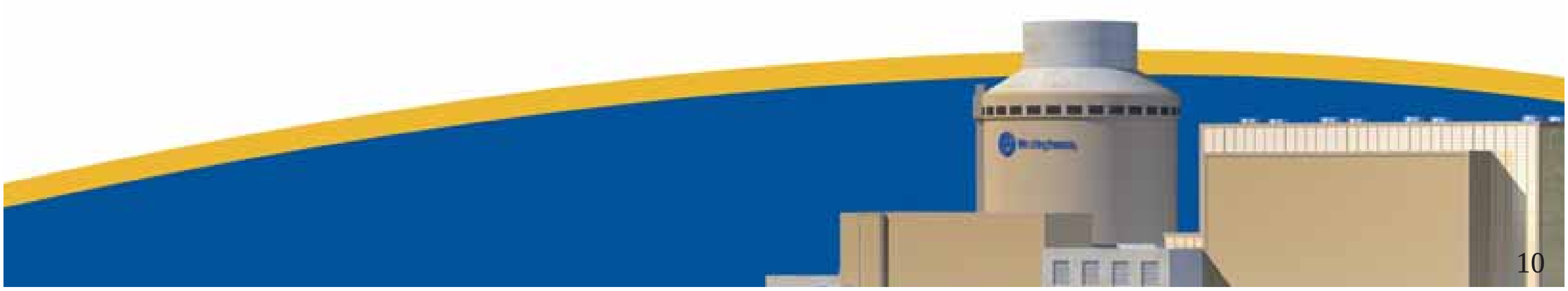
- Air inlet and exhaust paths that are incorporated in the shield building structure
- An air baffle that is located between the steel containment vessel and the concrete shield building
- A passive containment cooling water storage tank that is incorporated in the shield building structure above the containment
- A water distribution system
- An ancillary water storage tank and two recirculation pumps for onsite storage of additional PCS cooling water, heating to avoid freezing, and for maintaining proper water chemistry

Natural Circulation

The PCS is able to effectively cool the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced. The steel containment vessel itself provides the heat transfer surface that allows heat to be removed from inside the containment and rejected to the atmosphere. Heat is removed from the containment vessel by a natural circulation flow of air through the annulus formed by the outer shield building and the steel containment vessel it houses. Outside air is pulled in through openings near the top of the shield building and pulled down, around the baffle and then flows upward out of the shield building.



The flow of air is driven by the chimney effect of air heated by the containment vessel rising and finally exhausting up through the central opening in the shield building roof.

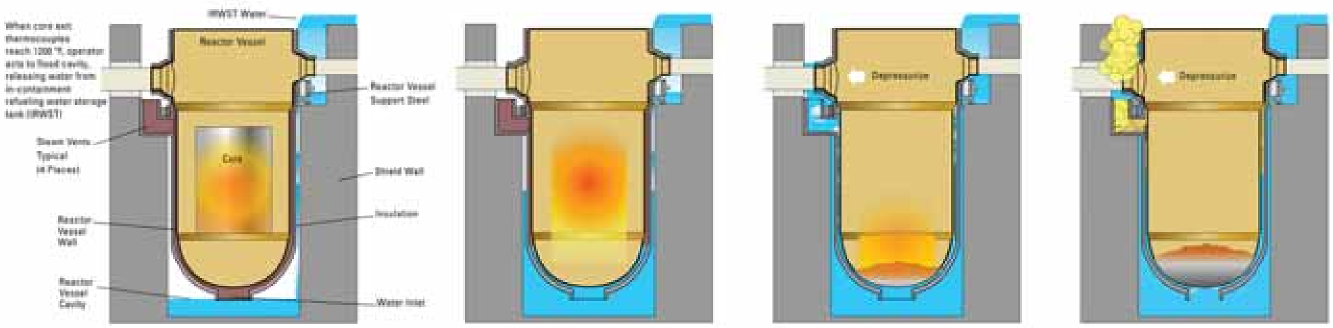


Water Evaporation

If needed, the air cooling can be supplemented by water evaporation on the outside of the containment shell. The water is drained by gravity from a tank located on top of the containment shield building. Three normally closed, fail-open valves will open automatically to initiate the water flow if a high containment pressure threshold is reached. The water flows from the top, outside, domed surface of the steel containment shell and down the side walls allowing heat to be transferred and removed from the containment by evaporation. The water tank has sufficient capacity for three days of operation, after which time the tank could be refilled, most likely from the ancillary water storage tank. If the water is not replenished after three days, the containment pressure will increase, but the peak pressure is calculated to reach only 90 percent of design pressure. After three days, air cooling alone is sufficient to remove decay heat.

In-Vessel Retention of Core Damage

The AP1000 is designed to mitigate a postulated severe accident such as core melt. In this event, the AP1000 operator can act to flood the reactor cavity- the space immediately surrounding the reactor vessel - with water from the in-containment refueling water storage tank (IRWST), submerging the lower portion of the reactor vessel. An insulating structure that surrounds the reactor vessel provides the pathway for water cooling to reach the vessel; flow around the bottom vessel head and up the vessel-insulation wall annulus; and to vent resulting steam from cooling the vessel from the reactor cavity. The cooling is sufficient to prevent molten core debris in the lower head from melting the steel vessel wall and spilling into the containment. Retaining the debris in the reactor vessel protects the containment integrity by simply avoiding the uncertainties associated with ex-vessel severe accident phenomena, such as ex-vessel steam explosion and core-concrete interaction with the molten core material.



Main Control Room Emergency Habitability System

The main control room can be isolated in case of high airborne radiation levels. The main control room (MCR) emergency habitability system is comprised of a set of compressed air tanks connected to a main and an alternate air delivery line. Components common to both lines include a manual isolation valve, a pressure-regulating valve, and a flow metering orifice. This system is designed to provide the ventilation and pressurization needed to maintain a habitable environment for up to 11 people in the MCR for 72 hours following any design-basis accident.

Economic Competitiveness

Construction costs of commercial nuclear generating plants must be reduced in order to expand the future use of nuclear energy. Two of the drivers of plant construction costs are the cost of financing during the construction phase and the substantial amount of skilled-craft-labor hours needed on site during construction. The AP1000™ pressurized water reactor's extensive use of modularization of plant construction mitigates both of these drivers.

Overnight Construction Cost

From the outset, the AP1000 was designed to reduce capital costs and to be economically competitive with contemporary fossil-fueled plants. This requires lower overnight construction costs and higher confidence in the construction schedule.

The AP1000 reduces the amount of safety-grade equipment required by using passive safety systems. Consequently, less Seismic Category I building volume is required to house the safety equipment (approximately 45 percent less than a typical reactor). The AP1000's modular construction design further reduces the construction schedule and the construction risks, with work shifted to factories with their better quality and cost control as well as labor costs that are less than those at the construction site. This also allows more work to be done in parallel. The use of heavy lift cranes enables an "open top" construction approach, which is effective in reducing construction time.

With new computer modeling capabilities, Westinghouse is able to optimize and choreograph the construction plan of an AP1000 in advance by simulation. The result is a very high confidence in the construction schedule.

AP1000 simplified plant arrangement:

- Much smaller seismic building volumes than current or evolutionary plants cut construction costs
- Major construction improvement by using large factory-made structural modules assembled and lifted into place with heavy duty cranes at site
- Large laydown areas in containment for ease of work and shortened outages

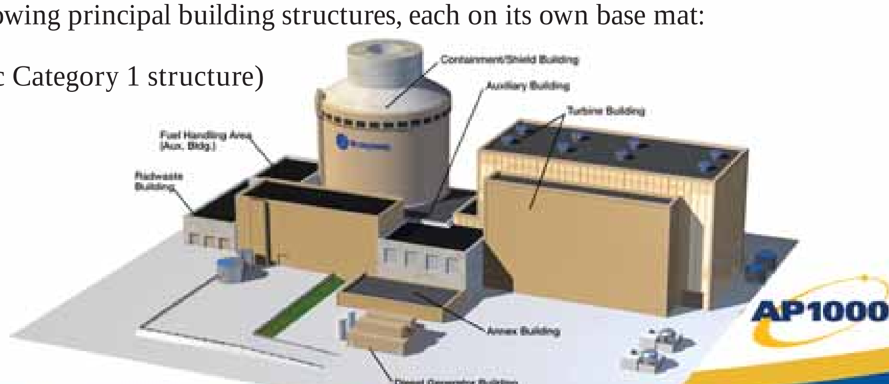
Simplified Plant Arrangement

The AP1000 has a smaller footprint than an existing nuclear power plant with the same generating capacity. The plant arrangement provides separation between safety-related and non safety-related systems to preclude adverse interaction between safety-related and non safety-related equipment.

Separation between redundant, safety-related equipment trains and systems provides confidence that the safety design functions of the AP1000 can be performed. In general, this separation is achieved by partitioning an area with concrete walls.

The AP1000 plant is arranged with the following principal building structures, each on its own base mat:

- Nuclear Island (the only Seismic Category 1 structure)
- Turbine Building
- Annex Building
- Diesel Generator Building
- Radwaste Building



Nuclear Island

The nuclear island is designed to meet Seismic Category I structural requirements. The volume of this building is much smaller than the buildings in previous nuclear power plant designs. This provides a large capital cost savings as seismic structures cost roughly three times as much as non-seismic structures. The nuclear island consists of the steel containment vessel, the concrete shield building and the auxiliary building. The nuclear island is designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

- The containment vessel is a high integrity, freestanding steel structure with a wall thickness of 1.75 inches (4.44 cm). The containment is 130 feet (39.6 m) in diameter. The ring sections and vessel heads are constructed of steel plates pre-formed in an off-site fabrication facility and shipped to the site for assembly and installation using a large-capacity crane.

  The primary containment prevents the uncontrolled release of radioactivity to the environment. It has a design leakage rate of 0.10 weight percent per day of the containment air during a design-basis accident and the resulting containment isolation.

  The AP1000 containment contains a 16-foot (4.9m) diameter main equipment hatch and a personnel airlock at the operating-deck level, and a 16-foot (4.9m) diameter maintenance hatch and a personnel airlock at grade level. These large hatches significantly improve accessibility to the containment during outages and, consequently, reduce the potential for congestion at the containment entrances. These containment hatches, located at the two different levels, allow activities occurring above the operating deck to be unaffected by activities occurring below the operating deck.

  The containment arrangement provides significantly larger laydown areas than most conventional plants at both the operating deck level and the maintenance floor level. Ample laydown space is provided for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting.

- Concrete Shield Building -- The AP1000 containment design incorporates a shield building that surrounds the containment vessel and forms the natural convection annulus for containment cooling. This building is a cylindrical, reinforced concrete structure with a conical roof that supports the water storage tank and air diffuser (or chimney) of the PCS. It shares a common base mat with the primary containment and auxiliary building, and is designed as a seismic Category 1 structure. It has an inner diameter of about 140 feet (43m), a height of 73.25 ft (83.3 m), and a wall thickness of 3 ft (0.9 m) in the cylindrical section.

  The two primary functions of the shield building during normal operation are 1) to provide an additional radiological barrier for radioactive systems and components inside the containment vessel and 2) to protect the containment vessel from external events, such as tornados and tornado-driven objects that might impinge on it. As described earlier, under design-basis accident conditions, the shield building serves as a key component of the PCS by aiding in the natural convective cooling of the containment.

- Auxiliary Building -- The auxiliary building is designed to provide protection and separation for the Seismic Category 1 mechanical and electrical equipment located outside the containment building. The building also provides protection for safety-related equipment against the consequences of internal and external events. Specifically, the auxiliary building houses the main control room, Class 1E I&C systems, Class 1E electrical systems, fuel handling and spent fuel handling area, mechanical equipment areas, liquid and gas radwaste areas, containment penetration areas, and main steam and feedwater isolation valve compartments. Large staging and laydown areas are provided outside the two equipment hatches.

Non-seismic Buildings

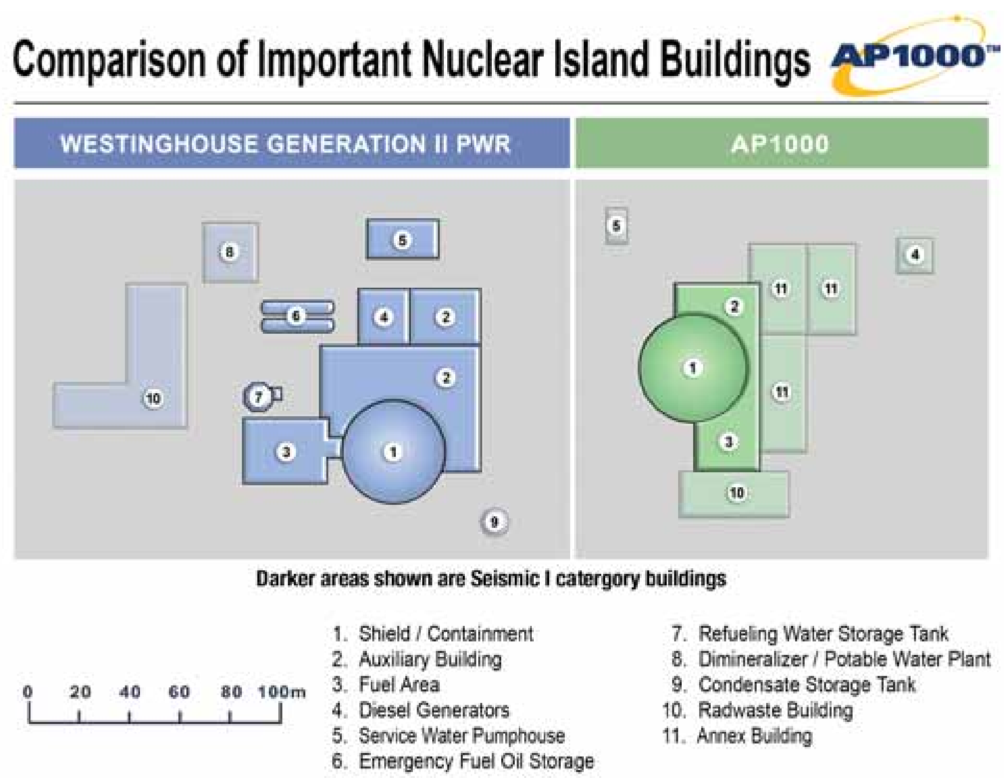
The following buildings are non-seismic Category 1 structures, and contain no safety-related equipment. They are designed for wind and seismic loads in accordance with the Uniform Building Code. The foundation of each building is a reinforced concrete mat on grade.

- The **annex building** serves as the main personnel entrance to the power generation complex. The building includes the health physics area, the non-Class 1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the technical support center, and various HVAC systems. The annex building provides large staging and laydown areas immediately outside the equipment hatches.

- The **turbine building** houses the main turbine, generator, and associated fluid and electrical systems. It also houses the makeup water purification system.

- The **diesel generator building** houses two diesel generators and their associated HVAC equipment.

- The **radwaste building** contains facilities for segregated storage of various categories of solid waste prior to processing, for processing by mobile systems, and for storing processed solid waste in shipping and disposal containers.



Modularization and Construction

Structural, piping and equipment modules provide:

- Shortened construction schedule
- Reduced field manpower
- Increased factory-based manufacturing and assembly of modules
- Improved quality - pre-testing and inspection of modules prior to shipment
- Reduced site congestion

Modular by Design

The AP1000 has been designed to make use of modern, modular construction techniques. The design incorporates vendor-designed skids and equipment packages, as well as large, multi-ton structural modules and special equipment modules. Modularization allows construction tasks that were traditionally performed in sequence to be completed in parallel. Factory-built modules can be installed at the site in a planned construction schedule of three years - from first concrete pour to fuel load. This duration has been verified by experienced construction managers through 4D (3D models plus time) reviews of the computer-simulated construction sequence.

Typical Breakdown of AP1000 Modules

	Structural Modules	Piping Modules	Mechanical Equipment Modules	Total
Containment	41	20	12	73
Auxiliary Building	42	34	29	105
Turbine Building	29	45	14	88
Annex Building	10			10
Total	122	99	55	276

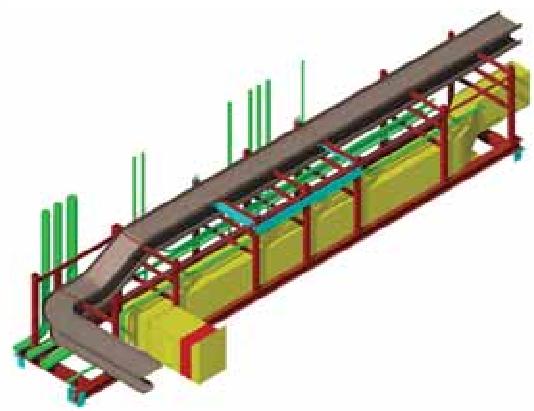
Parallel Work Processes in Controlled Environments

AP1000 modularization allows many more construction activities to proceed in parallel. This reduces the calendar time for plant construction, thereby reducing the cost of money and the exposure risks associated with plant financing. Furthermore, the reduced amount of work on site means the amount of skilled field-craft labor, which is more costly than shop labor, is greatly reduced. In addition to the labor cost savings, more of the welding and fabrication performed in a factory environment increases the quality of the work, improves the flexibility in scheduling, and reduces the amount of specialized tools on site.

To achieve proper interfaces with the rest of the plant systems and structures, interconnected piping between modules is represented in the 3D design model. This eliminates the interferance concerns of typical field-run commodities (e.g., piping, duct, raceway) and "stick-built" construction techniques.

Modularization Used to Reduce AP1000 Construction Cost

2 Weeks | 1 Month | 2 Months | 1 Year | 2 Years



The basic AP1000 module is a rail-shippable unit less than 12 feet high, 12 feet wide and 80 ft long, weighing less than 80 tons. Larger modules could be manufactured for shipment to a site accessible by barge.

AP1000 Construction Schedule

The standard AP1000 schedule is five years from order placement and three years from first concrete pour to fuel load.

Site Preparation	Site Construction	Start Up and Testing
18 Months	36 Months	6 Months

Improved and more efficient operations

Nuclear power remains a competitive part of our energy policies because of improved industry performance. Greater nuclear plant performance means more electricity for less money. The AP1000™ pressurized water reactor has several design features that improve worker safety and production, as well as availability and capacity factors.

Improved Plant Performance

- 18-month fuel cycle for improved availability and reduced overall fuel cost
- Significantly reduced maintenance, testing and inspection requirements and staffing
- Reduced radiation exposure, less plant waste
- 93 percent availability
- Sixty-year design lifetime

Operations & Maintenance

An important aspect of the AP1000 design philosophy focuses on plant operability and maintainability. The passive safety features use a much smaller number of valves than do the multiple trains of active pump-driven systems, and there are no safety pumps at all; so, there is less in-service testing to perform. In particular, simplified safety systems reduce surveillance requirements, significantly simplifying technical specifications and reducing the likelihood of forced shutdowns. Lower operating and maintenance requirements lead to smaller maintenance staffs.

The variable-speed canned-motor reactor coolant pumps (RCPs) simplify plant startup and shutdown operations because they are capable, for example, of reducing RCP speed during plant cooldown and providing the capability to vary RCP speed to better control shutdown operating-mode transitions. The RCPs operate at constant speed during power operations, simplifying control actions during load shifts.

The digital I&C design significantly reduces required I&C surveillance testing and simplifies trouble-shooting, repair and post-maintenance testing. The plant includes automation of some cooldown operations and improved steam-dump, low-pressure performance. The advanced control room design significantly improves the operator interfaces and plant operations capabilities.

Overall, the selection of proven components has been emphasized to ensure a high degree of reliability and reduced maintenance requirements. Component standardization reduces spare parts inventories, maintenance, training requirements, and allows shorter maintenance times. Built-in testing capability is provided for critical components.

Plant layout ensures adequate access for inspection and maintenance. Laydown space provides for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting, and computer data highway connections.

The AP1000 also incorporates radiation exposure reduction principles to keep worker dose as low as reasonably achievable (ALARA). Exposure length, distance, shielding, and source reduction are fundamental criteria that are incorporated into the design with the result of:

- Minimized operational releases
- Worker radiation exposure greatly reduced
- Total radwaste volumes minimized through features such as no boron load follow, ion exchange rather than evaporation, segregation of wastes at the source, minimization of active components, and packaging in high-integrity containers
- Other (non-radioactive) hazardous wastes minimized through such features as a simplified plant (e.g., elimination of many oil lubricated pumps), careful selection of processes (e.g., laboratory and turbine-side chemistry), and segregation of wastes

The AP1000 is designated for rated performance with up to 10 percent of the steam generator tubes plugged and with a maximum hot-leg temperature of 321.1°C (610°F). The plant is designed to accept a step-load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam-dump system actuation, provided that the rated power level is not exceeded. Further, the AP1000 is designed to accept a 100 percent load rejection from full power to house loads without a reactor trip or operation of the pressurizer or steam generator safety valves.

AP1000 - Operating and Maintenance (O&M) Costs

Operating nuclear plants in the U.S. are already competitive producers of electricity compared to coal-fired plants. That virtue is enhanced by fuel cost comprising only about 25 percent of the production costs of a nuclear plant. The remaining 75 percent of production cost is the fixed cost of operation and maintenance. That means that nuclear power production is less sensitive to changes in fuel cost than coal-fired plants where fuel costs can be more than 75 percent of the production cost. AP1000's modern design will engender even less expensive production by requiring less manpower for O&M than current plants for many reasons, including:

1) Less equipment and less safety-grade equipment to maintain and test
2) Improved equipment, such as the primary system canned motor pumps that are maintenance-free and do not need the complex seal injection systems of typical shaft seal coolant pumps
3) Features for faster head removal for refueling
4) Less waste produced
5) Improved protection from and fewer opportunities for radiation exposure (ALARA design)
6) Online-diagnosing electronics
7) A main control room featuring the latest human interface design, needing only an operator and supervisor for normal operation

An independent study by the Institute of Nuclear Power Operators (INPO) determined that a passive "single, mature Advanced Light Water Reactor" would require about one-third less O&M staff than a currently operating nuclear plant.

Availability

The AP1000 power generating system is familiar Westinghouse PWR technology updated from the substantial amount of operating experience accumulated over many decades to enhance plant reliability and operability. The AP1000 steam generators use long-life tube materials and a component design in a size that has recently been used for replacement steam generators. Canned motor pumps have significantly improved operational reliability in comparison with conventional shaft-seal pumps, and have now attained an experience base in sizes useful for application, again, in PWRs.

The advanced, digital instrumentation and control (I&C) systems feature an integrated control system that avoids reactor trips due to single-channel failure, and provides online diagnostics capabilities. In addition, the plant design provides large margins for plant operation before reaching the safety limits. This assures a stable and reliable plant operation with a goal of reducing the number of unplanned reactor trips to less than one per year. The AP1000 design incorporates design features that are essential to minimizing reactor trips. The design includes optimization of a number of plant variables that provide inputs to the reactor trip signals; increased margin between the normal operating range and the trip setpoint of safety variables; and a number of design features specifically incorporated to minimize unplanned automatic trips. In addition, a Design Reliability Assurance Program helps to focus on the structures, systems and components critical to reactor trip, and to identify new design features and maintenance methods to achieve the plant availability and reliability goals.

Based on the foregoing points, considering the short, refueling outage capability (17 days), and plans to use an 18- or 16-20-month alternating cycle for optimum economics, the AP1000 is expected to exceed the 93 percent availability goal.

The plant has a design life of 60 years based on the service life of the reactor vessel.

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Appendix E: CANDU 6E Technical Summary

An overview and technical summary of the CANDU 6E reactor design is provided in the following pages.

EC6™ Technical Summary

AECL

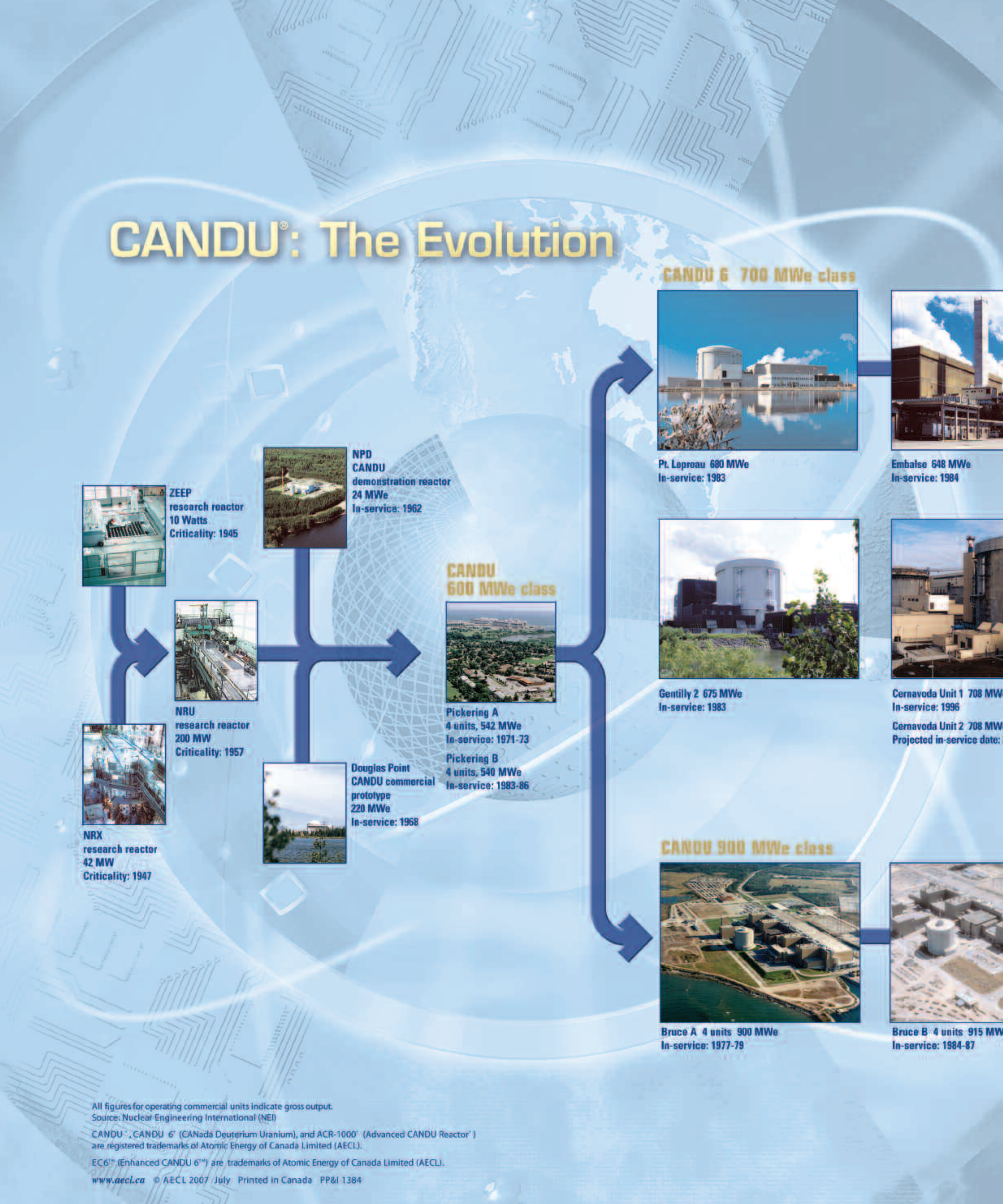


Figure S-1 CANDU: The Evolution

Wolsong Unit 1 679 MWe
In-service: 1983

Wolsong Unit 2 715 MWe
In-service: 1997

Wolsong Unit 3 715 MWe
In-service: 1998

Wolsong Unit 4 715 MWe
In-service: 1999

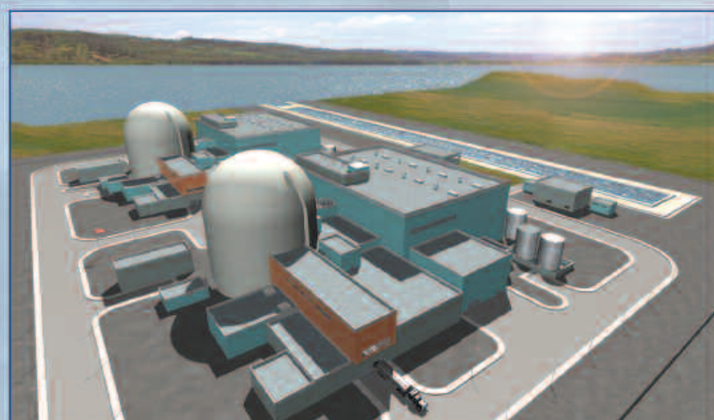
Qinshan Phase III
Unit 1 728 MWe
In-service: 2002

Qinshan Phase III
Unit 2 728 MWe
In-service: 2003

Ve

Ve
: 2007

Enhanced CANDU 6 GEN III 740 MWe class



Artist's impression of a 2-unit Enhanced CANDU 6 GEN III Nuclear Power Plant: 740 MWe class

ACR-1000* GEN III+ 1200 MWe class

Ve

Darlington 4 units 935 MWe
In-service: 1990-93

Artist's impression of a 2-unit ACR-1000 GEN III+ Nuclear Power Plant: 1200 MWe class

AECL

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FIGURES

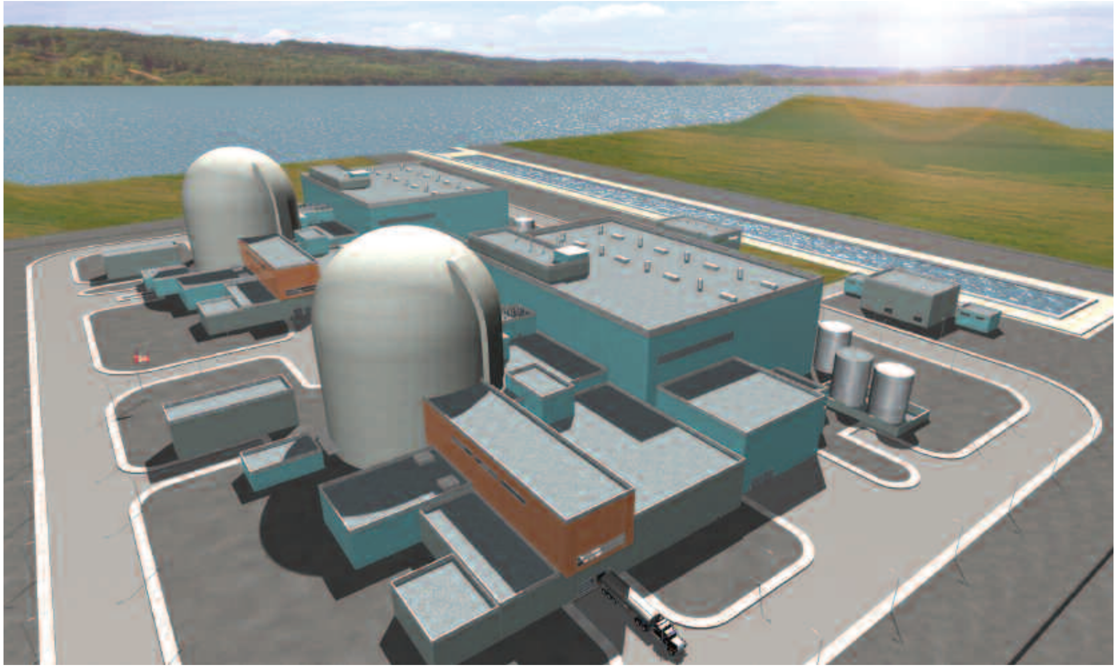


Figure 1-1 Pictorial View of Two-Unit Enhanced CANDU 6 Plant

1. Introduction

The Enhanced CANDU 6™ (EC6™) is a Generation III 740 MWe pressure tube reactor, designed to meet industry and public expectations for safe, reliable, environmentally-friendly, nuclear power generation. It has been enhanced by using the experience and feedback AECL has gained in the design, construction and operation of 10 CANDU 6 plants operating in five countries around the world.

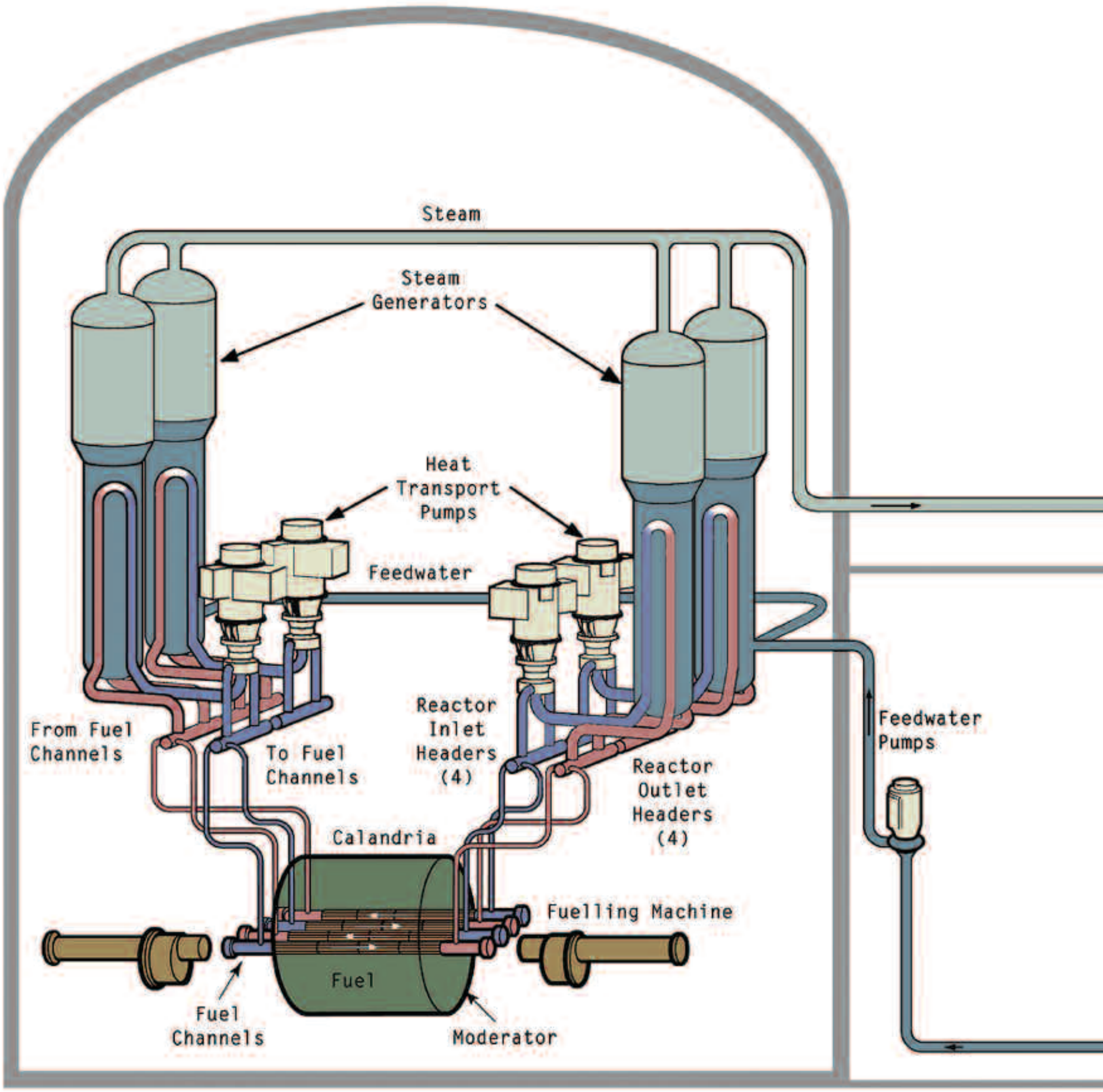
The Enhanced CANDU 6 is a heavy-water-cooled, heavy-water-moderated pressure-tube reactor retaining the basic features of the CANDU 6 design while incorporating a specific set of innovative features and state-of-the-art technologies to ensure that the safety, operation and performance of the Enhanced CANDU 6 meet the latest requirements.

The Enhanced CANDU 6 reactor has a projected annual capacity factor of more than 90%. For 2006, the global CANDU 6 fleet achieved an average gross capacity factor of 92.4%, ranking in the world's top reactor performance echelons. Three of the Wolsong CANDU units in Korea were in the top 10 reactors in the world over the last decade.

Enhanced safety features include design improvements based on insights gained from Probabilistic Safety Analysis (PSA) performed during the design process.

The latest design tools (CADDS) linking material management, documentation, safety analysis and project execution databases are used to ensure that accurate and complete configuration management can be readily maintained by the plant Owner.

Introduction



1.1 Enhanced CANDU 6 Design Features

The Enhanced CANDU 6 design benefits from the proven principles and characteristics of the CANDU 6 design and the extensive knowledge base of CANDU technology gained over many decades of operation.

Proven CANDU strengths:

- Modular, horizontal fuel channel core
- Separate low-temperature, low-pressure moderator provides inherently passive heat sinks
- Reactor vault filled with cool light water, which surrounds the core
- On-power refuelling
- Two independent safety shutdown systems
- Reactor building access for on-power maintenance

EC6 Enhancements from CANDU 6:

- Increased safety and operating margins
- Enhanced accident resistance and core damage prevention features
- SMART CANDU™ advanced operational and maintenance information systems for improved performance
- Thicker, steel-lined containment
- Improved fire protection

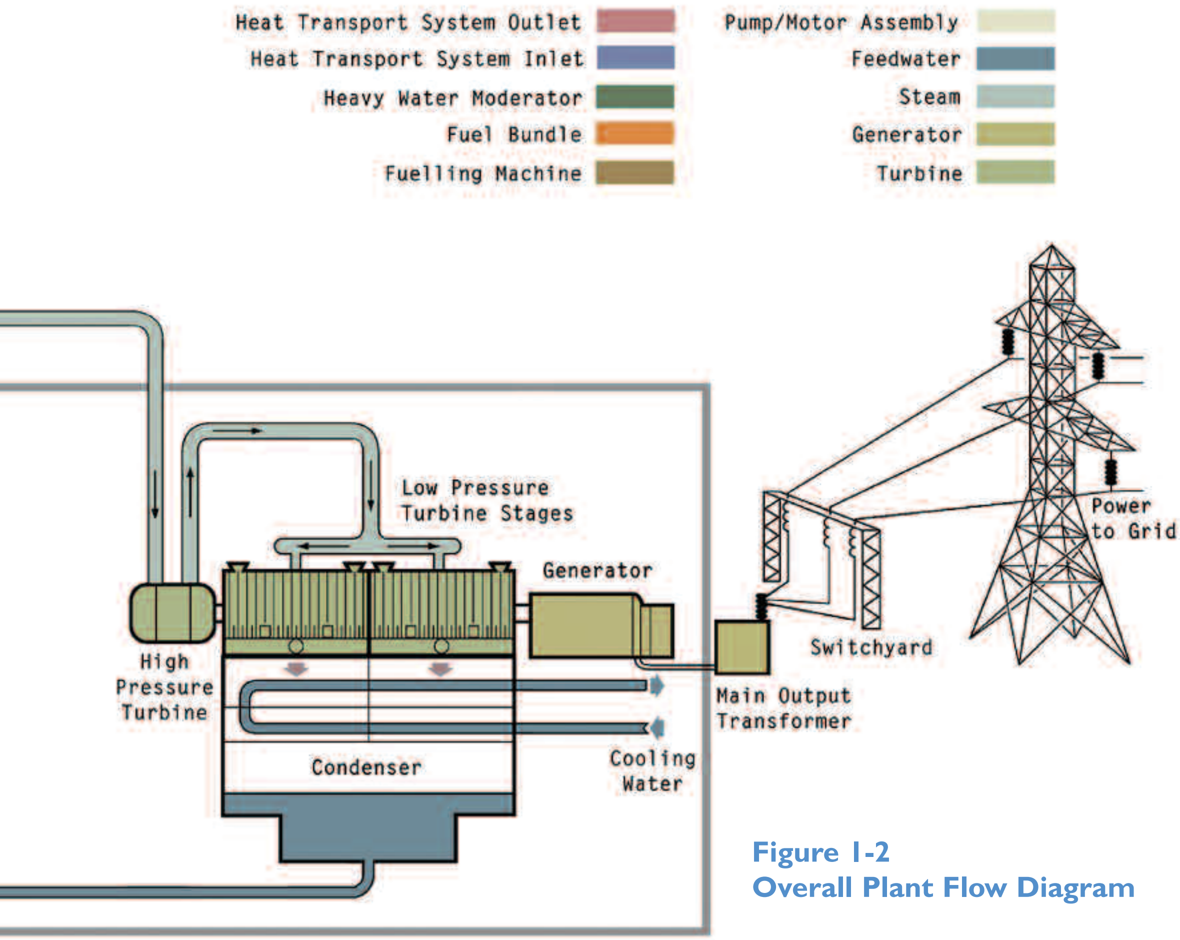


Figure 1-2
Overall Plant Flow Diagram

Significant design simplifications:

- Passive containment designed for all design basis events, eliminating need for large-flow dousing system
- Reduced inspections through improved design and feeder material selection for increasing resistance to flow-assisted corrosion (FAC) and robust fuel channel design margins
- Maintenance-based design with provision for built-in electrical, water and air supplies for on-power and normal shutdown maintenance

These technical improvements, along with advances in project engineering, manufacturing, and construction, result in significantly reduced capital cost and construction schedule, while enhancing the inherent safety features of the CANDU design.

1.2 Passive Safety Features

The Enhanced CANDU 6 design includes a number of "passive" safety features, some of which are design improvements over the already robust safety systems in existing CANDU plants. Examples of optimized passive safety features include:

- Two independent passive shutdown systems, each of which is capable of safely shutting down the reactor
- Increased safety margins
- Cool, low-pressure moderator, which serves as a passive heat sink for decay heat from the fuel channels in severe accident situations
- Large concrete reactor vault, surrounding the core in the calandria vessel and containing a large volume of light water to further slow down or arrest severe core damage progression by providing a second, passive, core heat sink
- Elevated reserve water tank (RWT) (located in the upper level of the containment building) designed to deliver passive make-up cooling water by gravity to the steam generators, to the moderator and to the calandria vault—delaying progression of severe accidents and providing even more time for mitigating actions
- Passive, robust, seismically-qualified containment consisting of:
    - Thickened pre-stressed concrete structure designed to withstand aircraft crashes
    - Leak-tight inner steel liner to reduce potential leakages
    - Passive spray system from elevated reserve water tank to reduce reactor building pressures in the event of a severe accident

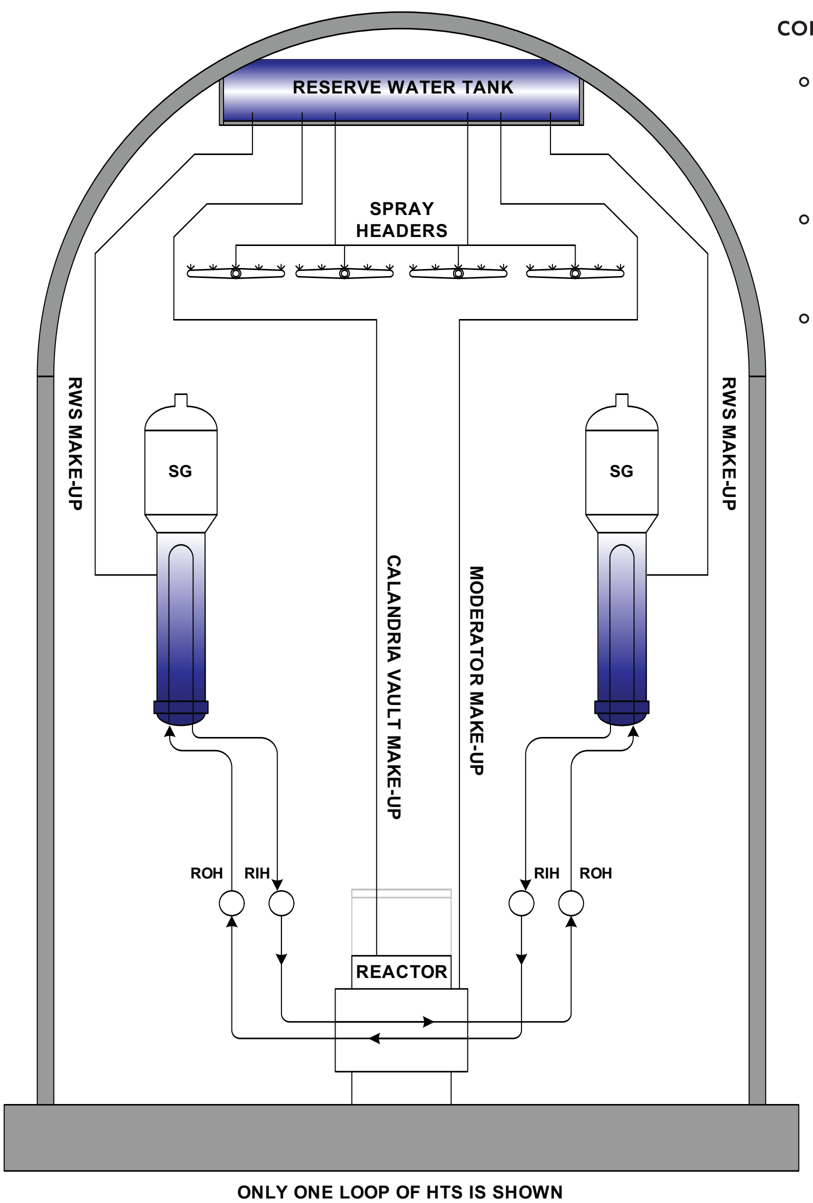


Figure 1-3 Reserve Water System

2. Plant Design

2.1 Plant Layout

Designed for efficient operation and increased safety, the plant is laid out to provide separation by distance, elevations, and the use of barriers for safety support structures, systems and components.

Security and physical protection have been enhanced to meet the latest criteria required in response to potential common mode events, such as fires, aircraft crashes and malevolent acts.

The plant layout is for a two-unit station designed to achieve the shortest practical construction schedule while supporting shorter maintenance durations with longer intervals in-between. The buildings are arranged to minimize interferences during construction, with allowance for on-site fabrication of module assemblies. Open-top construction, allows flexible equipment installation sequences.

A single-unit plant can be adapted from the two-unit layout with no significant changes to the reference design.

The major buildings and structures associated with the overall two-unit site arrangement are:

- Reactor Buildings (2)
- Service Buildings (2)
- Turbine Buildings (2)
- Secondary Control Areas (2)
- Condenser Cooling Water (CCW) Pumphouse
- Main Switchyard

The size of the power block for a 2 unit EC6 station is 31,000 m²* (actual area).

**Power block consists of 2 reactor buildings, 2 service buildings, 2 turbine buildings, 2 high pressure ECC buildings, 2 secondary control areas and 1 D_2O upgrader building.*

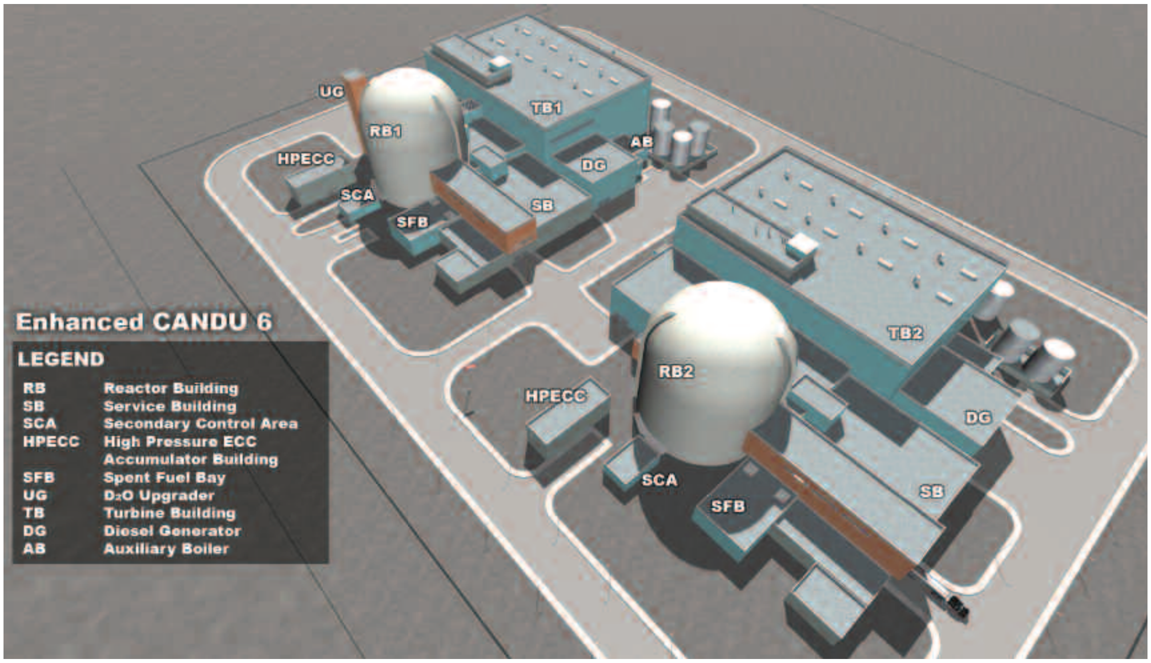


Figure 2-1 Two-Unit Plant Layout of Major Structures

Plant Design

Reactor Building

Strengthened over previous CANDU designs, the prestressed concrete reactor building is seismically qualified. The concrete perimeter wall has an inner steel liner which will achieve significantly reduced leak rates in the event of an accident. An isolation system ensures "button-up" in case of accidents.

The entire structure, including concrete internal structures, is supported on a reinforced concrete base slab to ensure a fully enclosed boundary for environmental protection and biological shielding.

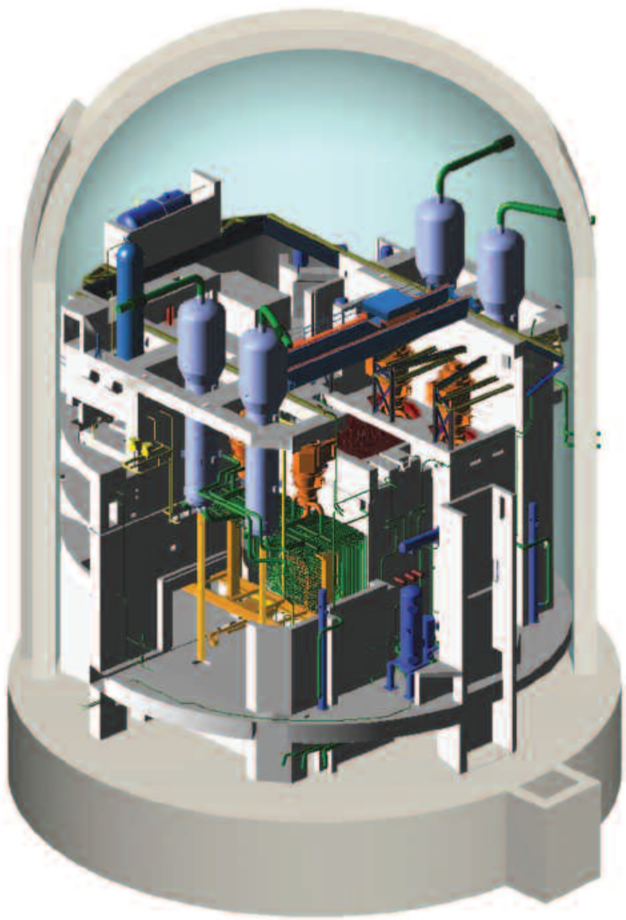


Figure 2-2 Reactor Building

Internal shielding permits personnel access, during operation, to specific areas for inspection and routine maintenance. These areas are designed to have temperatures suitable for personnel activities. Airlocks are designed as routine entry/exit doors.

Containment structure perimeter walls are separate from internal structures, so as to eliminate any interdependence and to provide flexibility in construction.

Service Building

The service building is a multi-level, reinforced concrete and steel structure that is seismically qualified and tornado protected. It accommodates the umbilicals that run between the principal structures, the electrical systems, and the spent fuel bay and associated fuel-handling facilities. It also houses the emergency core cooling (ECC) pumps and heat exchangers and the spent fuel bay cooling and purification system pumps and heat exchangers. The safety and isolation valves of the main steam lines are housed in a seismically-qualified concrete structure located on top of the building.

Containment Structure

Type	Prestressed concrete / steel liner
RB inside diameter	41.4 m
RB containment wall thickness	1.8 m
Building height (base slab to top of dome)	64.7 m

Turbine Building

The turbine building is located on one side of the service building. This is an optimum location for access to the main control room, the piping and cable tray runs to and from the service building, and the condenser cooling water ducts to and from the main pumphouse. Access routes are provided between the turbine building and the service building.

The turbine building houses the turbine generator and its auxiliary systems, the condenser, the condensate and feedwater systems, the building heating plant, and any compressed gas required for the balance of plant (BOP). Blowout panels in the walls and roof serve to relieve the internal pressure in the event of a steam line break.

Main Control Area

The main control area is located within each of the service buildings. The main control area in each unit contains the main control room (MCR) and associated controls.

Secondary Control Area

Each unit has a completely separate secondary control area (SCA) with sufficient control and monitoring equipment to shut down the unit, initiate the required cooling and ensure the unit remains in a safe shutdown state should the main control room (MCR) become uninhabitable or not functional. The SCA is located so that the MCR and the secondary control area cannot be simultaneously rendered inoperable due to any design basis event.

Condenser Cooling Water (CCW) Pumphouse

The condenser cooling water (CCW) pump-house has a reinforced concrete substructure and braced steelframe superstructure. It contains the CCW pumps, raw service water (RSW) pumps, screen wash pumps, trash racks, screens, and chlorination equipment, if required. Together with related intake and outfall structures, the pumphouse serves the two-unit Enhanced CANDU 6 plant, housing separate CCW and RSW systems with adequate separation for each unit. Sites with limited cooling water availability can use cooling towers instead of the conventional CCW system.

Main Switchyard

The switchyard is designed to provide a flexible switching arrangement to connect the plant main power output systems to the transmission grid. Traditionally, breaker and a half switching configurations have been used to interconnect the high voltage transmission lines to the generating units and system service supplies.

2.2 Nuclear Power Plant Siting

2.2.1 Unit Output

Each unit of the Enhanced CANDU 6 two-unit integrated plant design has a nominal gross electrical output of 740 MW(e). Output can be optimized by adjusting the turbine/condenser design to suit any site cooling water conditions.

2.2.2 Adaptation to Site Requirements

The Enhanced CANDU 6 can accommodate a wide range of geotechnical and meteorological data and conditions through its flexible design features:

- Cooling water systems for all nuclear steam plant cooling requirements can accommodate saltwater or fresh water sites. The plant can also accommodate the use of conventional cooling towers
- The ability to accommodate a range of cooling water temperatures, from those for a typical cold site to those for a typical warm site. A generic set of reference conditions has been developed to suit potential sites for the Enhanced CANDU 6
- The Design Basis Tornado (DBT) for the Enhanced CANDU 6 is selected to satisfy tornado design requirements for sites in North American and other potential sites overseas
- The ability to withstand Design Basis Earthquakes (DBE) at the plant site. This is the maximum ground motion of a potentially severe earthquake that has a low probability of being exceeded during the life of the plant

2.3 Nuclear Systems

Nuclear systems are located in the reactor building and the service building. These buildings are robust and shielded where necessary, to ensure that all radioactive substances are always secure. Systems include:

- Heat transport system with D_2O coolant in a two-loop, figure-of-eight configuration with four steam generators, four heat transport pumps, four reactor outlet headers, and four reactor inlet headers. This configuration is standard on all CANDU 6 reactors
- Heavy-water moderator system
- Reactor assembly, consisting of a calandria vessel installed in a concrete vault
- Fuel handling system, which consists of two fuelling machine heads each mounted on a fuelling machine bridge, supported by columns, located at each end of the reactor
- Two independent shutdown systems, emergency core cooling (ECC) system, containment system and associated safety support systems

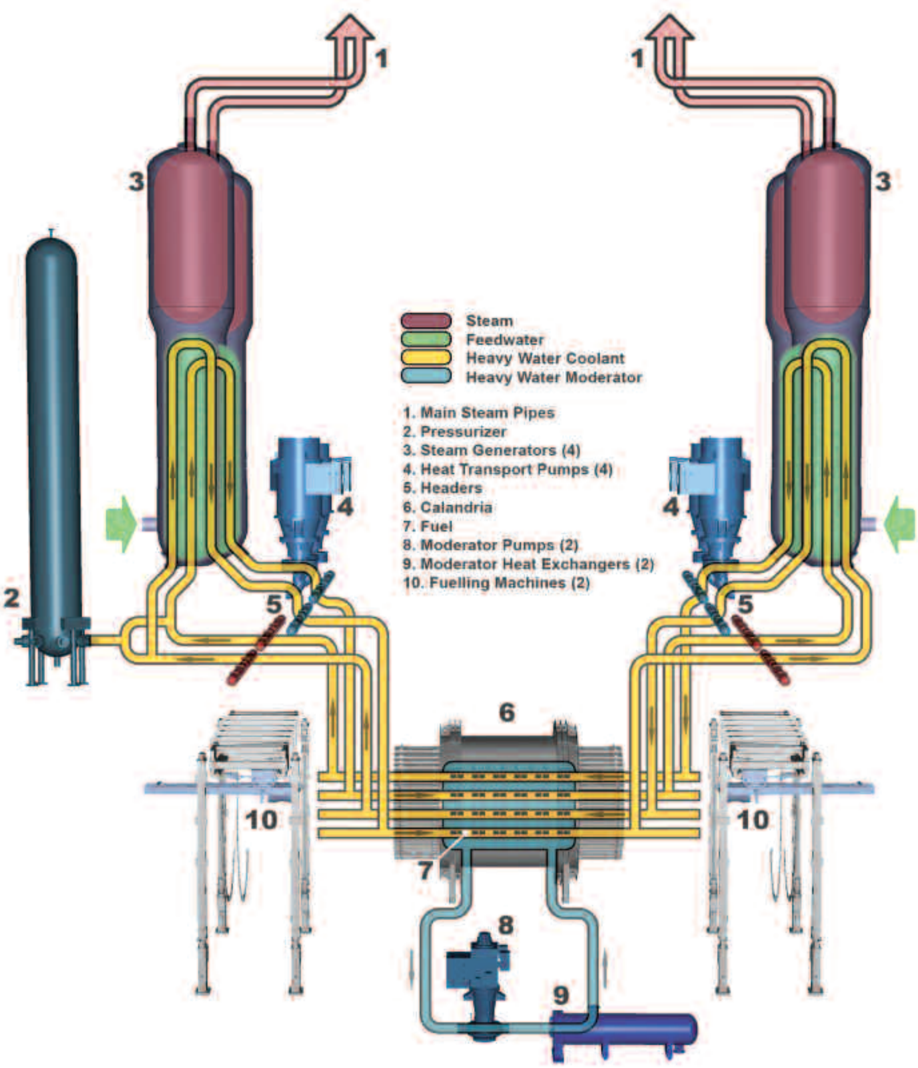


Figure 2-3 Nuclear Systems Schematic

2.4 Heat Transport System and Auxiliary Systems

The heat transport system (HTS) circulates pressurized heavy water coolant through the reactor fuel channels to remove heat produced by nuclear fission in the core.

The HTS consists of 380 reactor fuel channels with associated corrosion-resistant feeders, four inlet headers, four outlet headers and the interconnecting piping. The system includes four steam generators and four electrically-driven heat transport pumps in a two-loop, figure-of-eight configuration. The headers, steam generators, and pumps are all located above the reactor.

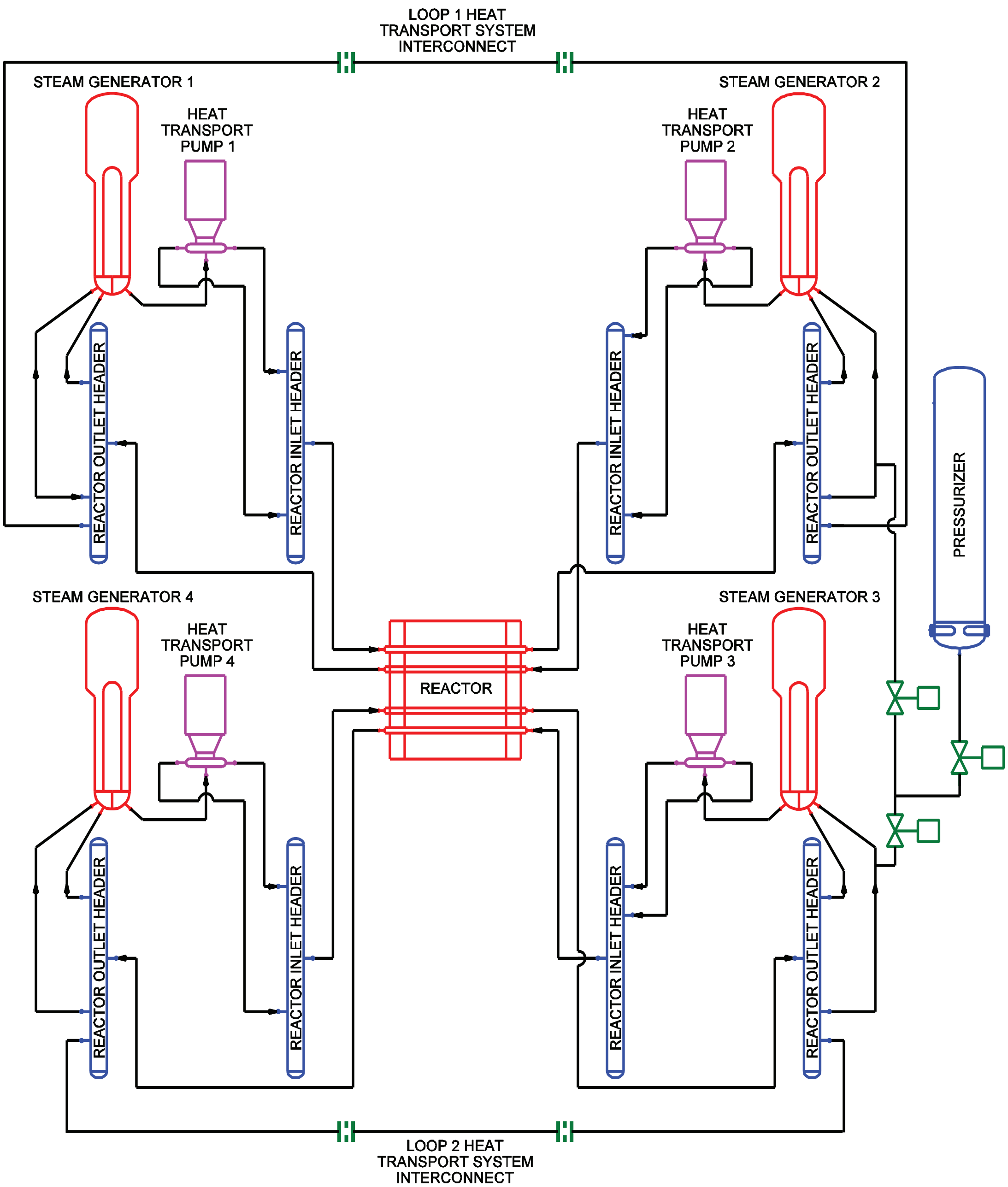


Figure 2-4 Heat Transport System Flow Diagram: Illustrating two loop flow paths

Heat Transport System Design Data

Reactor outlet header pressure [MPa (g)]	9.9
Reactor outlet header temperature [°C]	310
Reactor inlet header pressure [MPa (g)]	11.2
Reactor inlet header temperature [°C]	260
Single-channel flow (maximum) [kg/s]	28

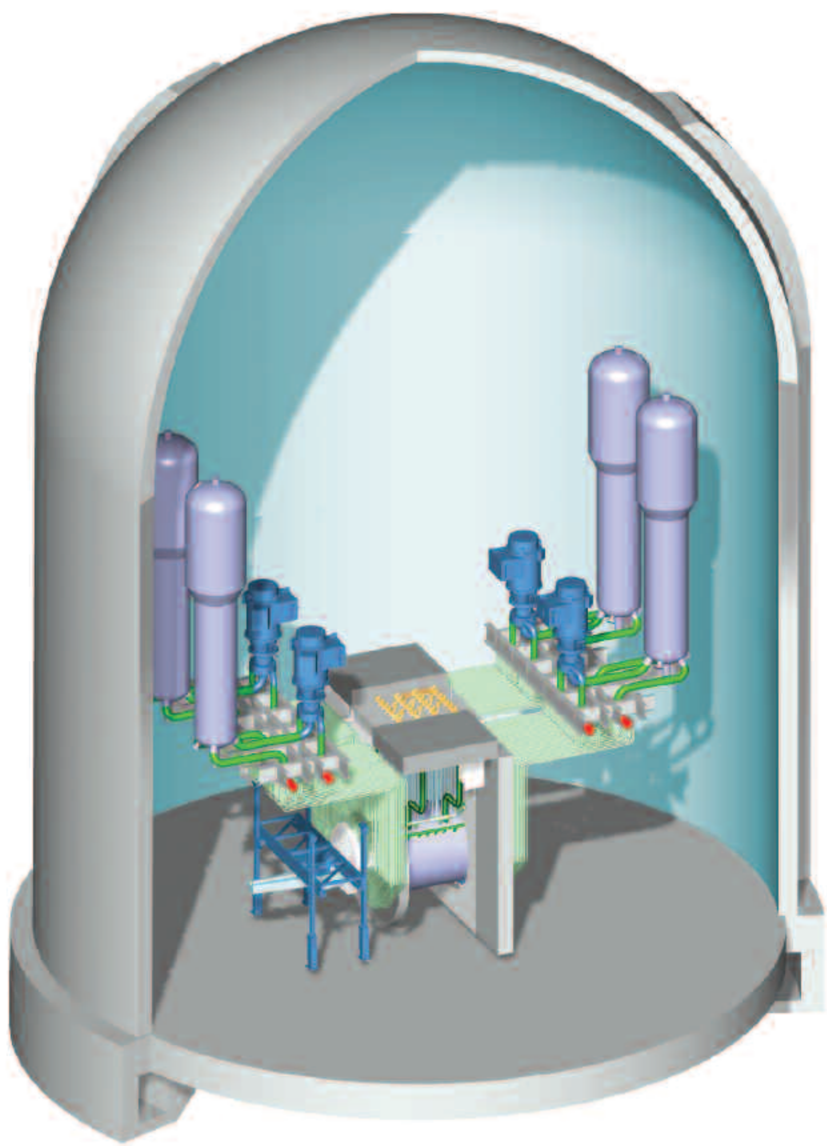


Figure 2-5 3D View of Heat Transport System in Reactor Building

Pressure and Inventory Control System

The heat transport pressure and inventory control system consists of a pressurizer, a degasser-condenser, two D_2O feed pumps, feed and bleed valves and a coolant storage tank. This system provides:

- Pressure and inventory control for each heat transport system loop
- Overpressure protection
- A controlled degassing flow

D_2O in the pressurizer is heated electrically to pressurize the vapour space above the liquid. This cushion pressure transients, without allowing excessively high or low pressures in the heat transport system.

The pressurizer also accommodates the change in volume of the reactor coolant in the heat transport system from zero power to full power. This permits reactor power to be increased or decreased rapidly, without imposing a severe demand on the coolant feed and bleed components of the system.

When the reactor is at power, (normal mode) pressure in the reactor outlet headers is controlled by the pressurizer; heat is added to the pressurizer with the electric heaters to increase pressure, and steam is bled from the pressurizer to the degasser-condenser to reduce pressure. The coolant inventory is adjusted by the feed and bleed circuit to maintain the pressurizer level at setpoint. Cool D_2O inventory is provided from the D_2O purification system via sprays, to further reduce temperatures and adjust inventory before and after maintenance.

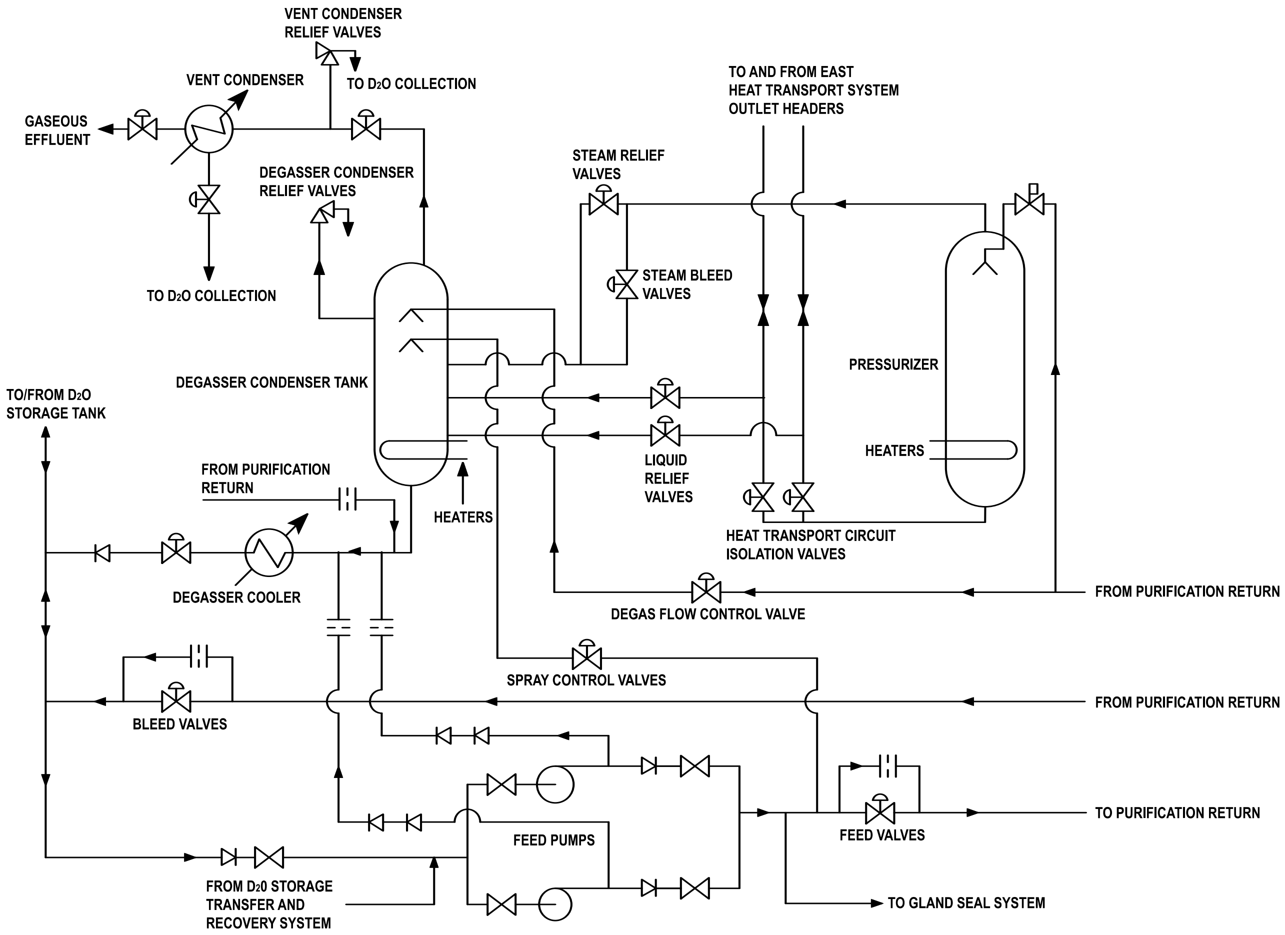


Figure 2-6 Pressure and Inventory Control Flow Diagram

2.4.1 Heat Transport Pumps

The EC6 heat transport pump retains the CANDU 6 mechanical multi-seal design, which allows for easy replacement.

Seal cooling lengthens pump service life and the time the pump will survive during accident conditions.

Heat Transport Pump Data

Number	4
Rated flow [L/s]	2228
Motor rating [MWe]	6.7

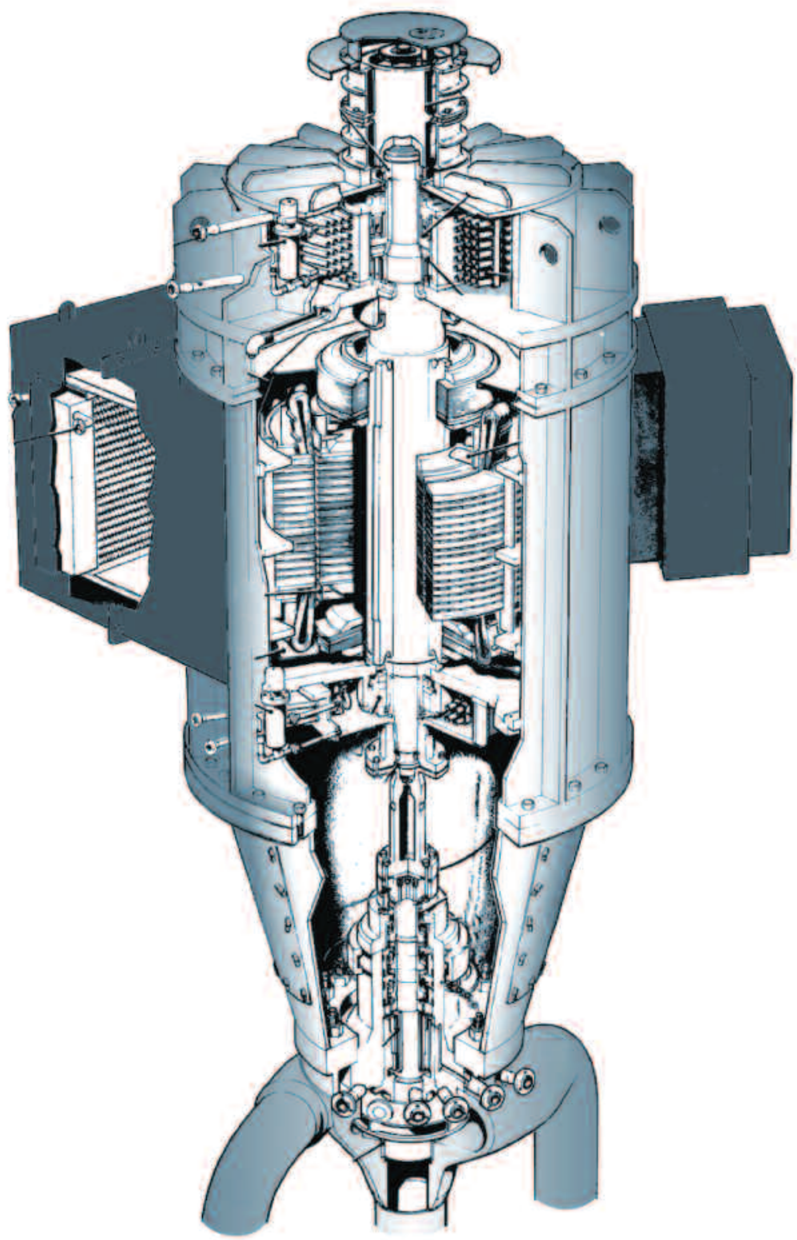


Figure 2-7 Heat Transport System Pump

2.4.2 Steam Generators

The steam generators are similar to those of the CANDU 6. The tubing is made of Incoloy-800, a material with proven operating performance and service at CANDU 6 stations.

Steam wetness at the steam nozzle has been reduced based on the latest steam separator technology, leading to improved turbine cycle economics.

Steam Generator Design Data

Number	4
Type	Vertical U-tube / integral preheater
Nominal Tube diameter [mm]	15.9 (5/8")
Steam temperature (nominal) [°C]	260
Steam pressure [MPa (g)]	4.6

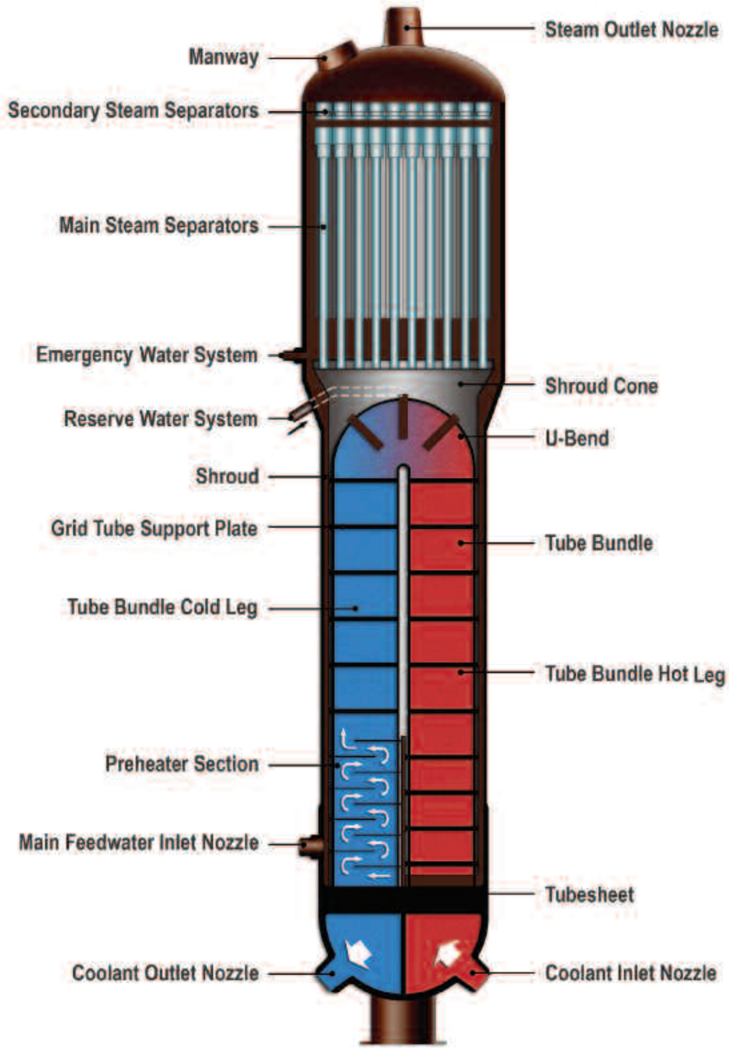


Figure 2-8 Steam Generator

2.5 Moderator System

The moderator system is a low-pressure and low-temperature system that is fully independent of the heat transport system. The moderator system consists of pumps and heat exchangers that circulate the heavy water moderator (D_2O) through the calandria vessel and remove the heat generated within the moderator during reactor operation. The heavy water acts as both a moderator and reflector for the neutron flux in the core.

The moderator system fulfills a safety function that is unique to CANDU. It serves as a backup heat sink in the event of loss of fuel cooling via the heat transport system, thereby mitigating core damage consequences.

An added safety improvement in the EC6 is a connection to the reserve water tank that provides additional passive gravity-fed inventory to the calandria vessel, extends core cooling and delays severe accident event progression.

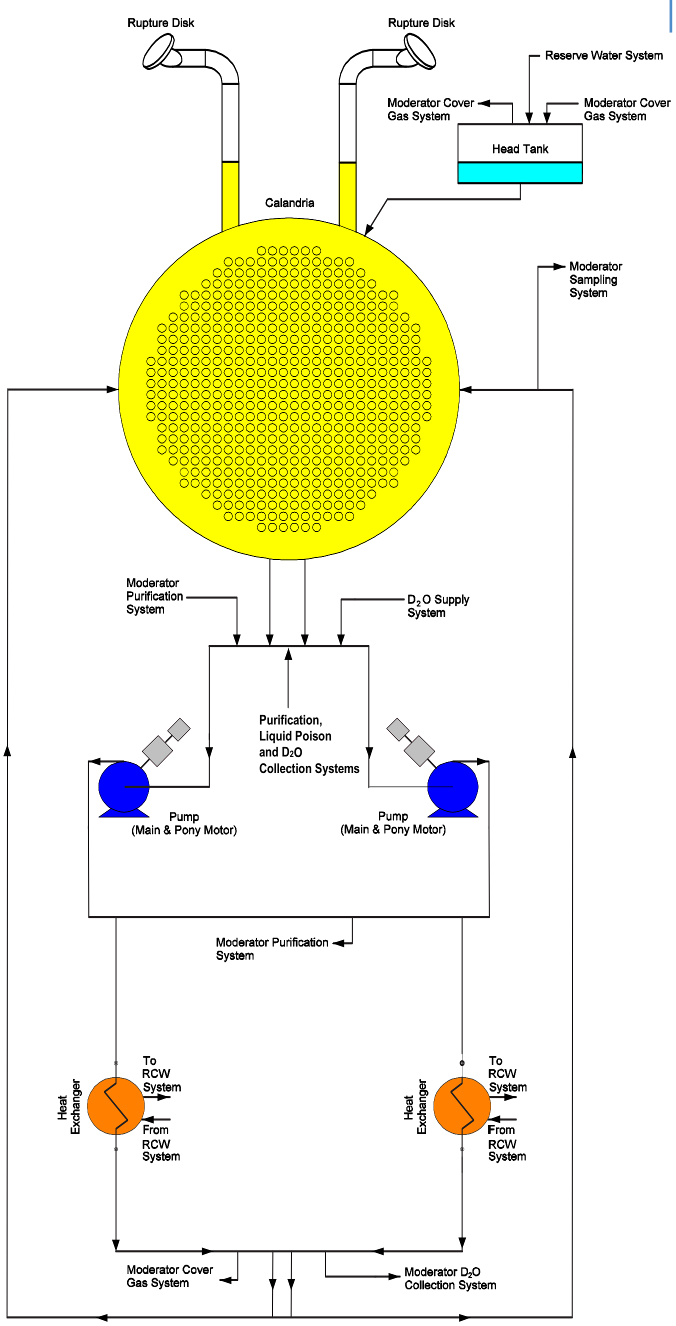


Figure 2-9 Moderator System Flow Diagram

Heavy Water Inventory Design Data

Moderator System [Mg D_2O]	265
Heat Transport System [Mg D_2O]	192
Total [Mg D_2O]	**457**

2.6 Reactor Assembly

The reactor assembly consists of the horizontal, cylindrical, low-pressure calandria and end-shield assembly. This enclosed assembly contains the heavy water moderator, the 380 fuel channel assemblies and the reactivity mechanisms. The reactor is supported within a concrete, light-water-filled calandria vault. Fuel is enclosed in the fuel channels that pass through the calandria and the end-shield assembly. Each fuel channel permits access for on-line fuelling operation while the reactor is at power.

The ability to replace fuel as required to maintain reactor power means there is minimal "excess" reactivity in the core at all times—an inherent safety feature. This feature also contributes to operational flexibility for improved outage planning since fixed cycle times are not required, and it allows the prompt removal of defective bundles without shutdown.

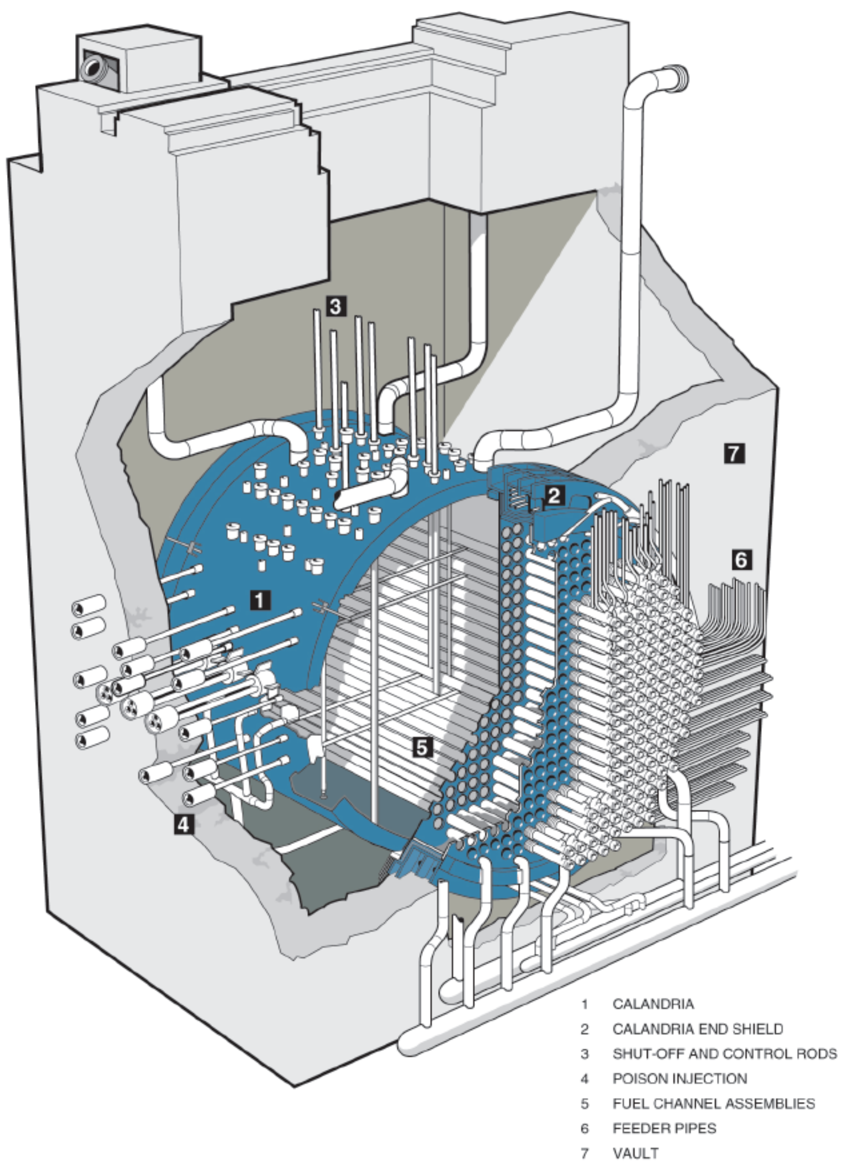


Figure 2-10 Reactor Assembly

Reactor Core Design Data

Output [MWth]	2084
Coolant	Pressurized D_2O
Moderator	D_2O
Calandria diameter [m]	7.6
Fuel channel	Horizontal Zr-2.5wt%Nb alloy pressure tubes with modified 403 SS end-fittings
Fuel channels	380
Lattice pitch (mm)	286

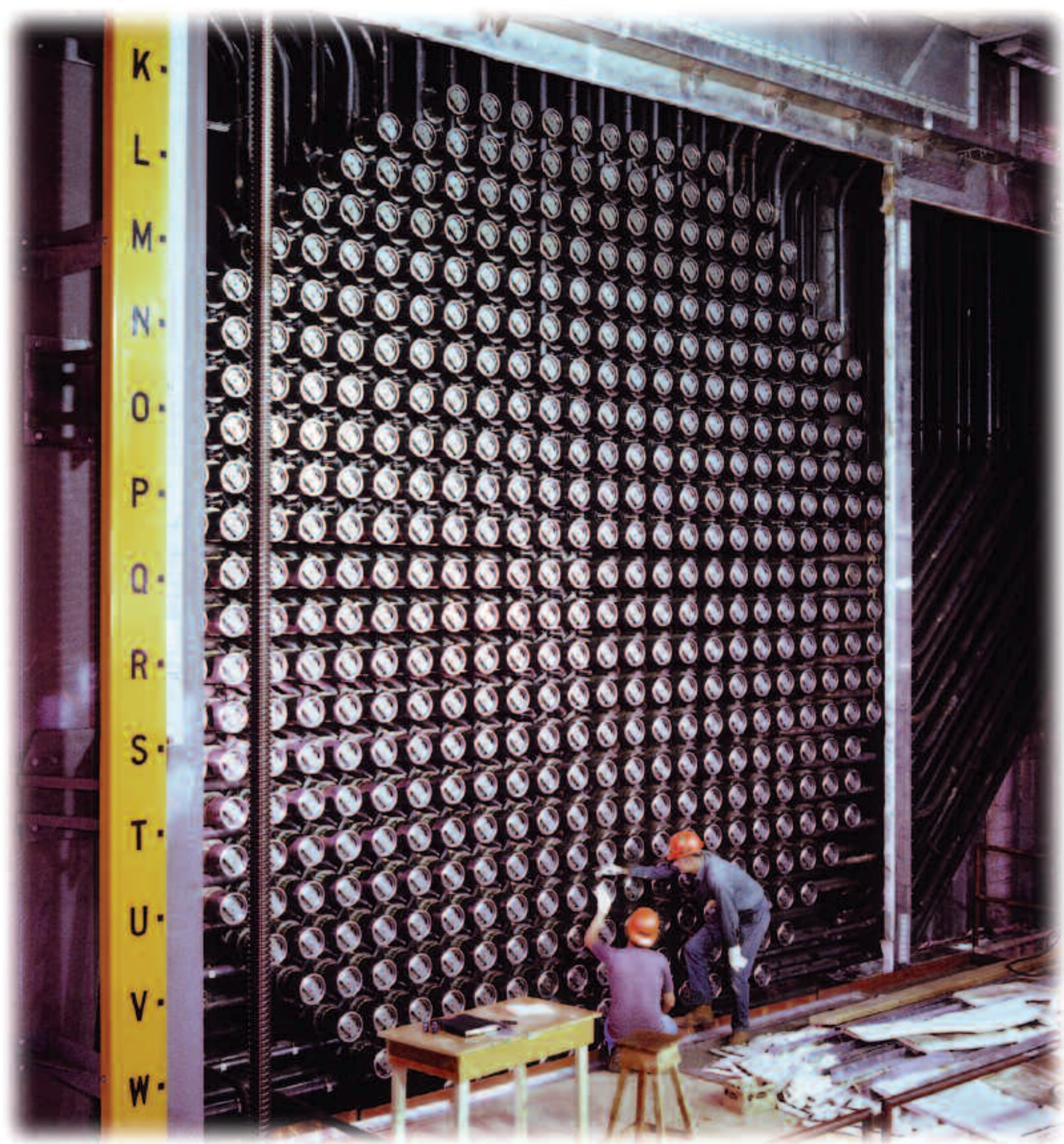


Figure 2-11 CANDU 6 Reactor Face

2.6.1 Reactor Control

Liquid zone control units provide the primary control in the Enhanced CANDU 6. Each zone control assembly consists of independently adjustable liquid zones. On-power refuelling and zone-control actions provide day-to-day reactivity control.

The reactor regulating system also includes control absorber units and adjusters that can be used to reduce reactor power if larger power reductions are required.

Each pressure tube is thermally insulated from the low-temperature moderator by the gas annulus between the pressure tube and the calandria tube. Tight-fitting spacers, positioned along the length of the pressure tube, maintain the annular space and prevent contact between the two tubes. Each end-fitting holds a liner tube, a shield plug, and a channel closure. The reactor coolant flows through adjacent fuel channels in opposite directions.

The Enhanced CANDU 6 is designed with provision for mid-life refurbishment, including replacement of pressure tubes.

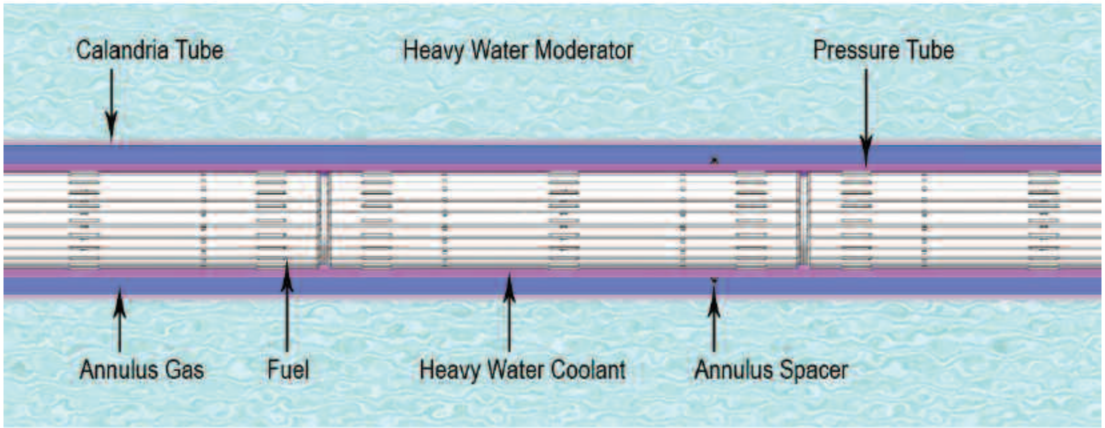


Figure 2-12 Fuel Channel

2.7 Fuel Handling Systems

The fuel handling systems consist of:

- New fuel handling and storage system
- Fuelling machines and their supports
- Spent fuel handling and storage

The fuel handling and storage system includes the storage of the natural uranium (NU) fuel with sufficient capacity to maintain full-power operation for at least six months.

Two fuelling machines are located on opposite sides of the reactor and mounted on bridges supported by columns.

The normal refuelling operation is an eight-bundle shift, in the direction of the coolant flow, in which spent bundles are removed from the outlet end of a fuel channel, while fresh bundles are inserted at the inlet end.

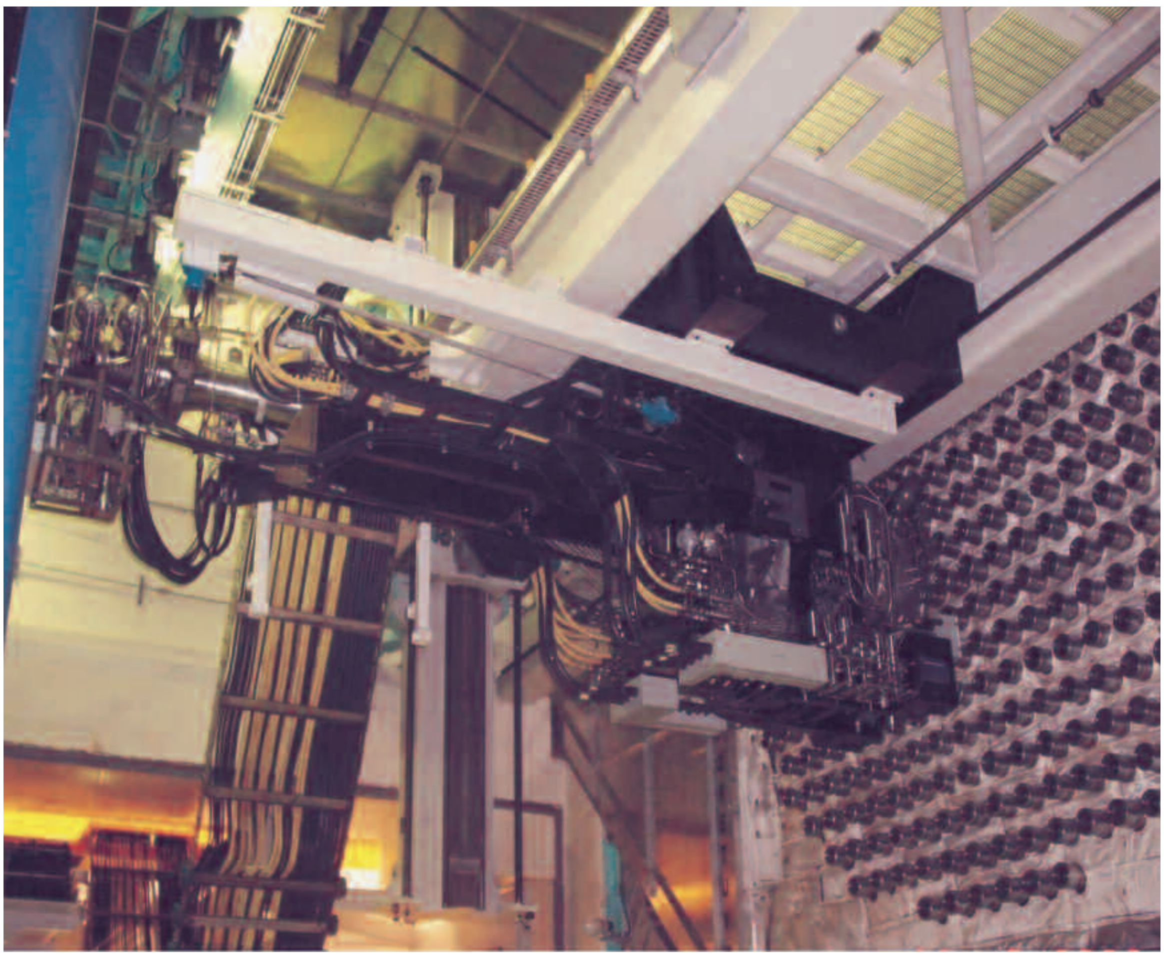


Figure 2-13 Fuelling Machine

The EC6 spent fuel transfer and storage system handles spent fuel from the time it is discharged from the fuelling machine to the time it is moved to the underwater spent fuel storage bay in the service building.

A storage bay man bridge and handling tools permit manipulation of spent fuel and containers.

From the loading of fresh fuel in the new-fuel mechanism to the discharge of spent fuel in the receiving bay, the fuelling process is automated and remotely controlled from the station control room.

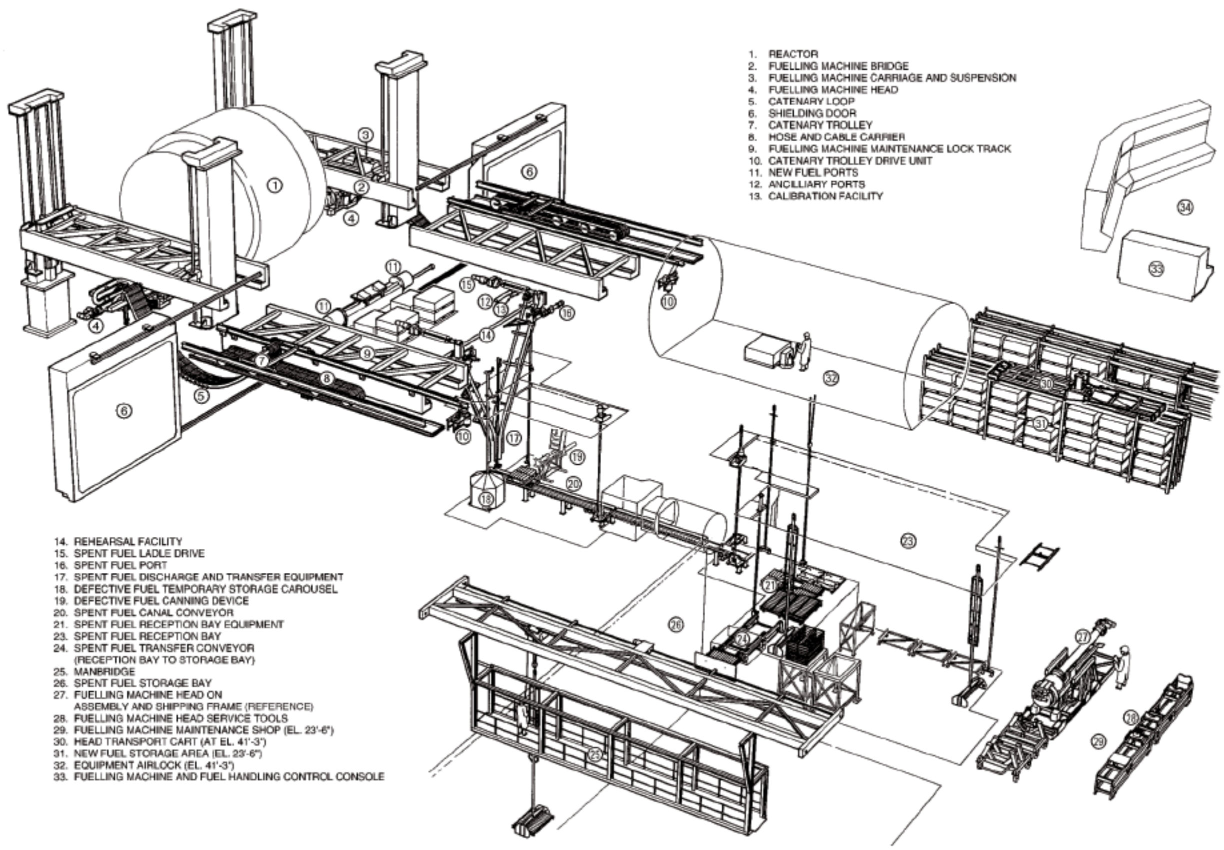


Figure 2-14 Fuel Handling Systems Arrangement

2.8 Fuel

The EC6 uses the proven 37-element natural uranium fuel bundle design.

Fuel	Natural UO_2
Enrichment level	0.71 wt% 235U
Fuel burn-up [MWd/te U]	7,500
Fuel bundle assembly	37 element
Bundles per fuel channel	12
Fuelling Scheme	8-bundle-shift

The natural uranium fuel cycle offers simplicity of fuel design, ease of fabrication, and benefits from the ready availability of natural uranium. This strategic feature facilitates localization of the EC6 technology. The Enhanced CANDU 6 retains the ability to adopt various fuel cycle options in the future.

- The first evolution of the CANDU fuel cycle can be the use of slightly enriched uranium (SEU), including recovered uranium from reprocessed Light Water Reactor (LWR) spent fuel. Relatively low enrichment (up to 1.2%) will result in a two- to three-fold reduction in the quantity of spent fuel per unit energy production, reductions in fuel-cycle costs, and greater flexibility in plant operations
- A high burnup CANDU Mixed Oxide (MOX) fuel design could utilize plutonium from conventional reprocessing or more advanced reprocessing options (such as co-processing)
- Long-term energy security can be assured either through the thorium cycle or through a CANDU / FBR (Fast Breeder Reactor) system, in which the FBR would be operated as a "fuel factory," providing the fissile material to power a number of lower-cost, high-efficiency CANDU reactors

The 43-element CANFLEX® (CANDU FLEXible) fuel bundle is available as an optimal fuel design for all these fuel cycles. Peak linear element ratings are reduced by 15-20% with CANFLEX fuel at current bundle power rating. Depending on burnup and fuel temperatures, the fission-gas release within the fuel element will be reduced. Allowable critical heat flux and critical channel powers can also be increased, due to optimized heat removal characteristics of the bundle.

Figure 2-15 Fuel Bundle

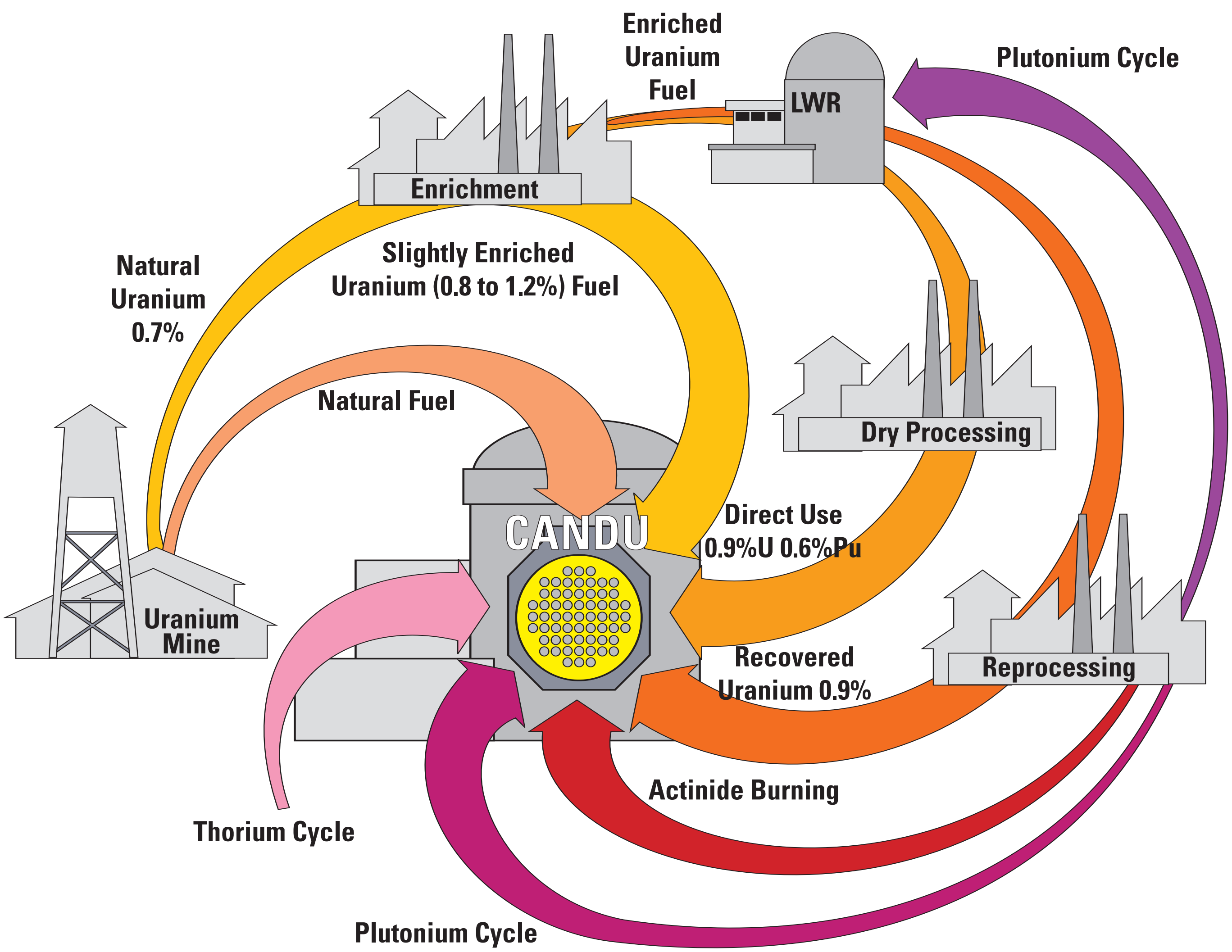


Figure 2-16 Flexible Fuel Cycle Options

2.9 Safety Systems

EC6 safety systems are designed to mitigate the consequences of plant process failures to ensure reactor shutdown, removal of decay heat, and prevention of radioactive releases.

The safety systems in the EC6 design follow the traditional CANDU practice of providing:

- Shutdown System 1 (SDS1)
- Shutdown System 2 (SDS2)
- Emergency Core Cooling (ECC) System
- Containment System

SDS1, SDS2, the ECC system, and the containment boundary meet high reliability requirements, established during system design and verified by reliability analysis.

Safety support systems are also provided to ensure reliable electrical power, cooling water, and instrument air supplies to the safety systems. Standby generators are provided as backup to the station power for postulated loss of station power events.

Safety systems and their support services are designed to perform their safety functions with a high degree of reliability. This is achieved through the use of redundancy, diversity, separation, testability, the application of appropriate quality assurance standards, and the use of stringent technical specifications, including seismic qualification and environmental qualification for accident conditions.

2.9.1 Shutdown Systems

The EC6 incorporates two passive, fast acting, fully capable, diverse, and separate shutdown systems, which are physically and functionally independent of each other.

SDS1 consists of mechanical shutoff rods that drop by gravity into the core when a trip signal de-energizes the clutches that hold the shutoff rods out of the core. The design of the shutoff rods is based on the proven CANDU 6 design.

SDS2 injects a concentrated solution of gadolinium nitrate into the low-pressure moderator to quickly render the core subcritical. The gadolinium nitrate solution is dispersed uniformly with pressurized gas, maximizing shutdown effectiveness.

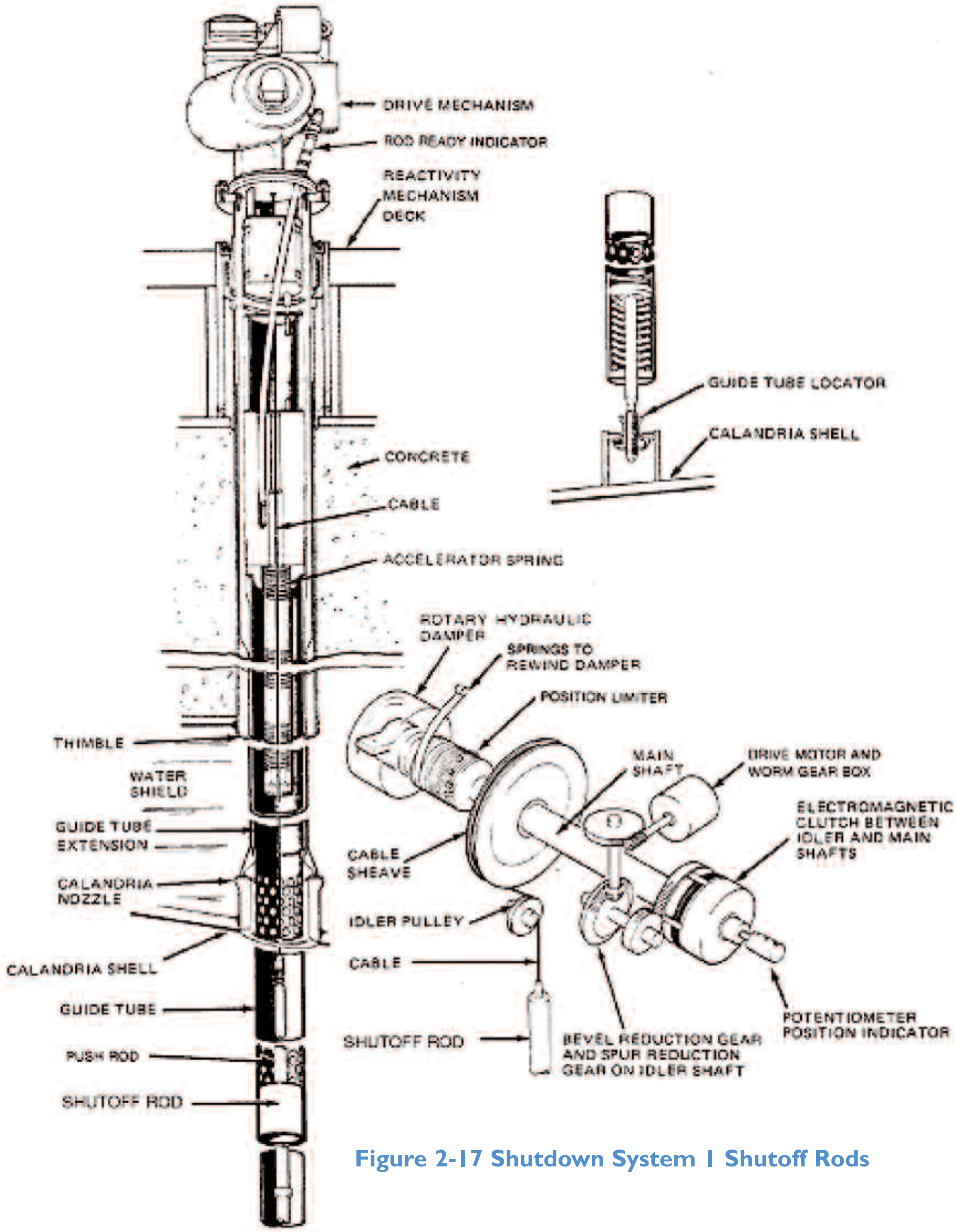


Figure 2-17 Shutdown System 1 Shutoff Rods

2.9.2 Emergency Core Cooling (ECC) System

The emergency core cooling (ECC) system consists of:

- Passive high-pressure emergency core cooling (HPECC) system:

  The HPECC system has accumulator tanks that will supply high-pressure water to the HTS and refill the fuel channels in the short term after a loss of coolant accident (LOCA)

  During normal operation the HPECC system is poised to detect any LOCA that results in a depletion of HTS inventory to such an extent that make-up by normal means is not assured. When the HTS pressure drops below the pressure of the HPECC accumulator tanks, water is injected into the heat transport system

  Valves on the HPECC interconnect lines between the reactor outlet headers (ROH) open upon detection of a LOCA to assist in establishing a sustainable cooling flow path

- Low-pressure emergency core cooling (LPECC) system:

  The LPECC system provides long-term recirculation and recovery. The LPECC system is used for cooling of the reactor including LOCA

  LPECC is initiated automatically when HTS is sufficiently depressurized, at which time the LPECC system begins operation in long-term recovery mode

2.9.3 Containment System

The containment system forms a continuous, pressure-retaining envelope around the reactor core and the heat transport system. This prevents releases of radioactive material to the external environment.

The containment boundary consists of a steel-lined, prestressed concrete reactor building structure, access airlocks and a containment isolation system. The containment design ensures a low leakage rate.

Heat removal from the containment atmosphere is also normally provided by the operation of local air coolers that are suitably located to maintain operating containment pressure and temperature.

Hydrogen control is provided in the reactor building by passive, autocatalytic recombiners and igniters to limit the hydrogen content to below the deflagration limit within the containment following a core damage accident.

Finally, the provision of a spray system connected to the elevated reserve water tank (RWT) will reduce reactor building pressures, if required, in the event of severe accidents.

2.10 Balance of Plant (BOP)

The balance of plant (BOP) comprises the turbine building, steam turbine, generator, condenser, and the feedwater heating system with associated auxiliary and electrical equipment. The BOP also includes the water treatment facilities, auxiliary steam facilities, condenser cooling water, pumphouse and/or cooling towers, and associated equipment to provide all conventional services to the plant.

2.10.1 Turbine Generator and Auxiliaries

The turbine generator system and the condensate and feedwater systems are based on conventional designs. They meet the design requirements specified by the NSP designer to assure the performance and integrity of the nuclear steam plant. These include requirements for: materials (i.e., titanium condenser tubes, absence of copper alloys in the feed train), chemistry control, feed train reliability, feedwater inventory, and turbine bypass capability.

In the event of loss of off-site power to the station, the reactors are designed to stay at power for the duration of the event with the turbine generators disconnected from the grid. In this mode of operation, power is only supplied to internal auxiliaries as needed for the safe operation of the plant.

Figure 2-18 Qinshan Low-Pressure Turbine Rotor

EC6 Turbine Generator

Steam Turbine Type	Hitachi impulse-type, tandem-compound
Steam Turbine Composition	One high-pressure cylinder, two low-pressure cylinders
Net thermal output to turbine (MWth)	2080
Gross/Net electrical output* (nominal) [MWe]	740*/690
Steam temperature at main stop valve [°C]	257 @ 4.5 MPa
Final feedwater temperature [°C]	187
Condenser Vacuum [kPa (a)]	4.9

(*) Site cooling water dependent

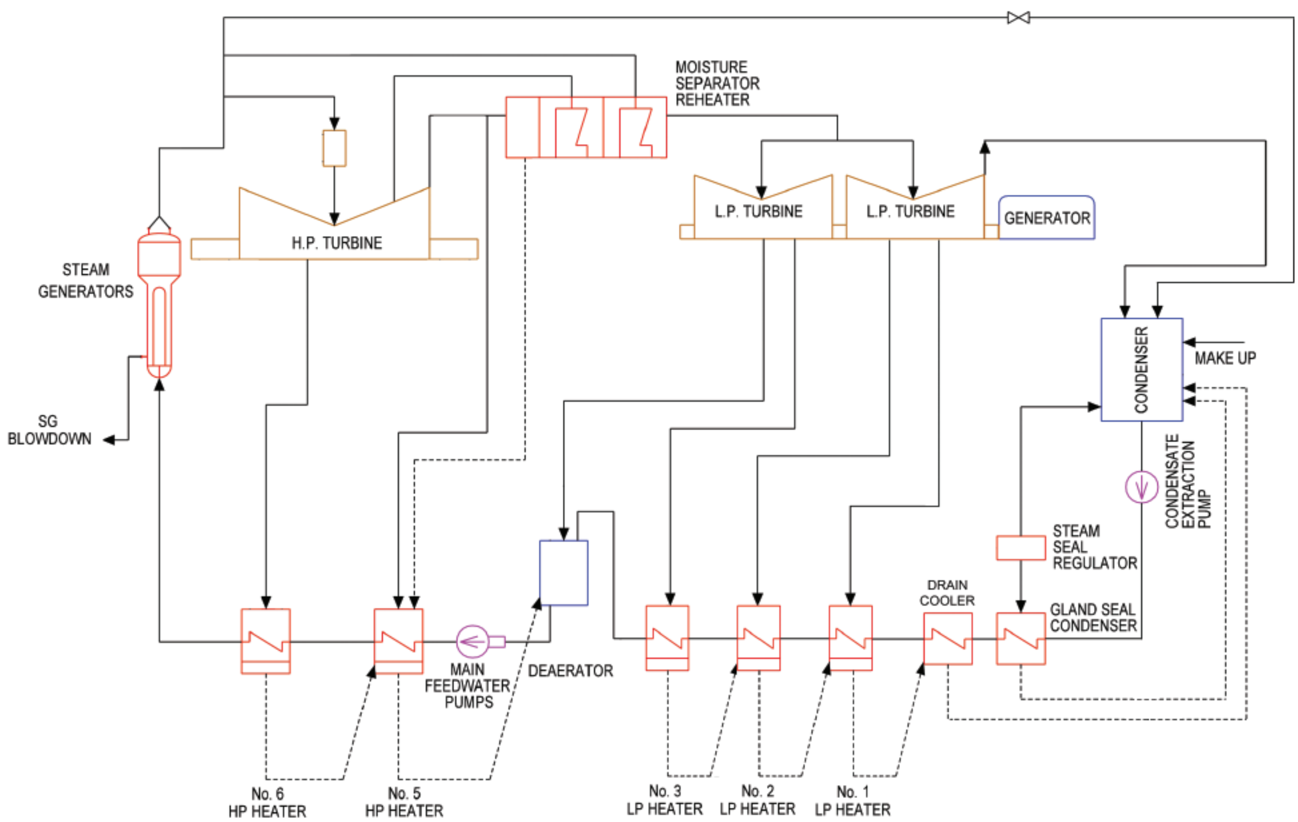


Figure 2-19 Turbine Generator and Auxiliaries Flow Diagram

2.10.2 Steam and Feedwater Systems

The EC6 main steam system supplies the steam from the steam generators in the reactor building to the turbine through the steam balance header. The feedwater system takes hot, pressurized feedwater from the feedwater train in the turbine building and discharges the feedwater into the preheater section of the steam generators. The feedwater system maintains the required steam generator level by controlling feedwater flow.

The condenser steam discharge valves (CSDVs) are designed to discharge up to 100% of steam flow directly to the condenser. This feature provides for operational flexibility in support of load following operation in conjunction with overall reactor control.

The safety functions of overpressure protection and cooling of the steam generator secondary side is provided by main steam safety valves (MSSVs). Main steam isolation valves (MSIVs) are provided and can be used to prevent releases in the event of steam generator tube leaks to the secondary side.

2.10.3 BOP Services

Conventional plant services include water supply, heating, ventilation, air conditioning, chlorination (if required), fire protection, compressed gases, and electric power systems.

Service Water Systems

The balance-of-plant water systems provide cooling water, demineralized water, and domestic water to plant users. The systems consist of the condenser cooling water (CCW) pumphouse, raw service water system, water treatment facility, and chlorination systems.

Heating, Ventilation, and Cooling Systems

Heating, ventilation, air conditioning, and chilled water (from the chilled water system) are supplied to the plant buildings to ensure a suitable environment for personnel and equipment during winter and summer.

The building heating plant provides the steam and hot water demands of the entire plant. Steam extracted from the turbine is used as the normal building heating steam source. Dedicated, separate ventilation systems are provided for the main control and secondary control areas.

Fire Protection System

Water supply for the main fire protection system comes from a fresh water source. The main system provides fire protection for the entire station (i.e., both NSP and BOP). In addition, a seismically-qualified water supply to the reactor building is provided.

The fire protection system also includes standpipe and fire hose systems, portable fire extinguishers for fire suppression, and a fire detection and alarm system covering all plant buildings and areas.

Fire-resistant barriers for mitigation purposes are provided where necessary to isolate and localize fire hazards and to prevent the spread of fire to other equipment and areas.

2.11 Instrumentation and Control

The EC6 unit control and monitoring systems apply modern distributed control, display, and network communication technologies. Safety system logic and control are based on redundancy to provide fault tolerance protection, and to minimize spurious reactor trips. This results in enhanced monitoring capability and contributes to lower operating and capital costs.

Most control functions are performed by a state-of-the art distributed control system (DCS) that uses small, programmable digital controller modules. The controllers communicate with one another by means of data highways, which use reliable, high-security data transmission methods.

Control Centre

The EC6 plant control centres enable operating staff to monitor, control, and effectively operate the units in both normal and abnormal modes.

A computerized plant display system (PDS) is used for monitoring the plant in both normal and abnormal modes. Integrated computer technology is used throughout the controls, displays, panels and consoles.

The control centre information system includes an advanced alarm annunciation capability based on the CANDU annunciation message list system (CAMLS) implemented on the Qinshan units.

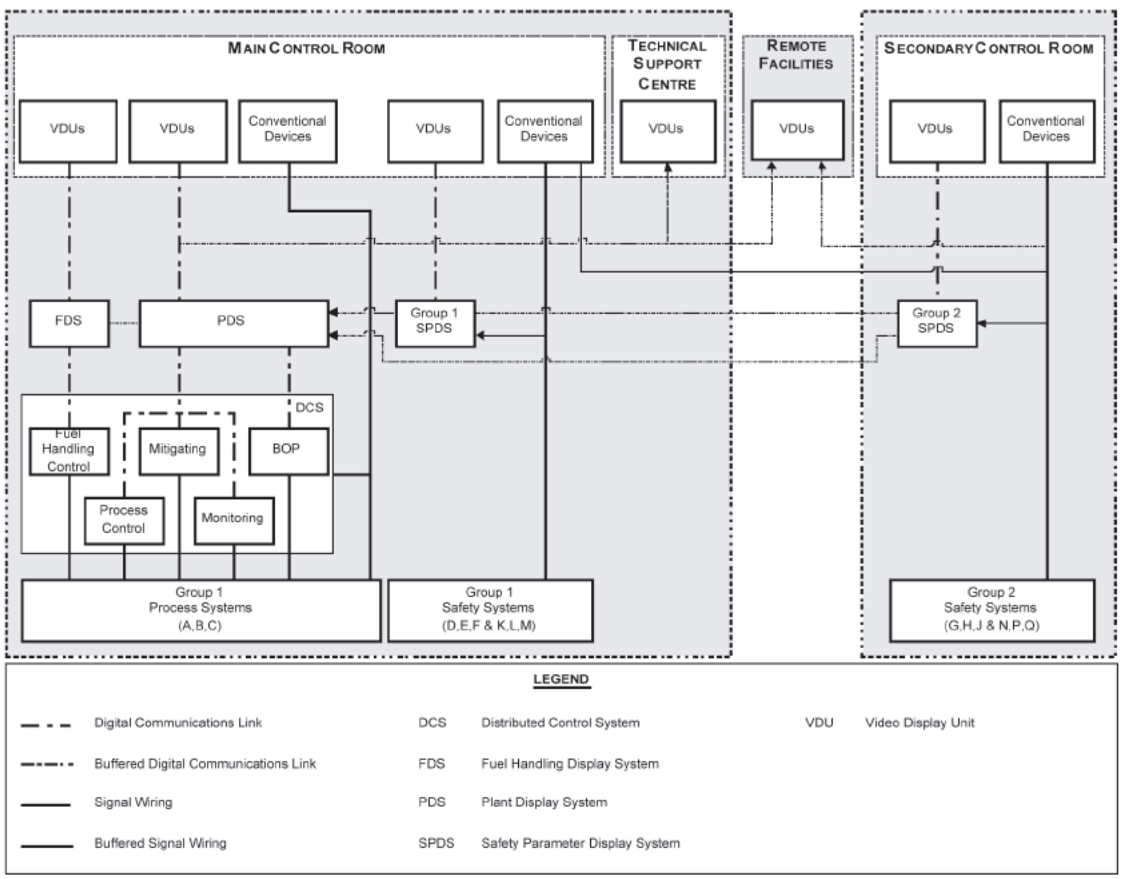


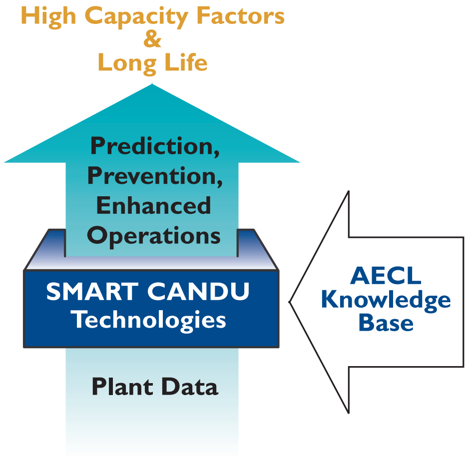
Figure 2-20 Plant Control and Monitoring Systems

Figure 2-21 Main Control Room (Qinshan)

It conveys up-to-date unit information through fault and status displays. The control centre information system also includes an alarm interrogation application that allows operations staff to view fault and status display and to interrogate alarm history from any of the control centre panels.

Each unit has a completely separate secondary control area (SCA) to control and monitor equipment required to shut down the unit, initiate the required fuel cooling, and monitor equipment and plant state to ensure the unit remains in a safe shutdown state should the station's main control room (MCR) become unavailable.

The EC6 will also provide an integrated package of software tools and work processes aimed at plant performance optimization throughout its life cycle. SMART CANDU technologies use the AECL knowledge base and plant data to predict, prevent and enhance operations. The SMART CANDU suite of tools includes ChemAND and other superior engineering tools.



CAMLS
Intelligent Annunciation Message List System that assists operators in coping with events such as blackouts.
ChemAND
Health monitor for plant chemistry. Predicts future performance of components, determines maintenance requirements and optimal operating conditions.
ThermAND
Health monitor for heat transfer systems and components. Ensures optimal margins and maximum power output.
MIMC
Maintenance Information Management Control system that links health monitor data to the plant work management system.

Figure 2-22 SMART CANDU

2.12 Electrical Power System

The electrical power system consists of connections to the off-site grid, the main turbine generator, the associated main output system, the on-site standby diesel generators, the battery power supplies, the uninterruptible power supplies (UPS) and the distribution equipment.

The electrical distribution system (EDS) supplies electrical power to all process and instrumentation and control loads within the unit. The EDS is divided into four classes of power based on availability: Class I is delivered from batteries, Class II from UPS, Class III from standby generators and Class IV from the main generator or grid. Seismically-qualified emergency standby generators are provided for backup power to safety loads that are required.

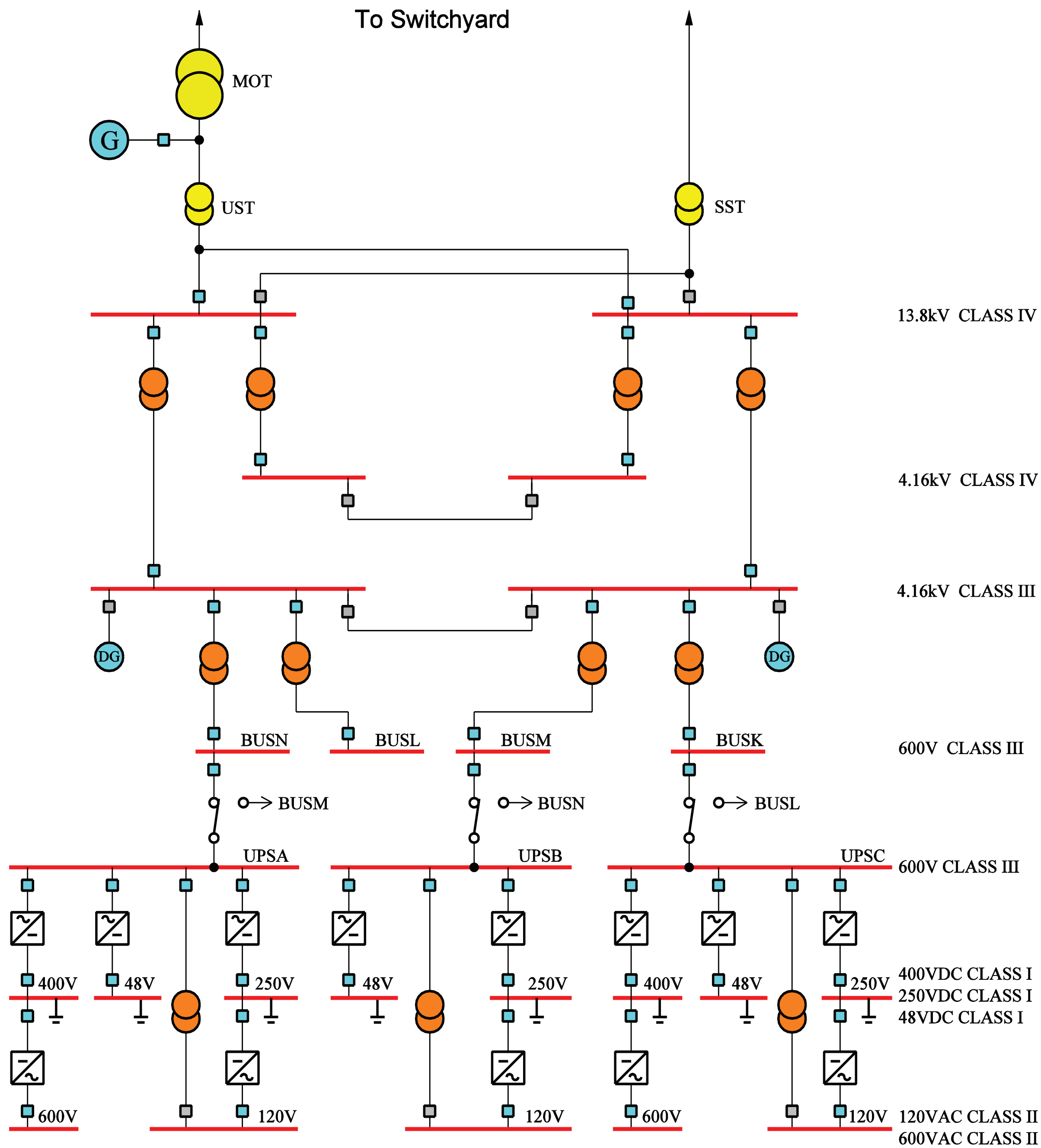


Figure 2-23 Unitized Electrical Power System

3.1 Safety Design

Nuclear safety requires that the radioactive products from the nuclear fission process be contained, both within the plant systems for worker protection and outside the plant structure to protect the public. This is achieved at all times by:

- Controlling the reactor power and, if necessary, shutting the reactor down
- Removing reactor heat, including decay heat following shutdown, in order to prevent heat-up of fuel
- Containing radioactive products that are normally produced and contained within the fuel
- Monitoring the plant to ensure that the above functions are being carried out and, if not, ensuring that mitigating actions are being taken

These nuclear safety functions are carried out to a high degree of reliability by applying the following principles:

- The use of high-quality components and installations
- Maximizing the use of inherent safety features of EC6
- Implementing multiple defense-in-depth barriers for prevention of radioactive release
- Providing enhanced features to mitigate and reduce consequences of design basis events and severe accidents

The implementation of these safety measures is provided by safety systems, safety support systems, safety-related systems and robust buildings and structures that meet high standards for diversity, reliability and protection against common-mode events such as seismic occurrences, tornados, fires, flooding and malevolent acts.

3.2 Defence-in-Depth

The EC6 is based on the CANDU principle of defence-in-depth by providing the following multiple, diverse barriers for accident prevention and mitigation of consequences:

- High-quality process systems to accommodate plant transients and to minimize the likelihood of accidents
- Reliable safety systems for reactor shutdown, emergency core cooling and containment
- Reliable safety support systems to provide services to the safety systems and other mitigating systems
- Back-up systems for heat sinks and essential controls
- Passive heat sinks to increase resistance against both design basis and severe accidents

As a result, the EC6 has at least seven barriers:

1. Fuel sheath which contains the radioactive material
2. Heat transport system, including pressure tubes
3. Calandria tubes
4. Cool, low-pressure moderator
5. Cool, low pressure reactor vault
6. Reserve water system
7. Steel-lined, concrete containment structure

The design of the safety systems follows the design principles of separation, diversity and reliability. High degrees of redundancy within systems are provided to ensure the safety functions can be carried out, even when systems or components are impaired. Protection against seismic events, tornados, flooding and fire is also provided, ensuring highly reliable and effective mitigation of postulated events, including severe accidents.

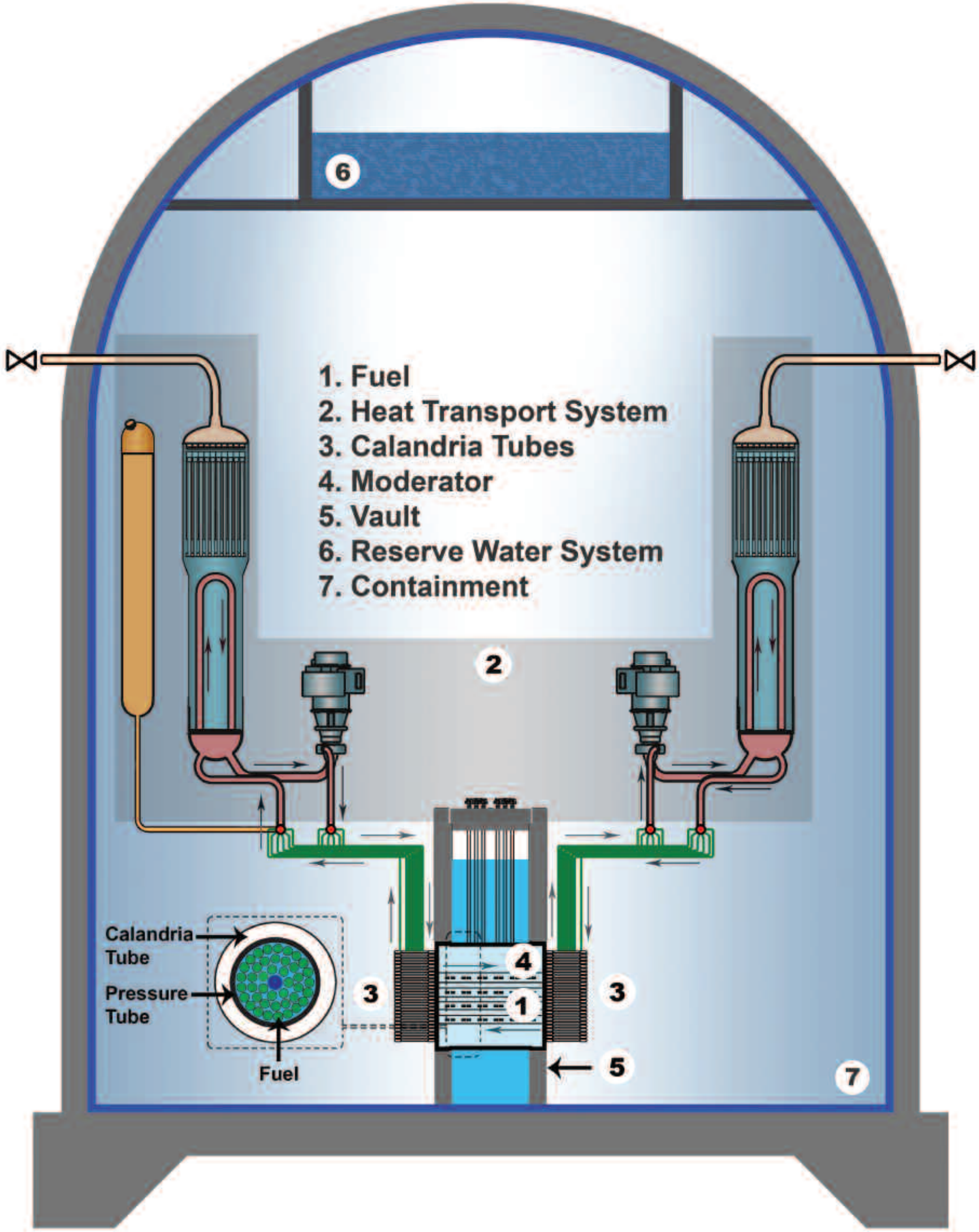


Figure 3-1 Barriers for Prevention of Releases

3.3 Inherent Safety Features

The EC6 maintains the traditional CANDU inherent safety characteristics:

- Low-pressure and low-temperature heavy water moderator, which is very efficient in slowing down neutrons, resulting in a fission process which is more than an order of magnitude slower than LWRs. Reactor control and shutdown are inherently easier to perform
- On-power refuelling reduces the 'excess' reactivity level needed for reactor control. Reactor characteristics are constant and no additional measures such as boron addition to the coolant (and its radioactive removal) are required
- Natural circulation capability in the reactor coolant system can cope with transients due to loss of forced flow
- Reactivity control devices are in the low-pressure moderator and do not penetrate the reactor coolant pressure boundary and therefore cannot be ejected
- Moderator back-up heat sink maintains core coolability for loss-of-coolant accidents even when combined with the unavailability of emergency core cooling (severe accident)

3.4 Severe Accidents

A severe accident is one in which the fuel is not cooled within the heat transport system. The CANDU design principle is to prevent severe accidents and to mitigate severe accident events, in addition to minimizing their consequences. This is achieved by providing a number of design measures:

- Normal heat removal systems
- Heat removal systems, using the Emergency Water System (EWS)
- Passive emergency feedwater make-up from reserve water system
- Emergency coolant injection
- Heat removal using moderator system
- Passive thermal capacity of moderator coolant
- Passive thermal capacity of reactor vault water
- Passive emergency make-up to reactor vault heat sink from reserve water system
- Passive containment cooling via spray
- Severe accident management monitoring capabilities

Severe accident management, as well as providing multiple mechanisms for fuel cooling and barriers to release, also includes mitigating measures within containment. In addition to the strong, concrete perimeter wall and inner steel liner, which by themselves can withstand the largest pipe breaks, containment is provided with:

- Passive, hydrogen recombiners and igniters to limit the hydrogen content to below the deflagration limit
- A spray system to reduce the build-up of containment pressure and reduce leakages

PSA studies estimate that the summed frequency of internal initiating events leading to reactor core damage during at-power operation is only 1.0 x 10-6 for the EC6. This exceeds the US NRC requirements and is comparable to latest LWR designs.

4. EC6 Deployment

Through feedback from past construction projects, AECL has been able to optimize key project elements. The Enhanced CANDU 6 construction schedule is 51 months from first Containment Concrete to fuel loading. The schedule from first Containment Concrete to in-service will be 57 months. The second unit can be in service nine months later. Deployment of the EC6 requires the coordination and timely delivery of key project elements including: licensing programs, environmental assessments, design engineering, and procurement, construction and commissioning start-up programs.

Design Engineering

Preliminary design and development programs are executed in parallel with the environmental assessment and licensing programs, ensuring continuous improvement and plant configuration is maintained. The final design program ensures plant reliability, equipment and component maintainability and constructability requirements are maximized to the fullest extent.

Licensing

The EC6 builds on the successful CANDU track record of accommodating requirements of off-shore jurisdictions in various host countries while retaining the standard nuclear platform. Licensing programs are executed and coordinated with the engineering design programs and environmental assessment, and are structured so as to support regulatory process requirements.

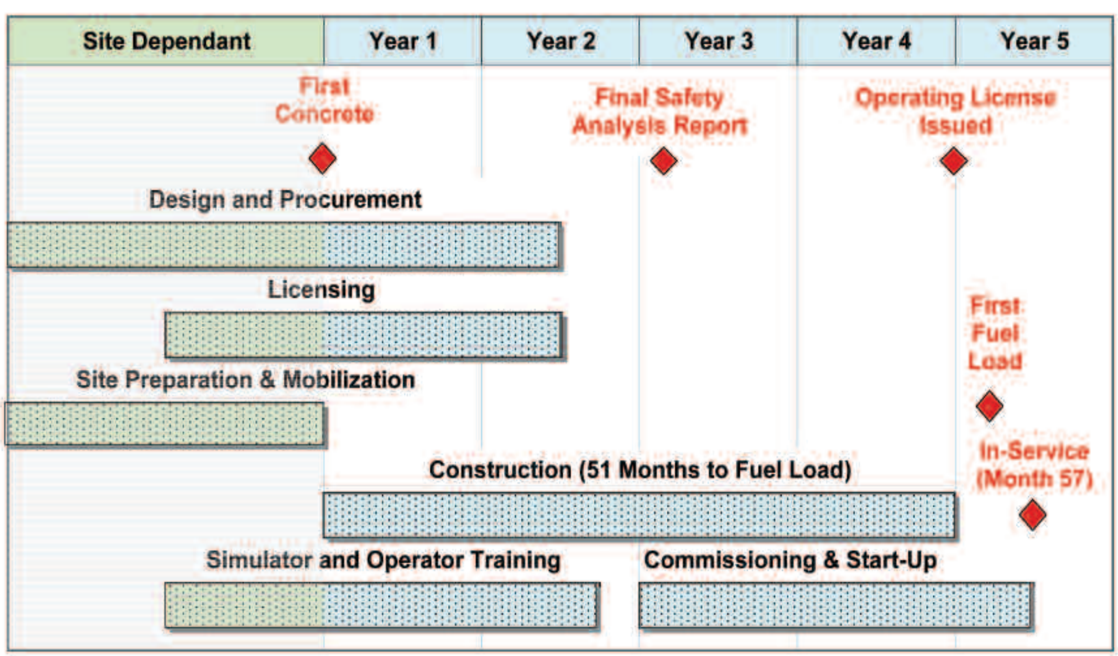


Figure 4-1 51-Month Deployment Schedule (Nominal)

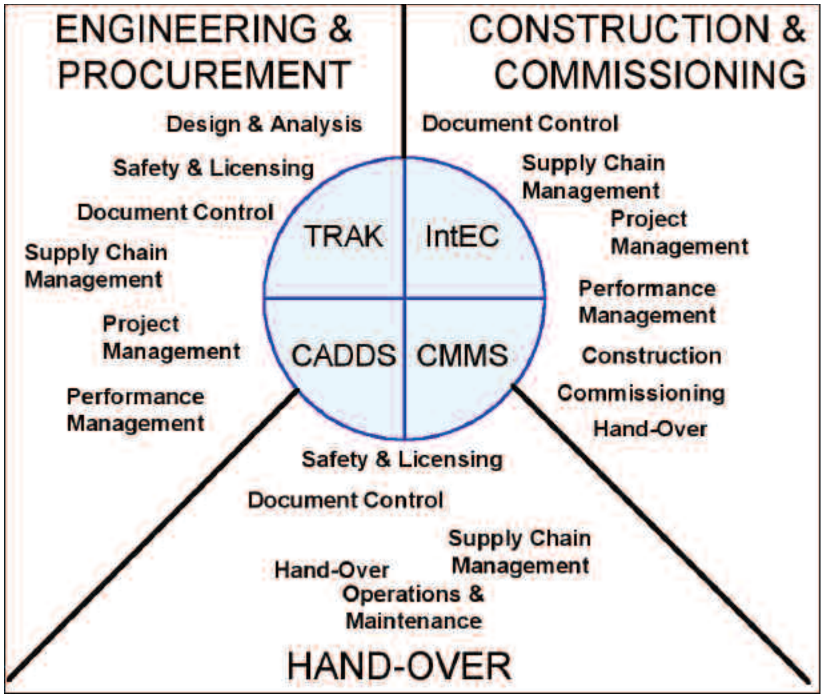


Figure 4-2 Design Engineering Applications

Configuration Management

The EC6 makes use of the latest computer technology for managing the complete plant configuration from design to construction and, finally, turnover to the Owner. State-of-the-art electronic drafting tools are integrated with material management, wiring and device design, and other technology applications.

Project Management

The EC6 project management structure provides fully integrated project management solutions. Performance management programs are executed from project concept, through a project readiness mode, and finally project closeout. The project management framework consists of three key elements: total project execution planning, a critical decision framework to control each phase of the project lifecycle and a comprehensive risk management program.

Procurement

Standardized procurement and supply processes are implemented to support time, cost and performance benefits to the project, including benefits such as efficiency through variety control (standardization), and economy in manufacturing and servicing.

Construction Programs

Constructability programs are implemented to ensure project simplification by:

- Maximizing concurrent construction to increase construction productivity
- Minimizing construction rework to decrease equipment costs
- Minimizing unscheduled activities to reduce capital costs and construction risk

Construction Strategy

The main elements of the Enhanced CANDU 6 construction strategy are:

- Open-top construction method using a very-heavy-lift crane
- Concurrent construction
- Modularization and prefabrication
- Use of advanced technologies to minimize interferences.

Figure 4-3 Module Lift Using VHL Crane

CANDU 6 units entering service in recent years

In-Service Date	Plant	Status
1997	**Wolsong Unit 2, S. Korea**	On budget, on schedule
1998	**Wolsong Unit 3, S. Korea**	On budget, on schedule
1999	**Wolsong Unit 4, S. Korea**	On budget, on schedule
2002	**Qinshan Phase III, Unit 1, China**	Under budget, six weeks ahead of schedule
2003	**Qinshan Phase III, Unit 2, China**	Under budget, 4 months ahead of schedule

CANDU 6: Commissioning

In-Service Date	Plant
2007	**Cernavoda, Unit 2, Romania**

Figure 4-4 World-Class CANDU 6/AECL Project Record

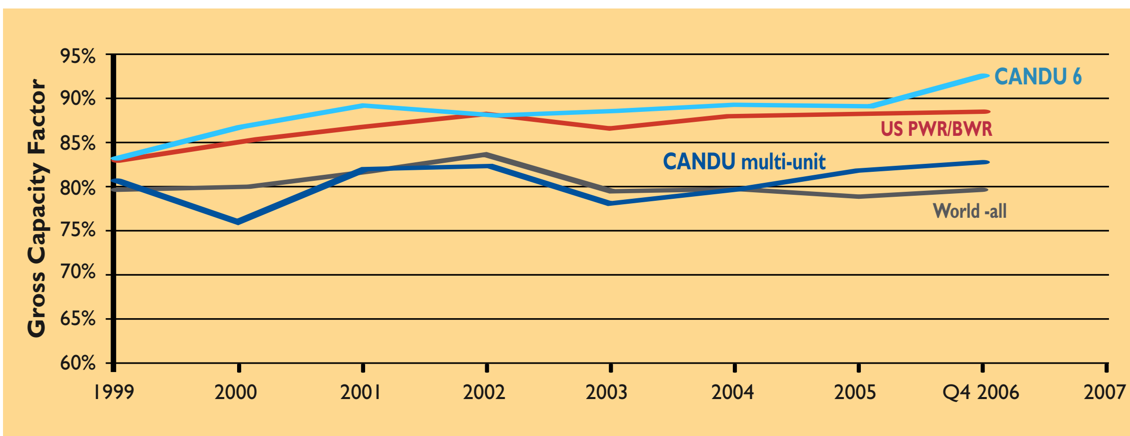
The construction strategy has contributed to the successful completion of CANDU 6 units around the world,—on budget and on or ahead of schedule.

5.1 Plant Performance

The annual capacity factor for Enhanced CANDU 6 is expected to be over 90%. This expectation is based on the proven track record of the CANDU 6 plants, which have collectively surpassed the U.S. PWR/BWR Gross Capacity Factor (GCF) with a combined average of 92.4% in 2006. CANDU 6 plants entering service in the last decade have lifetime capacity factors of 90.2%. These results are consistently better than LWRs around the world.

The EC6 has made a number of improvements to achieve these incremental performance targets.

Reference: CANDU Owners Group Newsletter



COGnizant Volume 12, Issue 6, June 2007. 2006 U.S. and world data based on Q4 results (courtesy of NEI)
The graph is for comparison of trends only.

Figure 5-1 Comparison of Gross Capacity Factors

5.2 Features to Enhance Operating Performance

Incorporation of feedback from operating reactors (both CANDU and other designs) is an integral feature of the design process. This addresses known operations and maintenance improvement opportunities and various features have been incorporated to enhance operating performance throughout the station life. Major enhancements include:

- Use of improved material and plant chemistry specifications, based on operating experience from CANDU plants; For example, life-limiting components such as heat transport system feeders and headers have been enhanced with higher chromium content to limit the effect of feeder corrosion.
- Implementation of advanced computer control and interaction systems for monitoring, display, diagnostics and annunciation.
- Provision of integrated SMART CANDU modules for monitoring plant chemistry of systems and components and providing predictive maintenance capability.
- Ensuring capability for return to full power on restoration of electrical grid. The EC6 has the capability to continue operation of house load without a grid connection, enabling a rapid return to full power upon reconnection.

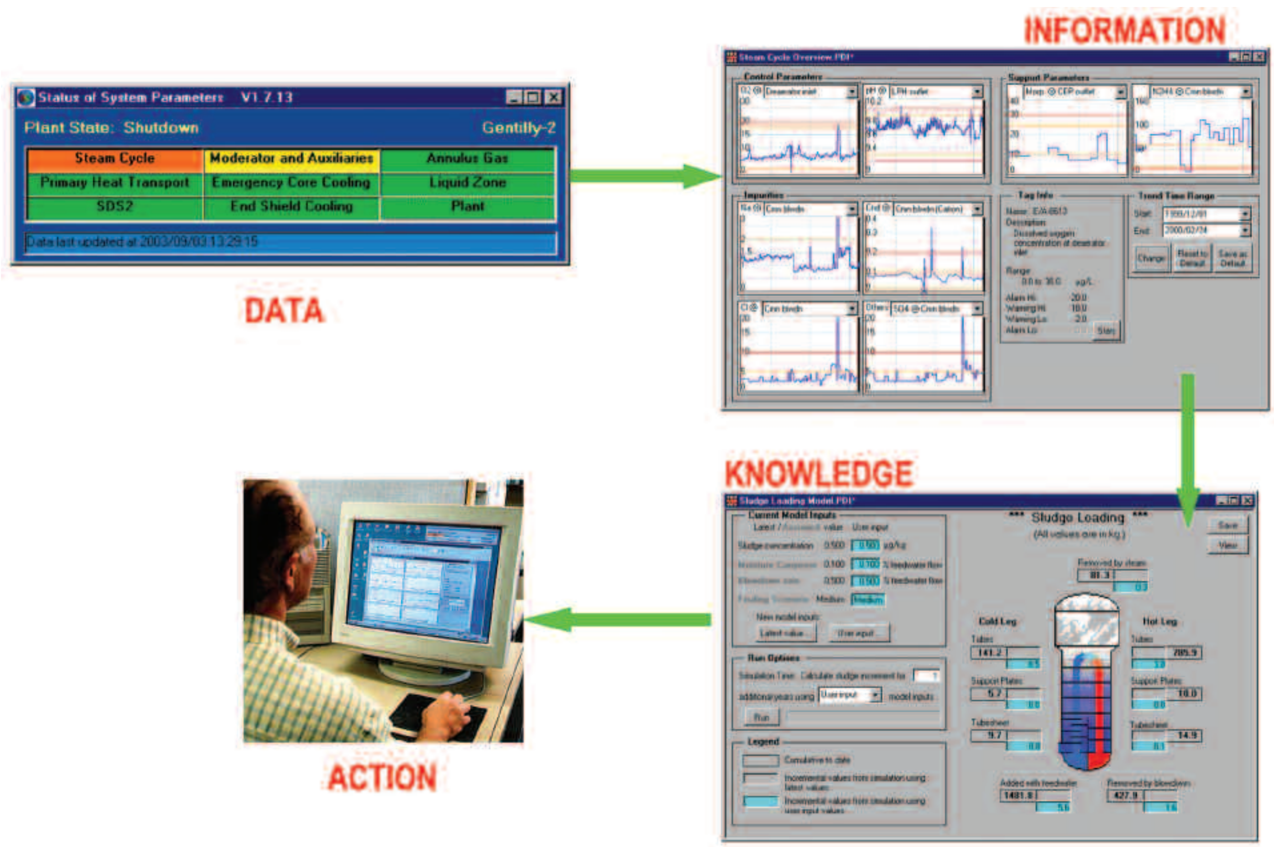


Figure 5-2 ChemAND – Performance Monitor for Plant Chemistry

5.3 Features that Facilitate Maintenance

Plant capacity factors are impacted by the number and duration of maintenance outages. The traditional 'annual' outage of up to one month for currently operating CANDU plants has been improved to a 'major' outage every two years for the Enhanced CANDU 6. To achieve this a number of enhancements have been incorporated into the reactor design:

- A maintenance-based design strategy. This program incorporates lessons learned and ensures maintainability of systems and components. It defines an improved maintenance program based on SMART CANDU technology to identify and take mitigating actions, if required, to ensure plant states are diagnosed and maintained within their design performance limits. This will lead to improved preventive maintenance and reduced forced outages at a rate of less than five days/year
- Improved plant maintenance with provisions for electrical, water and air supplies that are built-in for on-power and normal shutdown maintenance
- Enhanced shielding in radiologically controlled areas, minimizing worker exposure and occupational dose such that the dose to an individual member of the station staff is expected to be less than 50 mSv in any single year
- Improved equipment selection and system design, based on probabilistic safety evaluations using two-year outage intervals

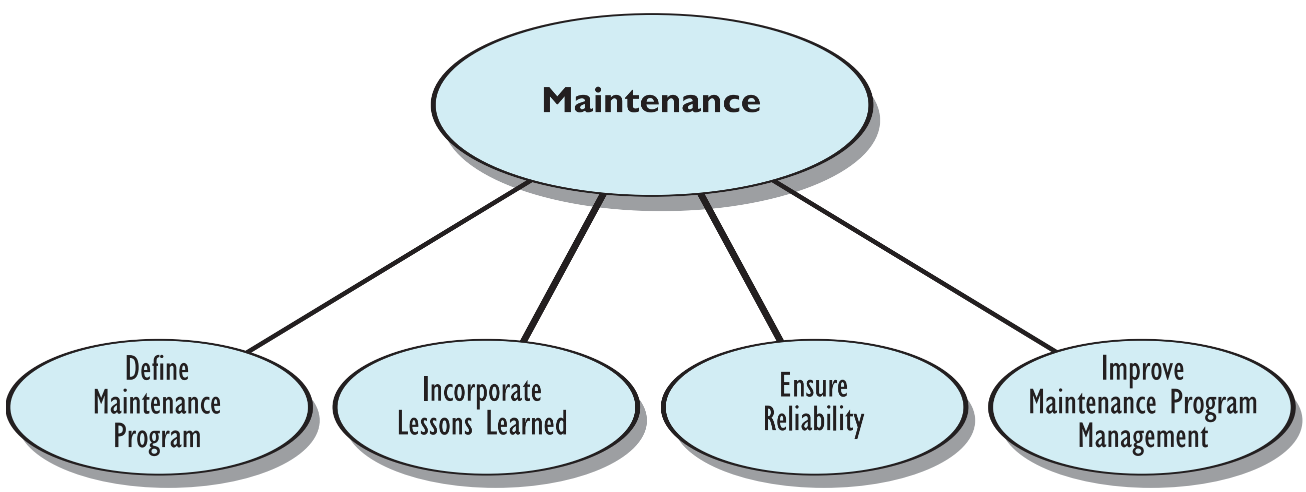


Figure 5-3 Maintenance Basis

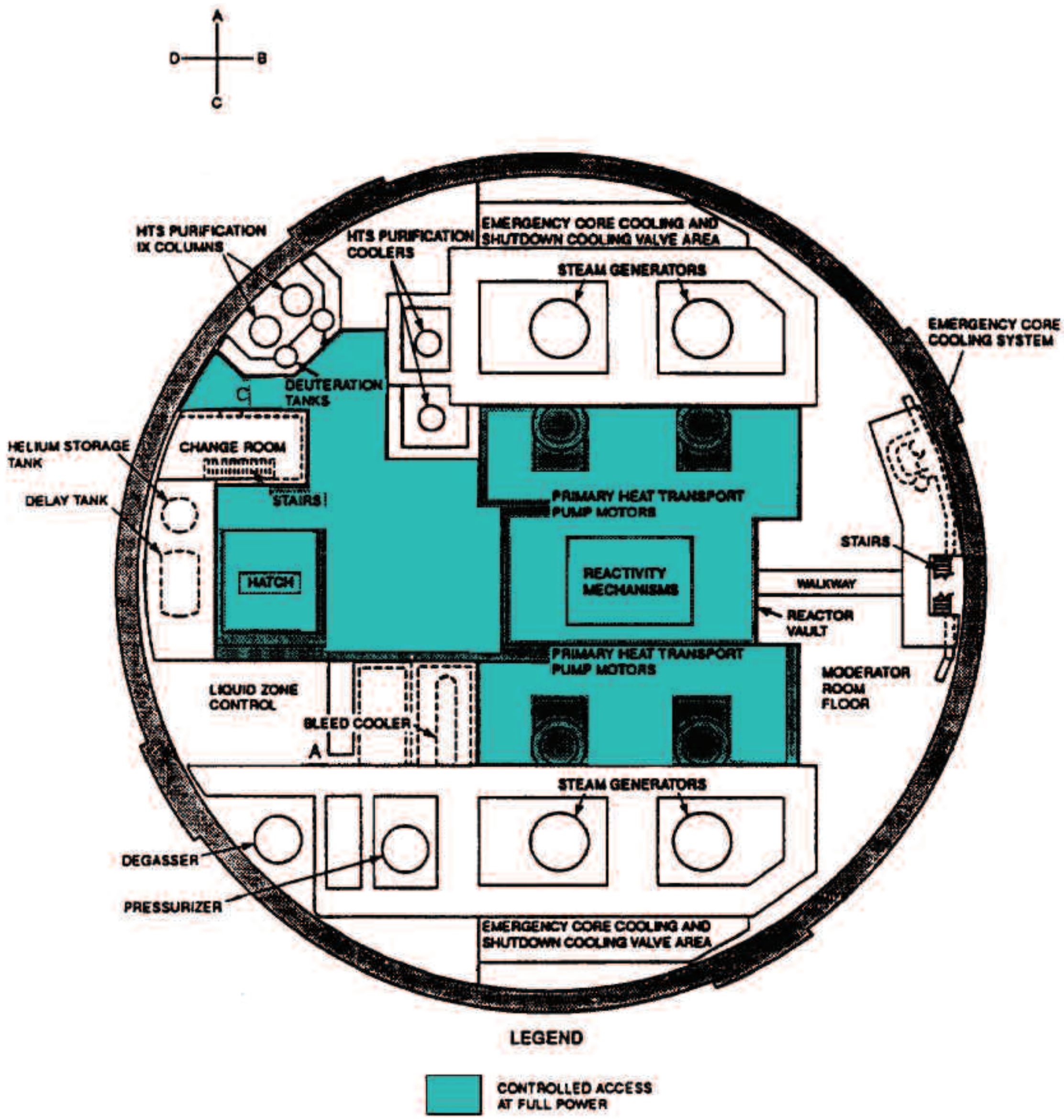


Figure 5-4 Accessible Areas in the Reactor Building – Level 125.0 m

6. Radioactive Waste Management

The waste management systems for the Enhanced CANDU 6 will minimize the radiological exposure to operating staff and the public. Exposures for workers from the plant are monitored and controlled to ensure they are within the limits recommended by the International Commission on Radiological Protection. The systems for the EC6 have been proven over many years at other CANDU sites. They provide for the collection, transfer and storage of all radioactive gases, liquid and solid, including spent fuel and wastes generated within the plant:

- Gaseous radioactive wastes gases, vapours or airborne particulates are monitored and filtered. Radioactive noble gases are treated by the off-gas management system (OGMS). Tritium releases are collected by a vapour recovery system and stored on site.
- Liquid radioactive wastes are stored in concrete tanks located in the service building. Any liquid, including spills, requiring removal of radioactivity is treated using cartridge filters and ion exchange resins.
- Solid radioactive wastes can be classified by five main groups: spent fuel, spent ion-exchange resins, spent filter cartridges, compactible, and non-compactible solids. Each type of waste is processed and moved using specially-designed transporting devices if necessary. After processing, the wastes are collected and prepared for on-site storage by the utility or for transport offsite.

AECL has developed MACSTOR®* (Modular Air-Cooled Storage) system for safe, above-ground storage of spent fuel. MACSTOR has been developed from more than 30 years of experience.

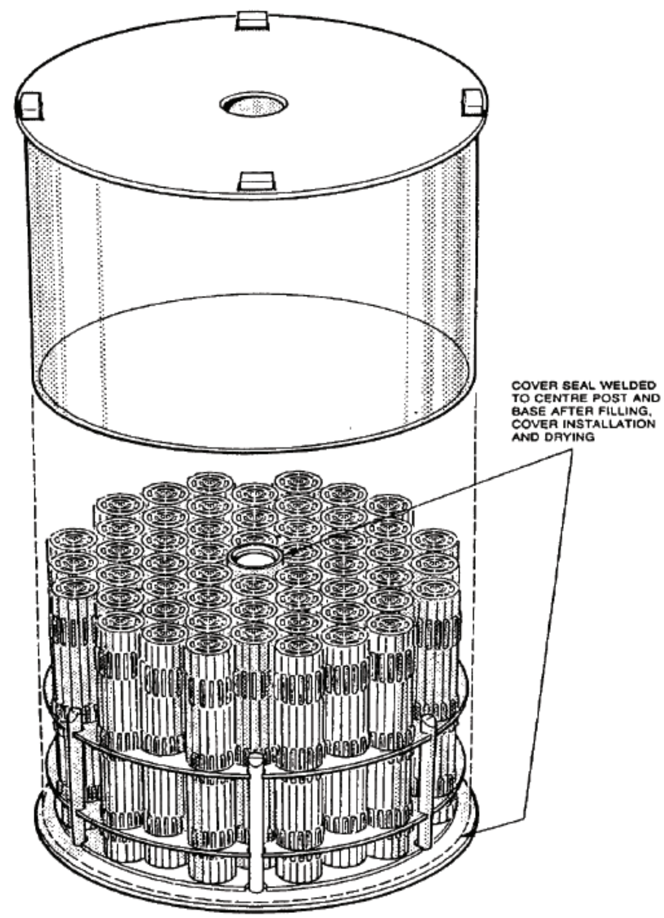


Figure 6-1 Spent Fuel Storage

MACSTOR® is a registered trademark of Atomic Energy of Canada Limited (AECL).

MACSTOR has highly efficient heat-rejection and shielding capabilities. It is constructed using multiple barriers to provide adequate radiation shielding for operators and the public, while being appropriately qualified and equipped with monitoring facilities. MACSTOR saves up to one-third of the space required for comparable systems, requires less manpower, has low operating and construction costs, and permits easy fuel retrieval.

Figure 6-2 MACSTOR Fuel Transfer

Figure 6-3 AECL's MACSTOR System

7. Conclusion

Evolution

Capitalizing on the proven features of CANDU technology, AECL has designed the EC6 to be cost-competitive with all forms of energy, including nuclear, while achieving high safety and performance standards consistent with customer expectations.

Proven CANDU Features

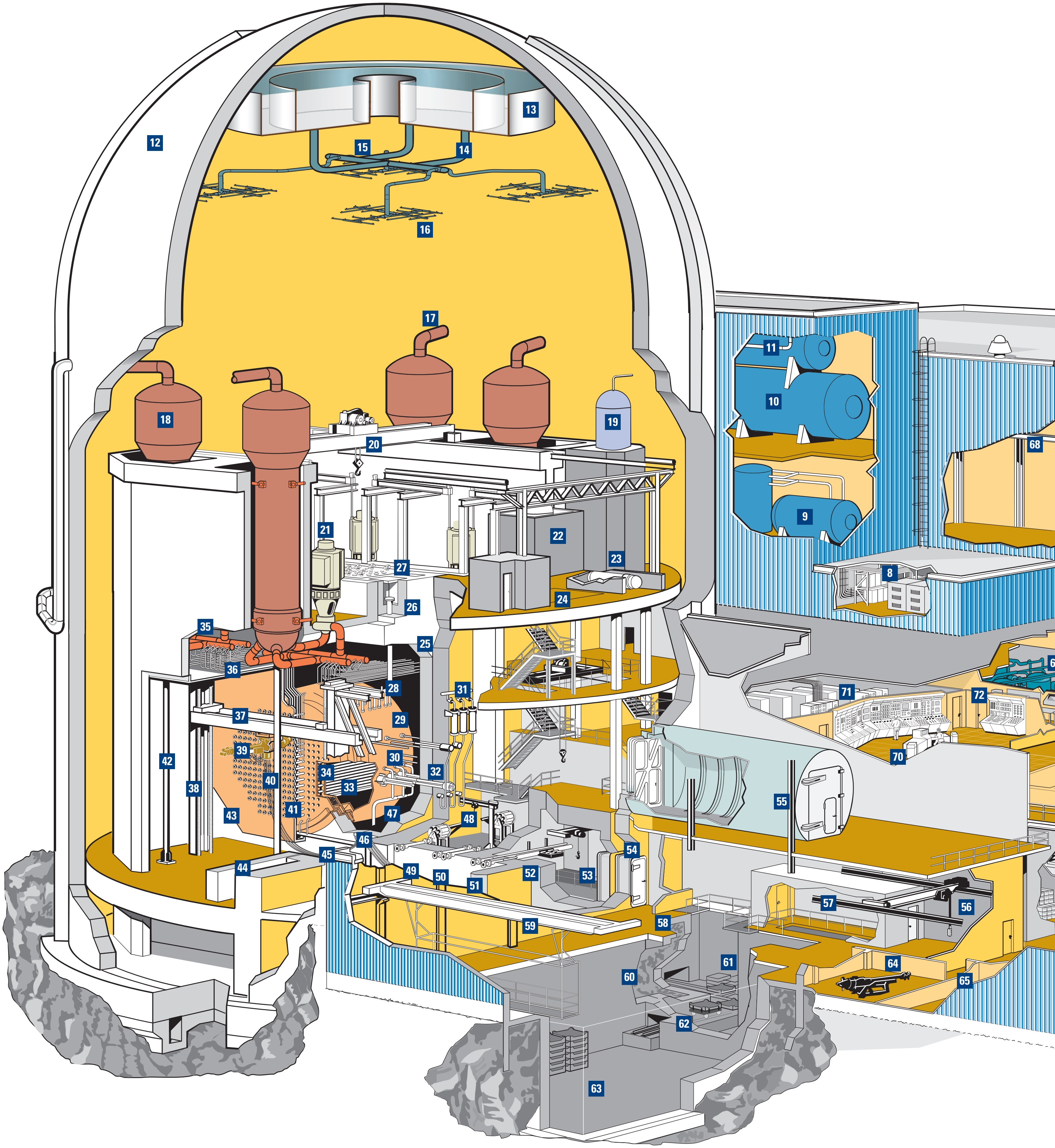
- Heavy water moderator and horizontal fuel channel design
- Series of parallel pressure tubes—rather than single pressure vessel allowing simpler manufacturing and reduced cost
- Two independent, passive, fast-acting safety shutdown systems and a unique inherent emergency-cooling capability
- On-power fuelling for flexible outage planning and minimal 'excess' reactivity burden
- Multiple heat removal systems to prevent and mitigate severe accidents

EC6 Enhancements

- Advanced construction techniques
- Enhanced safety design including addition of reserve water system for passive accident mitigation
- Steel liner and thicker containment
- Improved design for maintainability and operability

The EC6 will meet customer expectations for safe, reliable and economically competitive power production. It benefits from AECL's wealth of experience, technical excellence and innovations in engineering.

ENHANCED CANDU 6 NUCLEAR POWER PLANT



Key to Diagram

1. Diesel room
2. Water treatment plant *
3. Crane hall
4. Turbine building
5. Turbine building crane
6. Generator
7. Condenser
8. Battery room
9. Boiler feed water tanks
10. Deaerator storage tank
11. Deaerator
12. Reactor building
13. Reserve Water tank
14. Reserve Water supply pipes
15. Reserve Water valves
16. Low Flow Containment spray nozzles
17. Steam pipes
18. Steam generators
19. Pressurizer
20. Crane
21. Heat transport pumps
22. Bleed condenser
23. Bleed cooler
24. Hatch
25. Reactor vault
26. Pressure relief pipes
27. Reactivity mechanism deck
28. Reactivity mechanism guide tubes
29. Calandria
30. Poison injection nozzles
31. Poison tanks
32. Ion chambers
33. Fuel channel assemblies
34. End shield
35. Headers
36. Feeder pipes
37. Fuelling machine bridge
38. Bridge support column
39. Fuelling machine
40. Catenary
41. Fuel channel end fittings
42. Steam generator support column
43. Feeder pipe insulation cabinet
44. Fuelling machine vault door
45. End shield cooling
46. Fuelling machine track
47. Moderator inlet pipe
48. New fuel handling machine
49. New fuel port
50. Fuelling machine service ports
51. Rehearsal facility
52. Spent fuel port
53. Spent fuel elevator
54. Entrance to spent fuel area
55. Airlock
56. Crane
57. Spent fuel shipping area
58. Spent fuel handling area
59. Spent fuel bay gantry
60. Spent fuel bay
61. Spent fuel transfer baskets
62. Spent fuel transfer trolley
63. Spent fuel storage baskets
64. Fuelling machine maintenance area
65. Decontamination room
66. New fuel storage
67. Tool crib
68. Vapour recovery equipment
69. Office
70. Control room *
71. Control equipment room
72. Computer room

* Some items have been moved for clarity

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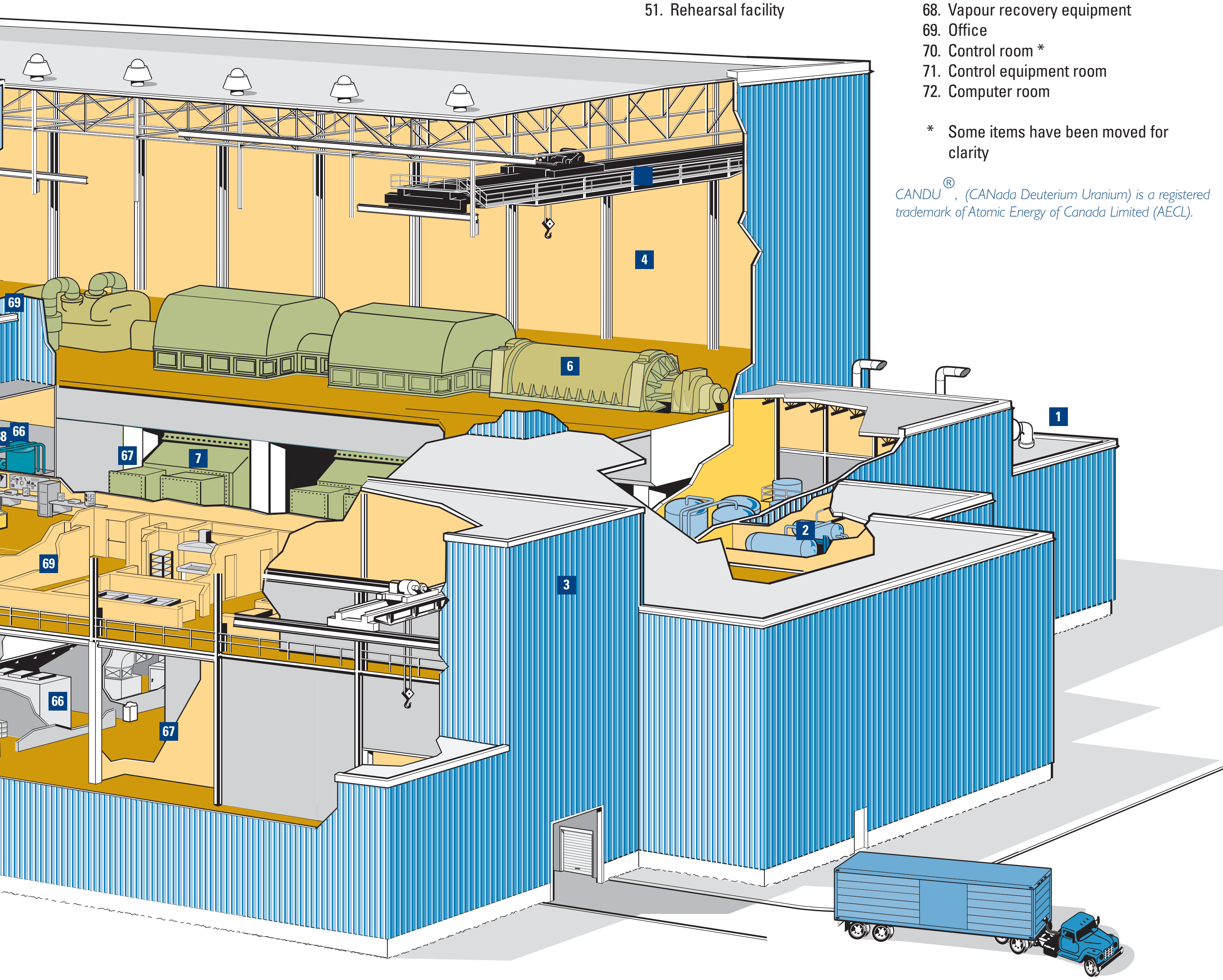

Figure 6-4: Enhanced CANDU 6 Nuclear Power Plant

AECL

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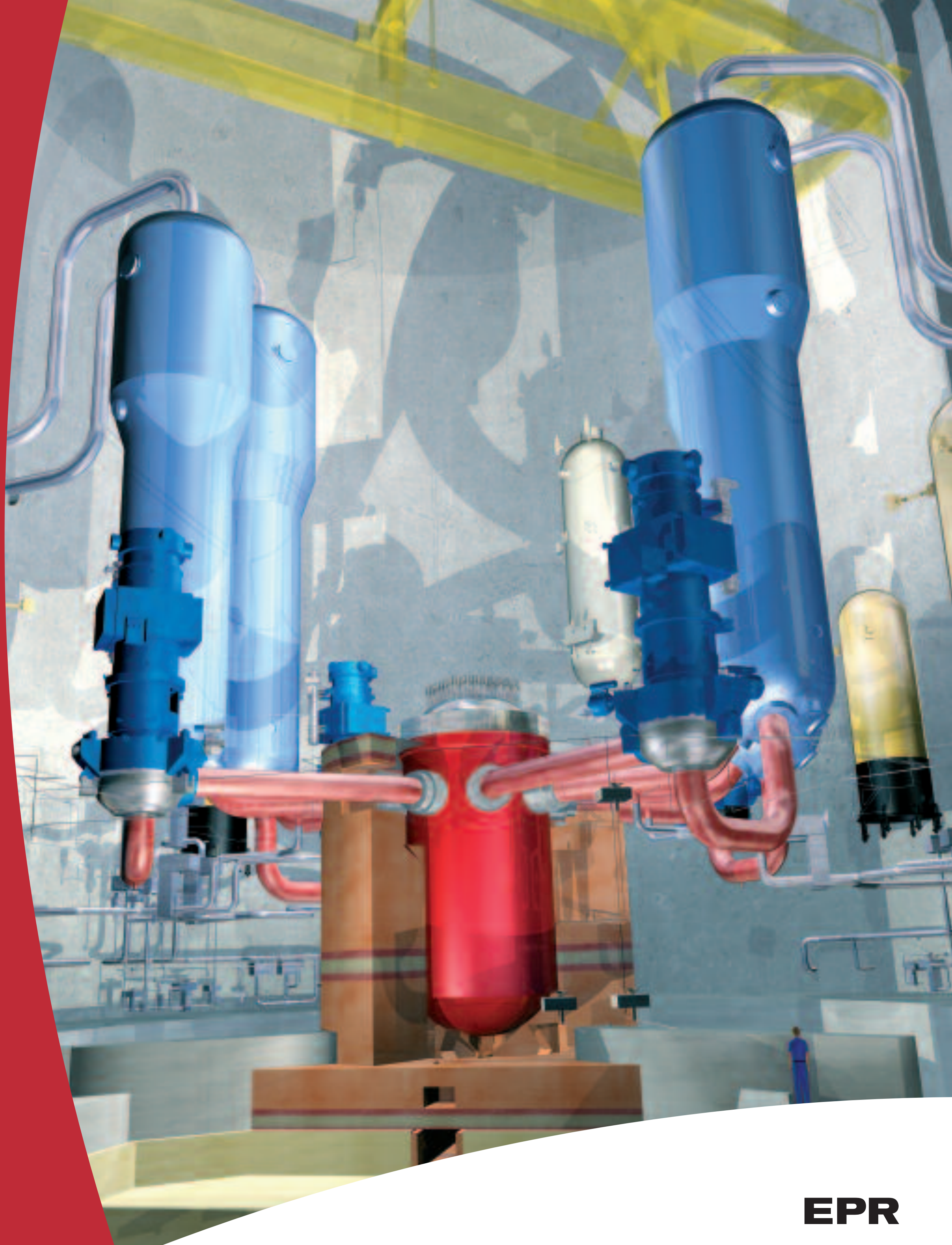
Appendix F: EPR Technical Summary

An overview and technical summary of the EPR reactor design is provided in the following pages.



EPR

English version

AREVA

> Readers accustomed to British units can use the following table to convert the main units from the International Metric System.

1 meter (m)	=	3.2808 feet
	=	39.370 inches
1 square meter (m^2)	=	10.764 square feet
1 cubic meter (m^3)	=	264.17 US gallons
1 kilogram (kg)	=	2.2046 pounds
1 tonne (t)	=	1.1023 short ton
1 bar	=	14.5 psi

> Conversion of temperature (°C into °F)

Temp. °C x 9/5 + 32	=	Temp. °F

> All pressures are expressed in absolute bar.

> FOREWORD

Security of energy supply and energy cost stability in the long term, plus the efforts to combat the greenhouse effect and potential global warming, argue in favor of a greater diversity in sources of energy supplies. Against this background nuclear power, which is more and more economically competitive, safe, reliable and environment friendly, has a vital role to play.

A world expert in energy, AREVA creates and offers solutions to generate, transmit and distribute electricity; its businesses cover on a long-term basis every sector in the use of nuclear power to support electricity needs: front end (Uranium ore mining and conversion, Uranium enrichment, fuel fabrication), reactor design and construction, reactor services, back end of the fuel cycle, transmission and distribution from the generator to the large end-users.

The EPR is a large advanced evolutionary reactor of the Pressurized Water Reactor (PWR) type offered by AREVA NP to satisfy electricity companies' needs for a new generation of nuclear power plants even more competitive and safer while contributing to sustainable development.

> **FOREWORD**

The EPR's key assets to support a strategic choice

An evolutionary, safe and innovative design

The EPR is a 1,600 MWe class PWR. Its evolutionary design is based on experience from several thousand reactor - years of operation of Light Water Reactors worldwide, primarily those incorporating the most recent technologies: the N4 and KONVOI reactors currently in operation in France and Germany respectively. The EPR design integrates the results of decades of research and development programs, in particular those carried out by the CEA (French Atomic Energy Commission) and the German Karlsruhe research center. Through its N4 and KONVOI filiation, the EPR totally benefits from the uninterrupted evolutionary and innovation process which has continuously supported the development of the PWR since its introduction in the Western marketplace in the mid-fifties.

Offering a significantly enhanced level of safety, the EPR features major innovations, especially in further preventing core meltdown and mitigating its potential consequences. The EPR design also benefits from outstanding resistance to external hazards, including military or large commercial airplane crash and earthquake. Together, the EPR operating and safety systems provide progressive responses commensurate with any abnormal occurrences.

Thanks to a number of technological advances, the EPR is at the forefront of nuclear power plants design. Significant progress has been incorporated into its main features:

- the reactor core and its flexibility in terms of fuel management,
- the reactor protection system,
- the instrumentation and control (I & C) system, the operator friendly man-machine interface and fully computerized control room of the plant,
- the large components such as the reactor pressure vessel and its internal structures, steam generators and primary coolant pumps.

These innovations contribute to the high level of performance, efficiency, operability and therefore economic competitiveness offered by the EPR to fully satisfy customers' expectations for their future nuclear power plants.

The straightforward answer to utilities' and safety authorities' requirements for new nuclear power plants

The French-German cooperation set up to develop the EPR brought together, from the start of the project:

- power plant vendors, Framatome and Siemens KWU (whose nuclear activities have since been merged to form Framatome ANP, now AREVA NP),
- EDF (Electricité de France), and the major German utilities now merged to become E.ON, EnBW and RWE Power,
- the safety authorities from both countries to harmonize safety regulations.

The EPR design takes into account the expectations of utilities as stated by the "European Utility Requirements" (EUR) and the "Utility Requirements Document" (URD) issued by the US Electric Power Research Institute (EPRI). It complies with the recommendations (1993) and positions on major issues (1995) that the French and German safety authorities jointly set up. The technical guidelines covering the EPR design were validated in October 2000 by the French standing group of experts in charge of reactor safety

> Building on Experience

Enhanced safety level and competitiveness

N4

Solid basis of experience
with outstanding performance

KONVOI

(*"Groupe Permanent Réacteurs"* which is the advisory committee for reactor safety to the French safety authority) supported by German experts.

On September 28, 2004, the French safety authority, on behalf of the French government, officially stated that the EPR safety options comply with the safety enhancement objectives established for new nuclear reactors.

On May 4, 2006, the Board of Directors of EDF decided to launch the building of its first EPR unit on the Flamanville site.

Continuity in technology

The N4 and KONVOI reactors are children of the earlier Framatome and Siemens KWU generation reactors which are themselves derivative of standard US type PWRs, first implemented in the US, then refined and expanded upon by Framatome and Siemens KWU. The EPR is the direct descendant of the well proven N4 and KONVOI reactors, guaranteeing a fully mastered technology. As a result, risks linked to design, licensing, construction and operation of the EPR are minimized, providing a unique certainty to EPR customers.

AREVA NP's Chalon/Saint-Marcel and Jeumont plants have gathered over 30 years of experience in the manufacturing of nuclear heavy components and are keeping it alive. This is why they have the know-how it takes to optimize the design and manufacturing of nuclear heavy components. The construction of the EPR stands to benefit from their unique capacity and expertise.

Operator expertise acquired through the operation of nuclear power plants using the same technology as the EPR is maintained and its value is increased.

Another major advantage is that the existing industrial capacities for design, engineering, equipment manufacturing, nuclear power plant construction and maintenance – including capacities resulting from previous technology transfers – can be easily deployed and utilized to carry out new nuclear plant projects based on EPR technology.

- **The EPR relies on a sound and proven technology.**
- **It complies with safety authorities requirements for new nuclear plants.**
- **Continuous in-house design and manufacturing cooperation for a better optimization.**
- **Design and licensing, construction and commissioning, operability and maintainability of EPR units benefit from AREVA NP long lasting and worldwide experience and expertise. Therefore, EPR customers uniquely minimize their technical risks and associated financial impacts.**

Enhanced economic competitiveness

The next generation of nuclear power plants will have to be even more competitive to successfully cope with deregulated electricity markets.

Thanks to an early focus on economic competitiveness during its design process, the EPR offers significantly reduced power generation costs. They are estimated to be 10% lower than those of the most modern nuclear units currently in operation. According to the most recent international study, OECD NEA/IEA (2005) *Projected Costs of Generating Electricity, 2005 Update*, in which several countries in Europe chose the EPR as the reference model for their future nuclear programs, the average cost of electricity generated by an EPR would be significantly less than that generated using combined cycle gas turbine (CCGT) technology; the cost savings amount to around 20% for a gas price of between 4 and 6 $/Mbtu and a weighted average capital cost (WACC) of 8 to 9% in real terms.

This high level of competitiveness is achieved through:

- **a unit power in the 1,600 MWe range (the highest unit power to date), providing an attractive cost of the installed kWe,**
- **a 36-37% overall efficiency depending on site conditions (presently the highest value ever for water reactors),**
- **a shortened construction time relying on experience feedback and continuous improvement of construction methodology and tasks sequencing,**
- **a design for a 60-year service life,**
- **an enhanced and more flexible fuel utilization,**
- **an availability factor up to 92%, on average, during the entire service life of the plant, obtained through long irradiation cycles, shorter refueling outages and in-operation maintenance.**

Significant advances for sustainable development

The EPR, due to its optimized core design and higher overall efficiency compared to the reactors in operation today, also offers many significant advantages in favor of sustainable development, typically:

- **17% saving on Uranium consumption per produced MWh,**
- **15% reduction on long-lived actinides generation per MWh,**
- **14% gain on the "electricity generation" versus "thermal release" ratio (compared to 1,000 MWe-class reactors),**
- **great flexibility to use MOX (mixed UO_2-PuO_2) fuel.**

> INTRODUCTION

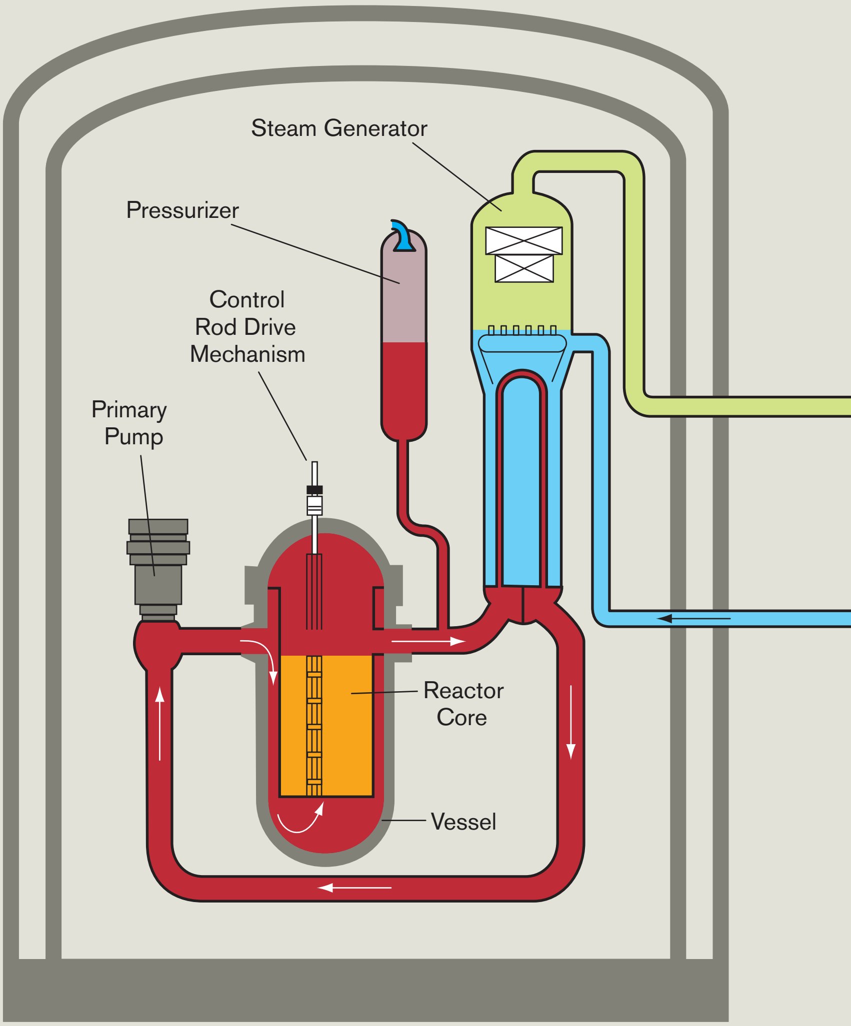
In a nuclear power plant, the reactor is the part of the facility in which the heat, necessary to produce steam, is generated by fission of atom nuclei.
The produced steam drives a turbine generator, which generates electricity.
The nuclear steam supply system is therefore the counterpart of coal, gas or oil-fired boilers of fossil-fuelled plants.

In a Pressurized Water Reactor (PWR) like the EPR, ordinary water is utilized to remove the heat formed inside the reactor core by the nuclear fission phenomenon. This water also slows down (or moderates) neutrons (constituents of atom nuclei that are released in the nuclear fission process). Slowing down neutrons is necessary to keep the chain reaction going (neutrons have to be moderated to be able to break down the fissile atom nuclei).

The heat produced inside the reactor core is transferred to the turbine through the steam generators. From the reactor core coolant circuit (primary circuit) to the steam circuit used to feed the turbine (secondary circuit), only heat is transferred and there is no water exchange.

The primary water is pumped through the reactor core and the primary side of the steam generators, in four parallel closed loops, by electric motor-powered coolant pumps. Each loop is equipped with a steam generator and a coolant pump.

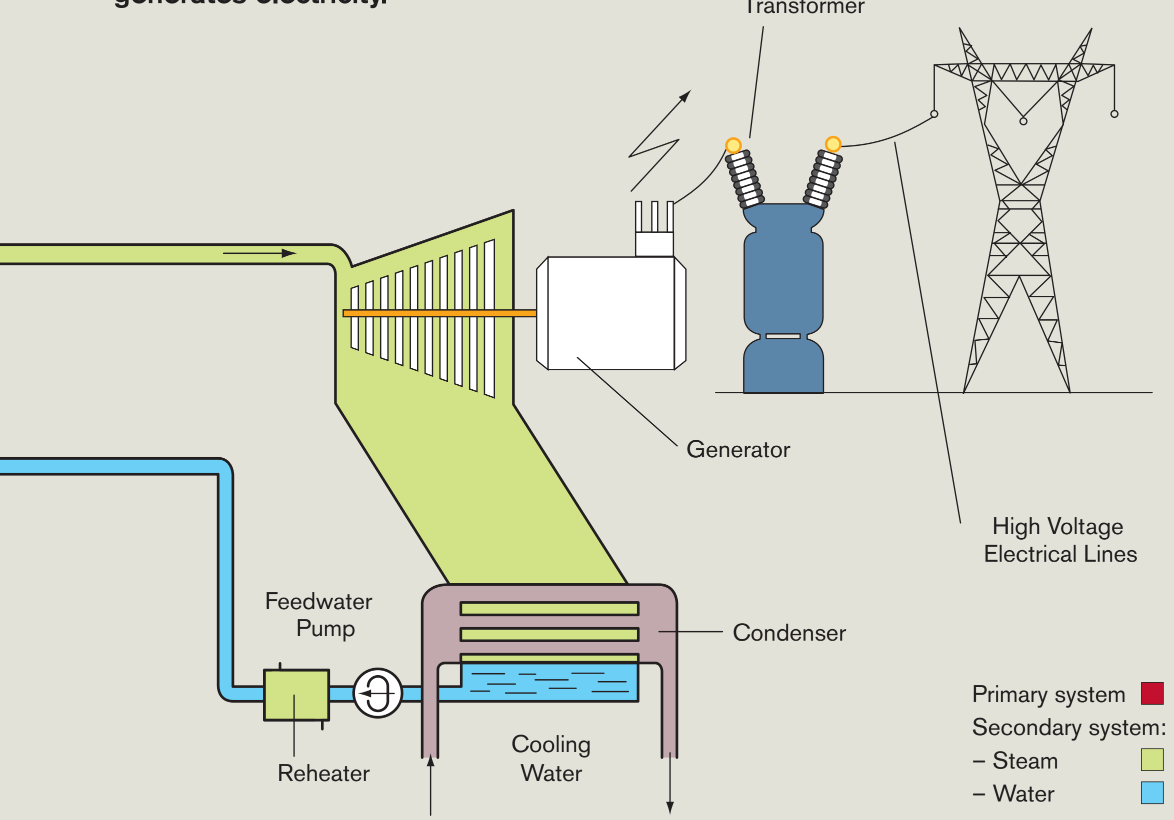
The reactor operating pressure and temperature are such that the cooling water does not evaporate and remains in the liquid state, which intensifies its cooling efficiency. A pressurizer controls the pressure; it is connected to one of the loops.



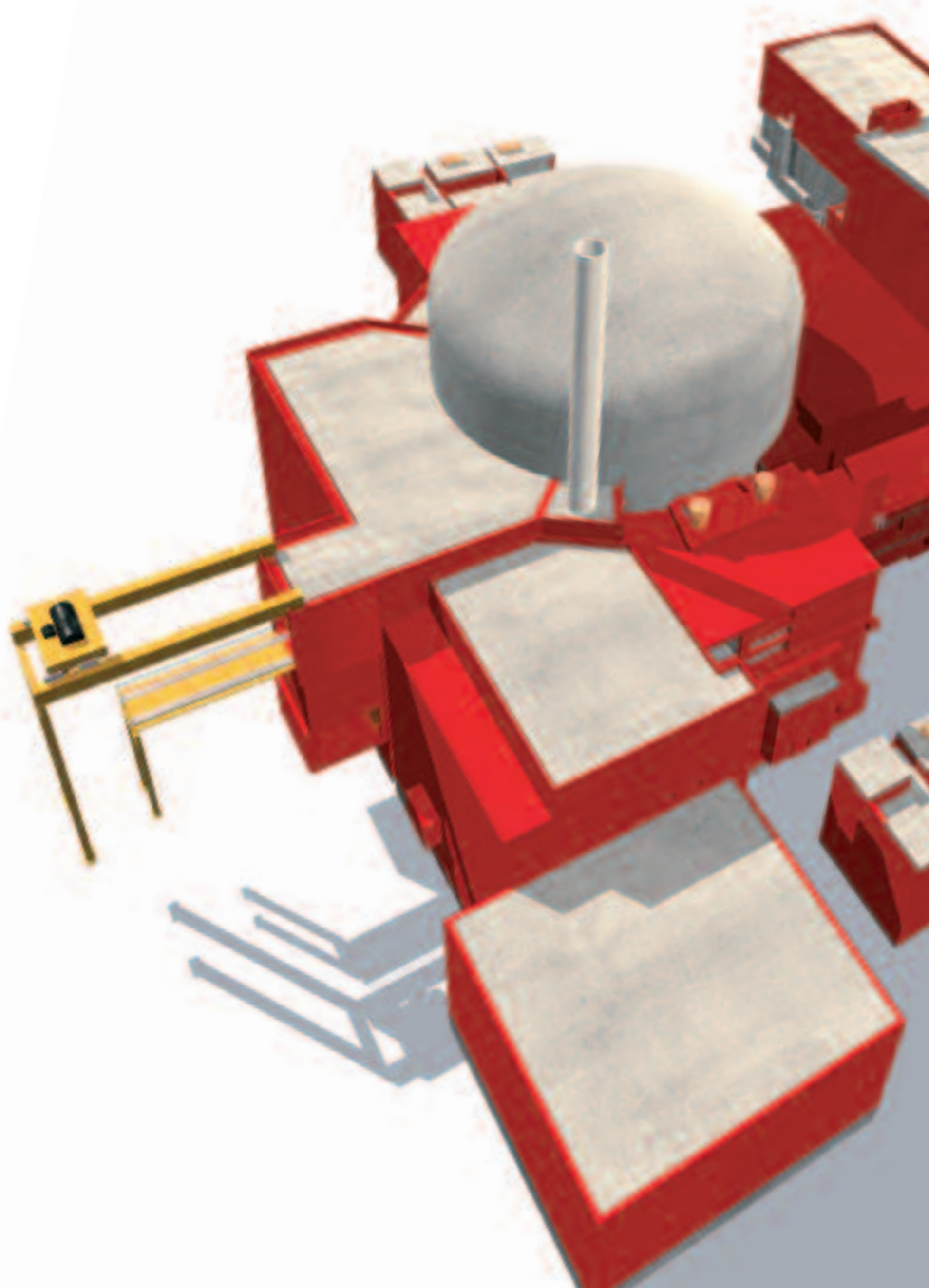
The feedwater entering the secondary side of the steam generators absorbs the heat transferred from the primary side and evaporates to produce saturated steam. The steam is dried in the steam generators then routed to the turbine to drive it. Then, the steam is condensed and it returns as feedwater to the steam generators.

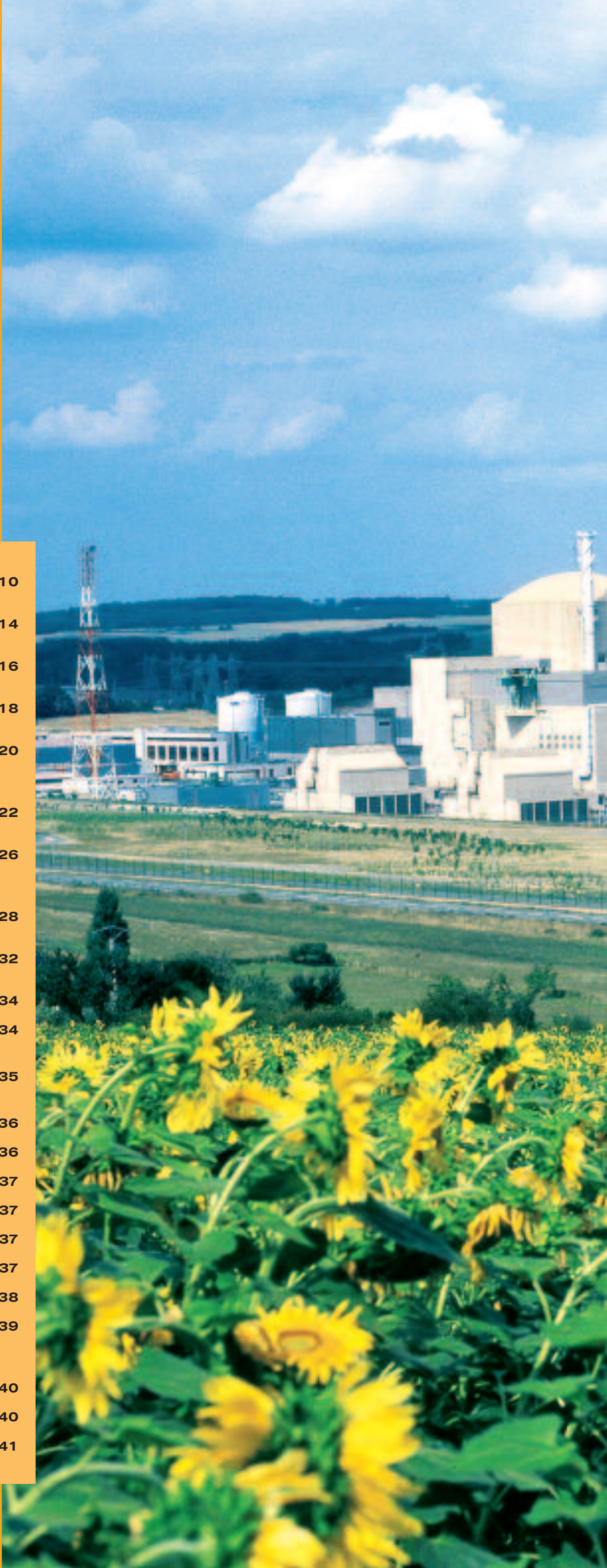
The generator, driven by the turbine, generates electricity.

➡ The following chapters will provide detailed explanation about the description and operation of PWR nuclear power stations based on the EPR reactor.



> TABLE OF CONTENTS





EPR NUCLEAR ISLAND

Civaux nuclear power plant, France
(N4, 1,500 MWe)

EPR LAYOUT

1 Reactor Building

The Reactor Building located in the center of the Nuclear Island houses the main equipment of the Nuclear Steam Supply System (NSSS) and the In-Containment Refueling Water Storage Tank (IRWST). Its main function is to ensure protection of the environment against internal and external hazards consequences under all circumstances. It consists of a cylindrical pre-stressed inner containment with a metallic liner surrounded by an outer reinforced concrete shell.

The main steam and feedwater valves are housed in dedicated reinforced concrete compartments adjacent to the Reactor Building.

The primary system arrangement is characterized by:
- pressurizer located in a separate area,
- concrete walls between the loops and between the hot and cold legs of each loop,
- concrete wall (secondary shield wall) around the primary system to protect the containment from missiles and to reduce the spread of radiation from the primary system to the surrounding areas.

2 Fuel Building

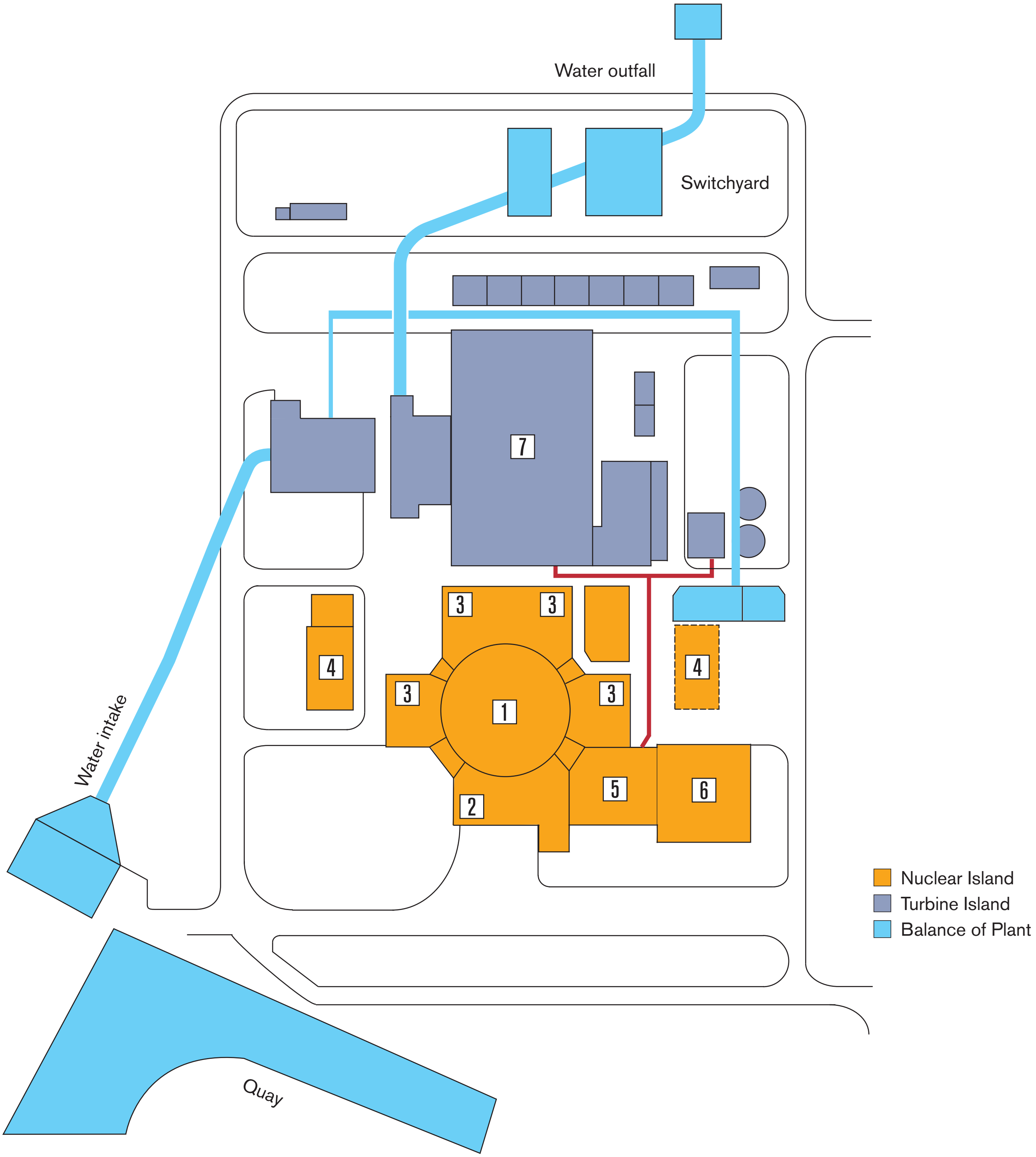
The Fuel Building, located on the same common basemat as the Reactor Building and the Safeguard Buildings, houses the fresh fuel, the spent fuel in an interim fuel storage pool and associated handling equipment. Operating compartments and passageways, equipment compartments, valve compartments and the connecting pipe ducts are separated within the building. Areas of high activity are separated from areas of low activity by means of shielding facilities. The mechanical floor houses the fuel pool cooling system, the emergency boration system, and the chemical and volume control system. The redundant trains of these systems are physically separated by a wall into two building parts.

3 The Safeguard Buildings

The four Safeguard Buildings house the safeguard systems such as the Safety Injection System and the Emergency Feedwater System, and their support systems. The four different trains of these safeguard systems are housed in four separate divisions, each located in one of the four Safeguard Buildings.
The Low Head Safety Injection System is combined with the Residual Heat Removal System. They are arranged at the inner areas in the radiologically controlled areas, whereas the corresponding Component Cooling and Emergency Feedwater Systems are installed at the outer areas in the classified non-controlled areas. The Main Control Room is located in one of the Safeguard Buildings.

4 Diesel Buildings

The two Diesel Buildings shelter the four emergency Diesel generators and their support systems, and supply electricity to the safeguard trains in the event of a complete loss of electrical power. The physical separation of these two buildings provides additional protection.



5 Nuclear Auxiliary Building

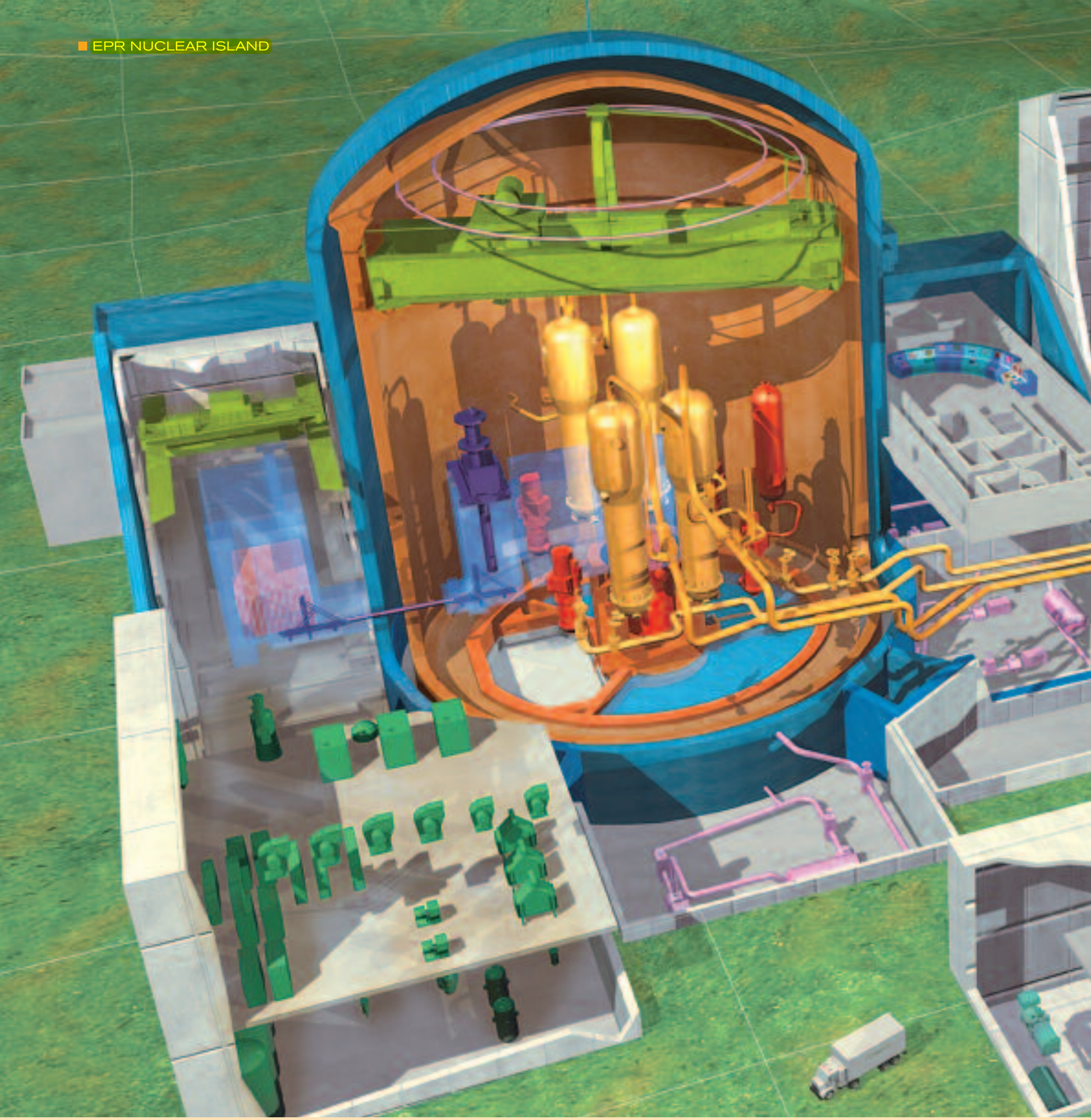
Part of the Nuclear Auxiliary Building (NAB) is designed as a radiological non-controlled area in which parts of the Operational Chilled Water System are located. Special laboratories for sampling systems are located at the lowest level. The maintenance area and some setdown areas used during the refueling phase are arranged on the highest level. All air-exhausts from the radiological controlled areas are routed, collected and controlled within the Nuclear Auxiliary Building prior to release through the stack.

6 Waste Building

The Waste Building is used to collect, store and treat liquid and solid radioactive waste.

7 Turbine Building

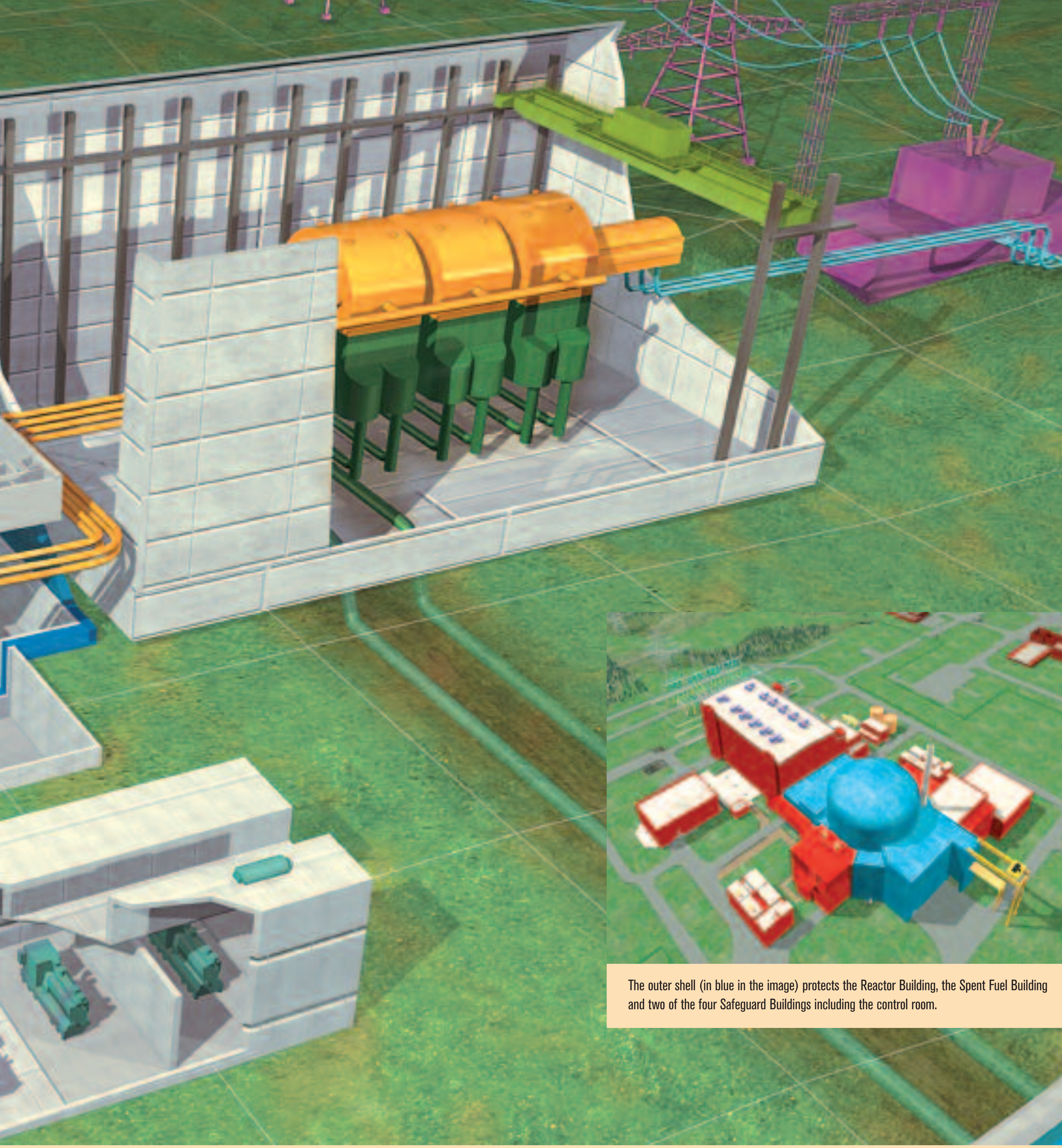
The Turbine Building houses all the main components of the steam-condensate-feedwater cycle. It contains, in particular, the turbine, the generator set, the condenser and their auxiliary systems.



The EPR layout offers exceptional and unique resistance to external hazards, especially earthquake and airplane crash.

- **To withstand major earthquake, the entire Nuclear Island stands on a single thick reinforced concrete basemat. Building height has been minimized and heavy components and water tanks are located at the lowest possible level.**

- **To withstand large airplane crash, the Reactor Building, Spent Fuel Building and two of the four Safeguard Buildings are protected by an outer shell made of reinforced concrete. The other two Safeguard Buildings are protected by a geographical separation. Similarly, the Diesel generators are located in two geographically separate buildings to avoid common failures.**

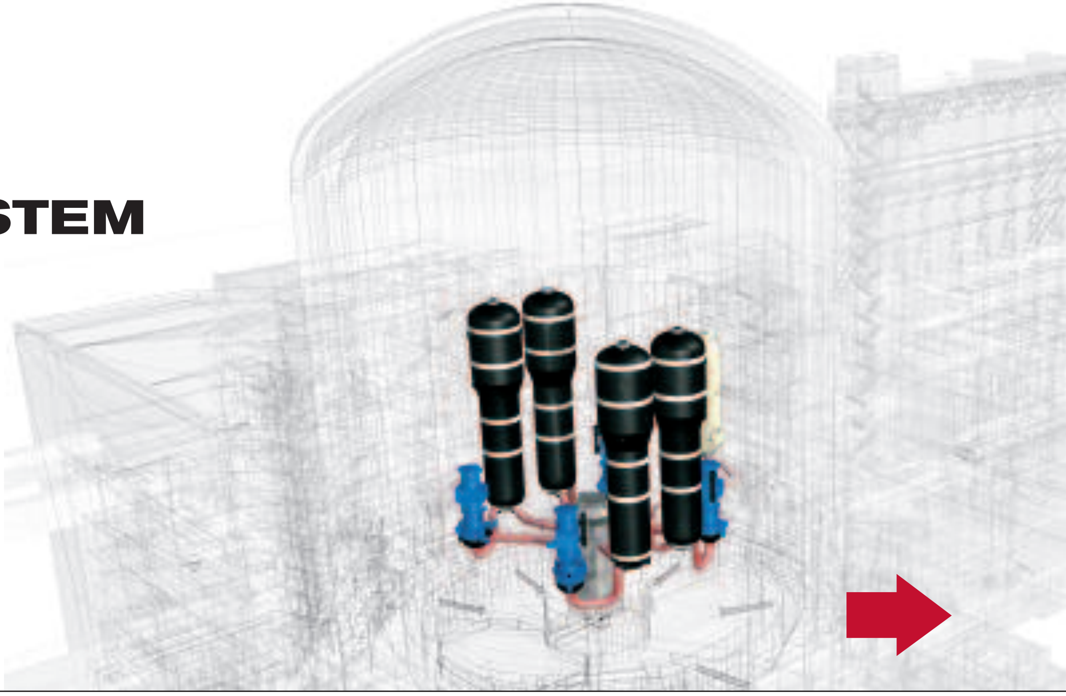


The outer shell (in blue in the image) protects the Reactor Building, the Spent Fuel Building and two of the four Safeguard Buildings including the control room.

➡ The EPR Nuclear Island design has undisputed advantages for operators, especially where radiation protection and ease of maintenance are concerned.

- **The layout is optimized and based on the strict separation of redundant systems.**
- **The distinction between access-controlled areas containing radioactive equipment and non-controlled areas significantly contributes to reduce exposure of the operating personnel.**
- **Maintenance requirements were systematically taken into account at the earliest stage of the design. For example, large setdown areas have been designed to make maintenance operations easier for operating personnel.**

PRIMARY SYSTEM



PRIMARY SYSTEM CONFIGURATION

The EPR primary system is of a well proven 4-loop design. The French 1,300 MWe and 1,500 MWe N4 reactors as well as the German KONVOI reactors are also of 4-loop design.

In each of the four loops, the primary coolant leaving the reactor pressure vessel through an outlet nozzle goes to a steam generator – the steam generator transfers heat to the secondary circuit –, then the coolant goes to a reactor coolant pump before returning to the reactor pressure vessel through an inlet nozzle. Inside the reactor pressure vessel, the coolant flows downward in the annular space between the core barrel and the vessel then it makes a U turn upward and flows through the core to extract the heat generated by the nuclear fuel.

A pressurizer, part of the primary system, is connected to one of the four loops. In normal operation, its main role is to automatically maintain the primary pressure within a specified range.

The EPR main reactor components: reactor pressure vessel, pressurizer and steam generators feature larger volumes than similar components from previous designs to provide additional operational and safety margins.

The increased free volume in the reactor pressure vessel, between the nozzles of the reactor coolant lines and the top of the core, provides a higher water volume above the core and thus additional margin with regard to the core "dewatering" time in the event of a postulated loss of coolant accident. Therefore, more time would be available to counteract such a situation.

Integration of design and manufacturing

Customers benefit greatly from the fact that heavy component design and manufacturing activities are brought together within the one group. The possibility, unique on the nuclear market place, of having a very close connection between the tow is an important asset for project performance. This setup, maintained by AREVA NP for many years, is a great advantage for utilities. It provides the opportunity to interact with a view to optimizing design, manufacturing, schedule and cost to obtain the best solutions.

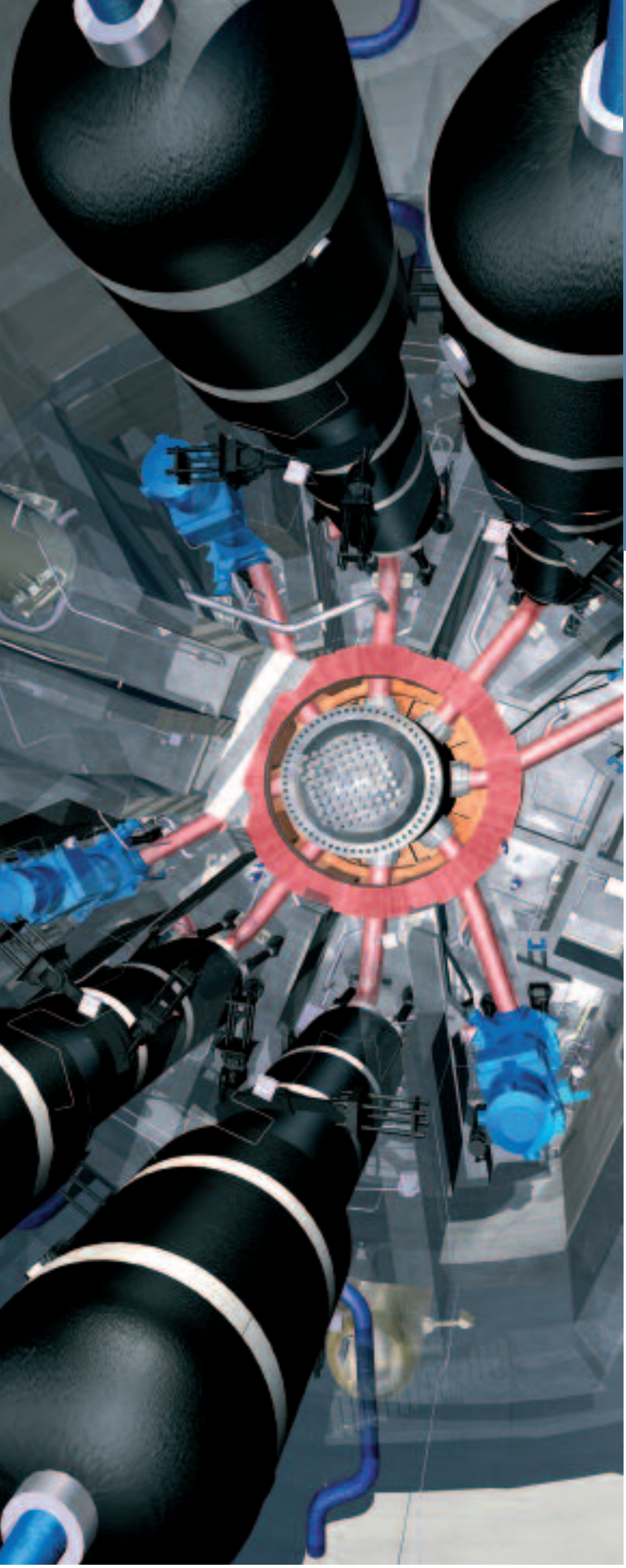
This increased volume would also be beneficial in shutdown conditions in case of loss of the Residual Heat Removal System function.

Larger water and steam phase volumes in the pressurizer smooth the response of the plant to normal and abnormal operating transients allowing extended time to counteract accident situations and extended equipment lifetime.

The larger volume of the steam generator secondary side results in increasing the secondary water inventory and the steam volume, which offers several advantages.

- During normal operation, smooth transients are obtained and thus the potential for unplanned reactor trips is reduced.
- Regarding the management of steam generator tube rupture scenarios, the large steam volume, in conjunction with a setpoint of the safety valves of the steam generators above the safety injection pressure, prevents liquid release outside the reactor containment.
- Due to the increased mass of secondary side water, in case of an assumed total loss of the steam generator feedwater supply, the dry-out time would be at least 30 minutes, sufficient time to recover a feedwater supply or to decide on other countermeasures.

Cattenom, France (4 X 1,300 MWe): inside a reactor building.



Computer-generated image
of the EPR primary system

CHARACTERISTICS	DATA
Reactor coolant system	
Core thermal power	4,500 MWth
Number of loops	4
Coolant flow per loop	28,330 m^3/h
Reactor pressure vessel inlet temperature	295.9 °C
Reactor pressure vessel outlet temperature	327.2 °C
Primary side design pressure	176 bar
Primary side operating pressure	155 bar
Secondary side	
Secondary side design pressure	100 bar
Saturation pressure at nominal conditions	78 bar
Main steam pressure at hot standby	90 bar

In addition, the primary system design pressure has been increased in order to reduce the actuation frequency of the safety valves which is also an enhancement in terms of safety.

OVERALL FUNCTIONAL REQUIREMENTS AND FEATURES

Activation of safety systems

Activation of the safety systems, including safety valves, does not occur prior to reactor trip, which means that best possible use is made of the depressurizing effect of the reactor trip. This approach also ensures maximum safety by minimizing the number of valve activations and the potential for valves sticking open after response.

Preventing reactor trip

Reactor trip is prevented by a fast reactor power cutback to part load when one of the following events occurs:
- loss of steam generator feedwater pumps, provided at least one of them remains available,
- turbine trip,
- full load rejection,
- loss of one reactor coolant pump.

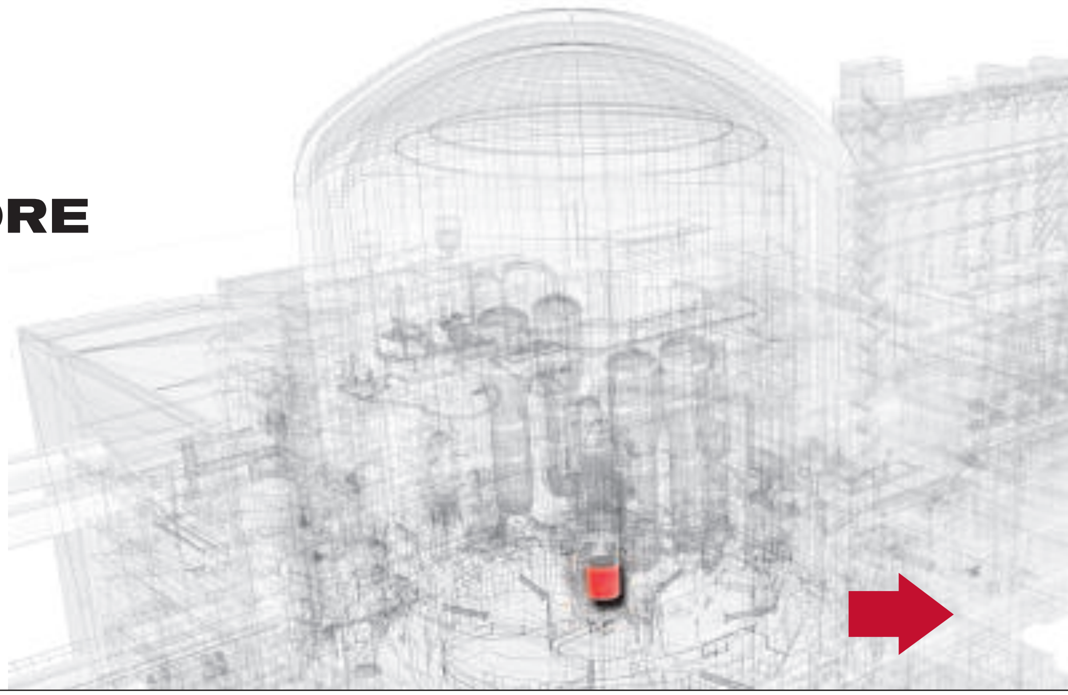
➡ **The increased volume of the primary system is beneficial for smoothing over many types of transients.**

➡ **The primary system design pressure has been increased to reduce the safety valve actuation frequency.**

➡ **The management of steam generator tube rupture scenarios prevents any liquid release outside the reactor containment.**

➡ **The large steam generator secondary side water inventory increases the time available to take action in case of assumed total loss of secondary feedwater.**

REACTOR CORE



The reactor core contains the fuel material in which the fission reaction takes place, releasing energy. The reactor internal structures support the fuel assemblies, channel the coolant and guide the control rods which control the fission reaction.

The core is cooled and moderated by light water at a pressure of 155 bar and a temperature in the range of 300 °C. The coolant contains soluble Boron as a neutron absorber. The Boron concentration in the coolant is varied as required to control relatively slow reactivity changes, including the effects of fuel burnup. Additional neutron absorbers (Gadolinium), in the form of burnable absorber-bearing fuel rods, are used to adjust the initial reactivity and power distribution. Instrumentation is located inside and outside the core to monitor its nuclear and thermal-hydraulic performance and to provide input for control functions.

The EPR core consists of 241 fuel assemblies. For the first core, assemblies are split into four groups with different enrichments (two groups with the highest enrichment, one of them with Gadolinium). For reload cores, the number and characteristics of the fresh assemblies depend on the type of fuel management scheme selected, notably cycle length and type of loading patterns. Fuel cycle lengths up to 24 months, IN-OUT and OUT-IN fuel management are possible. The EPR is designed for flexible operation with UO_2 fuel and/or MOX fuel. The main features of the core and its operating conditions have been selected to obtain not only high thermal efficiency of the plant and low fuel cycle costs, but also extended flexibility for different fuel cycle lengths and a high level of maneuverability.

Core design analysis

The core design analyses demonstrate the feasibility of different types of fuel management schemes to meet the requirements expressed by the utility companies in terms of cycle length and fuel cycle economy (reload fraction, burnup), and to provide the core characteristics needed for sizing of the reactor systems. The nuclear analyses establish physical locations for control rods, burnable poison rods, and physical parameters such as fuel enrichments and Boron concentration in the coolant. The thermal-hydraulic analyses establish coolant flow parameters to ensure that the margins against DNB are adequate to prevent burnout.

Core instrumentation

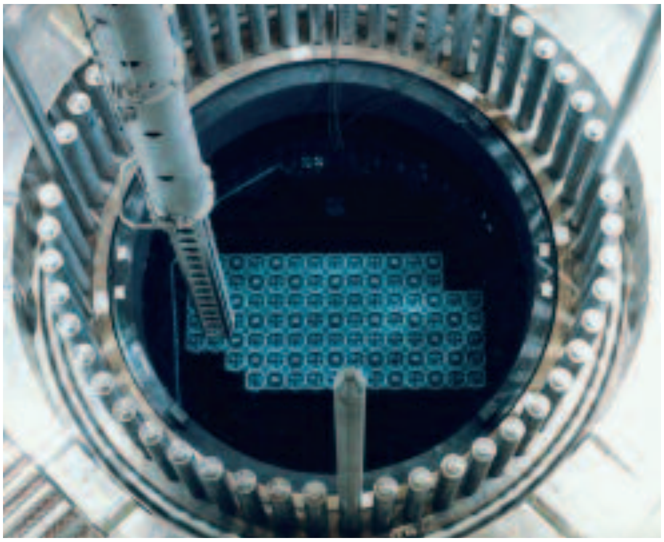
The core power is measured using the ex-core instrumentation, also utilized to monitor the process to criticality.

The reference instrumentation to monitor the power distribution in the core is an “aeroball” system. Vanadium balls are periodically inserted in the core. Their activation level is measured, giving values of the local neutron flux to construct the three-dimensional power map of the core.

The fixed in-core instrumentation consists of neutron detectors and thermocouples to measure the neutron flux distribution in the core and the temperature distribution at the core outlet.

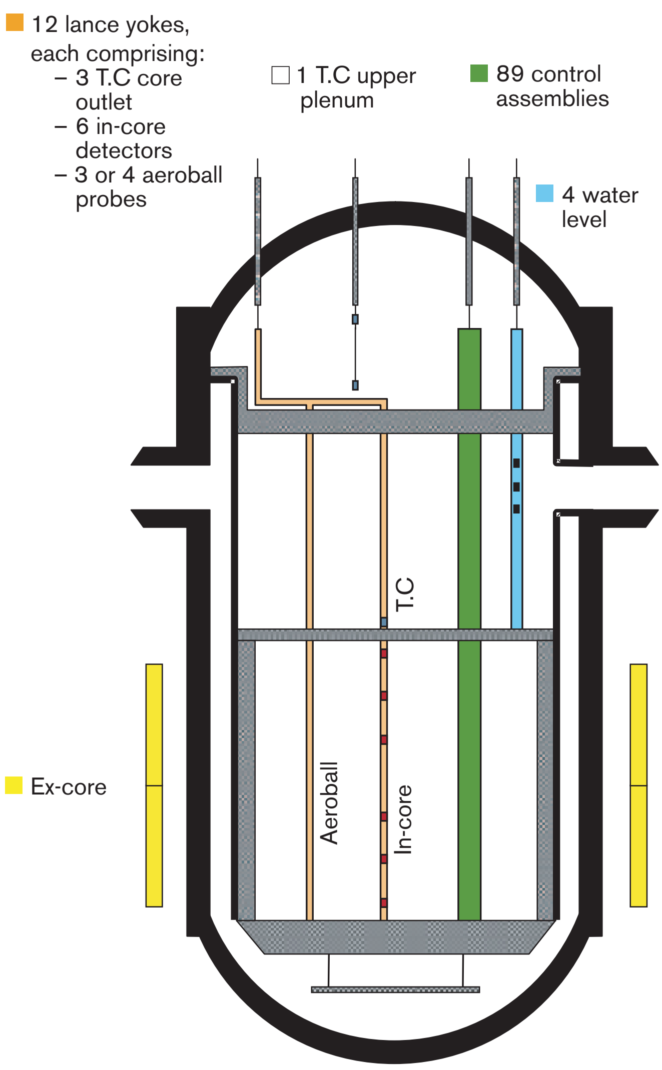
The in-core instrumentation is introduced through the vessel head. Therefore, the bottom of the reactor pressure vessel is free from any penetration.

For additional information see the “Instrumentation and Control systems” chapter, page 42.



Isar 2 unit, Germany (KONVOI, 1,300 MWe): fuel loading operation.

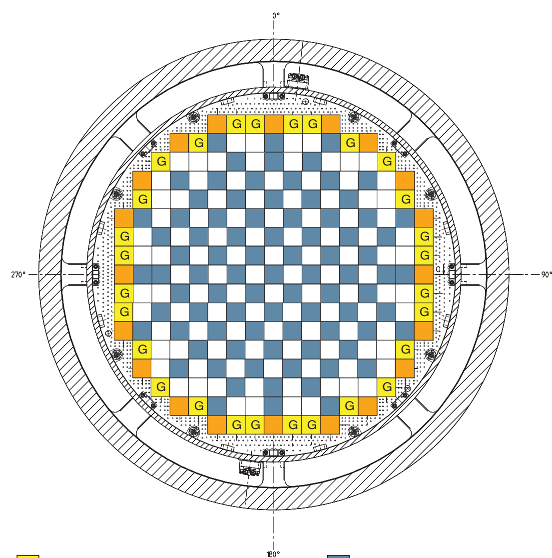
In-core instrumentation



T.C: Thermocouple

CHARACTERISTICS	DATA
Reactor core	
Thermal power	4,500 MWth
Operating pressure	155 bar
Nominal inlet temperature	295.6 °C
Nominal outlet temperature	328.2 °C
Equivalent diameter	3,767 mm
Active fuel length	4,200 mm
Number of fuel assemblies	241
Number of fuel rods	63,865
Average linear heat rate	163 W/cm

Typical initial core loading



G High enrichment with Gadolinium

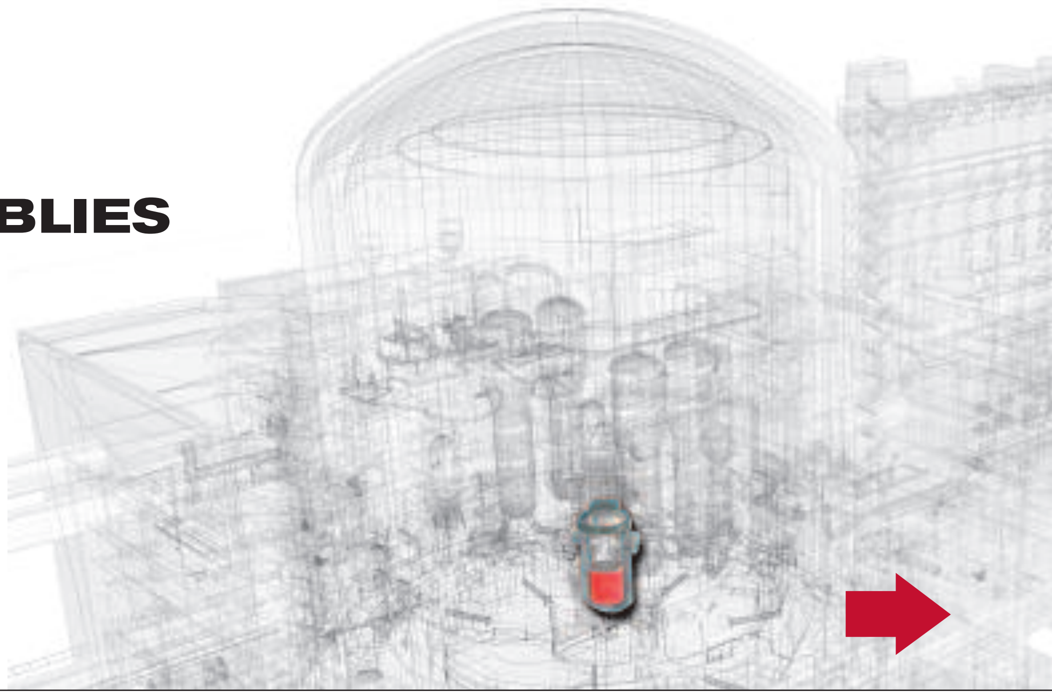
High enrichment without Gadolinium

Medium enrichment

Low enrichment

- **The EPR core is characterized by considerable margins for fuel management optimization.**
- **Several types of fuel management (fuel cycle length, IN-OUT/OUT-IN) are available to meet utilities' requirements.**
- **The main features of the core and its operating conditions give competitive fuel management cycle costs.**
- **The EPR core also offers significant advantages in favor of sustainable development:**
  - **17% saving on Uranium consumption per produced MWh,**
  - **15% reduction on long-lived actinides generation per MWh,**
  - **great flexibility for using MOX (mixed UO_2-PuO_2) fuel assemblies in the core, i.e. of recycling the plutonium extracted from spent fuel assemblies.**

FUEL ASSEMBLIES



Each fuel assembly is made up of a bundle of fuel rods that contain the nuclear fuel. The fuel rods and the surrounding coolant are the basic constituents of the active zone of the reactor core.

Fuel assembly structure

The fuel assembly structure supports the fuel rod bundle. It consists of a bottom and a top nozzles plus 24 guide thimbles and 10 spacer grids. The spacer grids are vertically distributed along the assembly structure. Inside the assembly, the fuel rods are vertically arranged according to a square lattice with a 17 x 17 array. 24 positions in the array are occupied by the guide thimbles, which are joined to the spacer grids and to the top and bottom nozzles. The bottom nozzle is equipped with an anti-debris device that almost eliminates debris-related fuel failures.

The guide thimbles are used as locations for the absorber rods of the Rod Cluster Control Assemblies (RCCA) and, when required, for fixed or moveable in-core instrumentation and neutron source assemblies. The bottom nozzle is shaped to direct and contributes to balance the coolant flow. It is also designed to trap small debris, which might circulate inside the primary circuit, in order to prevent damage to the fuel rods. The top nozzle supports the holddown springs of the fuel assembly. The spacer grids, except the top and bottom grids, have integrated mixing vanes to cause mixing of the coolant and improve the thermal exchange between the fuel rods and the coolant. The EPR spacer and mixing grids benefit from a proven design combining a mechanical robustness with a high level of thermal-hydraulic performance.

The guide thimbles and the structure of the mixing spacer grids are made of **M5™ alloy,** a Zirconium based alloy extremely resistant to corrosion and hydriding (the springs of the grids are made of Inconel 718).

Fuel rods

The fuel rods are composed of a stack of enriched Uranium dioxide (or Uranium and Plutonium Mixed Oxide, MOX) sintered pellets, with or without burnable absorber (Gadolinium), contained in a hermetically sealed cladding tube made of **M5™ alloy.** The fuel rod claddings, as the first of the three barriers against radioactive releases, isolate the fuel and fission products from the coolant. A plenum is provided inside the fuel rod to limit the build-up of pressure due to the release of fission gases by the pellets during irradiation. The fuel pellets are held in place by a spring which acts on the top end of the pellet stack. The fuel pellets consist of Uranium dioxide (UO_2) enriched in the fissile isotope U^{235} up to 5% or of Uranium-Plutonium mixed oxyde energetically equivalent.

Burnable poison

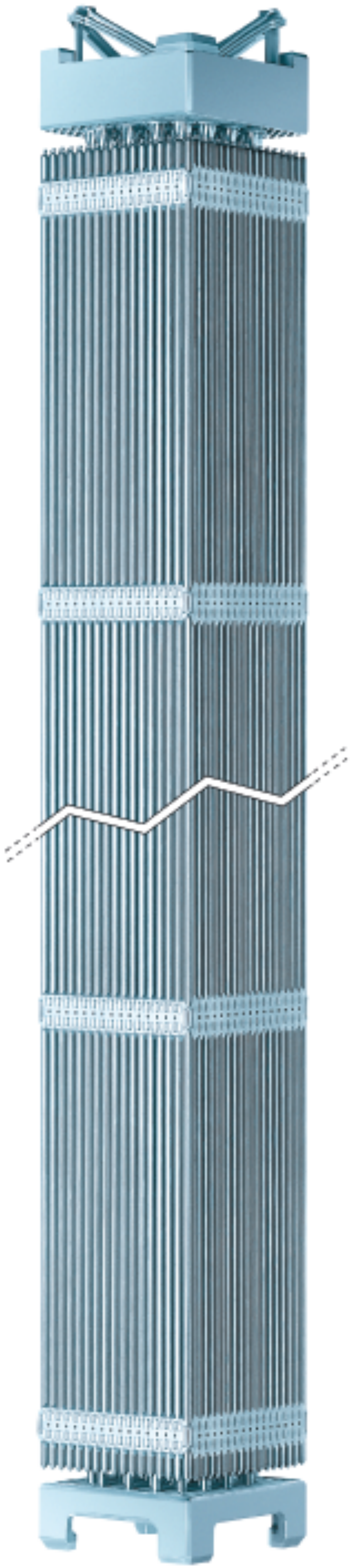
Gadolinium in the form of Gd_2O_3, mixed with the UO_2, is used as integrated burnable poison. The Gadolinium concentrations are in the range of 2% to 8% in weight. The number of Gadolinium-bearing rods per fuel assembly varies from 8 to 28, depending on the fuel management scheme. Enriched UO_2 is used as a carrier material for the Gd_2O_3 to reduce the radial power peaking factors once the Gadolinium has been consumed and makes it easier to meet the prescribed cycle length requirements.

The M5™ Zirconium based alloy

The M5™ alloy is a proven Zirconium based alloy which was developed, qualified and is industrially utilized by AREVA NP, mainly due to its outstanding resistance to corrosion and hydriding under PWR primary coolant system conditions. Under high duty and high burnup conditions, resistance to corrosion and hydriding is a crucial characteristic for PWR fuel rod claddings and fuel assembly structures as well. Consequently, EPR fuel rod claddings, guide thimbles and spacer grids are made of M5™ alloy. M5™ is presently the most advanced high performance PWR fuel material.

Fuel rod cutaway, showing fuel pellets, cladding, end-plugs and spring.

17 x 17 fuel assembly

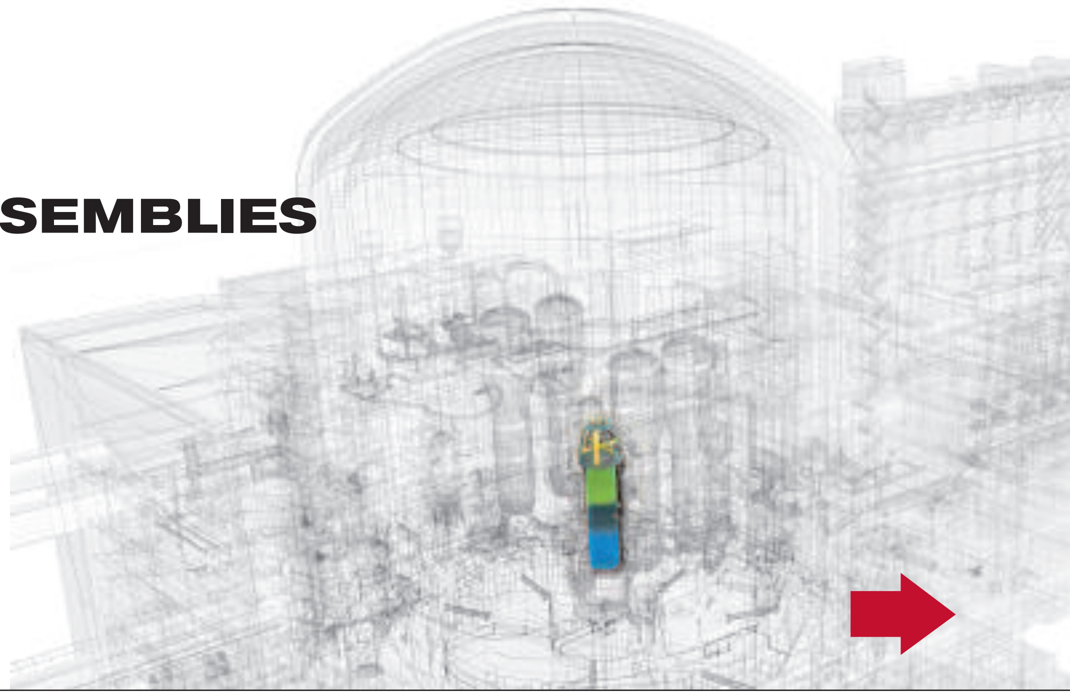


CHARACTERISTICS	DATA
Fuel assemblies	
Fuel rod array	17 x 17
Lattice pitch	12.6 mm
Number of fuel rods per assembly	265
Number of guide thimbles per assembly	24
Fuel assembly discharge burnup (maximum)	> 70,000 MWd/t
Materials	
– Mixing spacer grids	
• structure	M5™
• springs	Inconel 718
– Top & bottom spacer grids	Inconel 718
– Guides thimbles	M5™
– Nozzles	Stainless steel
– Holddown springs	Inconel 718
Fuel rods	
Outside diameter	9.50 mm
Active length	4,200 mm
Cladding thickness	0.57 mm
Cladding material	M5™

- **The U^{235} enrichment level up to 5% allows high fuel assembly burnups.**
- **The choice of M5™ for cladding and structural material results in outstanding resistance to corrosion and hydriding and excellent dimensional behavior at high burnup.**
- **The spacer grids design offers a low flow resistance and a high thermal performance.**
- **The use of an efficient anti-debris device almost eliminates debris-related fuel failures.**

Fuel manufacturing workshop, Lynchburg (Virginia, USA).

CONTROL ASSEMBLIES



The control assemblies, inserted in the core through the guide-thimbles of fuel assemblies, provide reactor power control and reactor trip.

RCCA manufacturing at the FBFC Pierrelatte (France) fuel fabrication plant.

Rod Cluster Control Assemblies

The core has a fast shutdown control system comprising 89 Rod Cluster Control Assemblies (RCCAs). All RCCAs are of the same type and consist of 24 identical absorber rods, fastened to a common head assembly. These rods contain neutron absorbing materials. When they are totally inserted in the core, they cover almost the whole active length of the fuel assemblies.

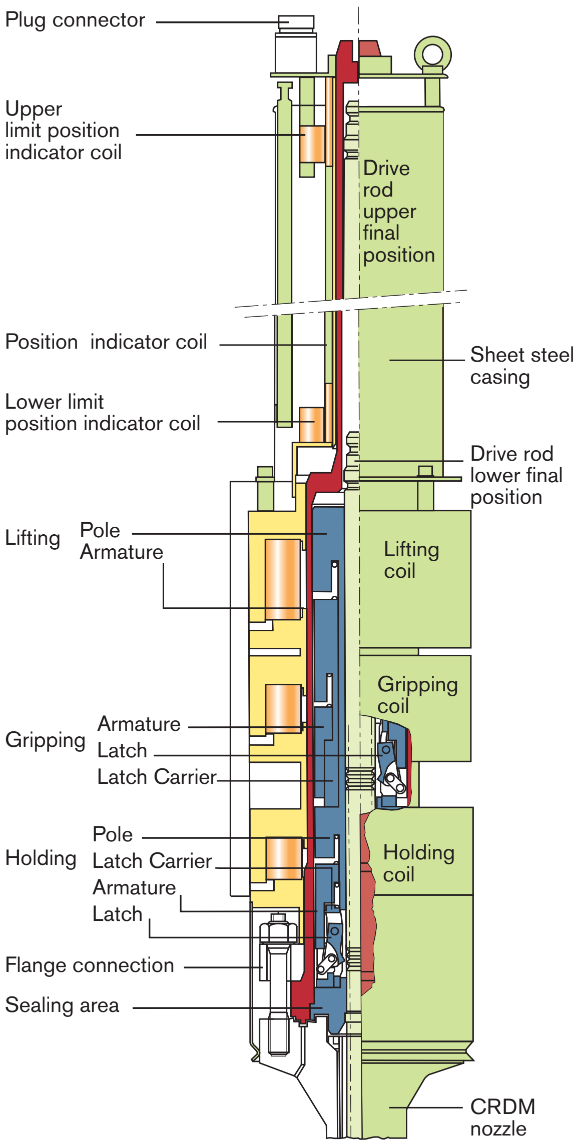
The EPR is equipped with RCCAs of the HARMONI™ type, a proven AREVA NP design. The neutron absorbing components are bars made of an Ag, In, Cd alloy and sintered pellets of Boron carbide (B_4C). Each rod is composed of a stack of Ag, In, Cd bars and B_4C pellets contained in a stainless steel cladding under a Helium atmosphere (for efficient cooling of the absorbing materials).

Because mechanical wear of the rod claddings happens to be a limiting factor for the operating life of RCCAs, the HARMONI™ claddings benefit from a specific treatment (ion-nitriding) that makes their external surface extremely wear-resistant and eliminates the cladding wear issue.

The RCCAs are assigned to different control bank groups. 37 RCCAs are assigned to control average moderator temperature and axial offset, and 52 RCCAs constitute the shutdown-bank. The first set is divided into five groups split into quadruplets. These quadruplets are combined to form four different insertion sequences depending on cycle depletion. This sequence can be changed at any time during operation, even at full power. A changeover is performed at regular intervals, approximately every 30 equivalent full power days, to rule out any significant localized burnup delay. At rated power the control banks are nearly withdrawn. At intermediate power level, the first quadruplet of a sequence can be deeply inserted and the second may be also inserted. Shutdown margins are preserved by the RCCA insertion limits.

➡ **The EPR is equipped with RCCAs of the proven HARMONI™ design that guarantees a long operating life whatever the operating mode of the reactor.**

CRDM cutaway



CHARACTERISTICS		DATA
Rod cluster control assemblies (RCCAs)		
Mass		82.5 kg
Number of rods per assembly		24
Absorber		
AIC part (lower part)		
– Weight composition (%): Ag, In, Cd		80, 15, 5
– Specific mass		10.17 g/cm^3
– Absorber outer diameter		7.65 mm
– Length		1,500 mm
B4C part (upper part)		
– Natural Boron		19.9% atoms of B^{10}
– Specific mass		1.79 g/cm^3
– Absorber diameter		7.47 mm
– Length		2,610 mm
Cladding		
Material		Austenitic stainless steel
Surface treatment (externally)		Ion-nitriding
Outer diameter		9.68 mm
Inner diameter		7.72 mm
Filling gas		Helium
Control rod drive mechanisms (CRDMs)		
Quantity		89
Mass		403 kg
Lift force		> 3,000 N
Travel range		4,100 mm
Stepping speed		375 mm/min or 750 mm/min
Max. scram time allowed		3.5 s
Materials	– pressure housing	Forged austenitic stainless steel
	– drive rod	Martensitic stainless steel
	– latch unit	Amagnetic austenitic stainless steel

Control Rod Drive Mechanisms

A function of the Control Rod Drive Mechanisms (CRDMs), for reactor control purposes, is to insert and withdraw the 89 RCCAs over the entire height of the core and to hold them in any selected position. The other function of the CRDMs is to drop the RCCAs into the core, to shut down the reactor in a few seconds by stopping the chain reaction, in particular in case of an abnormal situation.

The CRDMs are installed on the reactor pressure vessel head and bolted to adapters welded to the vessel head. Each CRDM is a self-contained unit that can be fitted or removed independently of the others. The CRDMs do not need forced ventilation of the coils, which saves space on the reactor head. The control rod drive system responds to the actuation signals generated by the reactor control and protection system or by operator action. The pressure housings of the CRDMs are part of the second barrier against radioactive releases, like the rest of the reactor primary circuit. Therefore, they are designed and fabricated in compliance with the same level of quality requirements.

The complete CRDM consists of:

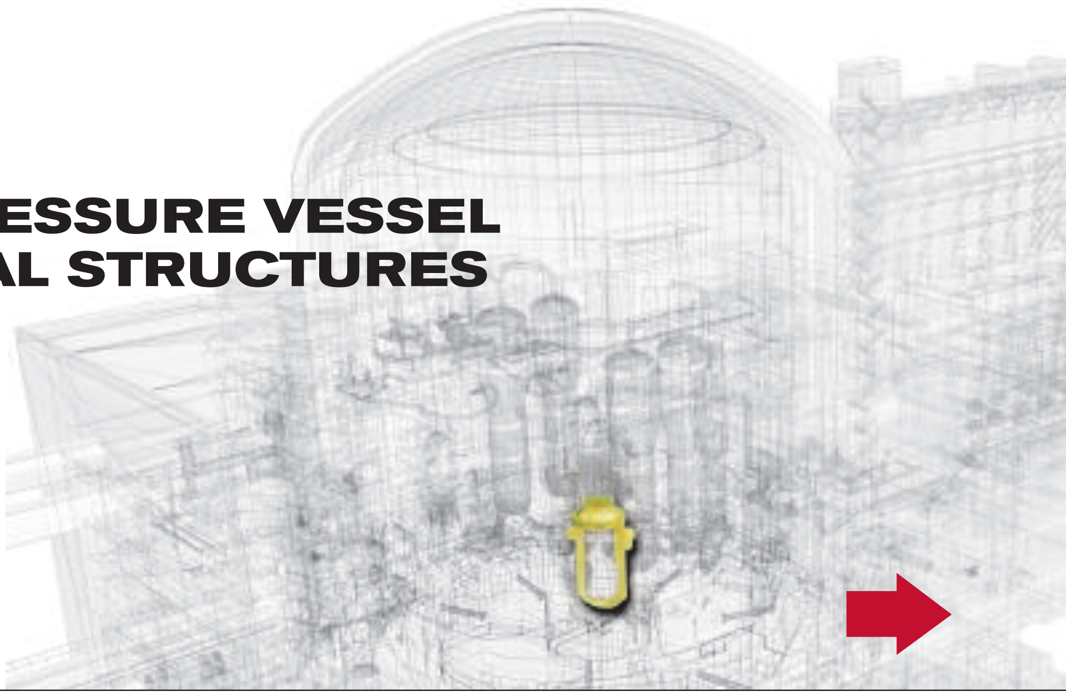
- the pressure housing with flange connection,
- the latch unit,
- the drive rod,
- the coil housing.

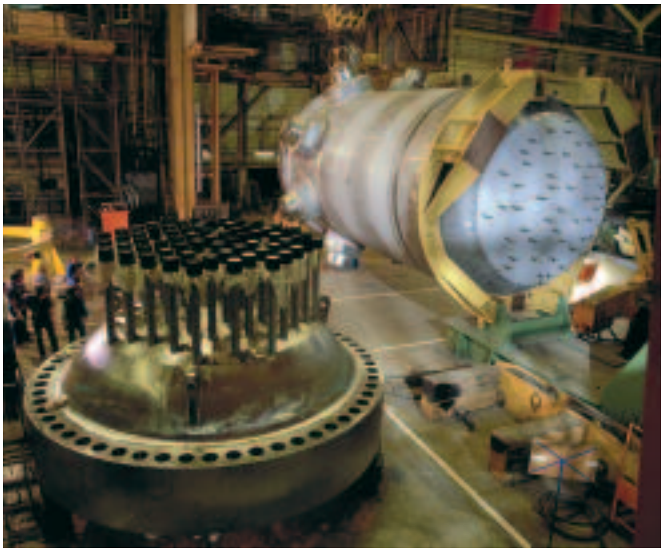
When the reactor trip signal is given, all operating coils are de-energized, the latches are retracted from the rod grooves and the RCCA drops freely into the reactor core under the force of gravity.

➡ **CRDMs are of the same type as those used in the KONVOI reactors, thus they are well proven and based on excellent track record.**

➡ **CRDMs are cooled by natural convection which saves space on the reactor head.**

REACTOR PRESSURE VESSEL AND INTERNAL STRUCTURES





Chalon manufacturing plant (France): Civaux 1 (N4, 1,500 MWe) reactor pressure vessel and its closure head.

Reactor Pressure Vessel

The Reactor Pressure Vessel (RPV) contains the core. The closure head is fastened to the top of the RPV by a set of studs.

The number of large welds which reduces their manufacturing cost and time for in-service inspection is minimized. Nozzles of the set-on type facilitate the welding of the primary piping to the RPV and the welds in-service inspection as well.

The lower part of the RPV consists of a cylindrical part at the core level, a transition ring and a spherical bottom piece. Because the in-core instrumentation is introduced through the closure head at the top of the RPV, there is no penetration through the bottom piece.

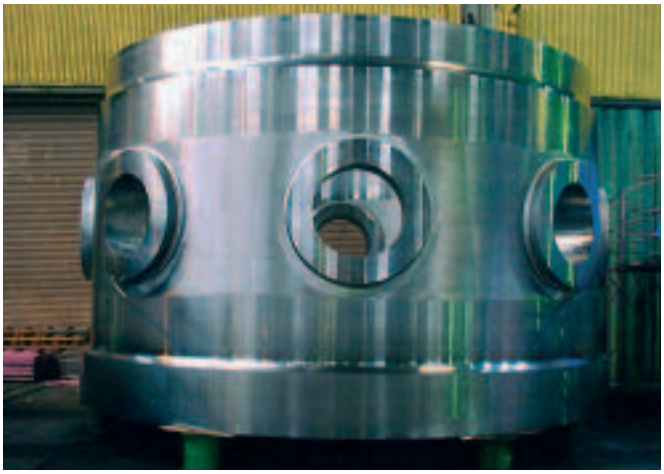
The RPV is designed to facilitate the non-destructive testing during in-service inspections. In particular, its internal surface is accessible to allow 100% visual and/or ultrasonic inspection of the welded joints from the inside.

The RPV closure head is a partly spherical piece with penetrations for the control rod drive mechanisms and the in-core instrumentation.

The RPV and its closure head are made of forged ferritic steel – 16 MND 5 – a material that combines adequate tensile strength, toughness and weldability. The entire internal surface of the RPV and its closure head are cladded with stainless steel for corrosion resistance. To contribute to the reduction of the corrosion products radiation source term, the cladding material is specified with a low Cobalt residual content.

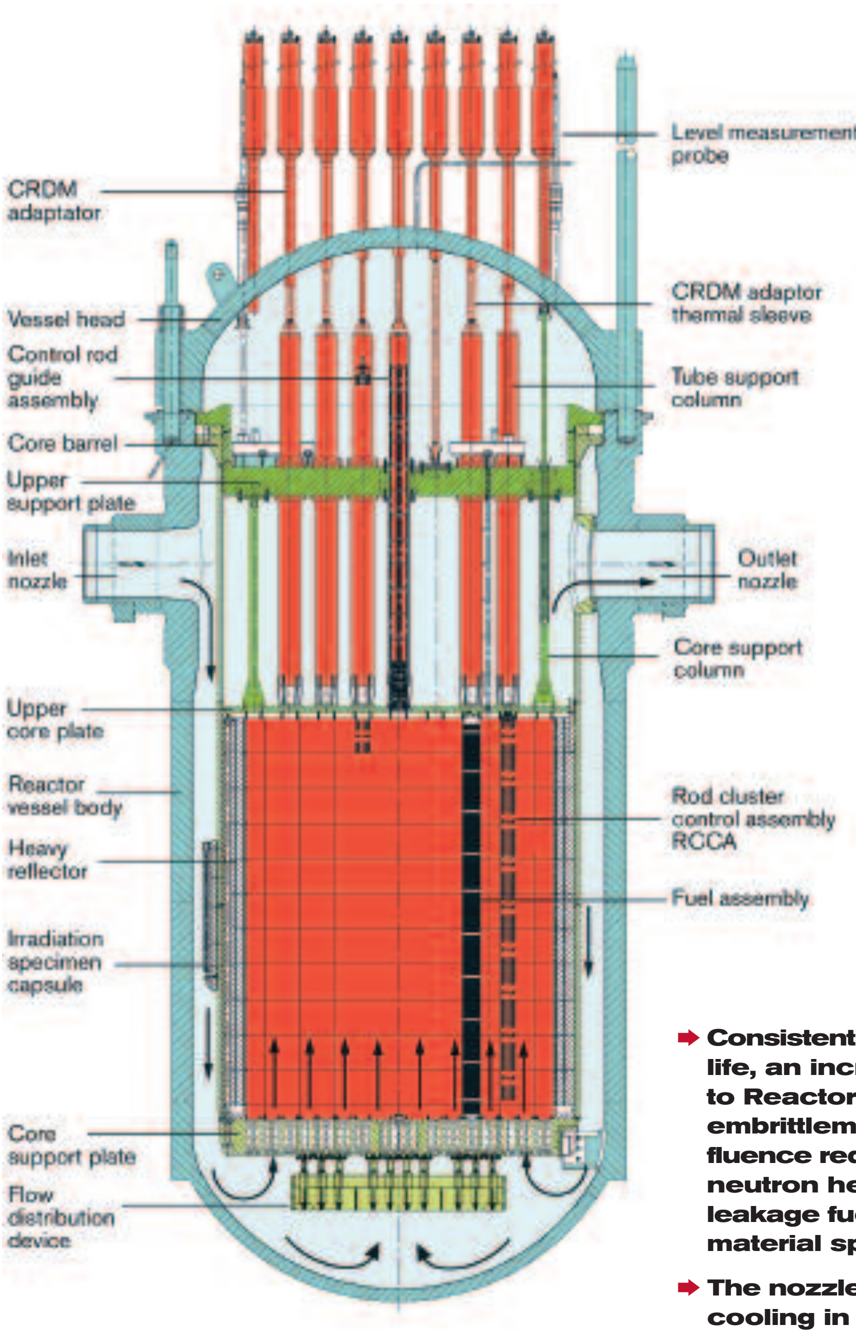
The RPV is supported by a set of integrated pads underneath the eight primary nozzles. These pads rest on a support ring which is the top part of the reactor pit.

Significant safety margin against the risk of brittle fracture (due to material aging under irradiation) during the RPV's 60 year design life is ensured.



Reactor pressure vessel monobloc upper shell for the Olkiluoto 3 (Finland) EPR.

Reactor pressure vessel and internals cutaway



The ductile-brittle transition temperature (RT_{NDT}) of the RPV material remains lower than 30 °C at the end of the design life. This result is obtained from the choice of the RPV material and its specified low content in residual impurities, and also thanks to a reduced neutron fluence to the RPV due to the implementation of a neutron reflector surrounding the core and protecting the RPV against the neutron flux.

The suppression of any weld between the flange and the nozzle shell course plus the set-on design of the nozzles allow an increase of the vertical distance between the nozzles and the top of the core. Therefore, in the assumption of a loss of coolant situation, more time is available for the operator to counteract the risk of having the core uncovered by the coolant.

- **Consistently with the EPR 60-year design life, an increased margin with regard to Reactor Pressure Vessel (RPV) embrittlement is obtained from neutron fluence reduction (RPV diameter enlarged, neutron heavy reflector, low neutron leakage fuel management) and from RPV material specifications (reduced RT_{NDT}).**
- **The nozzle axis raising improves the fuel cooling in the event of a loss of coolant accident.**
- **No penetration through the RPV bottom head strengthens its resistance in case of postulated core meltdown and prevents the need for in-service inspection and potential repairs.**
- **The reduced number of welds and the weld geometry decrease the need for in-service inspection, facilitate non-destructive examinations and reduce inspection duration as well.**
- **A low Cobalt residual content of the stainless steel cladding is specified to less than 0.06% to contribute to the radiation source term reduction.**

Reactor Internals

The Reactor Pressure Vessel Internals (RPVI) support the fuel assemblies and keep them properly aligned and spaced to ensure free motion of the control rods and core cooling by the primary coolant in any circumstances, including postulated accident circumstances.

The RPVI allow insertion and positioning of the in-core instrumentation as well as protection against flow-induced vibrations during reactor operation.

The internals also contribute to the integrity of the second of the three barriers (see page 45) by protecting the Reactor Pressure Vessel (RPV) against fast neutron fluence-induced embrittlement.

The internals accommodate the capsules containing samples of the RPV material which are irradiated then examined in the framework of the RPV material surveillance program.

The RPVI are removed partially from the RPV to allow fuel assembly loading/unloading, or are totally removed for complete access to the RPV inner wall for in-service inspection.

Chooz B1, France (N4, 1,500 MWe) upper internals.

The main parts of the RPVI

Upper internals

The upper internals house the Rod Cluster Control Assembly (RCCA) guides. The RCCA guide tube housings and columns are connected to an RCCA guide support plate and the upper core plate. In operation, the upper internals maintain axially the fuel assemblies in their correct position.

Core barrel assembly and lower internals

The core barrel flange sits on a ledge machined from the RPV flange and is preloaded axially by a large Belleville type spring. The fuel assemblies sit directly on a perforated plate, the core support plate. This plate is machined from a forging of stainless steel and welded to the core barrel. Each fuel assembly is positioned by two pins 180° apart.

Heavy reflector

To reduce neutron leakages and flatten the power distribution, the space between the polygonal core and the cylindrical core barrel is filled with a heavy neutron reflector. The **heavy reflector** is a stainless steel structure, surrounding the core, made of rings piled up one on top of the other. The rings are keyed together and axially restrained by tie rods bolted to the core support plate. The heat generated inside the steel structure by absorption of gamma radiation is removed by the primary coolant, through holes and gaps provided in the reflector structure.

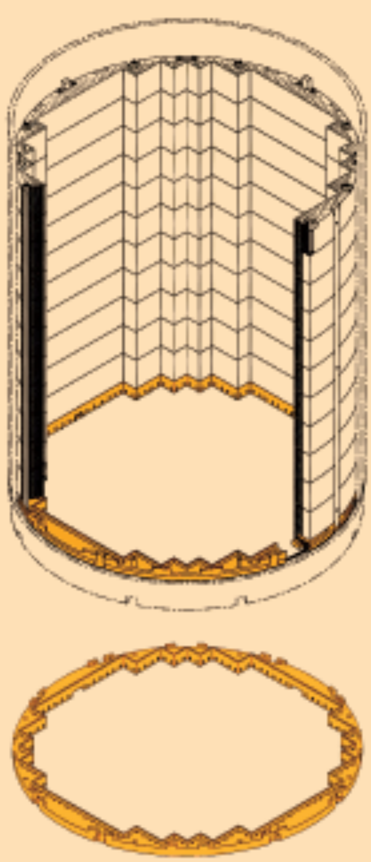
Materials

The base material of the internals is a low Carbon Chromium-Nickel stainless steel. The various connectors, such as bolts, pins, tie rods, etc., are made of cold-worked Chromium-Nickel-Molybdenum stainless steel. At some locations, hard-facing materials are used to prevent fretting wear. To contribute to the radiation source term reduction, stainless steels are specified with a very low Cobalt residual content and the use of Stellite hard-facing is reduced as much as possible.

Heavy reflector

The **heavy reflector** is an innovative feature with significant benefits:

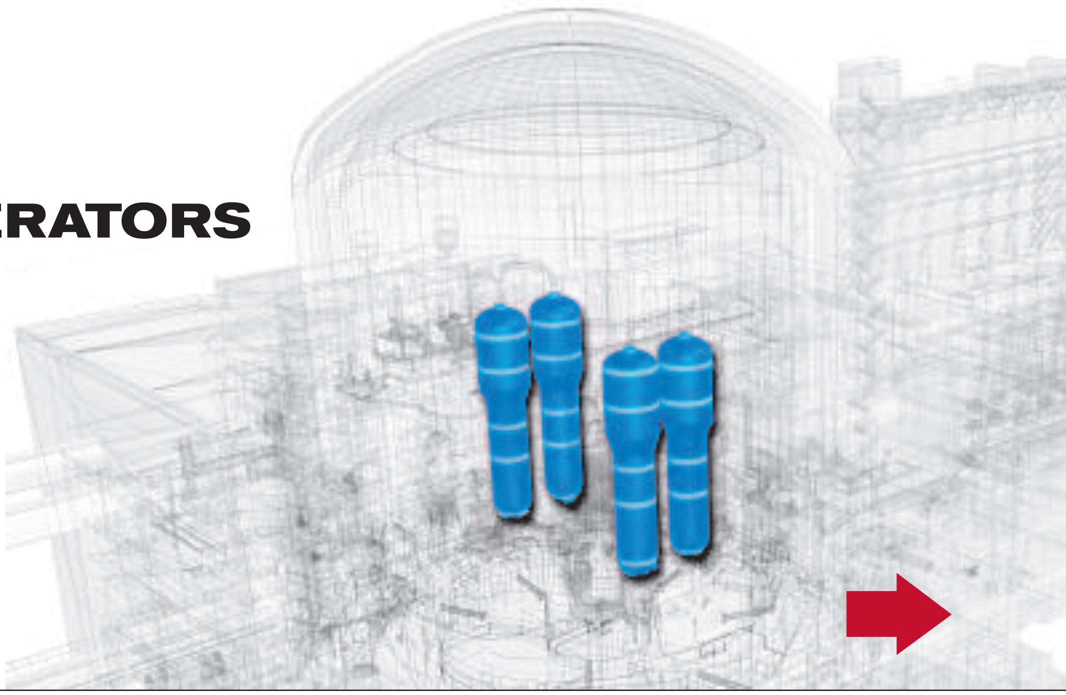
- By reducing the flux of neutrons escaping from the core, the nuclear fuel is better utilized (more neutrons are available to take part in the chain reaction process), thereby making it possible to decrease the fuel cycle cost by reducing the fuel enrichment necessary to reach a given burnup, or to increase burnup with a given enrichment.
- By reducing the neutron leakages from the core, the Reactor Pressure Vessel is protected against fast neutron fluence-induced aging and embrittlement, helping to ensure the 60-year design life of the EPR.
- The reactor also provides advances in terms of mechanical behavior of the internal structure surrounding the core:
  - a smooth stress distribution inside the structure, due to an efficient inside cooling of the reflector, limiting loads and avoiding deformation,
  - no discontinuities, like welds or bolts, in the most irradiated areas,
  - a large decrease of depressurization loads to take into account in case of assumed loss of coolant accident, because there is no significant quantity of water trapped in the structure around the core.



CHARACTERISTICS	DATA
Reactor pressure vessel	
Design pressure	176 bar
Design temperature	351 °C
Life time (load factor 0.9)	60 yrs
Inside diameter (under cladding)	4,885 mm
Wall thickness (under cladding)	250 mm
Bottom wall thickness	145 mm
Height with closure head	12,708 mm
Base material	Low alloy ferritic steel
Cladding material	Stainless steel (Cobalt $\leq$ 0.06%)
Mass with closure head	526 t
End of life fluence level (E > 1 MeV) IN-OUT fuel management scheme with UO_2	$\approx 1 \times 10^{19}$ n/cm^2
Base material final RT_{NDT} (final ductile-brittle transition temperature)	≈ 30 °C
Closure head	
Wall thickness	230 mm
Number of penetrations for:	
• Control rod mechanisms	89
• Dome temperature measurement	1
• Instrumentation	16
• Coolant level measurement	4
Base material	Low alloy ferritic steel
Cladding material	Austenitic stainless steel
Upper internals	
Upper support plate thickness	350 mm
Upper core plate thickness	60 mm
Main material	Austenitic stainless steel
Lower internals	
Lower support plate thickness	415 mm
Lower internals parts material	Austenitic stainless steel
Neutron heavy reflector	
Material	Austenitic stainless steel
Mass	90 t

- **The design of the EPR reactor pressure vessel internals is based on the N4 and KONVOI proven designs.**
- **The heavy neutron reflector brings an enhanced fuel utilization and protects the reactor pressure vessel against aging and embrittlement.**
- **A low Cobalt residual content of the stainless steels is specified and the use of Stellite hard-facing is optimized so as to reduce radiation source term.**

STEAM GENERATORS



The steam generators (SG) are the interface between the primary water heated by the nuclear fuel and the secondary water which provides steam to the turbine generator.

The EPR steam generator is a vertical, U-tube, natural circulation heat exchanger equipped with an **axial economizer**. It is an enhanced version of the N4 steam generator.

It is composed of two subassemblies:
• the lower section where the heat exchange process between the primary water and the secondary water takes place,
• the upper section where the steam-water mixture is mechanically dried before it is routed to the turbine.

In conjunction with an increased heat exchange area, the EPR axial economizer makes it possible to reach a saturation pressure of 78 bar and a plant efficiency of 36 to 37% (depending on site conditions). The tube bundle is made of a proven stress-corrosion resistant alloy: Inconel 690 with a specified mean value Co content less than 0.015%. The steam generator bundle wrapper is made of 18 MND 5 steel.

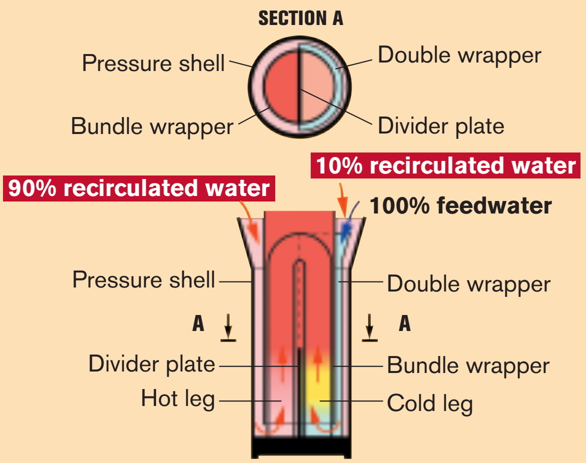
To increase the heat transfer efficiency, the axial economizer directs 100% of the cold feedwater to the cold leg of the tube bundle, and about 90% of the hot recirculated water to the hot leg. This is done by adding a wrapper to guide the feedwater to the cold leg of the tube bundle and a partition plate to separate the cold leg from the hot leg. This design improvement increases the steam pressure by about 3 bar compared to a conventional steam generator. There is an easy access to the tube bundle for inspection and maintenance.

Particular attention was given during the design of the EPR steam generator to prevent secondary cross-flows to protect the tube bundle against vibration risks.

The steam drum volume has been augmented. This feature, plus a safety injection pressure lower than the set pressure of the secondary safety valves, would prevent the steam generators from filling up with water in case of steam generator tube rupture to avoid liquid releases.

Compared to previous designs, the mass of water on the secondary side has been increased to get a dry-out time, in the event of a total loss of feedwater, of at least 30 minutes.

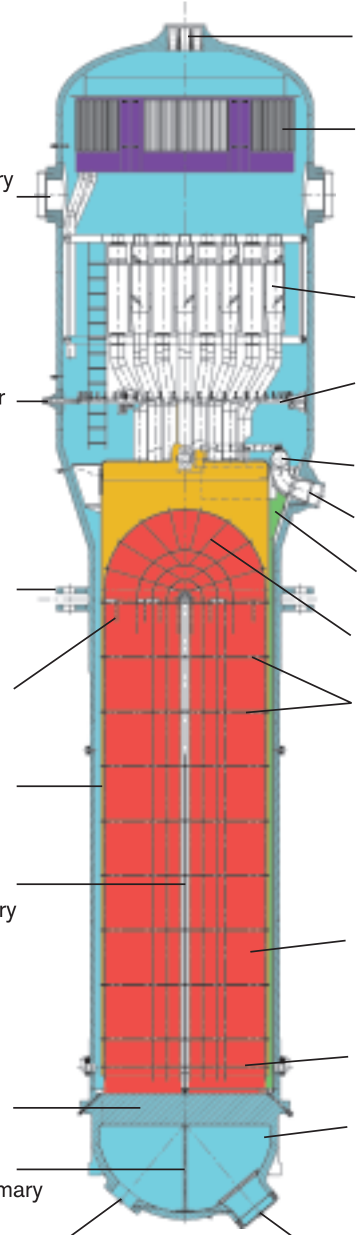
The steam generator is fully shop-built, transported to the plant site and installed in its reactor building cubicle in one piece.



The axial economizer

Its principle primarily consists in directing the feedwater to the cold leg of the tube bundle and about 90% of the recirculated water to the hot leg. In practice, this is done by adding to the standard natural circulation U-tube design a double wrapper in the cold leg of the downcomer to guide the feedwater to the cold leg of the tube bundle and a secondary side partition plate to separate the cold leg and the hot leg of the tube bundle. In conjunction with those two design features, the internal feedwater distribution system of the steam generator covers only the 180° of the wrapper on the cold side.

Steam generator cutaway



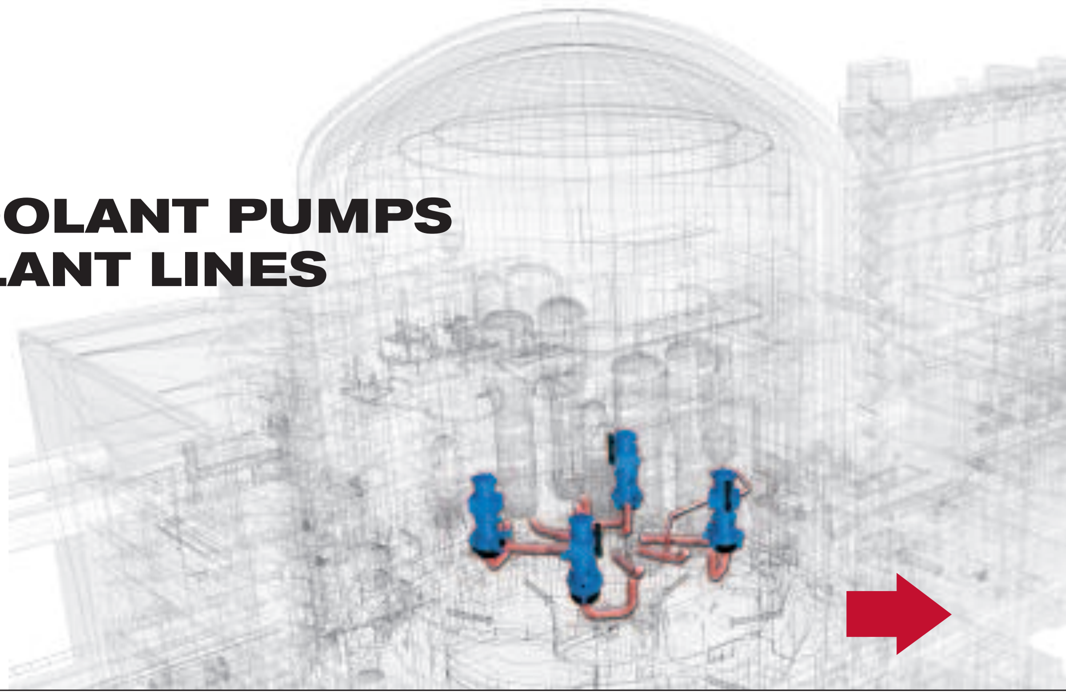
Transportation of a steam generator manufactured in China for Ling-Ao 2.

CHARACTERISTICS	DATA
Steam generators	
Number	4
Heat transfer surface per steam generator	7,960 m^2
Primary design pressure	176 bar
Primary design temperature	351 °C
Secondary design pressure	100 bar
Secondary design temperature	311 °C
Tube outer diameter/wall thickness	19.05 mm / 1.09 mm
Number of tubes	5,980
Triangular pitch	27.43 mm
Overall height	23 m
Total mass	500 t
Materials	
• Tubes	Alloy 690 TT* (Nickel base Alloy)
• Shell	Low alloy steel
• Cladding tube sheet	Ni Cr Fe alloy
• Tube support plates	13% Cr improved stainless steel
Miscellaneous	
Feedwater temperature	230 °C
Moisture carry – over	0.1%
Main steam flow at nominal conditions	2,554 kg/s
Main steam temperature	293 °C
Saturation pressure at nominal conditions	78 bar
Pressure at hot stand by	90 bar

* TT: Thermally treated

- **The steam generator is an enhanced version of the axial economizer steam generator implemented on N4 plants.**
- **The axial economizer allows increasing by 3 bar the steam pressure output compared to a conventional design, without impairing access to the tube bundle for inspection and maintenance.**
- **The very high steam saturation pressure at tube bundle outlet (78 bar) is a major contributor to the high efficiency of the EPR (37%).**
- **The secondary water mass is consistent with the 30 min. time period before steam generator dry-out in case of loss of all feedwater systems.**
- **The increase of the steam volume and the set pressure of the secondary safety valves prevent any liquid release to the environment in case of steam generator tube rupture.**

REACTOR COOLANT PUMPS & MAIN COOLANT LINES



Reactor Coolant Pumps

The Reactor Coolant Pumps (RCP) provide forced circulation of water through the reactor coolant system. This circulation removes heat from the reactor core to the steam generators, where it is transferred to the secondary system.

A reactor coolant pump is located between the steam generator outlet and the reactor vessel inlet of each of the four primary loops.

The reactor coolant pump design is an enhanced version of the model used in the N4 reactors. This pump model is characterized by the very low vibration level of its shaft line, due to the hydrostatic bearing installed at the end of the impeller. The pump capacity has been increased to comply with the EPR operating point. In addition, a new safety device, a **standstill seal**, has been added as shaft seal back-up.

The EPR coolant pump consists of three major components: the pump itself, the shaft seals and the motor.

• **The pump hydraulic cell** consists of the impeller, diffuser, and suction adapter installed in a casing. The diffuser, in one piece, is bolted to the closure flange. The whole assembly can be removed in one piece. The torque is transmitted from the shaft to the impeller by a "Hirth" assembly which consists in radial grooves machined on the flat end of the shaft and symmetrically on the impeller. The shaft is made of two parts rigidly connected by a "spool" piece bolted to each half and removable for maintenance of the shaft seals. It is supported by three radial bearings, two oil bearings on the upper part and one hydrostatic water bearing located on the impeller. The static part of the hydrostatic bearing is part of the diffuser. The axial thrust is reacted by a double acting thrust bearing located at the upper end of the motor shaft below the flywheel.

• **The shaft seal system** consists of three dynamic seals staggered into a cartridge and a standstill seal. The first dynamic seal is a hydrostatic-controlled leakage, film-riding face seal that takes the full primary pressure; the second one is a hydrodynamic seal that takes the remaining pressure in normal operation but can take the full primary pressure in the assumed event of a first stage failure; the third one is also a hydrodynamic seal with no significant differential pressure. Its purpose is to complete final leak tightness and prevent spillage of water.

The shaft seals are located in a housing bolted to the closure flange. The closure flange is clamped to the casing by a set of studs together with the motor stand.

In normal operation, the shaft seals are cooled by the seal injection water which is injected just under the shaft seals at a pressure slightly higher than that of the reactor coolant. A thermal barrier, a low-pressure water coil, would cool the primary water before it comes in contact with the shaft seals in the event of a disruption of the seal injection water.

The standstill seal

The shaft seals are backed up with a standstill seal that closes, once the pump is at rest and all seals of the leak-off lines are closed. It creates a sealing surface with a metal-to-metal contact ensuring the shaft tightness in case of:

- **simultaneous loss of water supply by the Chemical and Volume Control System and by the Component Cooling Water System used to cool the shaft sealing system,**
- **cascaded failure of all the stages of the shaft sealing system.**

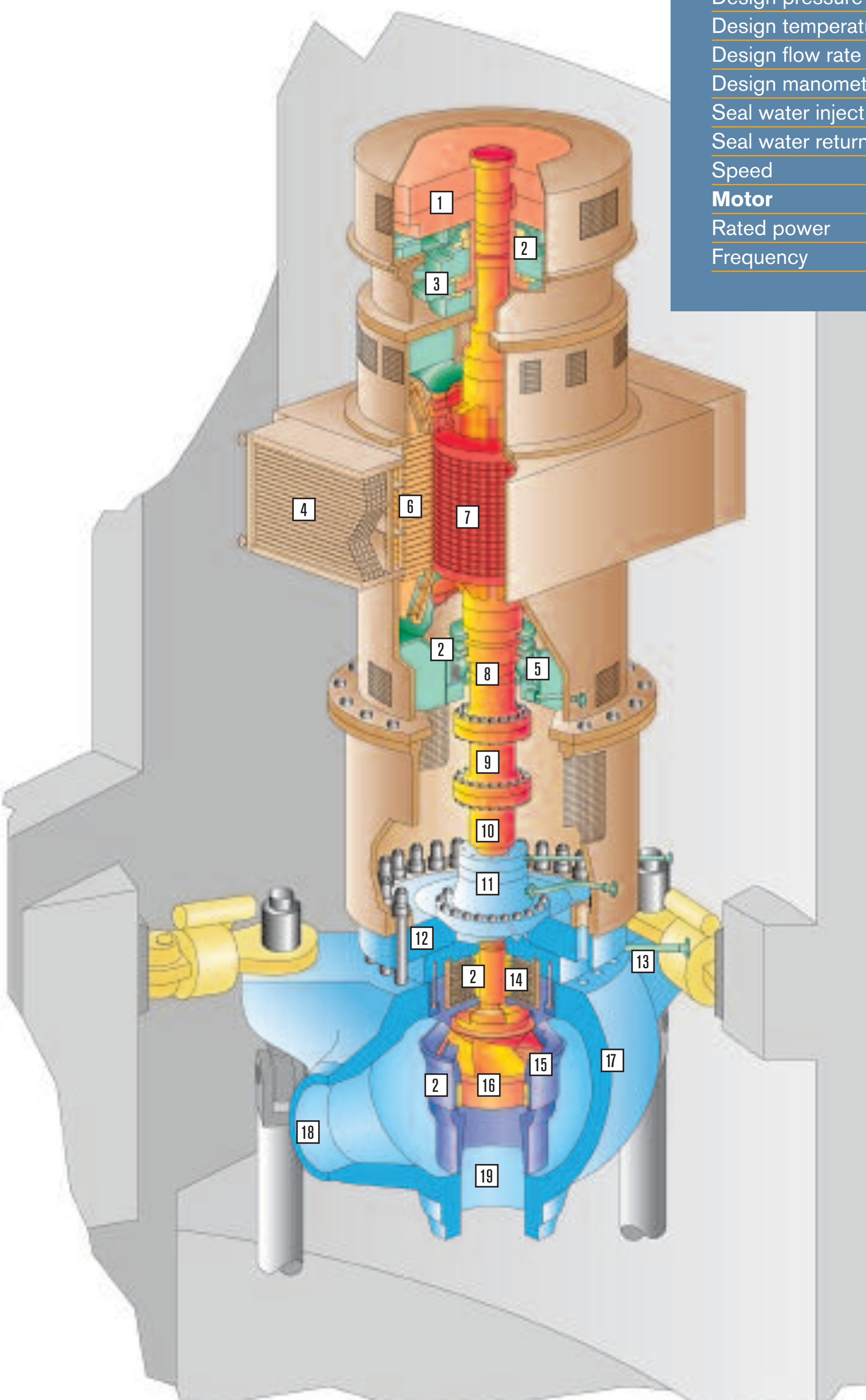
This feature ensures that even in case of total station blackout or failure of the main seals no loss of coolant would occur.

• **The motor** is a drip-proof squirrel-cage induction motor.

All parts of the reactor coolant pump are replaceable. Pump internals can be easily removed from the casing. The spool piece between the pump shaft and the motor shaft enables rapid maintenance of the controlled leakage seal with the motor in place.

CHARACTERISTICS	DATA
Reactor coolant pumps	
Number	4
Overall height	9.3 m
Overall mass w/o water and oil	112 t
Pump	
Design pressure	176 bar
Design temperature	351 °C
Design flow rate	28,330 m^3/h
Design manometric head	100.2 m ± 5%
Seal water injection	1.8 m^3/h
Seal water return	0.680 m^3/h
Speed	1,485 rpm
Motor	
Rated power	9,000 kW
Frequency	50 Hz

Reactor coolant pump cutaway



1 Flywheel
2 Radial bearings
3 Thrust bearing
4 Air cooler
5 Oil cooler
6 Motor (stator)
7 Motor (rotor)
8 Motor shaft
9 Spool piece
10 Pump shaft
11 Shaft seal housings
12 Main flange
13 Seal water injection
14 Thermal barrier heat exchanger
15 Diffuser
16 Impeller
17 Pump casing
18 Discharge
19 Suction

An enhanced version of the reactor coolant pump in operation on N4 plants which is characterized by the very low vibration level of its shaft line.

- **The shaft seal system consists of three dynamic seals staggered into a cartridge and a standstill seal.**
- **The standstill seal ensures that, in case of station blackout or failure of the shaft seals after the reactor coolant pump is at rest, no loss of coolant would occur.**
- **The shaft spool piece and the shaft seal cartridge design enable quick maintenance of the shaft seal with the motor in place.**

Jeumont manufacturing plant (France): reactor coolant pump (N4,1,500 MWe).

Chalon manufacturing plant (France): machining of primary piping elbow.

CHARACTERISTICS	DATA
Main coolant lines	
Primary loops	
Inside diameter of straight portions	780 mm
Thickness of straight portions	76 mm
Material	Low carbon austenitic stainless steel
Surge line	
Inside diameter	325.5 mm
Thickness	40.5 mm
Material	Low carbon austenitic stainless steel

Main Coolant Lines

The piping of the four primary loops and the pressurizer surge line are part of the Reactor Coolant System installed in the reactor building. The reactor main coolant lines convey the reactor coolant from the reactor pressure vessel to the steam generators and then to the reactor coolant pumps, which discharge it back to the reactor pressure vessel.
The surge line connects one of the four primary loops with the pressurizer.

Each of the four reactor coolant loops comprises:
– a hot leg, from the reactor pressure vessel to a steam generator,
– a cross-over leg, from the steam generator to a reactor coolant pump,
– a cold leg, from the reactor coolant pump to the reactor pressure vessel.

A large inner diameter of 780 mm was chosen for all the legs to minimize the pressure drop and to reduce the coolant flow velocity in the coolant lines.

The surge line routing has been designed to avoid thermal stratification during steady state operation.

The main coolant line materials and manufacturing processes have been selected to yield a high quality product with high toughness properties, and to improve inspectability and significantly reduce the number of welds.

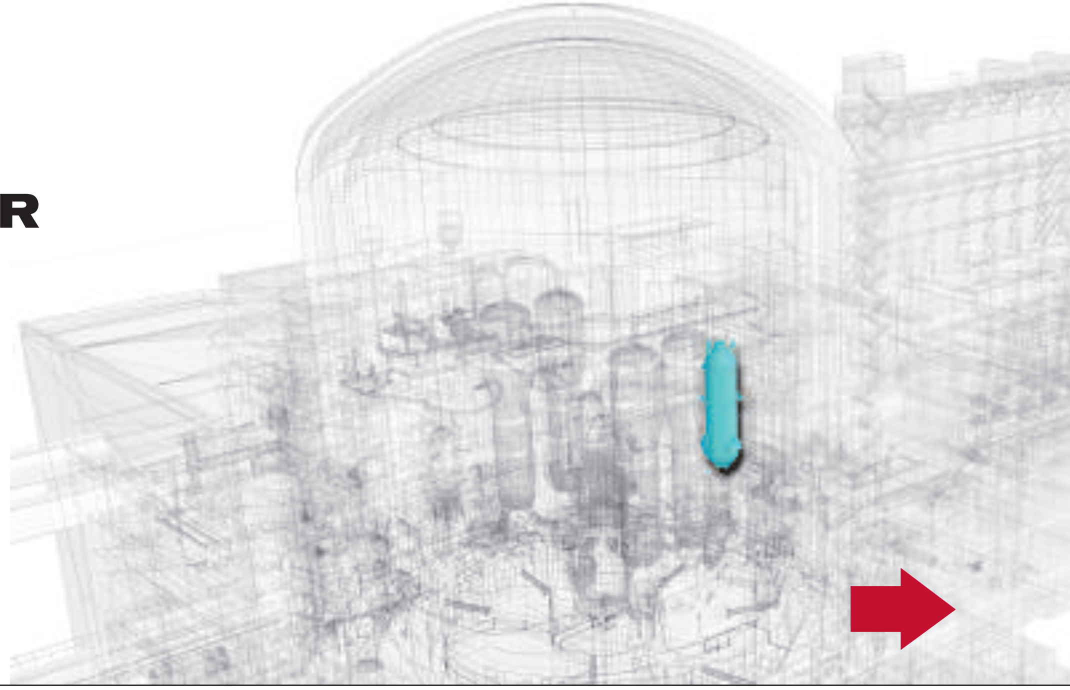
As already experienced on N4 reactors at the Civaux site, the material is a forged austenitic steel, which exhibits excellent resistance to thermal aging and permeability for ultrasonic testing. The hot leg is forged, with separate forged elbows. The cold leg is made using “one-piece technology” with an elbow machined out of the forging. The cross-over leg is made of three parts, mainly for erection convenience. The surge line also consists of several segments. Major advances concerning welding processes are implemented. The homogeneous circumferential welds are made using the orbital narrow gap TIG welding technology. The weld is made with an automatic TIG machine, which enables a large reduction of the volume of weld metal and an enhanced quality level. The bimetallic weld joining austenitic to ferritic parts (like reactor pressure vessel or steam generator nozzles) is made by direct automatic narrow gap welding of Inconel 52.

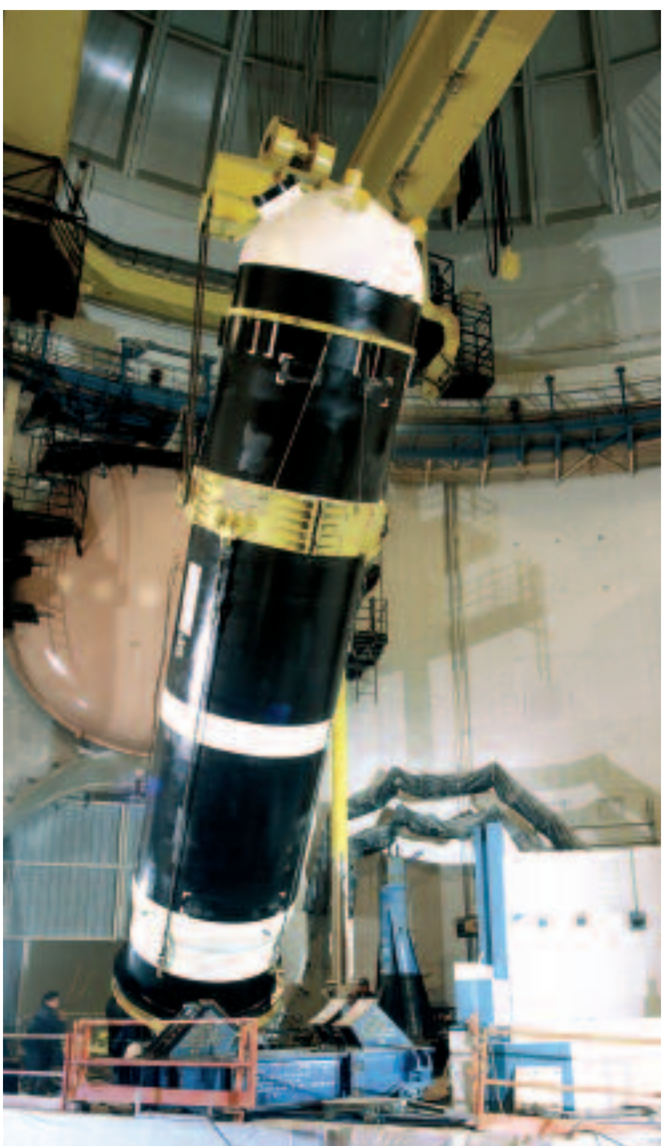
Several nozzles, branches and piping connections are mounted on each leg for auxiliary and instrumentation lines. Large nozzles are integral with the main coolant lines. They are machined out of the forging of the piping. Small nozzles are set on welded, except for the nozzles of the Chemical and Volume Control System, which are integral with the main coolant line to improve their resistance to thermal fatigue.

These design improvements strongly contribute to the capability for the main coolant lines to fulfill the Leak Before Break requirements.

- **The main coolant lines design and material are based on the technology already implemented on N4 reactor at the Civaux site.**
- **They are made of forged austenitic stainless steel parts (piping and elbows) with high mechanical strength, no sensitivity to thermal aging and are well suited to in-service ultrasonic inspection.**
- **Large nozzles for connection to auxiliary lines are integral and machined out of the forged piping (same for the Chemical and Volume Control System nozzles to avoid thermal fatigue effects).**
- **The main coolant lines design and material provide justification of the application of the Leak Before Break concept.**

PRESSURIZER





Pressurizer erection in a reactor building.

The pressurizer (PZR) role is to maintain the pressure of the primary circuit inside prescribed limits. It is a part of the primary circuit, and is connected through a surge line to the hot leg of one of the four loops of that circuit.

The pressurizer is a vessel containing primary water in its lower part, and steam water in its upper part. To accommodate some primary coolant volume variation, the pressurizer is equipped with electric heaters at its bottom to vaporize more liquid water, and with a spray system at its top to condense more steam. Compared to previous designs, the volume of the EPR pressurizer has been significantly increased in order to smooth the response to operational transients. This improvement provides a gain in terms of equipment life duration and a gain in terms of time available to counteract potential abnormal situations in operation.

Relief and safety valves at the top of the pressurizer protect the primary circuit against overpressure. Compared to previous designs, the EPR features an additional set of motorized valves; in case of postulated accident with a risk of core melting, these valves would provide the operator with an additional efficient mean to rapidly depressurize the primary circuit and avoid a high pressure core melt situation.

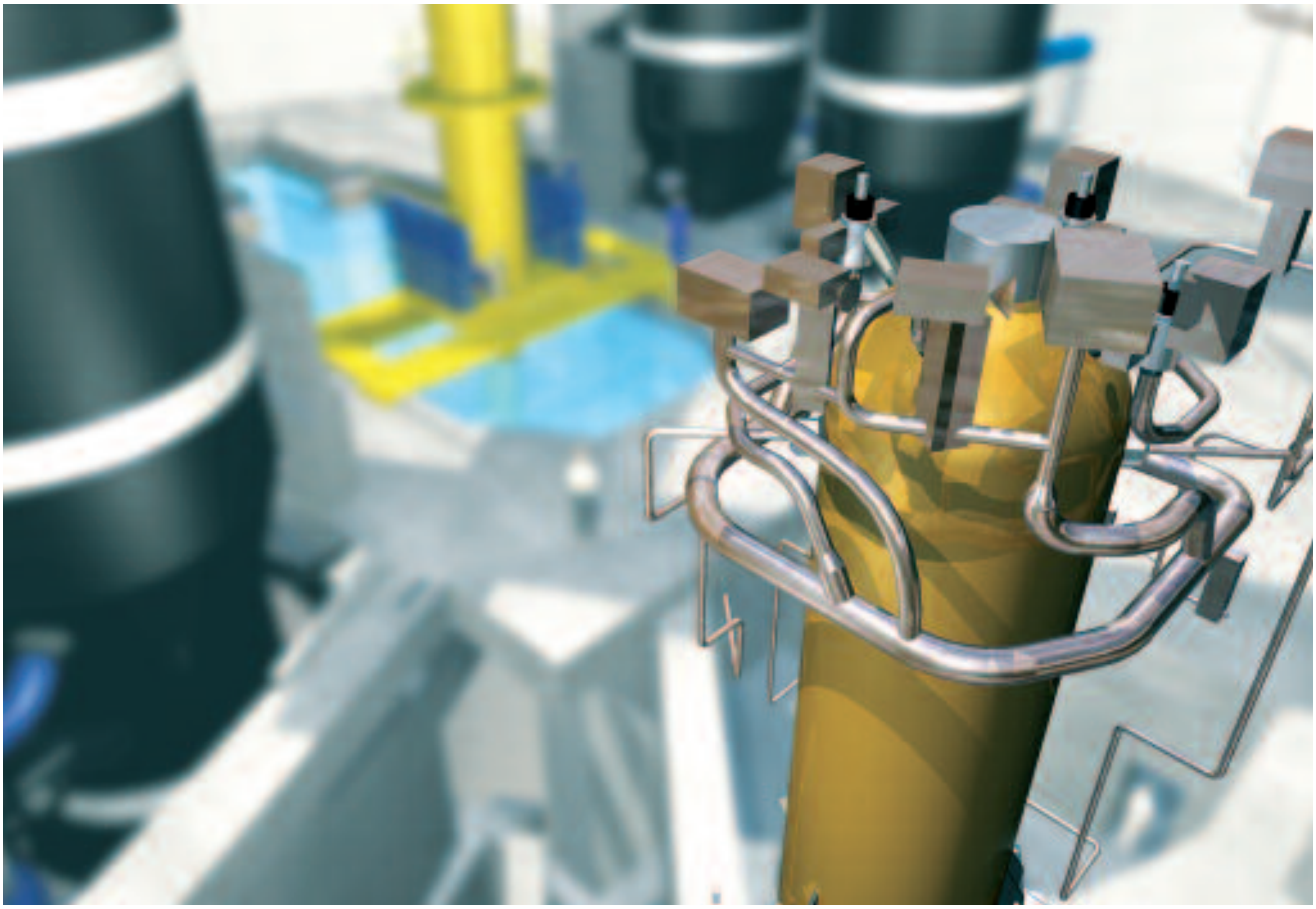
A number of construction provisions have improved maintainability. In particular, a floor between the pressurizer head and the valves eases heater replacement and reduces radiological dose during valve service.

All the pressurizer boundary parts, with the exception of the heater penetrations, are made of forged ferritic steel with two layers of cladding. The steel grade is the same as that for the reactor pressure vessel. The heater penetrations are made of stainless steel and welded with Inconel.

The pressurizer is supported by a set of brackets welded to the main body. Lateral restraints would preclude rocking in the event of a postulated earthquake or accident.

CHARACTERISTICS	DATA
Pressurizer	
Design pressure	176 bar
Design temperature	362 °C
Total volume	75 m^3
Total length	14.4 m
Base material	Low alloy ferritic steel
Cylindrical shell thickness	140 mm
Number of heaters	108
Total weight, empty	150 t
Total weight, filled with water	225 t
Number and capacity of safety valve trains	3 x 300 t/h
Depressurization valves capacity	900 t/h

Computer-generated image of the EPR pressurizer head with its safety and relief valves.



The pressurizer has a larger volume to smooth the operating transients in order to:

- **ensure the equipment 60-year design life,**
- **increase the time available to counteract an abnormal operating situation.**

Maintenance and repair (concerning safety valves, heaters) are facilitated and radiological doses are reduced.

A dedicated set of valves for depressurizing the primary circuit is installed on the pressurizer, in addition to the usual relief and safety valves, to prevent the risk of high pressure core melt accident.

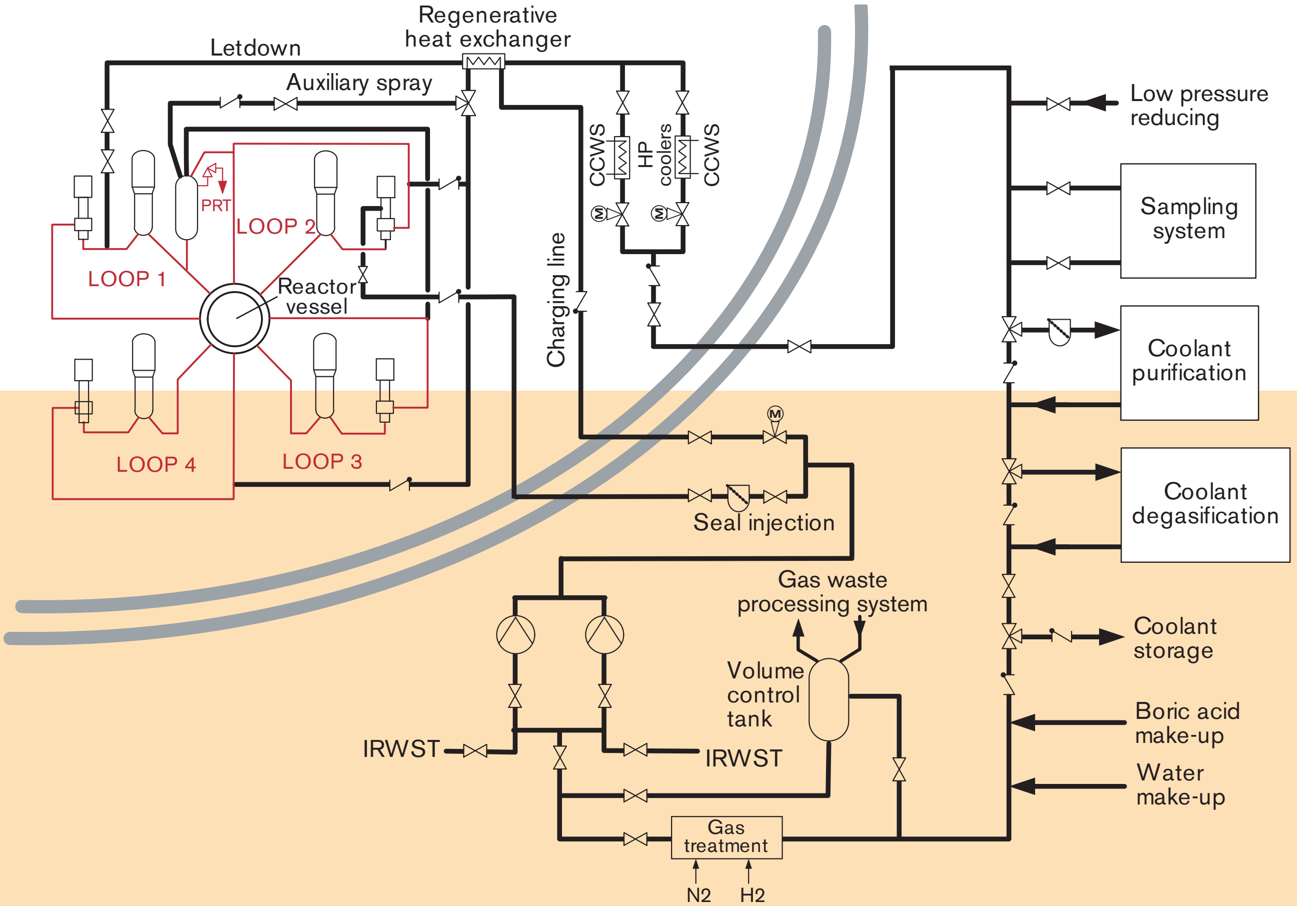
SYSTEMS

CHEMICAL AND VOLUME CONTROL

The Chemical and Volume Control System (CVCS) performs several operational functions.

- Continuous controls the water inventory of the Reactor Coolant System (RCS) during all normal plant operating conditions, using the charging and letdown flow.
- Adjusts the RCS Boron concentration as required for control of power variations and for plant start-up or shutdown, or to compensate for core burnup, using demineralized water and borated water.
- Ensures permanent monitoring of the Boron concentration of all fluids injected into the RCS, control of the concentration and the nature of dissolved gases in the RCS by providing the means of injecting the required Hydrogen content into the charging flow and allowing degassing of the letdown flow.
- Enables the adjustment of the RCS water chemical characteristics by allowing injection of chemical conditioning agents into the charging flow.
- Ensures a high flow rate capability for primary coolant chemical control with coolant purification, treatment, degassing and storage.
- Injects cooled, purified water into the Reactor Coolant Pump (RCP) seals system to ensure cooling and leaktightness and collection of the seal leakage flow.
- Supplies borated water to the RCS up to the concentration required for a cold shutdown condition and for any initial condition.
- Allows a reduction in pressure by condensing steam in the pressurizer by diverting the charging flow to the auxiliary pressurizer spray nozzle in order to reach Residual Heat Removal System (SIS/RHRS) operating conditions.
- Allows filling and draining of the RCS during shutdown.
- Provides a pressurizer auxiliary spray, if the normal system cannot perform its function, and make-up of the RCS in the event of loss of inventory due to a small leak.
- Ensures the feed and bleed function.

Chemical and Volume Control System



SAFETY INJECTION / RESIDUAL HEAT REMOVAL

The Safety Injection System (SIS/RHRS) comprises the Medium Head Safety Injection System, the Accumulators, the Low Head Safety Injection System and the In-Containment Refueling Water Storage Tank (IRWST). The system performs a dual function both during the normal operating conditions in RHR mode and in the event of an accident.

The system consists of four separate and independent trains, each providing the capability for injection into the RCS by an Accumulator, a Medium Head Safety Injection (MHSI) pump and a Low Head Safety Injection (LHSI) pump, with a heat exchanger at the pump outlet.

During normal operating conditions, the system in RHR mode:

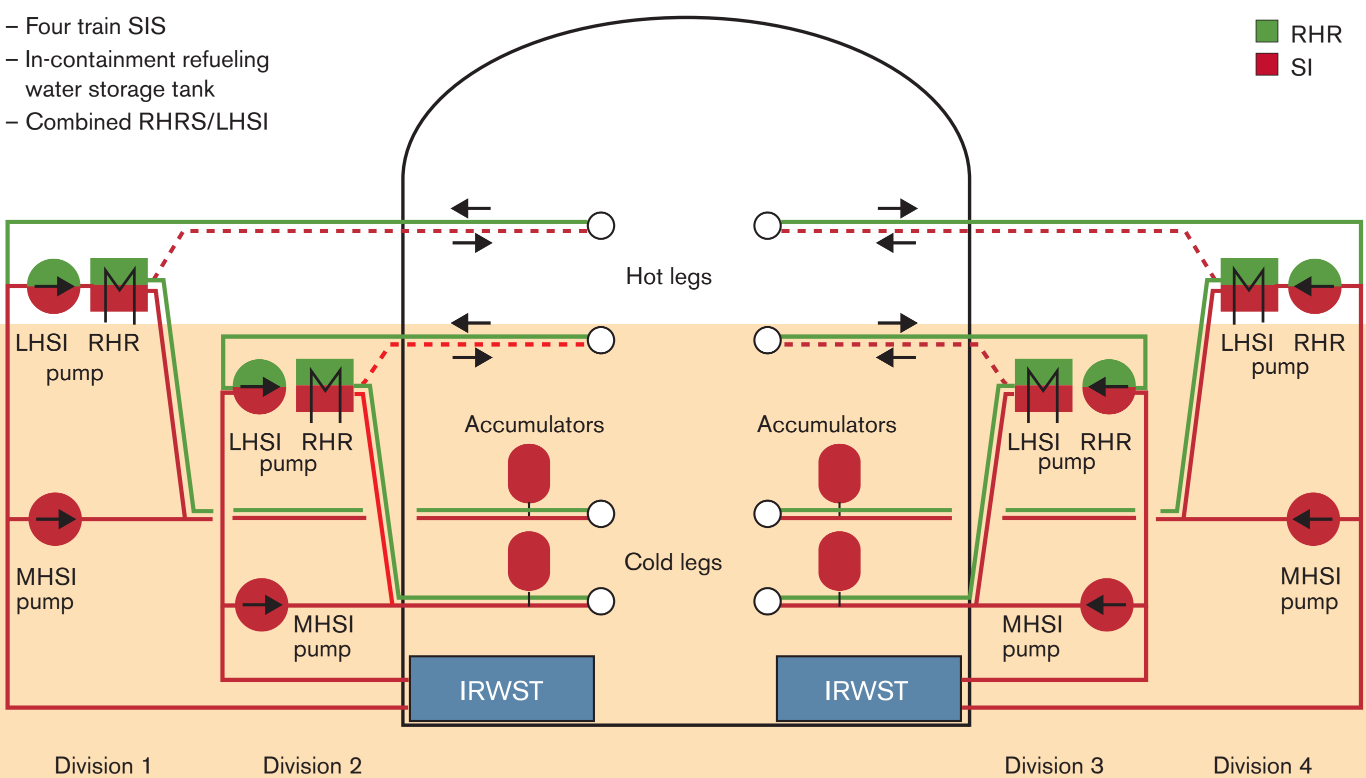
- provides the capability for heat transfer from the RCS to the Component Cooling Water System (CCWS) when heat transfer via the Steam Generators (SG) is no longer sufficiently effective (at an RCS temperature of less than 120 °C in normal operation),
- transfers heat continuously from the RCS or the reactor refueling pool to the CCWS during cold shutdown and refueling shutdown, as long as any fuel assemblies remain inside the containment.

In the event of an assumed accident and in conjunction with the CCWS and the Essential Service Water System (ESWS), the SIS in RHR mode maintains the RCS core outlet and hot leg temperatures below 180 °C following a reactor shutdown.

The four redundant and independent SIS/RHRS trains are arranged in separate divisions in the Safeguard Buildings. Each train is connected to one dedicated RCS loop and is designed to provide the necessary injection capability required to mitigate accident conditions. This configuration greatly simplifies the system design.

The design also makes it possible to have extended periods available for carrying out preventive maintenance or repairs. For example, preventive maintenance can be carried out on one complete safety train during power operation.

SI/RHR System



In safety injection mode, the main function of the SIS is to inject water into the reactor core following a postulated loss of coolant accident in order to compensate for the consequence of such events. It would be also activated during a steam generator tube rupture or during loss of a secondary-side heat removal function.

The MHSI system injects water into the RCS at a pressure (92 bar at mini-flow) set to prevent overwhelming the secondary side safety valves (100 bar) in the event of steam generator tube leaks. The accumulators and the LHSI system also inject water into the RCS cold legs when the primary pressure is sufficiently low (accumulator: 45 bar, LHSI: 21 bar at mini-flow).

IN-CONTAINMENT REFUELING WATER STORAGE TANK (IRWST)

The IRWST is a tank that contains a large amount of borated water, and collects water discharged inside the containment.

Its main function is to supply water to the SIS, Containment Heat Removal System (CHRS) and Chemical and Volume Control System (CVCS) pumps, and to flood the spreading area in the event of a severe accident.

The tank is located at the bottom of the containment below the operating floor, between the reactor cavity and the missile shield.

During the management of a postulated accident, the IRWST content should be cooled by the LHSI system.

Screens are provided to protect the SIS, CHRS and CVCS pumps from debris that might be entrained with IRWST fluid under accident conditions.

EMERGENCY FEEDWATER

The Emergency Feedwater System (EFWS) is designed to ensure that water is supplied to the steam generators when all the other systems that normally supply them are unavailable.

Its main safety functions are to:
- transfer heat from the RCS via the steam generators to the atmosphere, down to the connection of the RHRS following any plant incidents other than those involving a reactor coolant pressure boundary rupture; this is done in conjunction with the discharge of steam via the Main Steam Relief Valves (MSRV),
- ensure that sufficient water is supplied to the steam generators following a loss of coolant accident or a steam generator tube rupture accident,
- rapidly cool the plant down to LHSI conditions following a small loss of coolant associated with total MHSI failure, in conjunction with steam release from the Main Steam Relief Valves (MSRV).

This system consists of four separate and independent trains, each providing injection capability through an emergency pump that takes suction from an EFWS tank.

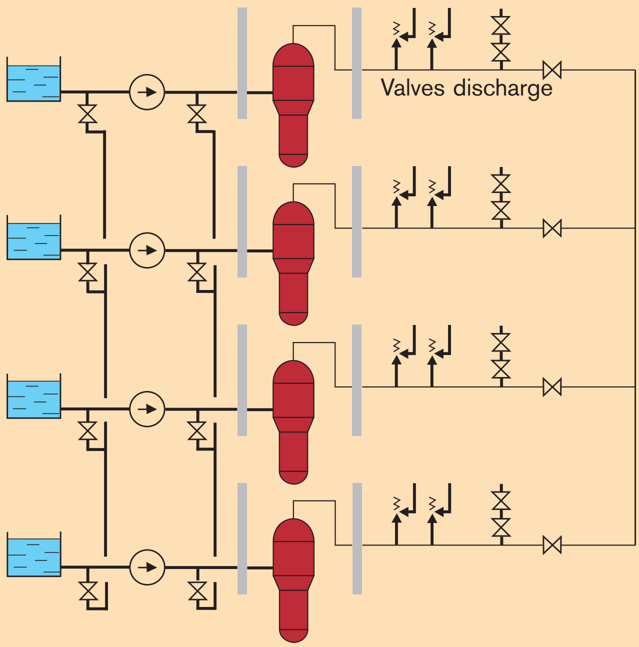
For start-up and operation of the plant, a dedicated system, separate from EFWS, is provided.

OTHER SAFETY SYSTEMS

The Extra Borating System (EBS) ensures sufficient boration of the RCS for transfer to the safe shutdown state with the Boron concentration required for cold shutdown. This system consists of two separate and independent trains, each capable of injecting the

Emergency Feedwater System (EFWS)

- Interconnecting headers at EFWS pump suction and discharge normally closed.
- Additional diverse electric power supply for 2/4 trains, using two smalls Diesel generator sets.



total amount of concentrated boric acid required to reach the cold shutdown condition from any steady state power operation.

Outside the containment, part of **the Main Steam System (MSS)** is safety classified. This part consists of four geographically separated but identical trains. Each includes one main steam isolation valve, one main steam relief valve, one main steam relief isolation valve and two spring-loaded main steam safety valves.

Outside the containment, part of **the Main Feedwater System (MFS)** is safety classified. It consists of four geographically separated but identical trains. Each includes main feedwater isolation and control valves.

In addition to the safety systems described above, other safety functions are performed **to mitigate postulated severe accidents,** as described in the section dealing with safety and severe accidents.

COMPONENT COOLING WATER

The Component Cooling Water System (CCWS) transfers heat from the safety related systems, operational auxiliary systems and other reactor equipment to the heat sink via the Essential Service Water System (ESWS) under all normal operating conditions.

The CCWS also performs the following safety functions:
- removes heat from the SIS/RHRS to the ESWS,
- removes heat from the Fuel Pool Cooling System (FPCS) to the ESWS for as long as any fuel assemblies are located in the spent fuel storage pool outside the containment,
- cools the thermal barriers of the Reactor Coolant Pump (RCP) seals,
- removes heat from the chillers in divisions 2 and 3 and cools the Containment Heat Removal System (CHRS) by means of two separate trains.

The CCWS consists of four separate safety trains corresponding to the four divisions of the safeguard buildings.

ESSENTIAL SERVICE WATER

The Essential Service Water System (ESWS) consists of four separate safety trains which cool the CCWS heat exchangers with water from the heat sink during all normal plant operating conditions and during incidents and accidents. This system also includes two trains of the dedicated cooling chain for conditions associated with the mitigation of postulated severe accidents.

OTHER SYSTEMS

Other systems include the Nuclear Sampling, Nuclear Island Vent and Drain, Steam Generator Blowdown, and Waste Treatment Systems.
- The **Nuclear Sampling System** is used for taking samples of gases and liquid from systems and equipment located inside the reactor containment.
- The **Vent and Drain System** collects gaseous and liquid waste from systems and equipment so that it can be treated.
- The **Steam Generator Blowdown System** prevents the build-up of solid matter in the secondary side water.
- The **Waste Treatment System** ensures the treatment of solid, gaseous and liquid wastes.

BACK-UP FUNCTIONS IN THE EVENT OF TOTAL LOSS OF THE REDUNDANT SAFETY SYSTEMS

➡ **the loss of secondary side heat removal is backed up by primary side feed and bleed through an appropriately designed and qualified primary side overpressure protection system,**

➡ **the combined function comprising secondary side heat removal, accumulator injection and the LHSI systems can replace the MHSI system in the event of a small break loss of coolant accident,**

➡ **similarly, complete loss of the LHSI system is backed up by the MHSI system and by the Containment Heat Removal System (CHRS) for IRWST cooling.**

SAFETY SYSTEMS AND FUNCTIONS

➡ **Simplification by separation of operating and safety functions.**

➡ **Fourfold redundancy applied to the safeguard systems and to their support systems. This architecture allows their maintenance during plant operation, thus ensuring a high plant availability factor.**

➡ **The different trains of the safety systems are located in four different buildings in which strict physical separation is applied.**

➡ **With systematic functional diversity, there is always a diversified system which can perform the desired function and bring the plant back to a safe condition in the highly unlikely event of a redundant system becoming totally unavailable.**

POWER SUPPLY

The outline design of the power supply system is shown below.

The Emergency Power Supply is designed to ensure that the safety systems are powered in the event of loss of the preferred electrical sources.

It is designed as four separate and redundant trains arranged in accordance with the four division concept. Each train is provided with an Emergency Diesel Generator (EDG) set.

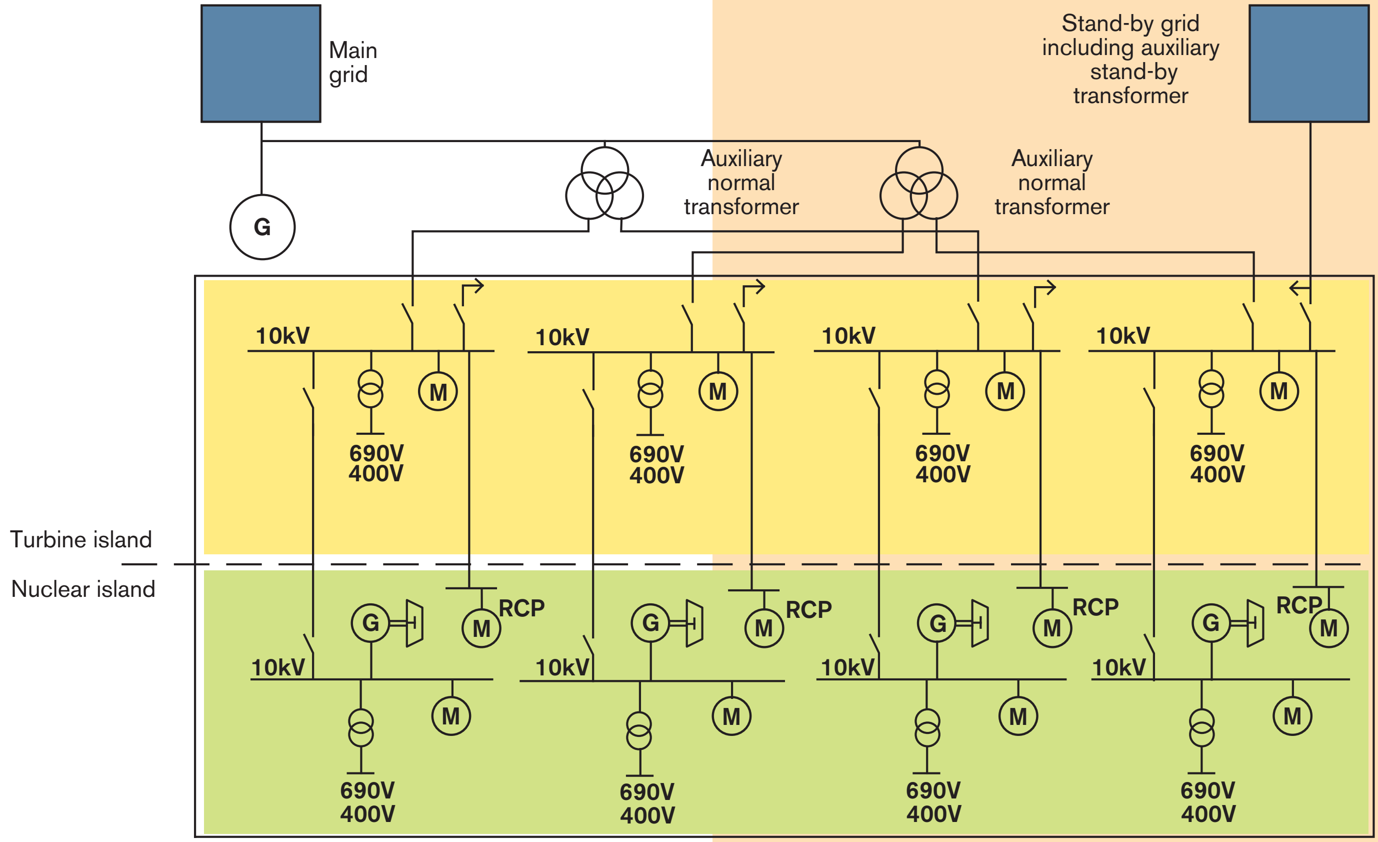
The emergency power supply system is designed to meet the requirements of the N+2 concept (i.e. assuming a single failure on one train and a maintenance operation on another).

The safety loads connected to the emergency power supply correspond to those required to safely shut down the reactor, remove the residual and stored heat and prevent release of radioactivity.

In the event of total loss of the four EDGs (Station BlackOut or SBO), two additional generators, the SBO Emergency Diesel Generators, provide the necessary power to the emergency loads. They are connected to the safety busbars of two divisions.

Isar 2, Germany (Konvoi, 1,300 MWe) emergency Diesel generator.

Electrical systems of an EPR nuclear power station



FUEL HANDLING AND STORAGE

The reactor core is periodically reloaded with fresh fuel assemblies. The spent fuel assemblies are moved to and stored in the Spent Fuel Pool (SFP). These operations are carried out using several handling devices and systems (fuel transfer tube, spent fuel crane, fuel elevator, refueling machine and spent fuel cask transfer machine).

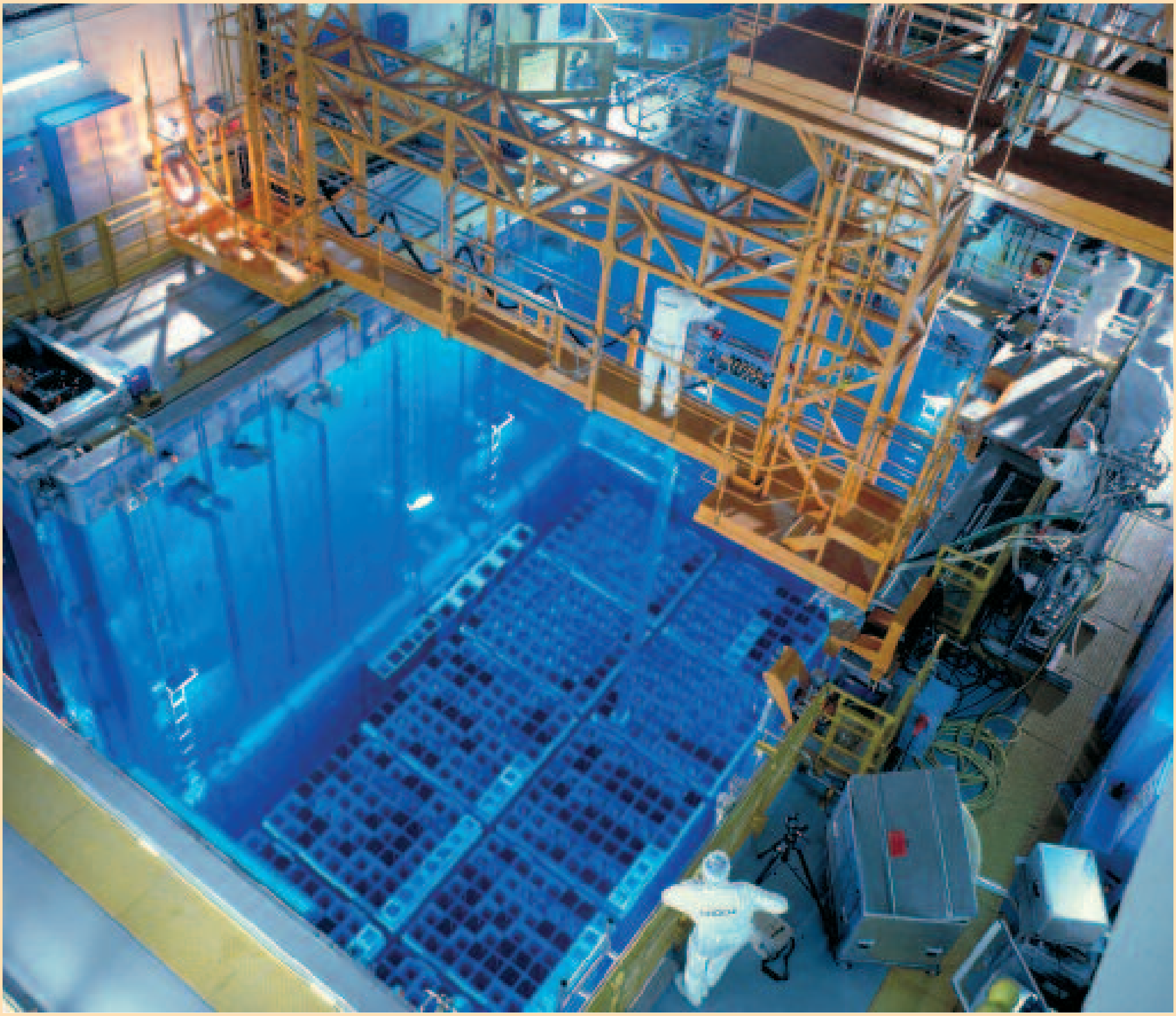
The underwater fuel storage racks are used for underwater storage of:

- fresh fuel assemblies, from the time they are delivered on site to the time they are loaded into the reactor core,
- spent fuel assemblies following fuel unloading from the core and prior to shipment out of the site.

The Fuel Pool Cooling and Purification System (FPCPS) is divided into two subsystems: the Fuel Pool Cooling System (FPCS) and the Fuel Pool Purification System (FPPS).

The FPCS provides the capability for heat removal from the SFP and is designed to keep the SFP temperature at the required level during normal plant operation (power operation and refueling outage). This system is arranged in a two separate and independent train configuration with two FPCS pumps operating in parallel in each train.

The FPPS comprises a purification loop for the SFP, a purification loop for the reactor pool and the IRWST, and skimming loops for the SFP and the reactor pool. The system includes two cartridge filters, a demineralizer and a resin trap filter used for purification of pool water.



Chooz B1, France (N4, 1,500 MWe) fuel building.

INSTRUMENTATION & CONTROL SYSTEM

A nuclear power plant, like any other industrial facility, needs technical means to monitor and control its processes and equipment. These means, as a whole, constitute the plant Instrumentation & Control (I & C) processes, which actually comprises several systems and their electrical and electronic equipment.

Basically, the I & C system is composed of sensors to transform physical data into electrical signals, programmable controllers to process these signals and control actuators, monitoring and control means at the disposal of the operators.

The overall design of the I & C system and associated equipment has to comply with requirements imposed by the process, nuclear safety and operating conditions.

To design the EPR and its I & C system, specific attention has been given to ensure a high level of operational flexibility in order to fit with electricity companies' needs. As a result, the EPR is particularly well adapted to load follow and remote control operation modes.

➡ **A plant I & C system, completely computerized, supported by the most modern digital technologies, for high-level operational flexibility**

EPR I & C OVERALL ARCHITECTURE

Inside the overall I & C architecture, each system is characterized depending on its functions (measurement, actuation, automation, man-machine interface) and its role in safety or operation of the plant.

A several level structure

Consideration of the different roles played by the different I & C systems leads to a several level structure for I & C architecture:
- level 0: process interface,
- level 1: system automation,
- level 2: process supervision and control.

(A level 3 deals with site management functions).
Different general requirements are assigned to each level.

The "process interface" (level 0) comprises the sensors, and the switchgears.

The "system automation" level (level 1) encompasses I & C systems to perform:
- reactor protection,
- reactor control, surveillance and limitation functions,
- safety automation,
- process automation.

The "process supervision and control" (level 2) consists of:
- the workstations and panels located in the Main Control Room, the Remote Shutdown Station and the Technical Support Centre, which are also called the Man-Machine Interface (MMI),
- the I & C systems which act as link between the MMI and the "system automation" level.

Safety classification

I & C functions and equipment are categorized into classes in accordance with their importance to safety. Depending on their safety class, I & C functions must be implemented using equipment having the appropriate quality level.

Redundancy, division, diversity and reliability

I & C systems and equipment of the EPR comply with the principles of redundancy, division and diversity enforced for designing EPR safety-related systems. As an illustration, the Safety Injection System and the Emergency Feedwater System, which consist of four redundant and independent trains, have four redundant and independent I & C channels.

Each safety-related I & C system is designed to satisfactorily fulfil its functions even if one of its channels is not available due to a failure and if, at the same time, another of its channels is not available for preventive maintenance reasons or due to an internal hazard (e.g. fire).

I & C systems and equipment participating in safety functions are specified with a level of availability in compliance with the safety probabilistic targets adopted to design the EPR.

➡ **A quadruple redundant safety-related I & C for a further increased level of safety.**

Description of the I & C architecture

Functional safety class		Equipment quality level
F1A	Functions required in case of accident to bring the reactor to controlled state.	E1A
F1B	Functions required after an accident to bring the reactor to safe state. Functions intended to avoid the risk of radioactive releases.	E1B
F2	Other functions contributing to plant safety (adherence to limit operating conditions, surveillance of safety system availability, protection against the effects of internally-generated hazards, detection/monitoring of radioactive releases, functions used in post-accident operation…).	E2
NC	Non-classified functions.	NC

A computerized screen-based control room designed to maximize operator efficiency. Chooz B1, France (N4, 1,500 MWe).

I & C technology

Concerning I & C technology, AREVA NP uses a consistent I & C system based on its TELEPERM-XS technology for safety applications and on a diversified technology for standard applications.

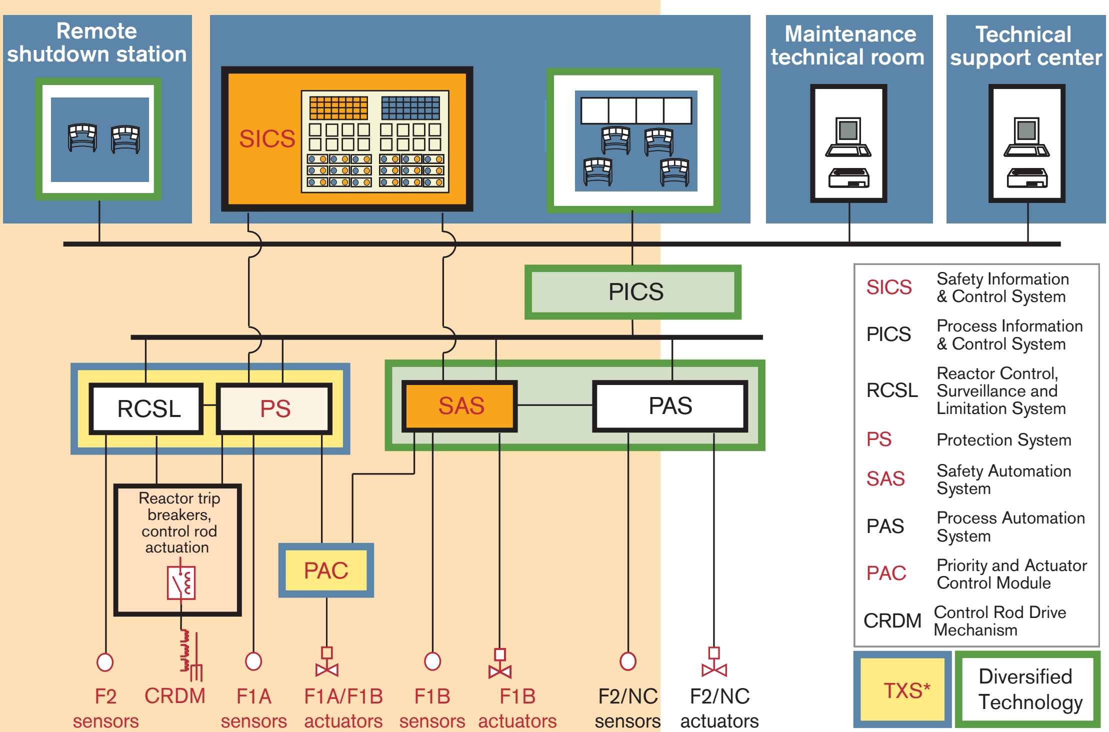
ROLE OF THE I & C SYSTEMS

The I & C systems act in accordance with the “defense in depth” concept.

Three lines of defense are implemented:
- the control system maintains the plant parameters within their normal operating ranges,
- in case a parameter leaves its normal range, the limitation system generates appropriate actions to prevent protective actions from having to be initiated,
- if a parameter exceeds a protection threshold, the reactor protection system generates the appropriate safety actions (reactor trip and safeguard system actuation).

Normally, to operate and monitor the plant, the operators use workstations and a plant overview panel in the Main Control Room. In case of unavailability of the Main Control Room, the plant is monitored and controlled from the Remote Shutdown Station.

I & C architecture



*TELEPERM-XS AREVA NP technology.

Instrumentation (level 0)

A number of instrumentation channels supply measured data for control, surveillance and protection systems and for information of the control room staff. Multiple-channel acquisition is used for important controls such as control of pressure and temperature of the primary coolant, liquid level in the reactor pressure vessel. Multiple-channel and diversified data acquisition means are implemented.

Concerning the protection of the reactor, a major aspect is the capacity to predict and measure the nuclear power (or neutron flux) level and the three dimensional distribution of power in the core.

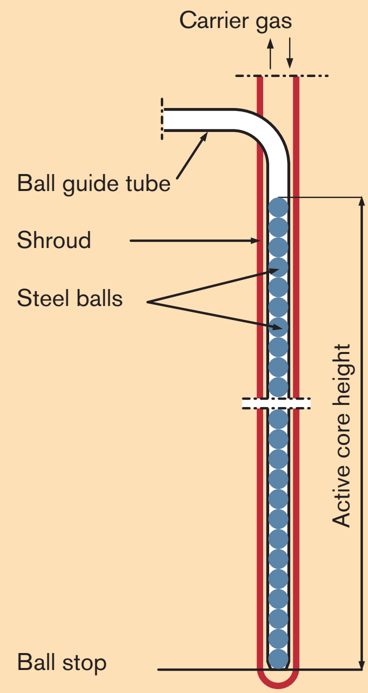
The measurement of the power level is performed using ex-core instrumentation which also provides signals to monitor the core criticality. Relying on temperature measurements in the cold and hot legs of the four primary loops, a quadruple-redundant primary heat balance is achieved and complemented by neutron flux measurements with very short response time.

Prediction and measurement of the three-dimensional power distribution relies on two types of in-core instrumentation:
- "movable" reference instrumentation to validate the core design and to calibrate the other sensors utilized for core surveillance and protection purposes,
- "fixed" instrumentation to deliver online information to the surveillance and protection systems which actuate appropriate actions and countermeasures in case of anomalies or exceeding of predefined limits.

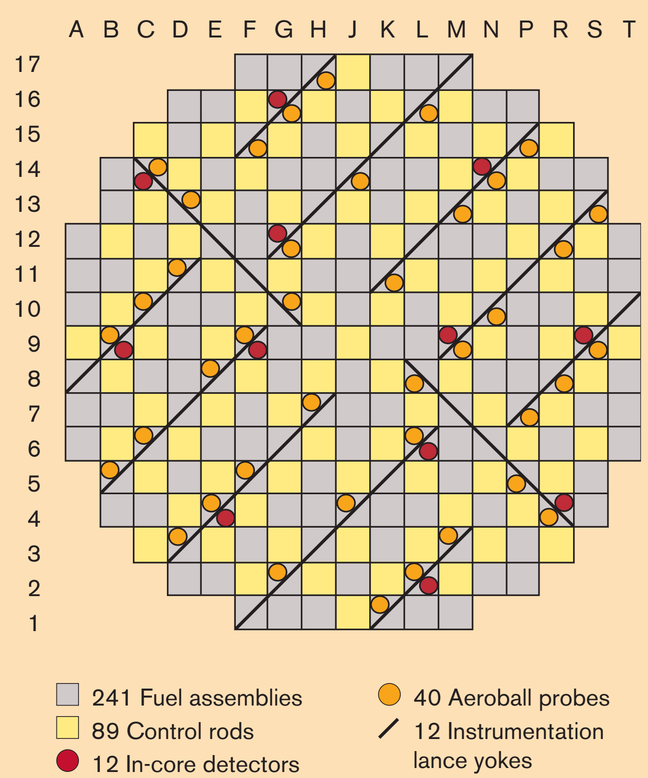
The movable reference instrumentation for power distribution assessment is an "aeroball" system. Stacks of vanadium-alloy balls, inserted from the top of the pressure vessel, are pneumatically transported into the reactor core (inside guide thimbles of fuel assemblies), then, after three minutes in the core, to a bench where the activation of each probe is measured at 30 positions in five minutes. This gives values of the local neutron flux in the core, which are processed to construct the three-dimensional power distribution map.

The fixed in-core instrumentation consists of neutron detectors and thermocouples to measure the neutron flux radial and axial distribution in the core and temperature radial distribution at the core outlet. The neutron flux signals are utilized to control the axial power distribution, and for core surveillance and protection. The core outlet thermocouples continuously measure the fuel assembly outlet temperature and provide signals for core monitoring in case of loss of coolant event. They also provide information on radial power distribution and thermal-hydraulic local conditions.

Aeroball system



EPR in-core instrumentation



Limitation functions and protection of the reactor (level 1)

Four-channel limitation functions are implemented to rule out impermissible operational conditions that would otherwise cause reactor trip actions to be initiated. They also ensure that process variables are kept within the range on which the safety analysis is based, and they initiate actions to counteract disturbances that are not so serious as to require the protection system to trip the reactor.

The protection system counteracts accident conditions, first by tripping the reactor, then by initiating event-specific measures. As far as reasonably possible, two diverse initiation criteria are available for every postulated accident condition.

Reactor trip is actuated by cutting off the power to the electromagnetic gripping coils of the control rod drive mechanisms. All the control assemblies drop into the core under their own weight and instantaneously stop the chain reaction.

➡ **An enhanced and optimized degree of automated plant control, associated to an advanced Man-Machine interface for operator information and action.**

Man-Machine interface (level 2)

At the design stage of the EPR, due consideration has been given to the human factor for enhancing the reliability of operators' actions, during operation, testing and maintenance phases. This is achieved by applying appropriate ergonomic design principles and providing sufficiently long periods of time for the operators' response to encountered situations or events.

Sufficient and appropriate information is made available to the operators for their clear understanding of the actual plant status, including in the case of a severe accident, and for a relevant assessment of the effects of their actions.

The plant process is supervised and controlled from the Main Control Room which is equipped, regarding information and control, with:

- two screen-based workstations for the operators,
- a screen-based workstation for presenting information to the shift supervisor and the safety engineer,
- an additional workstation for a third operator to monitor auxiliary systems,
- an auxiliary panel enabling to bring the plant to cold shutdown using safety-grade displays and controls,
- large plant overview panel which gives information on the status and main parameters of the plant.

The Remote Shutdown Station is provided with the same information and data on the process as the Main Control Room.

The plant also comprises a Technical Support Centre. It is a room with access to all the data concerning the process and its control, to be used, in case of accident, by the technical team in charge of analysing the plant conditions and supporting the post accident

The EPR's computerized control room features control screens providing relevant summary information on the process (computer-generated picture).

SAFETY

Golfech 2, France (1,300 MWe): reactor pressure vessel and internals.

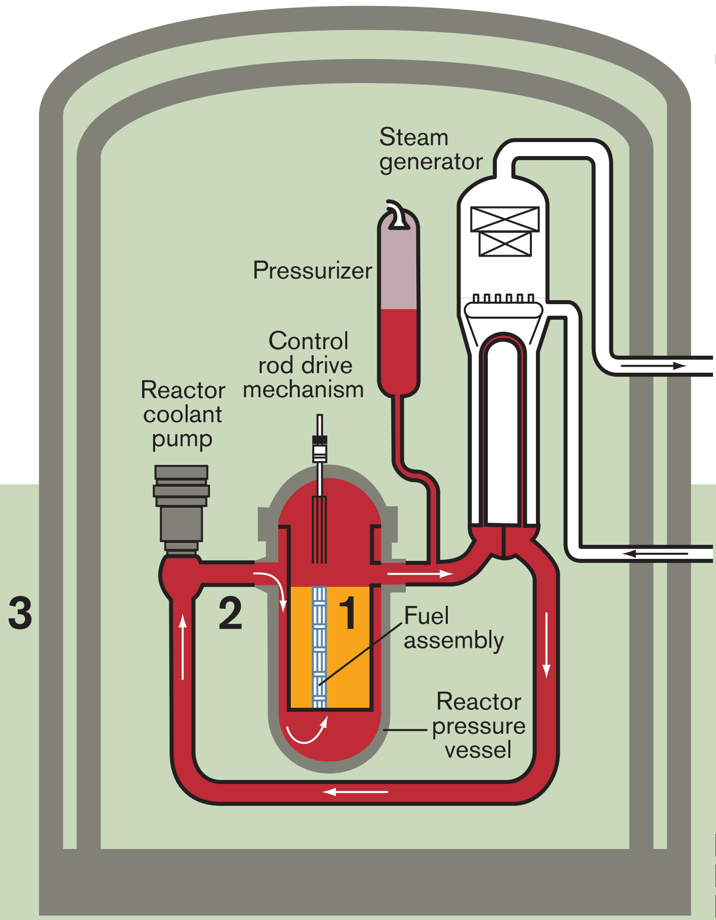
NUCLEAR SAFETY

The fission of atomic nuclei, performed in reactors to generate heat, brings into play large quantities of radiation-emitting radioactive substances from which people and the environment must be protected.
This explains the need for nuclear safety, which consists of the set of technical and organizational provisions taken at each stage in the design, construction and operation of a nuclear plant to ensure normal service, prevent the risks of an accident and limit its consequences in the unlikely event of its occurrence.

Nuclear reactor safety requires that three functions should be fulfilled at all times:
- control of the chain reaction, and therefore of the power generated,
- cooling of the fuel, including after the chain reaction has stopped, to remove residual heat,
- containment of radioactive products.

It relies upon two main principles:
- the three protective barriers,
- defense in depth.

The three protective barriers



1 Fuel cladding
2 Reactor coolant boundary
3 Reactor containment

THREE PROTECTIVE BARRIERS

The concept of the "three protective barriers" involves placing, between the radioactive products and the environment, a series of strong, leak-tight physical barriers to contain radioactivity in all circumstances:
- **first barrier:** the fuel, inside which most of the radioactive products are already trapped, is enclosed within a metal cladding,
- **second barrier:** the reactor coolant system is housed within a metal enclosure which includes the reactor vessel containing the core constituted by the fuel within its cladding,
- **third barrier:** the reactor coolant system is also enclosed within a high-thickness concrete construction (for the EPR, this construction is a double shell resting upon a thick basemat, whose inner wall is covered with a leak-tight metal liner).

➡ **The resistance and leaktightness of just one of these barriers is sufficient to contain the radioactive products.**

DEFENSE IN DEPTH

The concept of "defense in depth" involves ensuring the resistance of the protective barriers by identifying the threats to their integrity and by providing successive lines of defense which will guarantee high effectiveness:

- **first level:** safe design, quality workmanship, diligent operation, with incorporation of the lessons of experience feedback in order to prevent occurrence of failures,
- **second level:** means of surveillance for detecting any anomaly leading to departure from normal service conditions in order to anticipate failures or to detect them as soon as they occur,
- **third level:** means of action for mitigating the consequences of failures and prevent core melt down; this level includes use of redundant systems to automatically bring the reactor to safe shutdown; the most important of these systems is the automatic shutdown by insertion of the control rods into the core, which stops the nuclear reaction in a few seconds; in addition, a set of safeguard systems, also redundant, are implemented to ensure the containment of the radioactive products,
- **beyond,** the defense in depth approach goes further, as far as postulating the failure of all these three levels, resulting in a "severe accident" situation, in order to provide all the means of minimizing the consequences of such a situation.

➡ **By virtue of this defense in depth concept, the functions of core power and cooling control are protected by double or triple systems – and even quadruple ones as in the EPR.**

➡ **These systems are diversified to prevent a single failure cause from concurrently affecting several of the systems providing the same function.**

➡ **In addition, the components and lines of these systems are designed to automatically go to safe position in case of failure or loss of electrical or fluid power supply.**

The training for steam generator inspection illustrates:

➡ **the first level of defense in depth relating to the quality of workmanship,**

➡ **the second barrier, as the training relates to steam generator tubes which form part of the primary system.**

Lynchburg technical center (Va, USA): training for steam generator inspection.

EPR SAFETY

The first important choice, in line with the recommendations of the French and German Safety Authorities, was to build the EPR design upon an evolutionary approach based on the experience feedback from around 100 reactors previously built by AREVA NP. This choice enables AREVA NP to offer an evolutionary reactor based on the latest constructions (N4 reactors in France and KONVOI in Germany) and to avoid the risk arising from the adoption of unproven technologies.

This does not mean that innovative solutions, backed by the results of large-scale research and development programs, have been left out; indeed, they contribute to the accomplishment of the EPR progress objectives, especially in terms of safety and in particular regarding the prevention and mitigation of hypothetical severe accidents.

These progress objectives, motivated by the continuous search for a higher safety level, involve reinforced application of the defense in depth concept:

- by improving the preventive measures in order to further reduce the probability of core melt,
- by simultaneously incorporating, right from the design stage, measures for limiting the consequences of a severe accident.

➡ **A two-fold safety approach against severe accidents:**

- **further reduce their probability by reinforced preventive measures,**
- **drastically limit their potential consequences.**

DESIGN CHOICES FOR REDUCING THE PROBABILITY OF ACCIDENTS LIABLE TO CAUSE CORE MELT

In order to further reduce the probability of core melt, which is already extremely low for the reactors in the current nuclear power plant fleet, the advances made possible with the EPR focus on three areas:

- extension of the range of operating conditions taken into account right from design,
- the choices regarding equipment and systems, in order to reduce the risk of seeing an abnormal situation deteriorate into an accident,
- the advance in reliability of operator action.

The EPR complies with the safety objectives set up jointly by the French and German safety authorities for future PWR power plants:

➡ **further reduction of core melt probability,**

➡ **practical elimination of accident situations which could lead to large early release of radioactive materials,**

➡ **need for only very limited protective measures in area and time*, in case of a postulated low pressure core melt situation.**

* No permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long-term restriction in the consumption of food.

Extension of the range of operating conditions taken into account right from design

Provision for the shutdown states in the dimensioning of the protection and safeguard systems

The probabilistic safety assessments highlighted the importance that should be given to the reactor shutdown states. For the EPR, these shutdown states were systematically taken into account, both for the risk analyses and for the dimensioning of the protection and safeguard systems.

The use of the probabilistic safety assessments

Although the EPR safety approach is mainly based on the defense in depth concept (which is part of a deterministic approach), it is reinforced by probabilistic analyses. These make it possible to identify the accident sequences liable to cause core melt or to generate large radioactive releases, to evaluate their probability and to ascertain their potential causes so that they can be remedied. In their large scale right from the design phase, the probabilistic assessments conducted for the EPR constitute a world first. They have been a decisive factor in the technical choices intended to further strengthen the safety level of the EPR.

With the EPR, the probability of an accident leading to core melt, already extremely small with the previous-generation reactors, becomes infinitesimal:

- smaller than 1/100,000 (10^{-5}) per reactor/year, for all types of failure and hazard, which fully meets the objective set for the new nuclear power plants by the International Nuclear Safety Advisory Group (INSAG) with the International Atomic Energy Agency (IAEA) – INSAG 3 report,
- smaller than 1/1,000,000 (10^{-6}) per reactor/year for the events generated inside the plant, making a reduction by a factor 10 compared with the most modern reactors currently in operation,
- smaller than 1/10,000,000 (10^{-7}) per reactor/year for the sequences associated with early loss of the radioactive containment function.

Greater provision for the risk arising from internal and external hazards

The choices taken for the installation of the safeguard systems and the civil works minimize the risks arising from the various hazards (earthquake, flooding, fire, aircraft crash).

The safeguard systems are designed on the basis of a quadruple redundancy, both for the mechanical and electrical portions and for the I & C. This means that each system is made up of four sub-systems, or "trains", each one capable by itself of fulfilling the whole of the safeguard function. The four redundant trains are physically separated from each other and geographically shared among four independent divisions (buildings).

Each division includes:

- **for borated water safety injection into the reactor vessel in case of loss of coolant accident, a low-head injection system and its cooling loop, together with a medium-head injection system,**
- **a steam generator emergency feedwater system,**
- **the electrical systems and I & C linked to these systems.**

The building housing the reactor, the building in which the spent fuel is interim-stored, and the four buildings corresponding to the four divisions of the safeguard systems, are given special protection against externally-generated hazards such as earthquakes and explosions.

This protection is further strengthened against an airplane crash. The reactor building is covered with a double concrete shell: an outer shell made of 1.30 m thick reinforced concrete and an inner shell made of pre-stressed concrete and also 1.30 m thick which is internally covered with a 6 mm thick metallic liner. The thickness and the reinforcement of the outer shell on its own have sufficient strength to absorb the impact of a military or large commercial aircraft. The double concrete wall protection is extended to the fuel building, two of the four buildings dedicated to the safeguard systems, the main control room and the remote shutdown station which would be used in a state of emergency.

The other two buildings dedicated to the safeguard systems, those which are not protected by the double wall, are remote from each other and separated by the reactor building, which shelters them from simultaneous damage. In this way, should an aircraft crash occur, at least three of the four divisions of the safeguard systems would be preserved.

The choices regarding the equipment and systems, in order to reduce the risk of an abnormal situation deteriorating into an accident

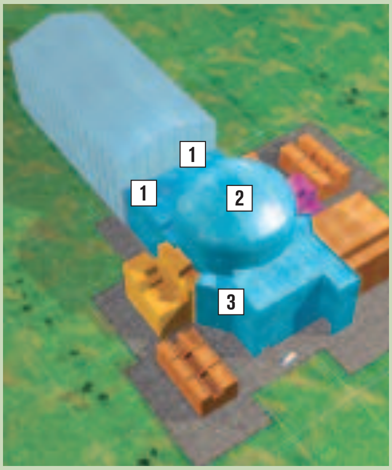
Elimination of the risk of a large reactor coolant pipe break

The reactor coolant system design, the use of forged pipes and components, construction with high mechanical performance materials, combined with the measures taken to detect leaks at the earliest time and to promote in-service inspections, practically rule out any risk of large pipe rupture.

The major safety systems comprise four sub-systems or trains, each capable of performing the entire safety function on its own. There is one train in each of the four safeguard buildings (1) surrounding the reactor building (2) to prevent a simultaneous failure of the trains.

➡ **A set of quadruple redundant safeguard systems, with independent and geographically separated trains, minimize consequences of potential internal and external hazards.**

➡ **This protection is even reinforced against the airplane crash risk by the strong double concrete shell implemented to shelter the EPR.**



The outer shell (5) covers the reactor building (2), the spent fuel building (3) and two of the four safeguard buildings (1). The other two safeguard buildings are separated geographically.

The reactor containment building has two walls: an inner prestressed concrete housing (4) internally covered with a metallic liner and an outer reinforced concrete shell (5), both 1.30 m thick.

Optimized management of accidental steam generator tube break

Steam generator tube break is an accident which, if it occurs, leads to a transfer of water and pressure from the primary system to the secondary system. The primary side pressure drop automatically induces a reactor shutdown then, if a given pressure threshold is reached, the activation of the safety injection of water into the reactor vessel. The choice, for the EPR, of a safety injection pressure (medium-head injection) lower than the set pressure of the secondary system safety valves prevents the steam generators from filling up with water in such a case. This has a dual advantage: it avoids the production of liquid releases and considerably reduces the risk of a secondary safety valve locking in open position.

Simplification of the safety systems and optimization of their redundancy and diversification

The safety-important systems and their support systems are – as already set out – quadrupled, each featuring four trains shared among four separate divisions.

The structure of these systems is straightforward and minimizes the changes that have to be made to their configuration depending on whether the reactor is at power or in shutdown; the design of the EPR safety injection system and residual heat removal system is an illustration of this.

The safety injection system, which would be activated in case of a loss of coolant accident, is designed to inject water into the reactor core to cool it down. In a first phase, water would be injected into the core via the cold legs of the reactor coolant system loops (legs located between the reactor coolant pumps and the reactor vessel). In the longer term, the water would be simultaneously injected via the cold and hot legs (legs located between the steam generators and the reactor vessel). The water reserve intended to feed the safety injection system is located on the inside and at the bottom of the reactor containment, and the injection pumps only take suction from this reserve. Therefore, there is no need (compared to previous designs) for switching over from a so-called “direct injection” phase to a “recirculation” phase. The EPR safety injection system is equipped with heat exchangers in its low-head portion, to be capable of ensuring core cooling on its own. The EPR is further equipped with a severe accident dedicated system for cooling the inside of the reactor containment, which would be only activated in the eventuality of an accident leading to core melt.

Residual heat removal is provided by the four trains of the low head portion of the safety injection system, which are then configured to remove the residual heat in closed loop (suction via the hot legs, discharge via the cold legs). Safety injection remains available for action in the eventuality of a leak or break occurring on the reactor coolant system.

➨ **The safety-related systems are simple, redundant and diversified to ensure reliability and efficiency.**

Increased reliability of operator action

Extension of action times available to the operator

The protection and safeguard actions needed in the short term in the eventuality of an incident or accident are automated. Operator action is not required before 30 minutes for an action taken in the control room, or one hour for an action performed locally on the plant.

The increase in the volumes of the major components (reactor pressure vessel, steam generators, pressurizer) gives the reactor extra inertia which helps to extend the time available to the operators to initiate the first actions.

Increased performance of the Man-Machine Interface

The progress accomplished in the digital I & C field and the analysis of the experience feedback from the design and operation of the N4 reactors, among the first plants to be equipped with a fully-computerized control room, have conferred on the EPR a high-performance, reliable and optimized solution in terms of Man-Machine Interface. The quality and relevance of the summary data on the reactor and plant status made available in real time to the operators further boost the reliability of their actions.

The ergonomics of the EPR control room benefits from the latest developments (computer-generated picture).

➨ **Design of components, high degree of automation, advanced solutions for I & C and Man-Machine Interface combine to further add to reliability of operator actions.**

DESIGN CHOICES FOR LIMITING THE CONSEQUENCES OF A SEVERE ACCIDENT

➡ **Although highly unlikely, a core melt accident would cause only very limited off-site measures in time and space.**

In response to the new safety model for the future nuclear power plants, introduced as early as 1993 by the French and German safety authorities, the plant design must be such that a core melt accident, although highly unlikely, causes only very limited off-site measures in time and space.

The policy of mitigation of the consequences of a severe accident, which guided the design of the EPR, therefore aimed to:

- ➡ **practically eliminate the situations which could lead to early important radiological releases, such as:**
  - **high-pressure core melt,**
  - **high-energy corium/water interaction,**
  - **Hydrogen detonation inside the reactor containment,**
  - **containment by-pass,**
- ➡ **ensure the integrity of the reactor containment, even in the eventuality of a low-pressure core melt followed by ex-vessel progression, through:**
  - **retention and stabilization of the corium inside the containment,**
  - **cooling of the corium.**

➡ **Practically, situations which could generate a significant radioactivity release are eliminated.**

Prevention of high-pressure core melt

In addition to the usual reactor coolant system depressurization systems on the other reactors, the EPR is equipped with valves dedicated to preventing high-pressure core melt in the eventuality of a severe accident. These valves would then ensure fast depressurization, even in the event of failure of the pressurizer relief lines.

Controlled by the operator, they are designed to safely remain in open position after their first actuation.

Their relieving capacity guarantees fast primary depressurization down to values of a few bars, precluding any risk of containment pressurization through dispersion of corium debris in the event of vessel rupture.

Prevention of high-energy corium/water interaction

The high mechanical strength of the reactor vessel is sufficient to rule out its damage by any reaction, even high-energy, which could occur on the inside between corium* and coolant.

The portions of the containment with which the corium would come in contact in the eventuality of a core melt exacerbated by ex-vessel progression – namely the reactor pit and the core spreading area – are kept "dry" (free of water) in normal operation. Only when it is spread inside the dedicated area, therefore already partially cooled, surface-solidified and less reactive, would the corium be brought into contact with the limited water flow intended to cool it down further.

*Corium: product which would result from the melting of the core components and their interaction with the structures they would meet.

Containment design with respect to the Hydrogen risk

In the unlikely case of a severe accident, Hydrogen would be released in large quantities inside the containment. This would happen first of all by reaction between the coolant and the Zirconium which is part of the composition of the fuel assembly claddings, then, in the event of core melt and ex-vessel progression, by reaction between the corium and the concrete of the corium spreading and cooling area.

For this reason, the pre-stressed concrete inner shell of the containment is designed to withstand the pressure which could result from the combustion of this Hydrogen. Further, devices called catalytic Hydrogen recombiners are installed inside the containment to keep the average concentration below 10% at all times, to avoid any risk of detonation. Besides, the pressure in the containment does not exced 5.5 bar, assuming an Hydrogen deflagration.

Corium retention and stabilization aiming to protect the base mat

The reactor pit is designed to collect the corium in case of ex-vessel progression and to transfer it to the corium spreading and cooling area. The reactor pit surface is protected by "sacrificial" concrete which is backed-up by a protective layer consisting of zirconia-type refractory material.

In the event of core meltdown, molten core escaping from the reactor vessel would be passively collected and retained, then cooled in a specific area inside the reactor building.

➡ **Even in case of extremely unlikely core melt accident with piercing of the reactor pressure vessel, the melted core and radioactive products would remain confined inside the reactor building whose integrity would be ensured in the long term.**

The dedicated corium spreading and cooling area is a core-catcher equipped with a solid metal structure and covered with "sacrificial" concrete. It aims to protect the nuclear island basemat from any damage, its lower section features cooling channels in which water circulates. The aim of its large spreading surface area (170 m^2) is to promote the cooling of the corium.

The transfer of the corium from the reactor pit to the spreading area would be initiated by a passive device: a steel "plug" melting under the effect of the heat from the corium.

After spreading, the flooding of the corium would also be initiated by a passive fusible plug-based device. It would then be cooled, still passively, by gravity injection of water from the tank located inside the containment and by evaporation.

The effectiveness of the cooling would then provide stabilization of the corium in a few hours and its complete solidification in a few days.

Containment heat removal system and long-term residual heat removal device

In the eventuality of a severe accident, to prevent the containment from losing its long-term integrity, means would have to be provided to control the pressure inside the containment and to stop it from rising under the effect of residual heat. A dedicated dual-train spray system with heat-exchangers and dedicated heat sink is provided to fulfil this function. A long time period would be available for the deployment of this system by the operators: at least 12 hours owing to the large volume of the containment (80,000 m^3).

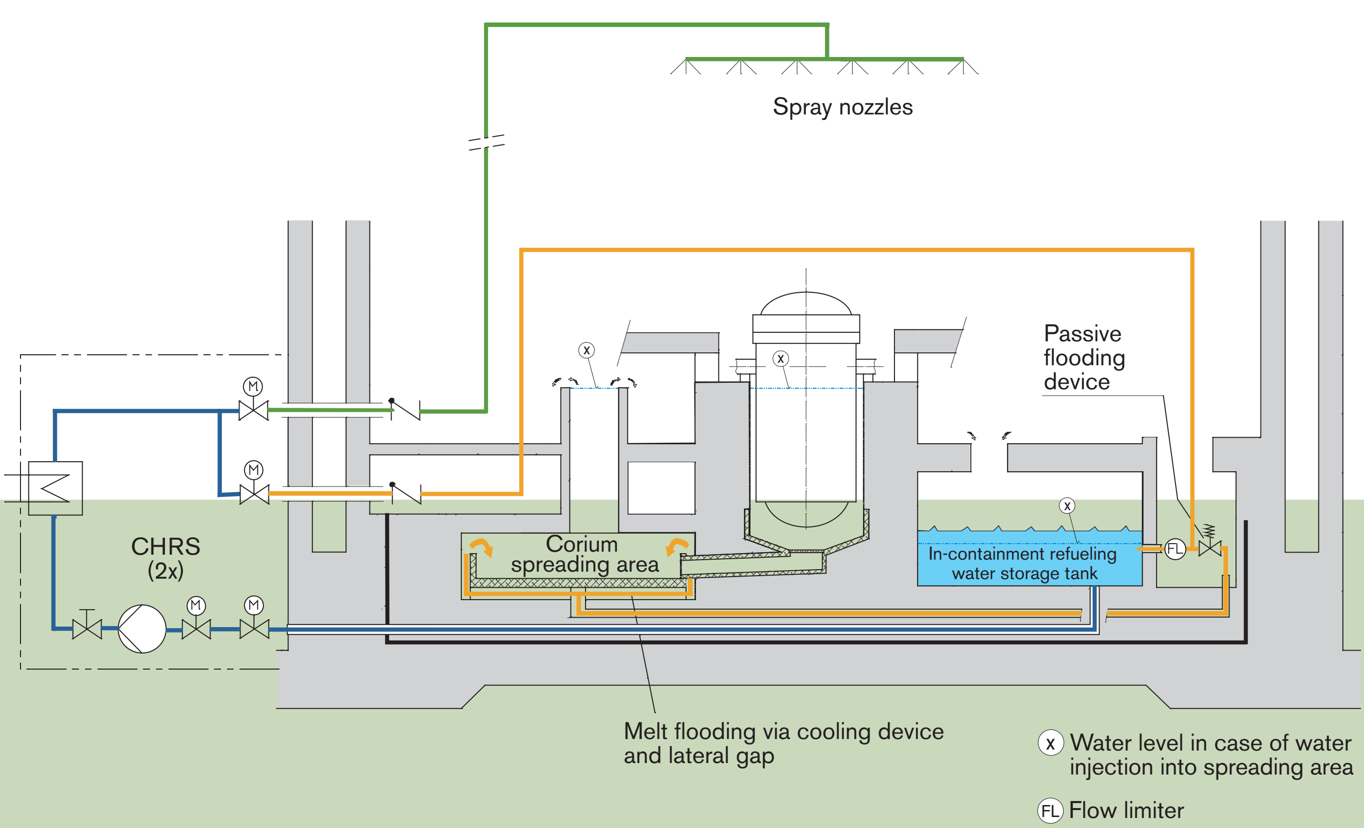
A second mode of operation of the containment heat removal system enables to feed water directly into the core-catcher, instead of into the spray system.

Collection of inter-containment leaks

In the eventuality of a core melt leading to vessel failure, the containment remains the last of the three containment barriers; this means that provisions must be taken to make sure that it remains undamaged and leak-tight. For the EPR, the following measures have been adopted:

- a 6 mm thick metal liner internally covers the pre-stressed concrete inner shell,
- the internal containment penetrations are equipped with redundant isolation valves and leak recovery devices to avoid any containment bypass,
- the architecture of the peripheral buildings and the sealing systems of the penetrations rule out any risk of direct leakage from the inner containment to the environment,
- the space between the inner and outer shells of the containment is passively kept at slight negative pressure to enable the leaks to collect there,
- these provisions are supplemented by a containment ventilation system and a filter system upstream of the stack.

Containment heat removal system



EPR CONSTRUCTION

Emsland nuclear power plant, Germany (KONVOI, 1,300 MWe).

EPR CONSTRUCTION TIME SCHEDULE

The evolutionary approach adopted for the EPR allows its construction schedule to benefit from vast construction experience feedback and from the continuous improvement process of the methodologies and tasks sequencing implemented by AREVA NP worldwide.
Provisions have been made in the design, construction, erection and commissioning methods to further shorten the EPR construction schedule as far as possible. Significant examples can be given as follows.

DESIGN FEATURES

The general layout of the main safety systems in four trains housed in four separate buildings simplifies, facilitates and shortens performance of the erection tasks for all work disciplines.

Location of electromechanical equipment at low levels means that it can be erected very early on in the program, thus shortening the critical path of the construction schedule.

CONSTRUCTION AND ERECTION METHODS

Three main principles are applied to the EPR construction and erection: minimization of the interfaces between civil works and erection of mechanical components, modularization and piping prefabrication.

Minimization of the interfaces between civil works and erection. The on-going search for the optimization of interfaces between civil and erection works results in the implementation of a construction methodology “per level” or “grouped levels” enabling equipment and system erection work at level “N”, finishing construction works at level “N + 1” and main construction work at levels “N + 2” and “N + 3” to be carried out simultaneously; this methodology is used for all the different buildings except for the reactor building, where it cannot apply.

Use of modularization for overall schedule optimization. Modularization techniques are systematically considered, but retained only in cases where they offer a real benefit to the optimization of the overall construction schedule without inducing a technical and financial burden due to advanced detailed design, procurement or prefabrication. This approach enables the site preparation schedule to be optimized, delays investment costs with regard to start of operation, and so offers financial savings.

For instance, modules are mainly implemented for the civil works of the reactor building, such as the reactor pit, the internal structures and the containment dome, as well as for the structures of the reactor building (and fuel building) pools, as they are all on the critical path for the construction of the reactor building.

Maximization of piping and support prefabrication. Piping and support prefabrication is maximized in order to minimize erection man-hours and especially welding and controls at erection places; this measure also results in an even better quality of the piping spools with lower cost.

MAIN COMPONENTS MANUFACTURING

AREVA NP’s Chalon/Saint-Marcel and Jeumont plants have clocked up over 30 years of experience in the manufacturing of heavy nuclear components and are keeping it alive. This is why they have the know-how it takes to optimize heavy nuclear component production time. The construction of the EPR stands to benefit from their unique manufacturing capability and expertise.

COMMISSIONING TESTS

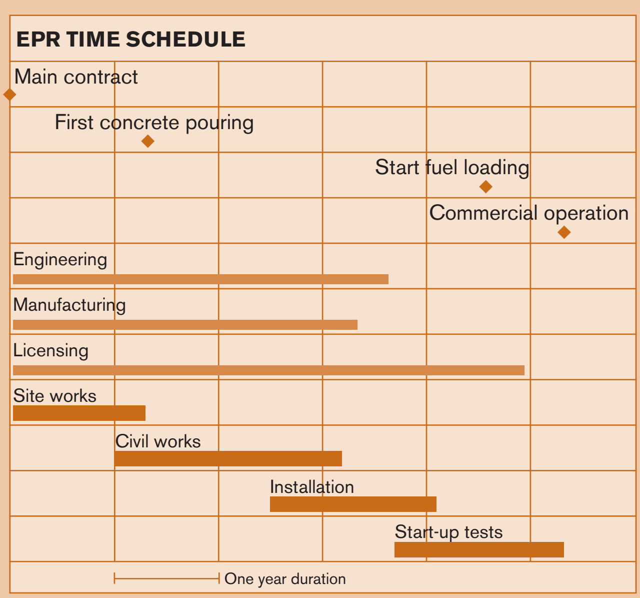
As with the interfaces between civil and erection works, the interfaces between erection and tests have been carefully reviewed and optimized. For instance, teams in charge of commissioning tests are involved in the finishing works, flushing and conformity checks of the systems, so that these activities are only carried out once.

Instrumentation & Control factory acceptance tests are carried out on a single test platform with all cabinets interconnected, which ensures a shorter on-site test period together with improved overall quality.

The benefits drawn from the unique experience feedback gained from AREVA NP’s past achievements, associated with the systematic analysis of possible improvements and optimization of construction, erection and test activities together with their interfaces, results in an optimal technical and economical construction schedule for the implementation of the EPR projects. This experience and current EPR projects show that the EPR time schedule is totally realistic.

Indicative planning and overall time-scale

The overall construction schedule of a unit in the series depends largely on site conditions, industrial organization and policies, and local working conditions. So accurate figures are valid only for the specific project to which they are related.



PLANT OPERATION, MAINTENANCE & SERVICES

Neckarwestheim nuclear power plant (Germany): unit 2 (right foreground) is of the KONVOI type (1,300 MWe).

PLANT OPERATION, MAINTENANCE & SERVICES

From the beginning, the EPR and its equipment and systems have been designed to allow for efficient refueling outages and to simplify and optimize inspection and maintenance in order to increase plant availability and reduce maintenance costs, two major objectives of plant operators worldwide to meet the demands of more and more competitive power markets.

A 92% AVAILABILITY FACTOR OVER THE ENTIRE PLANT LIFE

Regarding availability, the EPR is designed to reach up to 92% over the entire 60 years of its design lifetime. This is made possible by short-scheduled outages for fuel loading/unloading and in-service inspections and maintenance, and also through reduced downtimes attributable to unscheduled outages.

The high degree of equipment reliability on the one hand, and the decrease in reactor trip causes (in particular due to the deployment of the limitation system related to reactor operation) on the other hand lead to an unscheduled unavailability not exceeding 2%.

The quadruple redundancy of the safeguard systems allows a large part of the preventive maintenance operations to be performed while the reactor is at power.

Chooz B1, France (N4, 1,500 MWe): removal of the hydraulic section of a reactor coolant pump for maintenance.

Moreover, the reactor building is designed to be accessible, under standard safety and radiation protection conditions, while the reactor is at power. This enables the outage and maintenance operations to be prepared and demobilized with no loss of availability. This possibility of access with the reactor on line also facilitates field services which could be needed outside scheduled outage periods. Based on experience feedback, standardization and ease of access of the components of the reactor allow simple and rapid performance of inspection and maintenance work.

Access to the reactor building during power operation allows to start preventive maintenance and refueling tasks up to seven days before reactor shutdown and to continue their demobilization up to three days after reactor restart.

The duration of the plant shutdown phase is reduced by a time gain for reactor coolant system cooldown, depressurization and vessel head opening. Similarly the length of the restart phase is reduced as well and benefits from the reduction in the time needed to run the beginning-of-cycle core physics tests (gain supplied by the "aeroball" in-core instrumentation system). Durations of about 70 and 90 hours are respectively scheduled for the shutdown and restart phases. For the fuel loading/unloading operations, a time period of about 80 hours is scheduled.

➡ **Typical outage duration: the duration of a regular outage for preventive maintenance and refueling is reduced to 16 days. Duration of an outage for refueling only does not exceed 11 days. Decennial outages for main equipment in-service inspection, turbine overhaul and containment pressure test are planned to last 40 days.**

The EPR is designed to:

- ➡ **maximize plant availability and maneuverability,**
- ➡ **ease operation and maintenance and reduce their costs,**
- ➡ **enhance radiological protection of the personnel,**
- ➡ **protect the environment and contribute to a sustainable development.**

A HIGH LEVEL OF OPERATIONAL MANEUVERABILITY

In terms of operation, the EPR is designed to offer the utilities a high level of maneuverability. It has the capacity to be permanently operated at any power level between 20 and 100% of its nominal power in a fully automatic way, with the primary and secondary frequency controls in operation.

The EPR capability regarding maneuverability is a particularly well adapted response to scheduled and unscheduled power grid demands for load variations, managing of grid perturbations or mitigation of grid failures.

AN ENHANCED RADIOLOGICAL PROTECTION

Allowance for operating constraints and for the human factor, with the aim of improving worker radiation protection and limiting radioactive releases, together with radwaste quantity and activity, was a set objective as soon as EPR design got underway. For this purpose, the designers drew heavily upon the experience feedback from the operation of the French and German nuclear power plant fleets.

Accordingly, major progress has been made, particularly in the following areas:
- the choice of materials, for example the optimization of the quantity and location of the Cobalt-containing materials and liners, in order to obtain a gain on the Cobalt 60 "source term",
- the choices regarding the design and layout of the components and systems liable to convey radioactivity, taking into account the various plant operating states,
- the optimization of the radiation shielding thicknesses in response to forecast reactor maintenance during outages or in service.

Thanks to these significant advances, collective doses less than 0.4 Man.Sievert per reactor/year can be expected for operation and maintenance staff (to date, for the major nuclear power plant fleets of OECD countries like France, Germany, the United States and Japan, the average collective dose observed is about 1 Man.Sievert per reactor/year).

PLANT SERVICES

Optimization of plant processes and implementation of innovative maintenance technologies and concepts are also significant contributors to the achieving of operators' cost and availability objectives. In this area, AREVA NP supplies the most comprehensive range of nuclear services and technologies in the world.

Thanks to its experience from designing and constructing 96 nuclear power plants worldwide, its global network of maintenance and services centers with highly trained teams (more than 3,000 specialists mainly based in France, Germany and the USA) committed to excellence, AREVA NP provides a full range of inspection, repair and maintenance services for all types of nuclear power plants, based on the most advanced techniques available today. Its field of expertise covers the whole scope of customers' needs from unique one-of-a-kind assignments to the implementation of integrated service packages.

AREVA NP's offer of power plant services encompasses:
- in-service inspection and non destructive testing,
- outage services,
- component repair and replacement (including steam generators, reactor pressure vessel heads),
- supply of spare parts,
- off-site maintenance of components in "hot" workshops,
- fuel inspection, repair and management,
- services in the fields of instrumentation and diagnosis, I&C and electrical systems, chemistry,
- plant engineering and plant upgrading,
- plant decommissioning and waste management,
- training of operating personnel,
- expert consultancy.

The "FROG" Owners Group (see page 57) offers member electricity companies a cost-effective means for exchange of information and experience. FROG's members have access to broad operational and maintenance feedback. They also benefit from the results of study programs jointly decided to deal with issues of shared interest.

In-service inspection machine for ultrasonic testing of reactor pressure vessels.

Operators have developed ambitious outage optimization plans to decrease outage duration. Their objectives are even more ambitions and include plant upgrades and component replacement for life extension of plant operation. Aware of the strategic importance of the operators' goal of reducing outage duration, AREVA NP has created an International Outage Optimization Team that spans all regions and capabilities of the company for customer benefit in terms of quality, safety and costs.

AREVA NP'S SPIRIT OF SERVICE

➡ **To satisfy customers and help them to succeed in a highly competitive energy market, by:**

- **reducing operating and maintenance costs,**
- **improving safety and performance,**
- **extending plant life,**
- **reducing radiation exposure.**

CONTINUOUSLY IMPROVING SERVICE TO CUSTOMERS

To continuously improve service to customers, with particular attention to respect of local cultures and practices, especially in geographical areas outside its European and American bases, AREVA NP has established special links and partnerships with entities well positioned to locally propose and perform power plant services. A significant illustration is the company's long-lasting and successful cooperation with Chinese companies and institutes involved in the extensive long-term nuclear program currently underway in China. An excellent example of this cooperation is the tight links with the ShenZhen Nuclear Company Ltd (SNE), which is mainly engaged in maintenance and refueling outages of commercial power stations in China and has also diversified its activities to cover other industrial projects.

SNE was created in the Guangdong province at the end of 1998. Since July 2003, SNE is a joint venture between Company 23 of China Nuclear Engineering and Construction Corporation (CNEC) and AREVA NP, which fully benefits from AREVA NP's expertise and technologies in its activity field.

AREVA NP Technical Center (TC), with its locations in France, Germany and the USA, is the first link for the development of new technologies. A major objective of the TC is to provide support in solving technical issues in specific fields. More than 300 scientific engineers and technicians work in the TC laboratories which are equipped with the most up-to-date technology and test loops. Their fields of excellence cover material engineering, welding, chemistry and radiochemistry, corrosion, non-destructive examination, thermal-hydraulics and fluid dynamics, testing of components and systems, manufacture of special components.

AREVA NP'S COMMITMENT

➡ **Flexibility to accommodate customers' needs, cultures and practices, through:**

- **optimized organization and processes,**
- **consolidation of expertise and experience,**
- **rapid mobilization of skilled and highly qualified multi-cultural teams,**
- **technical and contractual innovation,**
- **partnerships with customers and local service partners.**

THE "FROG" OWNERS GROUP

The FROG (formely Framatome Owners Group) is dedicated to building strong and efficient teaming for mutual cooperation, assistance and sharing of its members' experience and expertise, to support the safe, reliable, cost-effective operation of its members' nuclear power units.

The FROG was set up in October 1991 by five utility companies that were either operating or building nuclear power plant units incorporating a Framatome nuclear steam supply system or nuclear island.

These utility companies are Electrabel from Belgium, Electricité de France, Eskom from the Republic of South Africa, GNPJVC from the People's Republic of China and KHNP from the Republic of Korea.

Later on, Ringhals AB from Sweden (in June 1997), LANPC, owner of the Ling Ao plant in China (in October 2000), British Energy owner of Sizewell B in the United Kingdom (in October 2002) joined the FROG as members. In 2003, GNPJVC and LANPC merged operation of their plants in one company DNMC.

The Owners group provides a forum for its members to share their experiences in all domains of nuclear power plant operation, enabling a cost-effective exchange of information to identify and solve common issues or problems.

Several working groups and technical committees are actively dealing with specific technical and management issues. Among them, a specific Steam Generator Technical Committee, has been formed by utilities having steam generators served by AREVA NP. Committee participants are the FROG members plus the companies NSP and AmerenUE from the USA, NOK from Switzerland and NEK from Slovenia.

> CONCLUDING REMARKS

Let us summarize the advantages offered by the EPR from an electricity utility point of view:

- culminating from the legacy of Western PWR technology,
- evolutionary design, uniquely minimizing design, licensing, construction and operation technical risks and their financial impacts,
- assurance to be backed in the long run by the world's largest company comprising the entire nuclear cycle,
- continuity in the mastery of PWR technology,
- outstanding efficiency thanks to the integration of design and manufacturing within AREVA NP,

On December 18, 2003, the Finnish electricity utility, Teollisuuden Voima Oy (TVO) signed a contract with the consortium set up by AREVA NP and Siemens for the construction of the Olkiluoto 3 EPR in Finland.

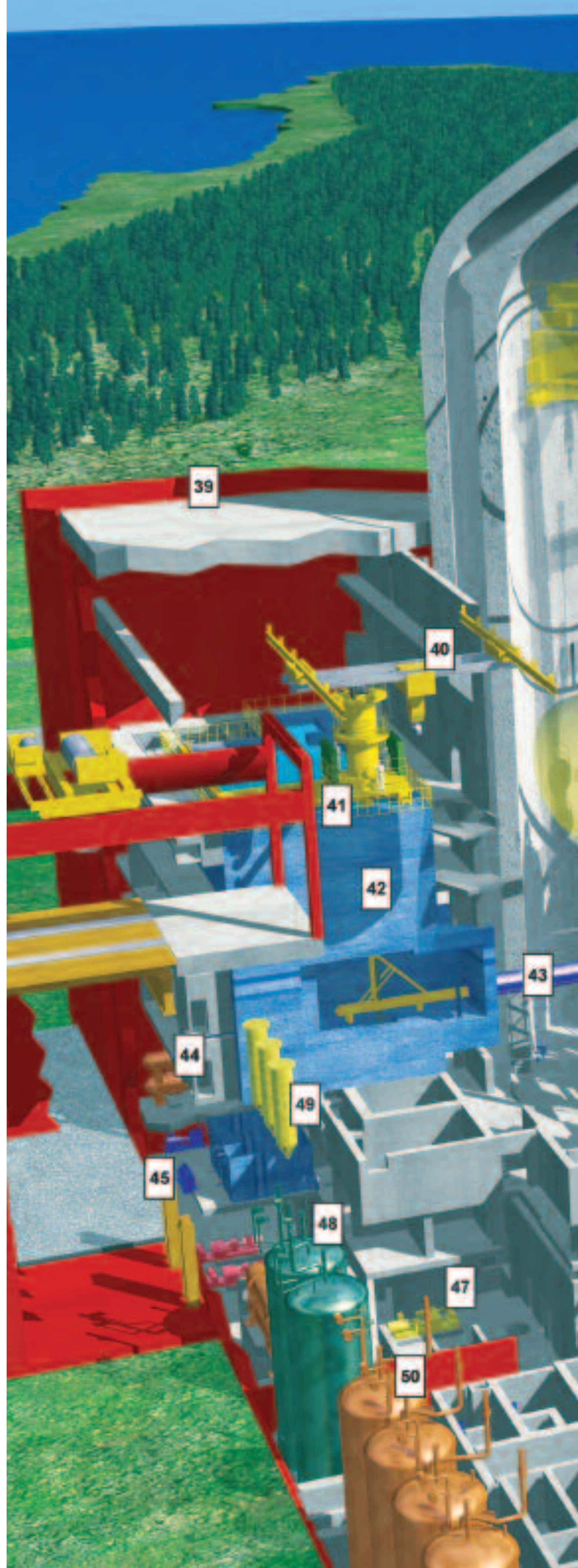
- competitiveness in terms of installed kW cost and kWh production cost: a 1,600 MWe-class reactor, with high efficiency, reduced construction time, extended service life, enhanced and more flexible fuel utilization, increased availability,
- safety:
  - heightened protection against accidents, including core meltdown, and their radiological consequences,
  - robustness against external hazards, in particular airplane crash and earthquake,
- optimized operability,
- enhanced radiological protection of operating and maintenance personnel,
- efficiency in the use of nuclear fuel, fostering sustainable development.

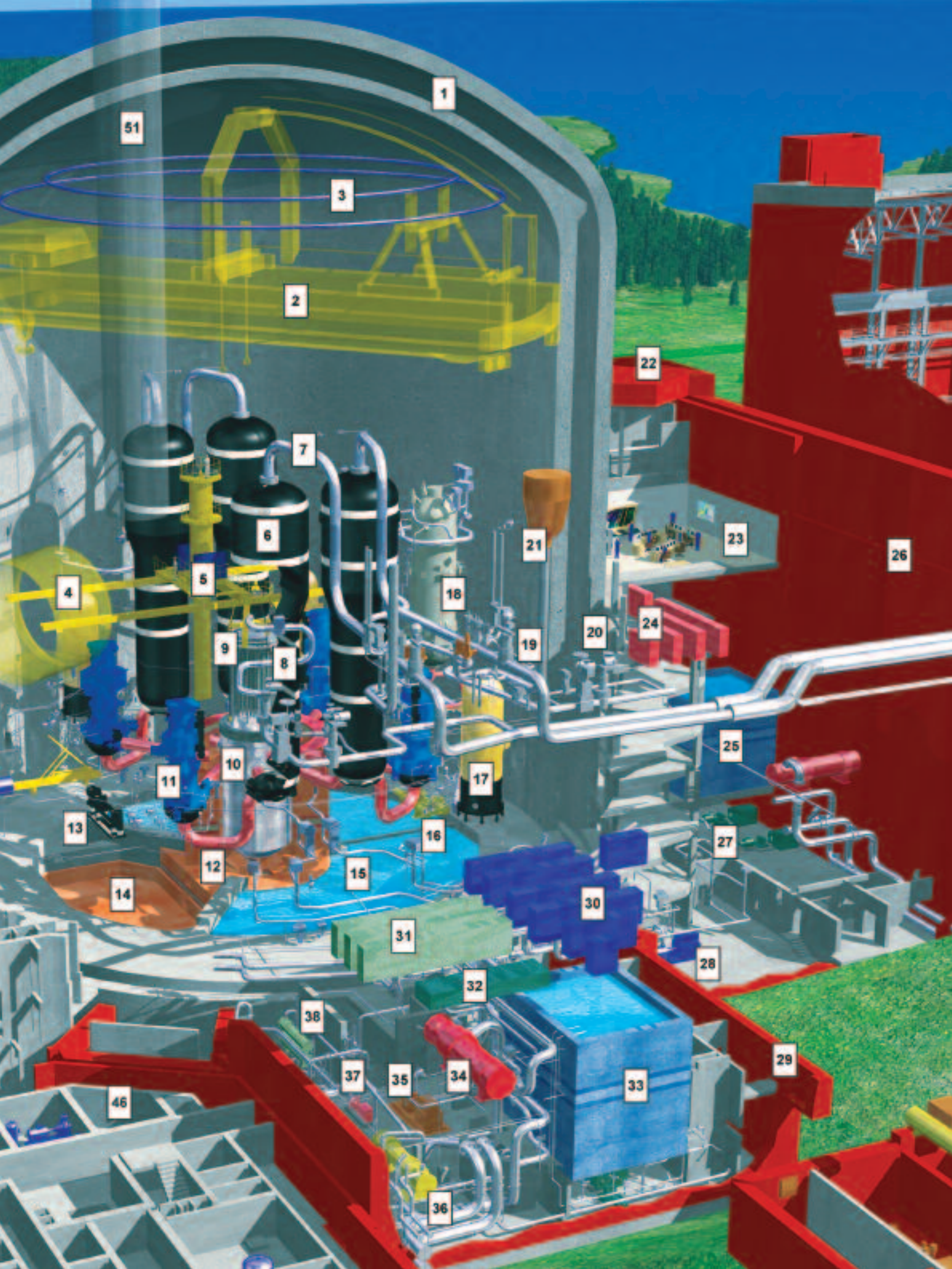
On May 4, 2006, the Board of Directors of EDF decided to launch the building of its first EPR unit in France on the Flamanville site.

> EPR

Key to power station cutaway

1 **Reactor building: inner and outer shell**
2 Polar crane
3 Containment heat removal system: sprinklers
4 Equipment hatch
5 Refueling machine
6 Steam generator
7 Main steam lines
8 Main feedwater lines
9 Control rod drives
10 Reactor pressure vessel
11 Reactor coolant pump
12 Reactor coolant piping
13 CVCS heat exchanger
14 Corium spreading area
15 In-containment refueling water storage tank
16 Residual heat removal system, heat exchanger
17 Safety injection accumulator tank
18 Pressurizer
19 Main steam isolation valves
20 Feedwater valves
21 Main steam safety and relief valve exhaust silencer
22 **Safeguard building division 2**
23 Main control room
24 Computer room
25 Emergency feedwater storage, division 2
26 **Safeguard building, division 3**
27 Emergency feedwater pump, division 3
28 Medium head safety injection pump, division 3
29 **Safeguard building, division 4**
30 Switchgear, division 4
31 I & C cabinets
32 Battery rooms, division 4
33 Emergency feedwater storage, division 4
34 CCWS heat exchanger, division 4
35 Low head safety injection pump, division 4
36 Component cooling water surge tank, division 4
37 Containment heat removal system pump, division 4
38 Containment heat removal system heat exchanger, division 4
39 **Fuel building**
40 Fuel building crane
41 Spent fuel pool bridge
42 Spent fuel pool and fuel transfer pool
43 Fuel transfer tube
44 Spent fuel pool cooler
45 Spent fuel pool cooling pump
46 **Nuclear auxiliary building**
47 CVCS pump
48 Boric acid tank
49 Delay bed
50 Coolant storage tank
51 Vent stack





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With manufacturing facilities in 40 countries and a sales network in more than 100, AREVA offers customers reliable technological solutions for CO_2-free power generation and electricity transmission and distribution. We are the world leader in nuclear power and the only company to cover all industrial activities in this field.
Our 58,000 employees are committed to continuous improvement on a daily basis, making sustainable development the focal point of the group's industrial strategy.
AREVA's businesses help meet the 21st century's greatest challenges: making energy available to all, protecting the planet, and acting responsibly towards future generations.

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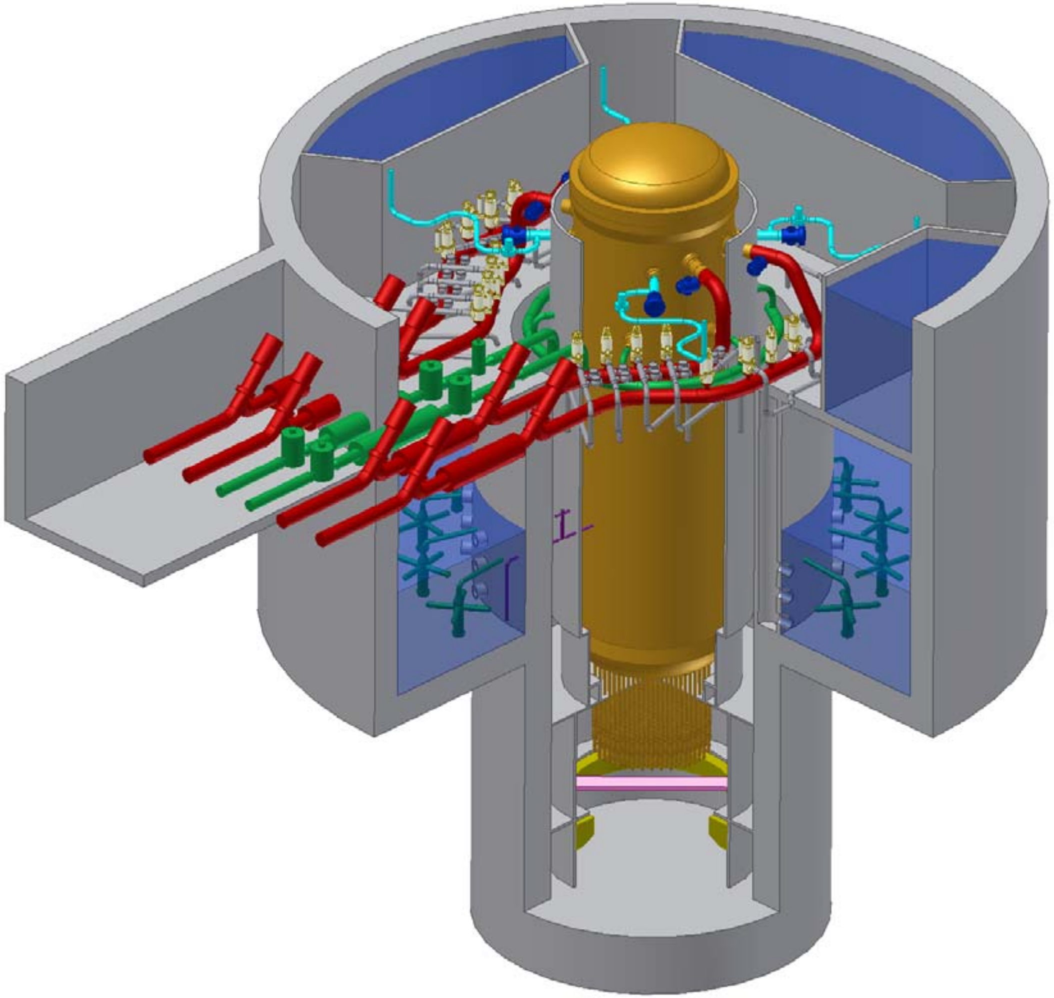
Appendix G: ESBWR Technical Summary

An overview and technical summary of the ESBWR reactor design is provided in the following pages.

ESBWR Overview



J. Alan Beard
September 15, 2006

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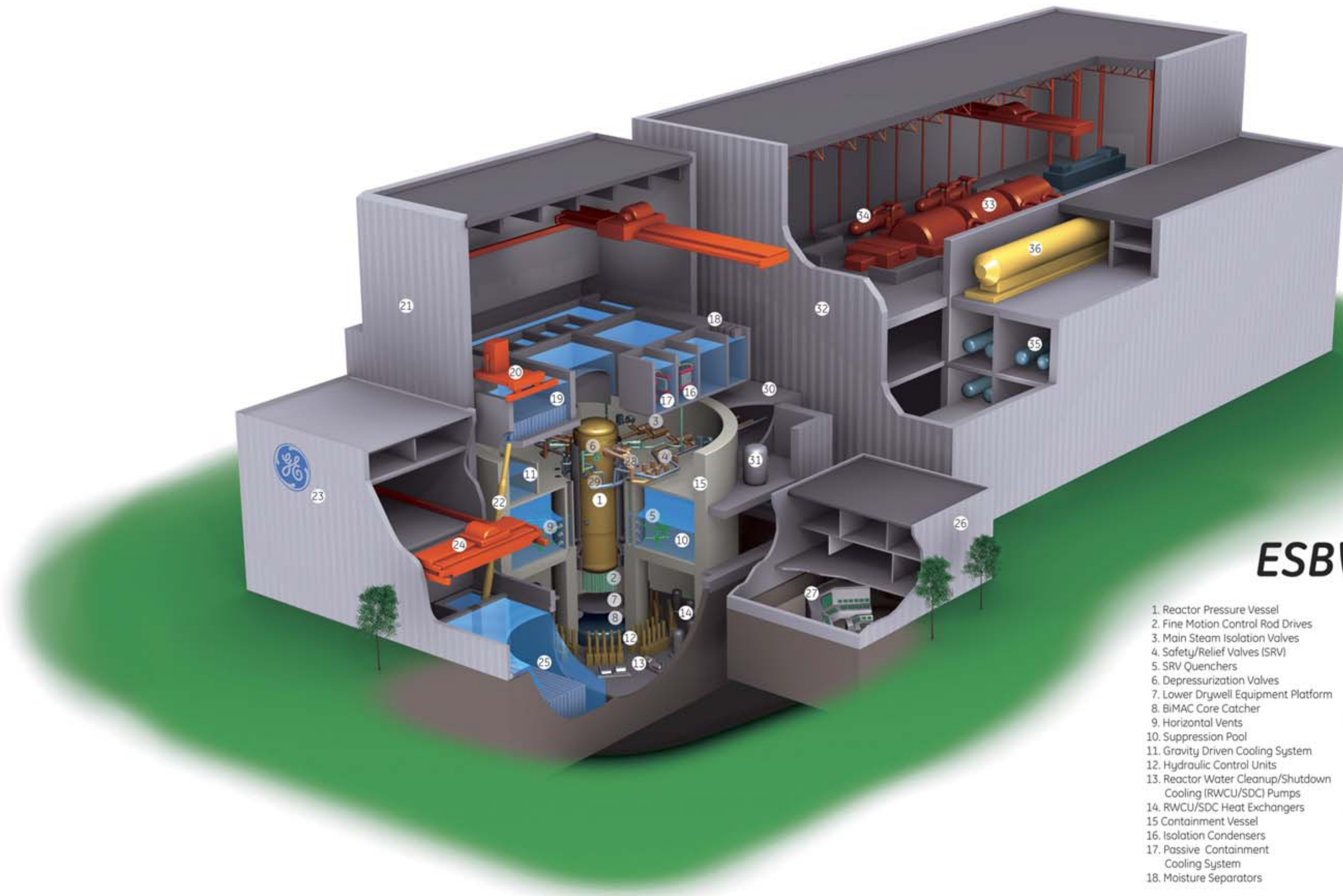
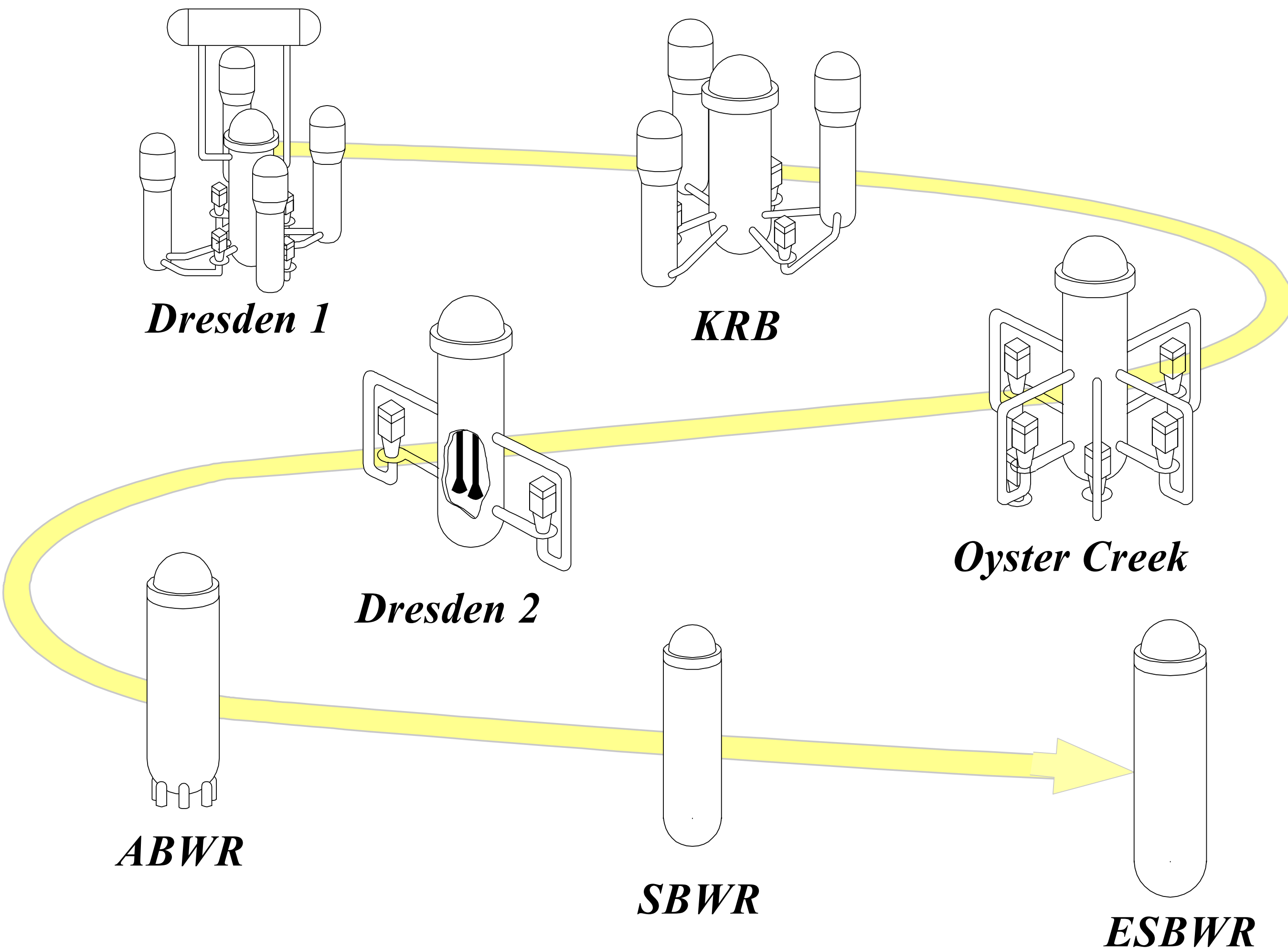
Presentation Content

- BWR Design Evolution
- ESBWR Primary Characteristics
- ESBWR Passive Systems
- Differences from previous BWRs
- ESBWR Active Systems

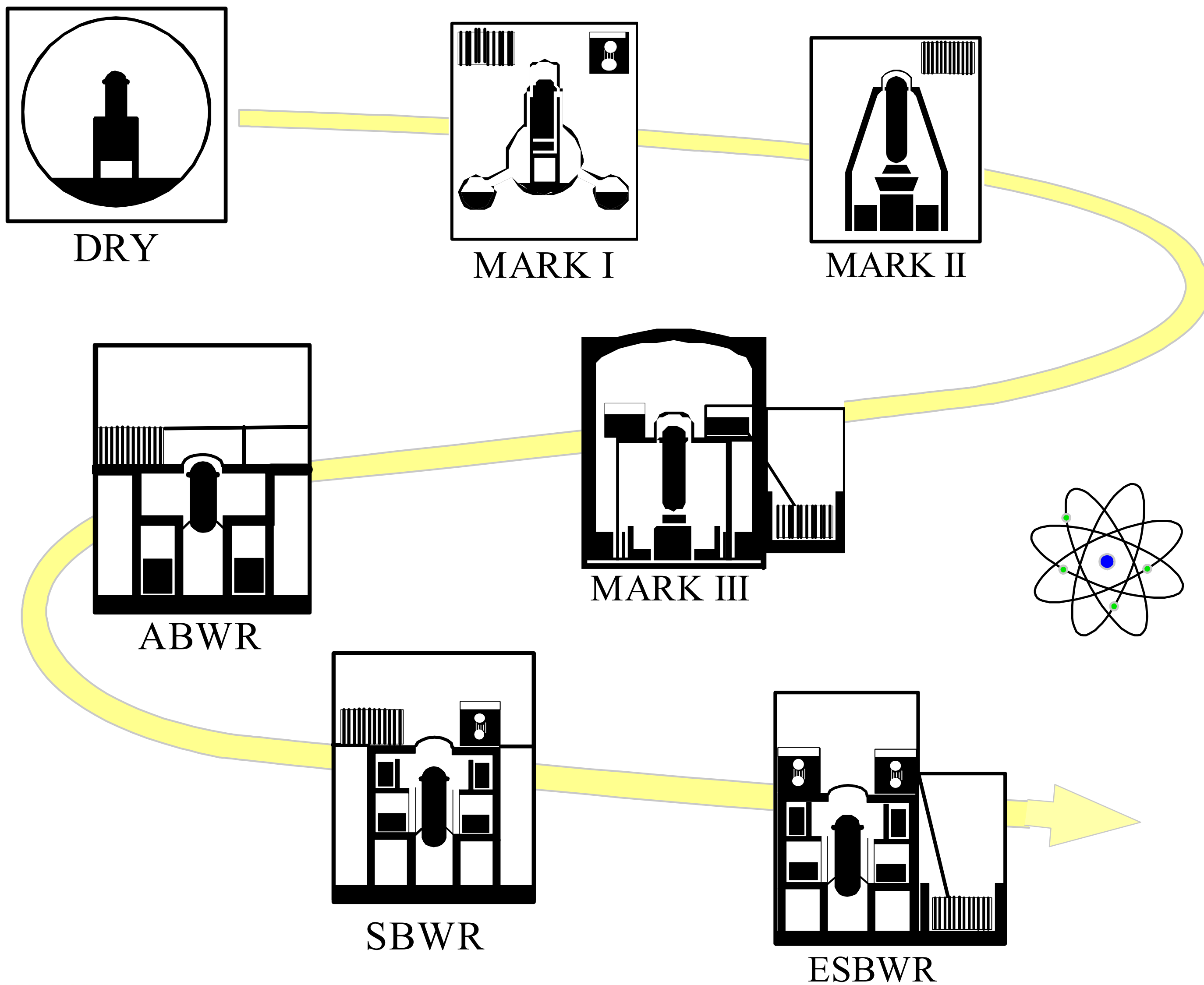


1. Reactor Pressure Vessel
2. Fine Motion Control Rod Drives
3. Main Steam Isolation Valves
4. Safety/Relief Valves (SRV)
5. SRV Quenchers
6. Depressurization Valves
7. Lower Drywell Equipment Platform
8. BiMAC Core Catcher
9. Horizontal Vents
10. Suppression Pool
11. Gravity Driven Cooling System
12. Hydraulic Control Units
13. Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) Pumps
14. RWCU/SDC Heat Exchangers
15. Containment Vessel
16. Isolation Condensers
17. Passive Containment Cooling System
18. Moisture Separators
19. Buffer Fuel Storage Pool
20. Refueling Machine
21. Reactor Building
22. Inclined Fuel Transfer Machine
23. Fuel Building
24. Fuel Transfer Machine
25. Spent Fuel Storage Pool
26. Control Building
27. Main Control Room
28. Main Steam Lines
29. Feedwater Lines
30. Steam Tunnel
31. Standby Liquid Control System Accumulator
32. Turbine Building
33. Turbine-Generator
34. Moisture Separator Reheater
35. Feedwater Heaters
36. Direct Contact Feedwater Heater and Tank

BWR Evolution



Containment Evolution



Site Parameters

- EPRI Utility Requirements Document Plus
  - > Tornado
    - – 330 mph
  - > Extreme Winds
    - – 140 mph for safety-related
  - > Temperatures
    - – Bound the 3 ESP sites
  - > Seismic
    - – Reg Guide 1.60 plus a CEUS hard rock site

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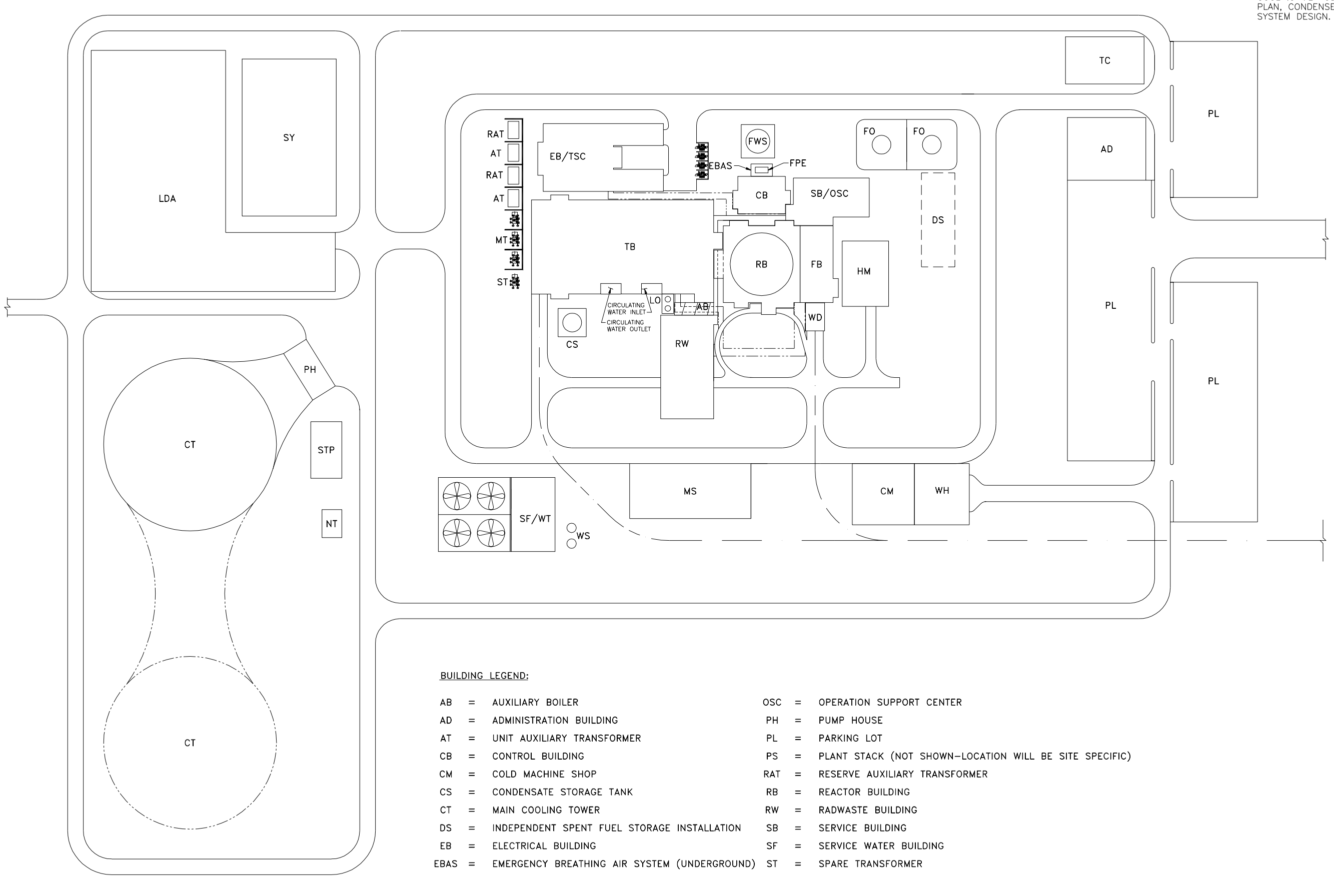
Site Plan

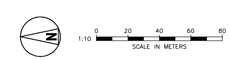
NOTES:

1. THIS PLOT PLAN REPRESENTS THE STANDARD ESBWR CONFIGURATION. THIS CONFIGURATION WILL BE MODIFIED FOR SITE SPECIFIC REQUIREMENTS DURING COMBINED OPERATING LICENSE APPLICATIONS.
2. THE REFERENCE NORMAL HEAT SINK IS SHOWN AS NATURAL DRAFT COOLING TOWERS; HOWEVER, SITE SPECIFIC AVAILABLE WET BULB AND COOLING WATER TEMPERATURES, ENVIRONMENTAL LIMITATIONS AND SPECIFIC TURBINE CONFIGURATION MAY DICTATE EITHER ONCE THROUGH OR MECHANICAL DRAFT TOWER COOLING. THESE SITE SPECIFIC ALTERNATE COOLING METHODS MAY ALSO CHANGE THE PLOT PLAN, CONDENSER AND CIRCULATING WATER SYSTEM DESIGN.



BUILDING LEGEND:

AB	= AUXILIARY BOILER	OSC	= OPERATION SUPPORT CENTER
AD	= ADMINISTRATION BUILDING	PH	= PUMP HOUSE
AT	= UNIT AUXILIARY TRANSFORMER	PL	= PARKING LOT
CB	= CONTROL BUILDING	PS	= PLANT STACK (NOT SHOWN–LOCATION WILL BE SITE SPECIFIC)
CM	= COLD MACHINE SHOP	RAT	= RESERVE AUXILIARY TRANSFORMER
CS	= CONDENSATE STORAGE TANK	RB	= REACTOR BUILDING
CT	= MAIN COOLING TOWER	RW	= RADWASTE BUILDING
DS	= INDEPENDENT SPENT FUEL STORAGE INSTALLATION	SB	= SERVICE BUILDING
EB	= ELECTRICAL BUILDING	SF	= SERVICE WATER BUILDING
EBAS	= EMERGENCY BREATHING AIR SYSTEM (UNDERGROUND)	ST	= SPARE TRANSFORMER
FB	= FUEL BUILDING	STP	= SEWAGE TREATMENT PLANT
FO	= DIESEL FUEL OIL STORAGE TANK	SY	= SWITCH YARD
FPE	= FIRE PUMP ENCLOSURE	TB	= TURBINE BUILDING
FWS	= FIRE WATER STORAGE TANK	TC	= TRAINING CENTER
HM	= HOT MACHINE SHOP & STORAGE	TSC	= TECHNICAL SUPPORT CENTER
LDA	= LAY DOWN AREA	WD	= WASH DOWN BAYS (EQUIPMENT ENTRY)
LO	= DIRTY/CLEAN LUBE OIL STORAGE TANK	WH	= WAREHOUSE
MS	= MISCELLANEOUS SERVICE AREA	WS	= WATER STORAGE
MT	= MAIN TRANSFORMER	WT	= WATER TREATMENT
NT	= NITROGEN STORAGE TANK		



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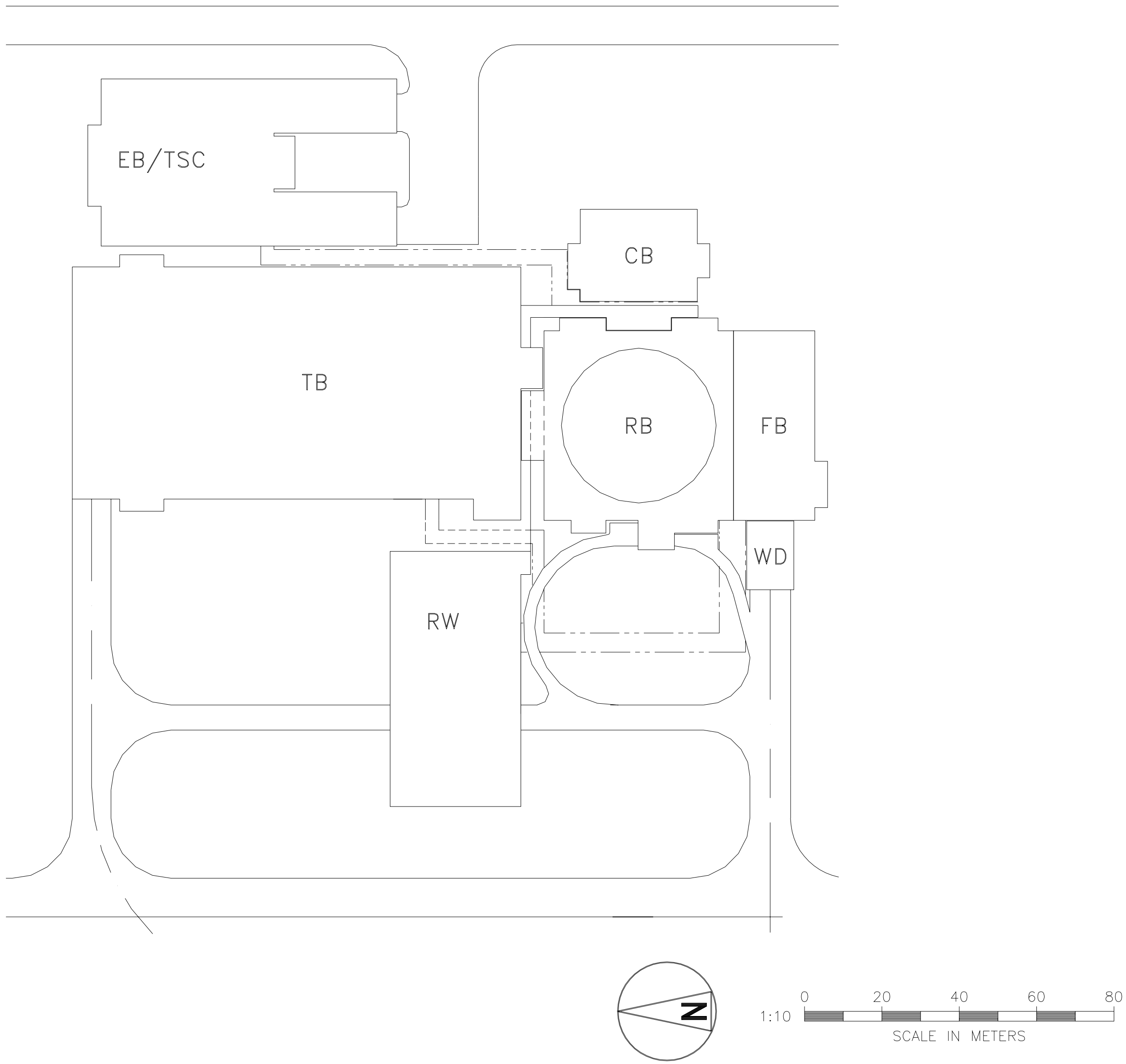
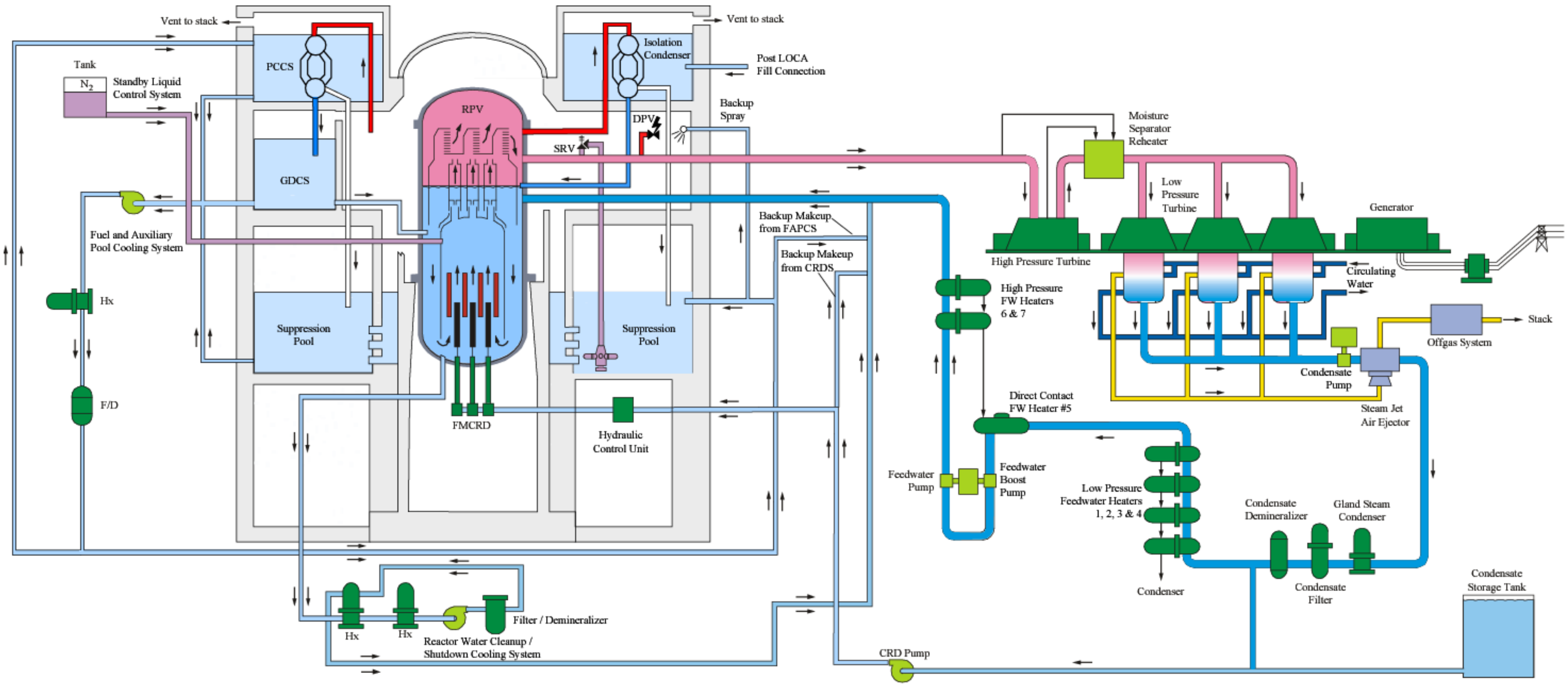
Power Block Arrangement



ESBWR Basic Parameters

- 4,500 Megawatt Core Thermal Power
- ~1, 575 to 1,600 Megawatt Electric Gross
  - Nominal Summer Rating
- Natural Circulation
  - No recirculation pumps
- Passive Safety Systems
  - 72 hours passive capability



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What's different about ESBWR

ABWR	ESBWR
Recirculation System + support systems	Eliminated
HPCF System (2 each)	Eliminated need for ECCS pumps
LPFL (3 each)	Utilize passive and stored energy
Residual Heat Removal (3 each)	Non-safety, combined with cleanup system
Safety Grade Diesel Generators (3 each)	Eliminated – only 2 non-safety grade diesels
RCIC	Replaced with IC heat exchangers
SLC –2 pumps	Replaced pumps with accumulators
Reactor Building Service Water (Safety Grade) And Plant Service Water (Safety Grade)	Made non-safety grade

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Optimized Parameters for ESBWR

Parameter	BWR/4-Mk I(Browns Ferry 3)	BWR/6-Mk III (Grand Gulf)	ABWR	ESBWR
Power (MWt/MWe)	3293/1098	3900/1360	3926/1350	4500/1590
Vessel height/dia. (m)	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel Bundles (number)	764	800	872	1132
Active Fuel Height (m)	3.7	3.7	3.7	3.0
Power density (kw/l)	50	54.2	51	54
Recirculation pumps	2(large)	2(large)	10	zero
Number of CRDs/type	185/LP	193/LP	205/FM	269/FM
Safety system pumps	9	9	18	zero
Safety diesel generator	2	3	3	zero
Core damage freq./yr	1E-5	1E-6	1E-7	3E-8
Safety Bldg Vol (m^3/MWe)	115	150	160	< 130

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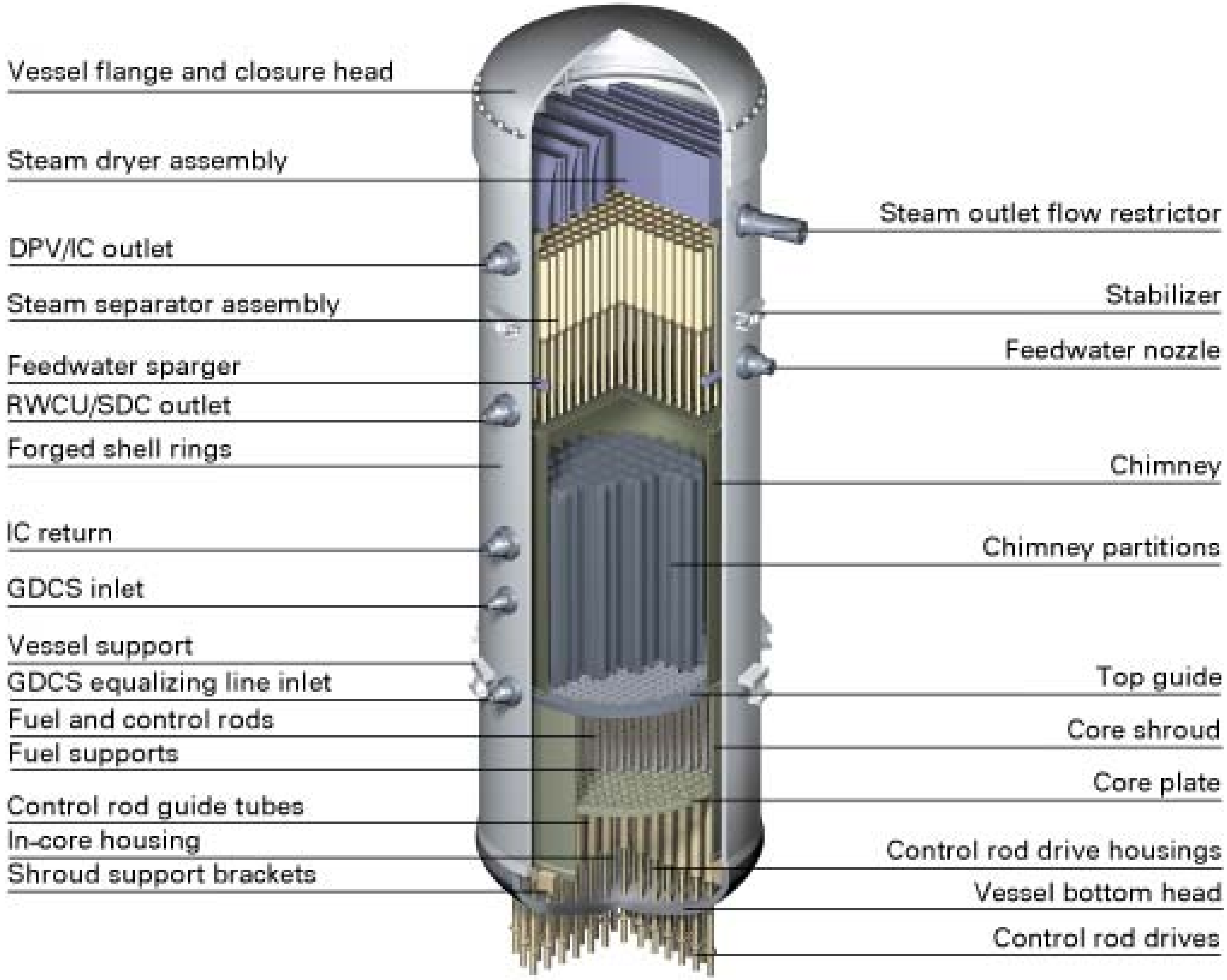

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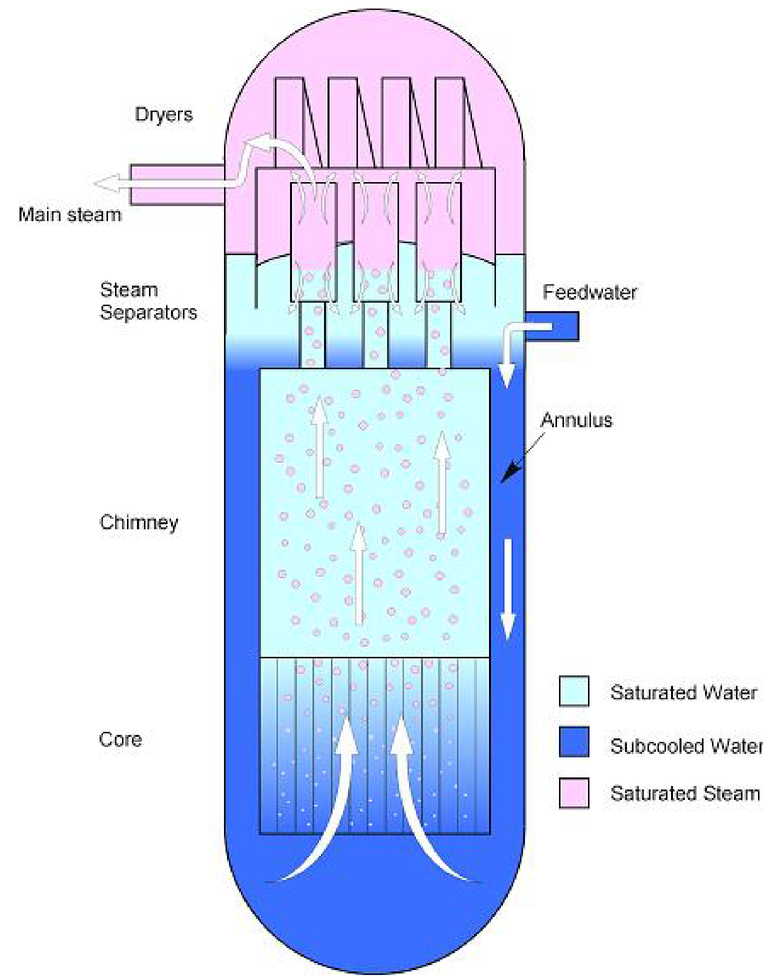
Other Design Improvements

- 100% Steam Bypass
  > Island Mode of Operation
- Fine Motion Control Rod Drives (FMCRD)
- Shoot-out Steel Eliminated
- Integrated Head Vent Pipe
- Improved Incore Instrumentation
  > Start-up Range Neutron Monitor (SRNM)
  > Gamma Thermometer
    – No Traversing Incore Probe (TIP)

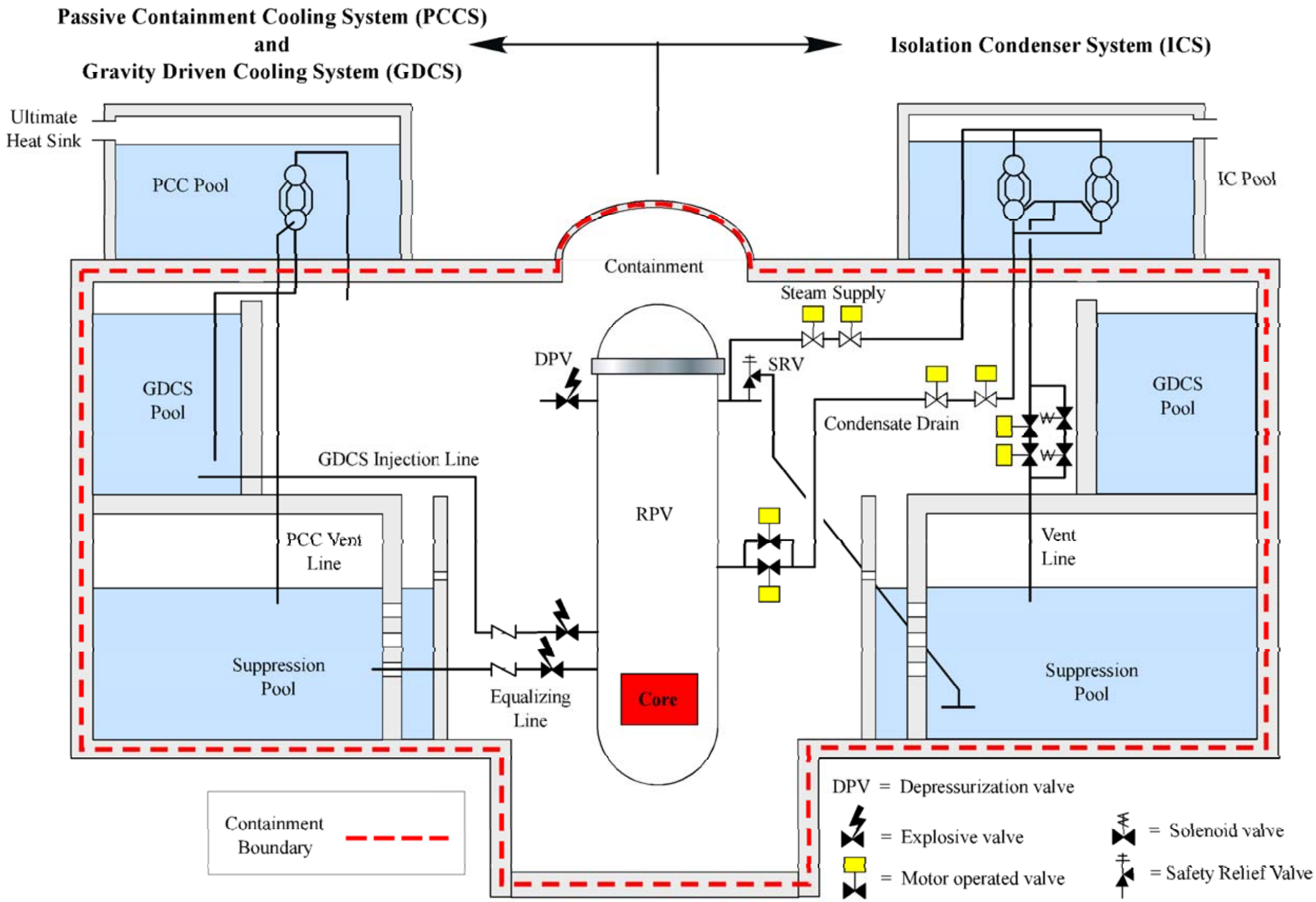
Natural Circulation

Simplification without performance loss ..

- **Passive safety/natural circulation**
  - Increase the volume of water in the vessel
  - Increase driving head
- **Significant reduction in components**
  - Pumps, motors, controls, HXers
- **Power Changes with Control Rod Drives**
  - Minimal impact on maintenance



Passive Safety



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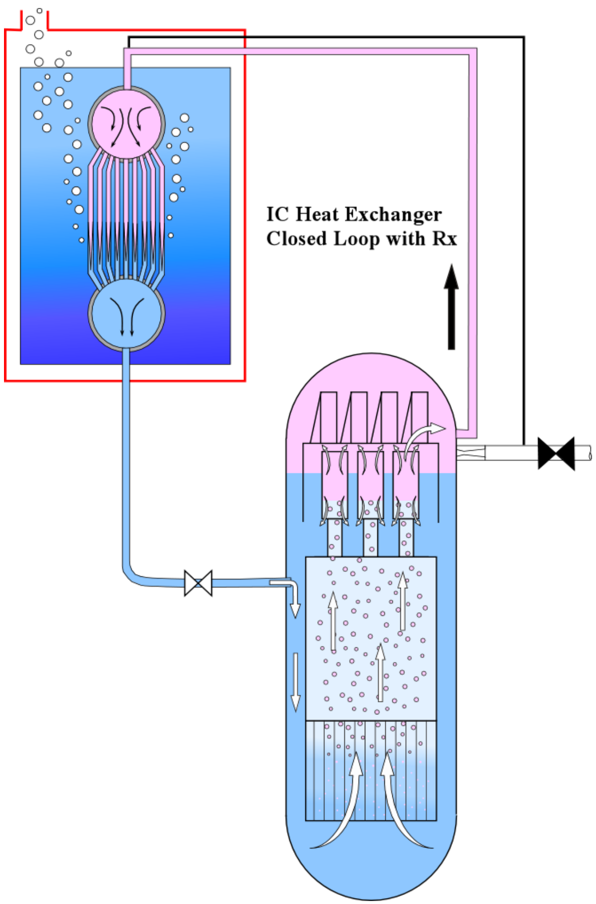
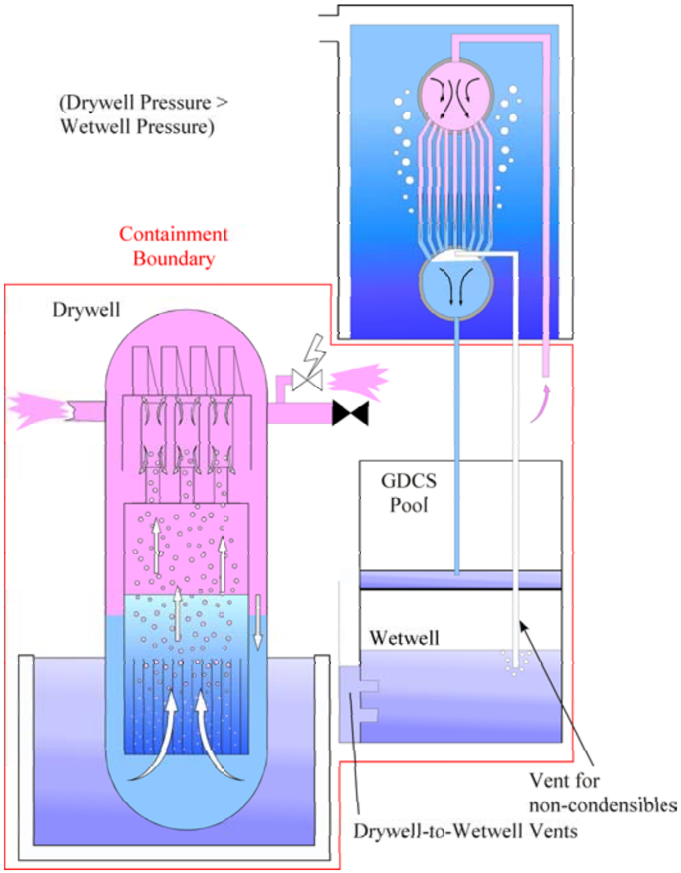
Passive Safety Systems …

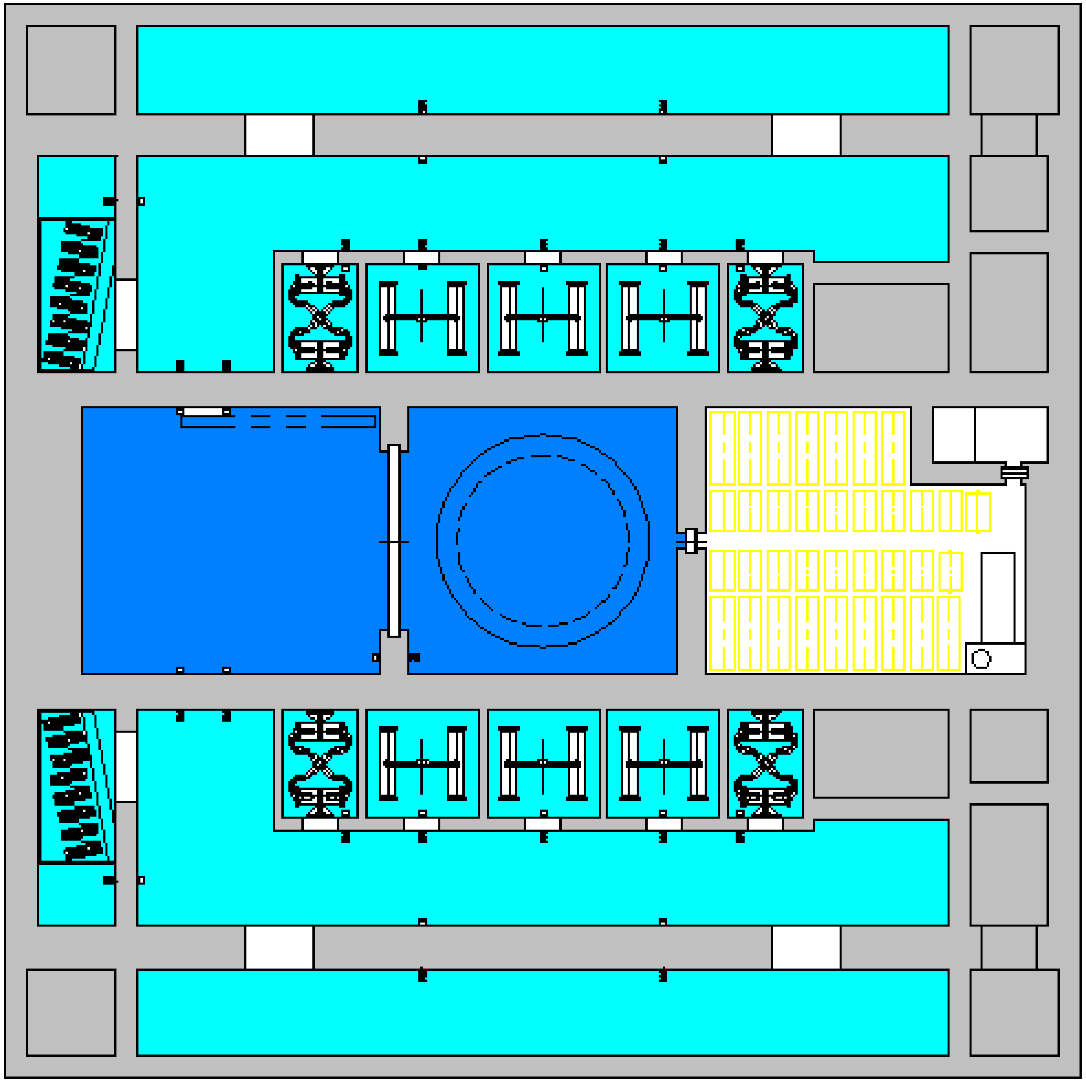
Isolation Condenser System



Passive Containment Cooling



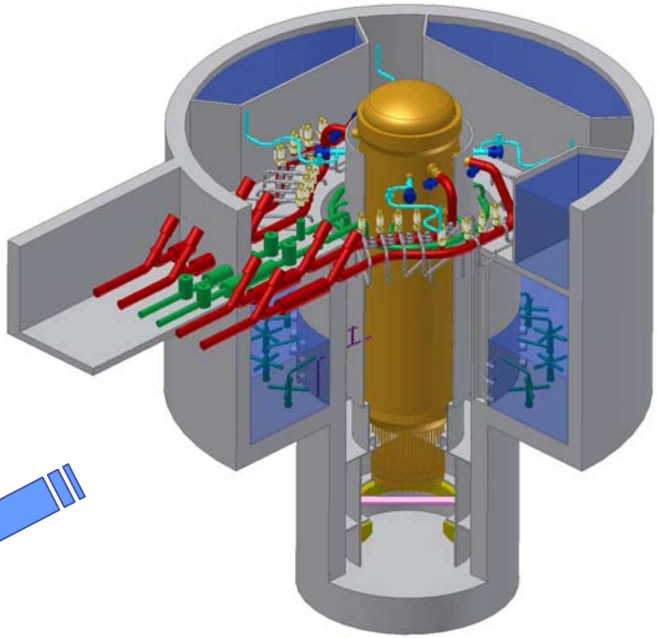
72 Hours Passive Capability

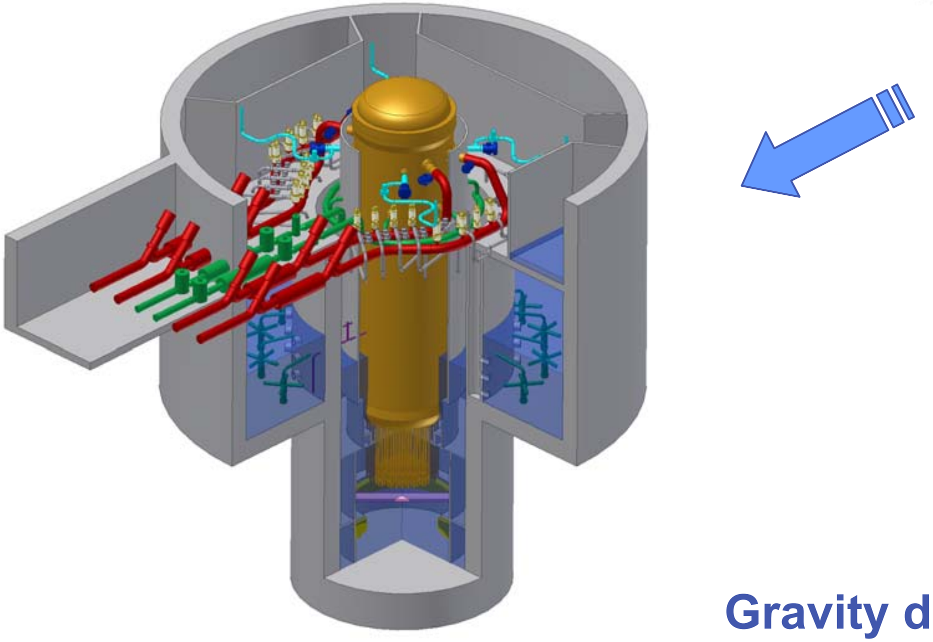


Gravity Driven Cooling System …

Simple design
Simple analyses

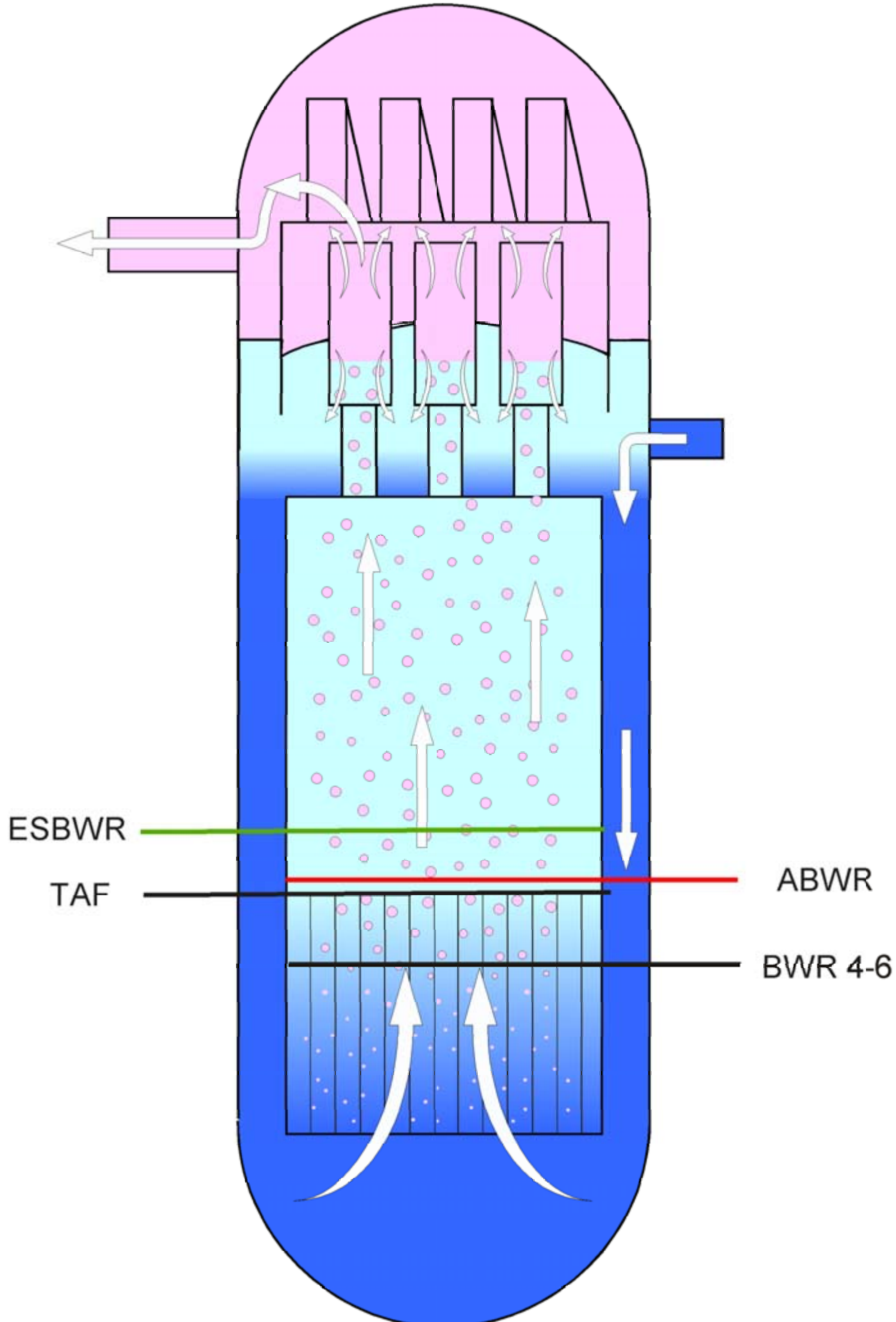
Extensive testing
Large safety margins



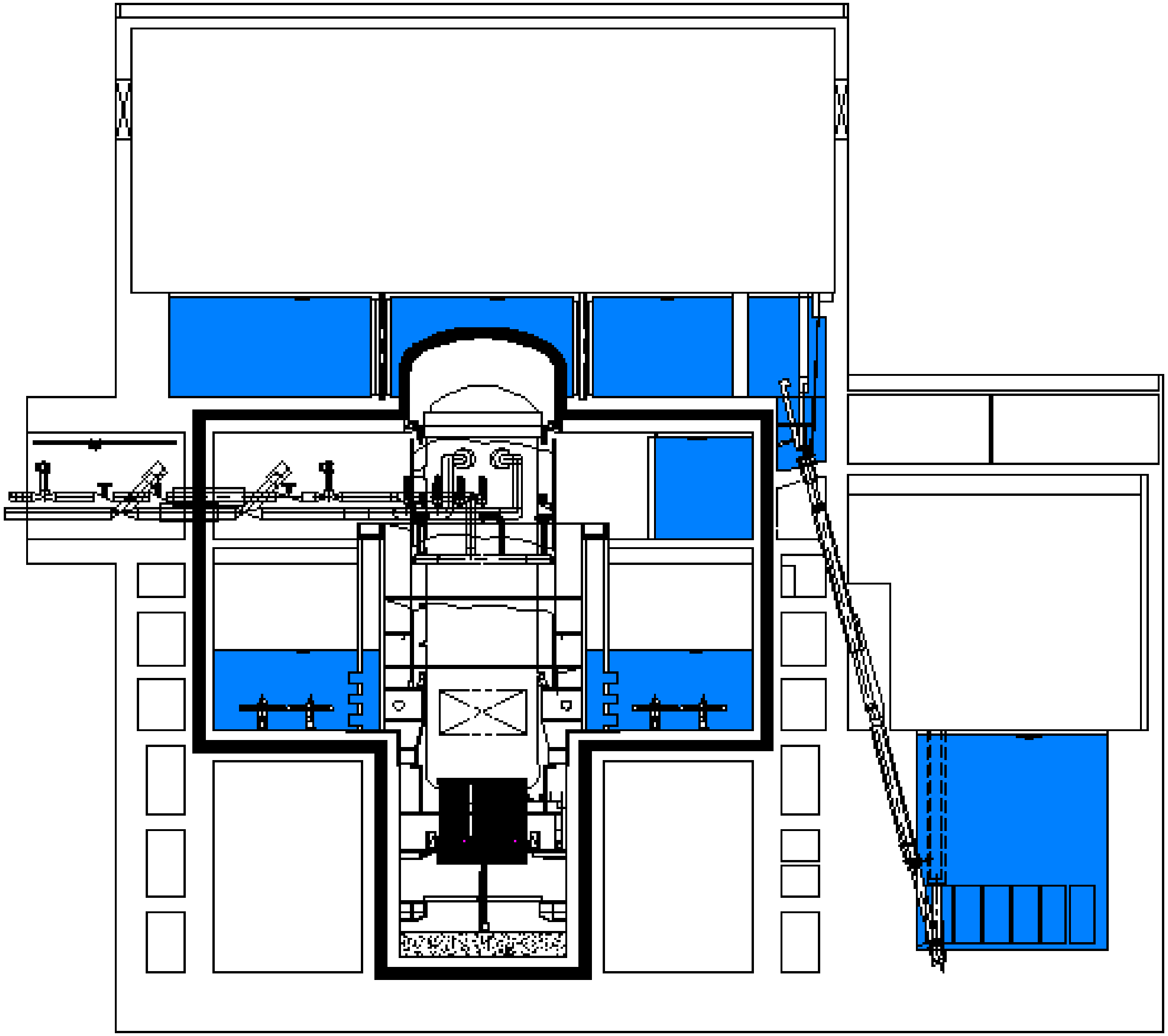


Gravity driven flow keeps core covered

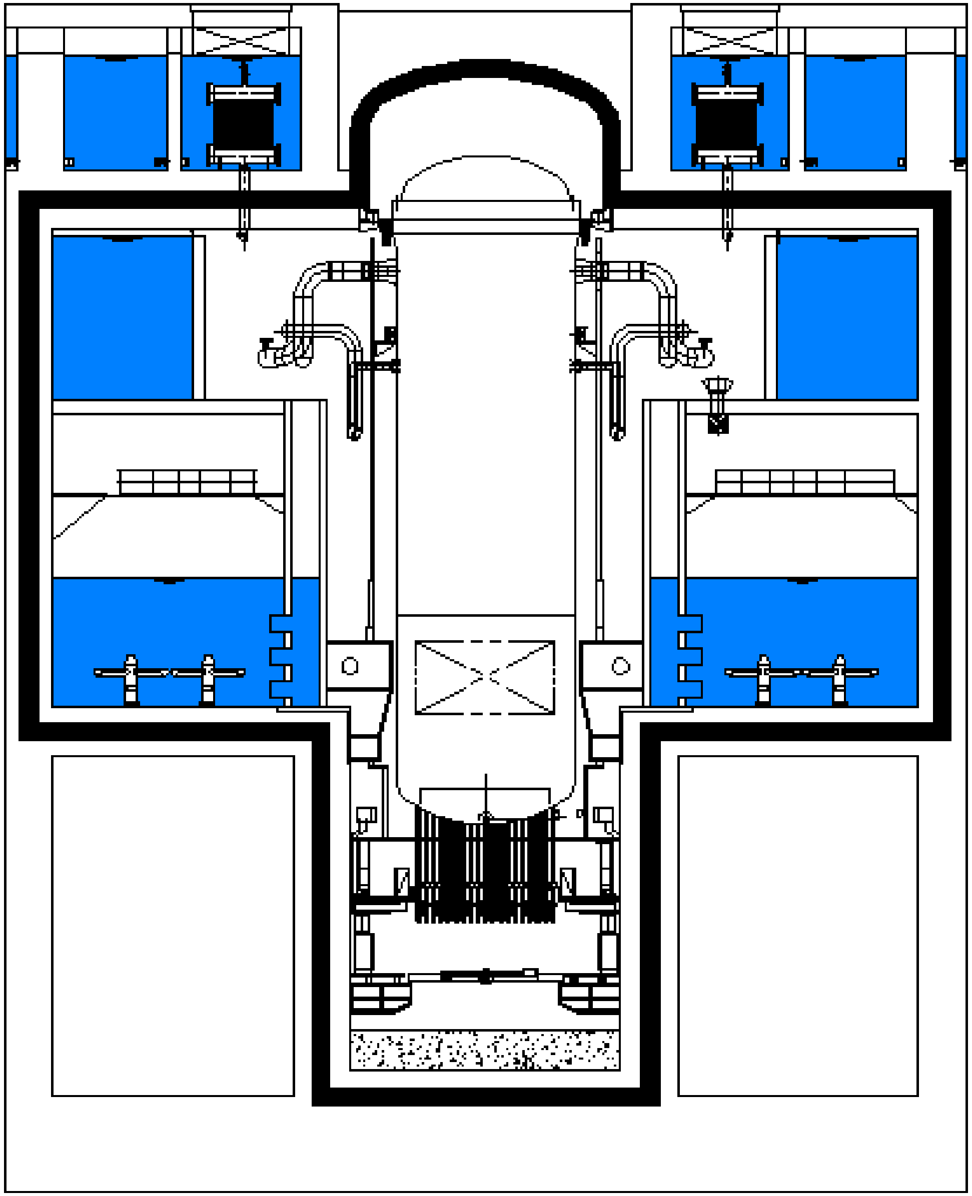
LOCA Water Level Response



Reactor and Fuel Building



Containment



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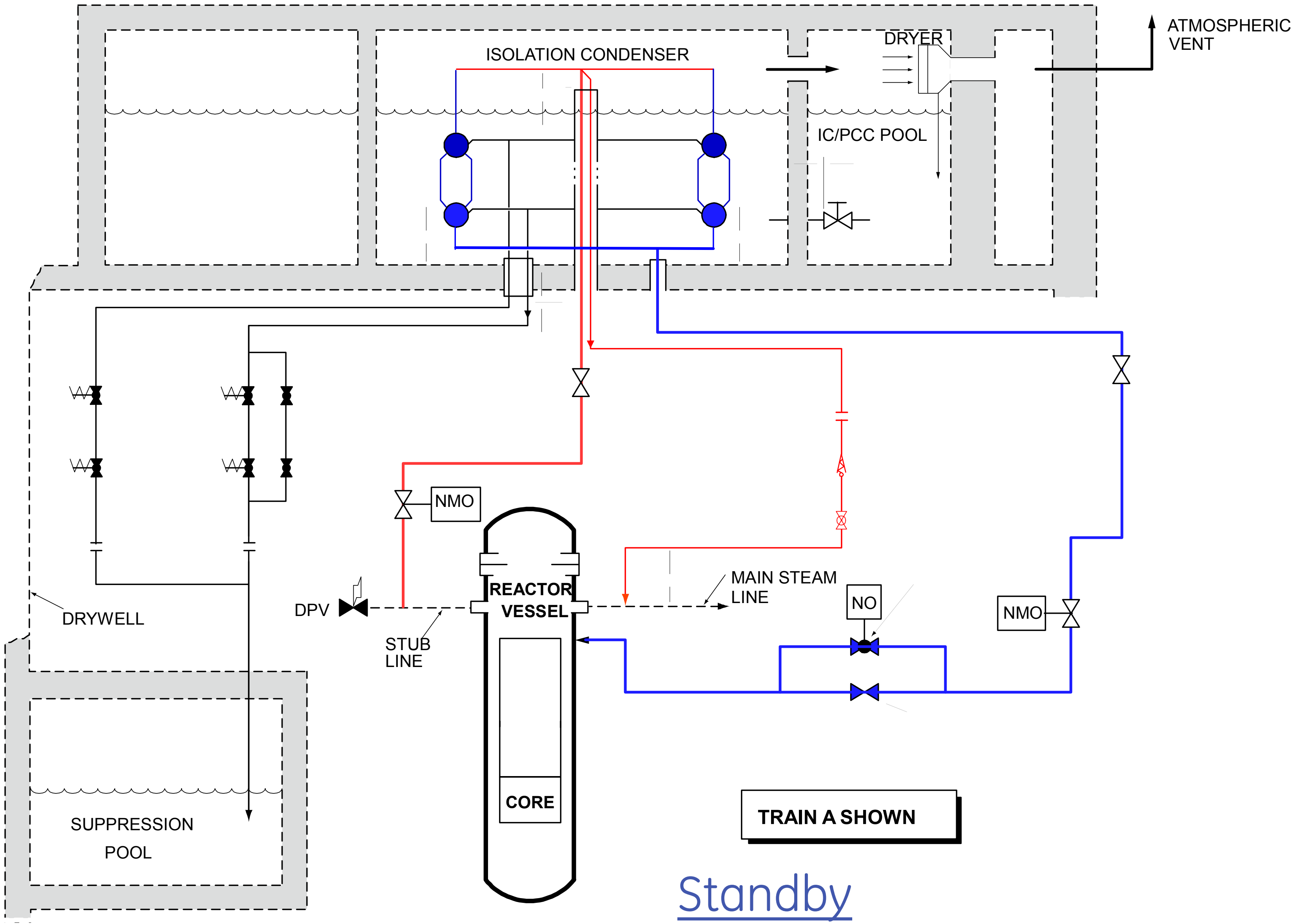
Isolation Condensers

- ICs provide passive decay heat removal
  - Single Failure Criteria apply
  - No lift of the Safety Relief Valves (SRVs)
  - Operates in all Design Basis Conditions except medium and large break LOCAs
  - ICs transport decay heat direct from NSSS to the Ultimate Heat Sink
    - No steaming in the primary containment
  - Rapidly reduces RPV pressure
  - Redundant Active Components


ISOLATION CONDENSER
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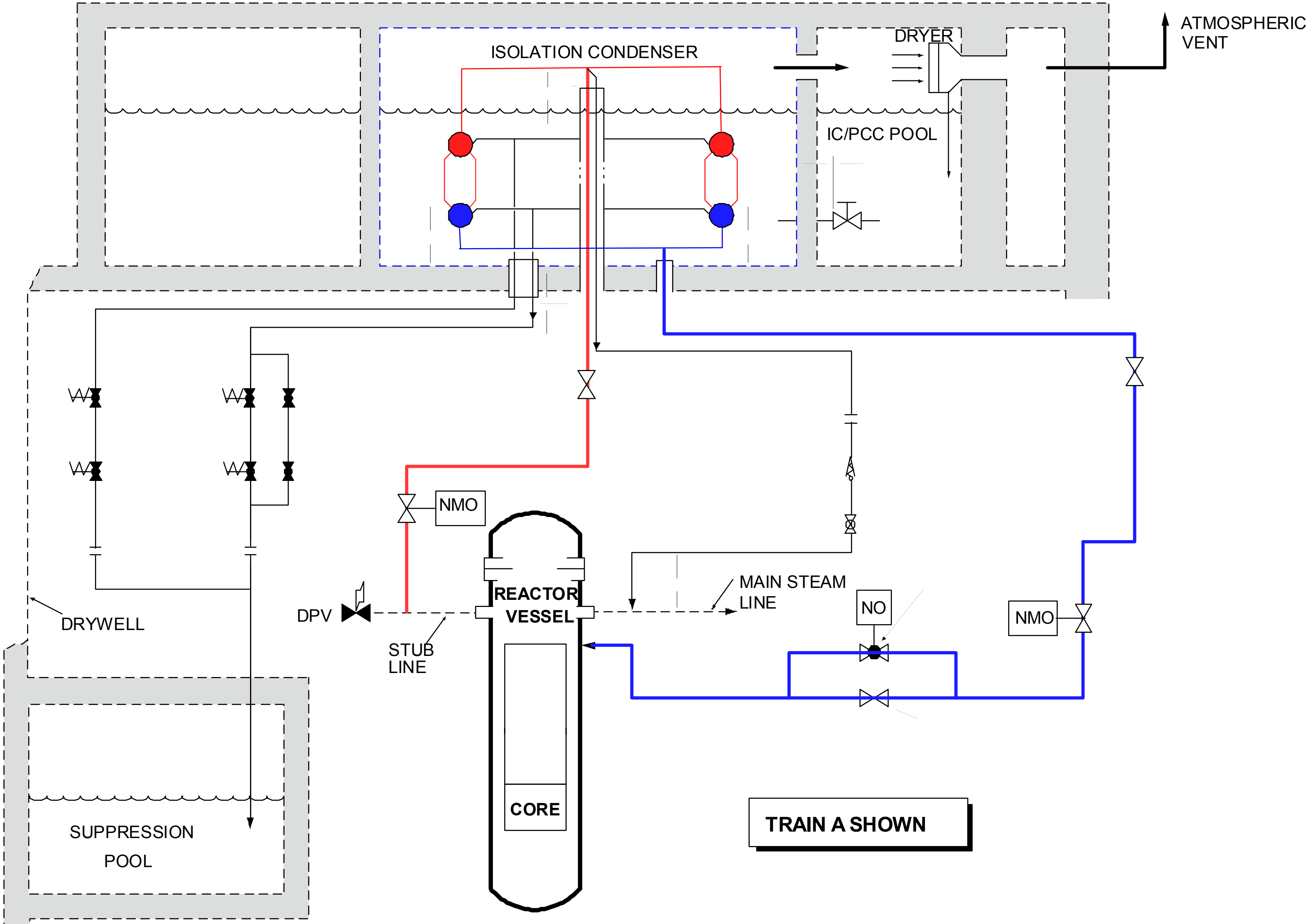
ATMOSPHERIC
VENT
IC/PCC POOL
NMO
REACTOR
VESSEL
MAIN STEAM
LINE
NO
NMO
DPV
STUB
LINE
DRYWELL
CORE
SUPPRESSION
POOL
TRAIN A SHOWN

Standby

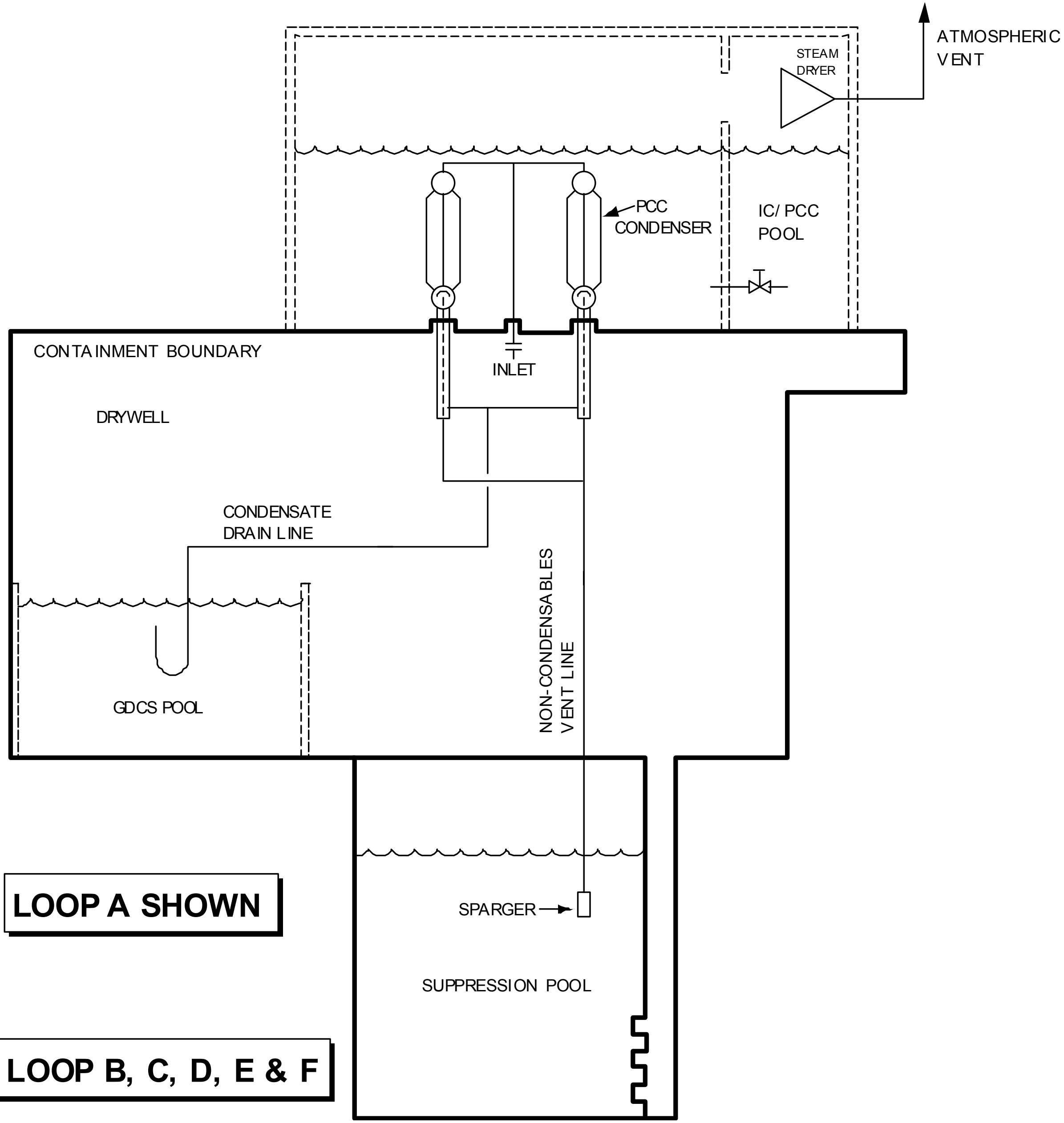


ISOLATION CONDENSER
DRYER
ATMOSPHERIC VENT
IC/PCC POOL
NMO
DPV
REACTOR VESSEL
STUB LINE
MAIN STEAM LINE
NO
NMO
DRYWELL
CORE
SUPPRESSION POOL
TRAIN A SHOWN

Operation

Passive Containment Cooling

- PCCs provide passive decay heat removal from the primary containment
  - > Operates in medium and large break LOCAs
  - > Provides backup of ICs if needed
    - – RPV is depressurized using DPVs
  - > Entirely Passive
    - > ~40 hours with demineralized water
  - > PCCs transport decay heat direct from Primary Containment to the Ultimate Heat Sink



LOOP A SHOWN

TYP LOOP B, C, D, E & F

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Emergency Core Cooling (ECC)

- Gravity Driven Cooling System (GDCS)
  - Three Pools
  - Four Trains
- Automatic Depressurization System (ADS)
  - 10 of 18 Safety Relief Valves (SRV)
    - Pneumatic actuation
  - 8 Depressurization Valves (DPV)
    - Squib actuated

Emergency Core Cooling (cont)

- Core remains covered for entire range of Design Basis Accidents
  - > No fuel heat-up
- Complies with 10 CFR 50.46
  - > Codes have been approved by NRC
- Stored water is sufficient to flood containment and RPV to above the top of fuel
  - > 1 meter above TAF

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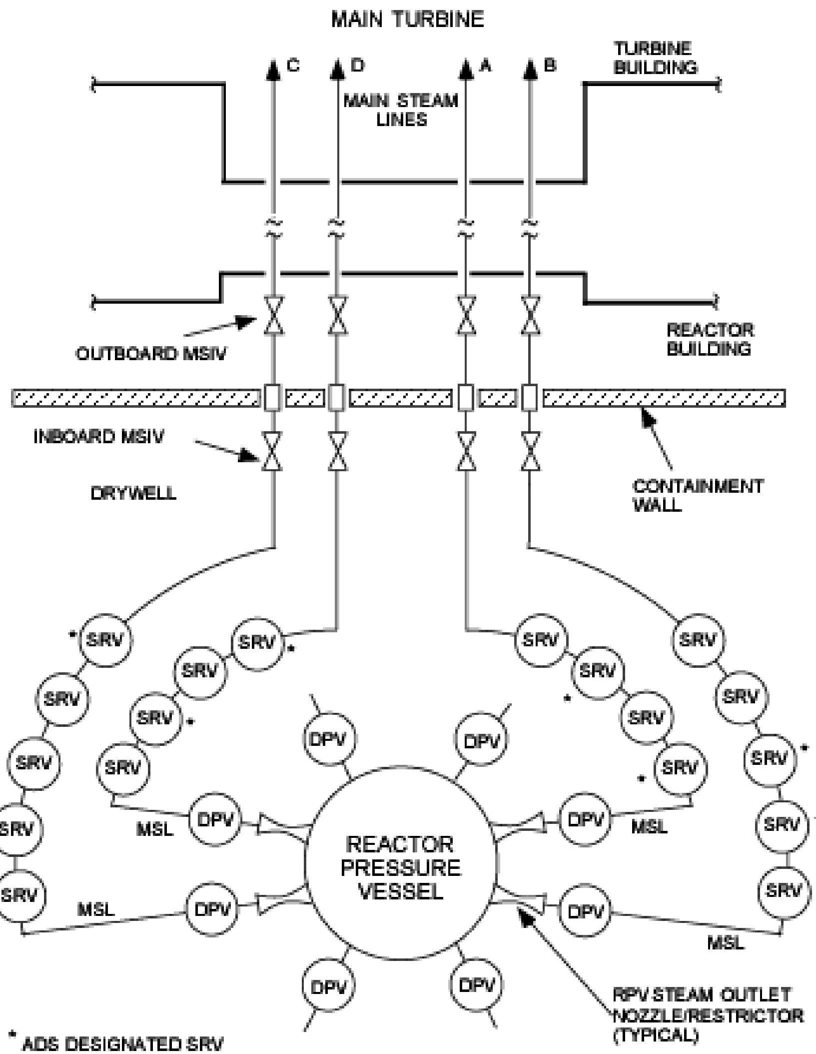
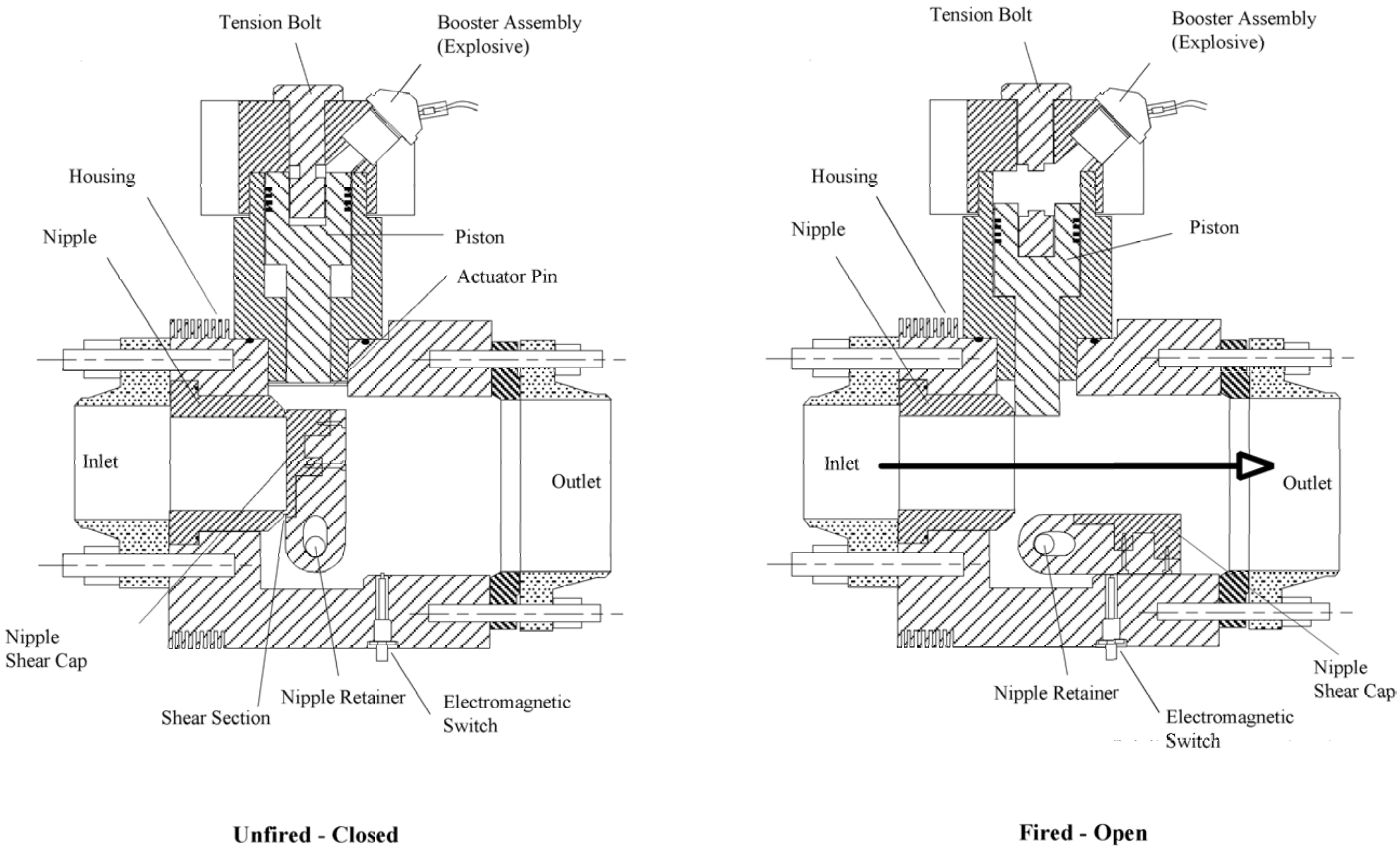
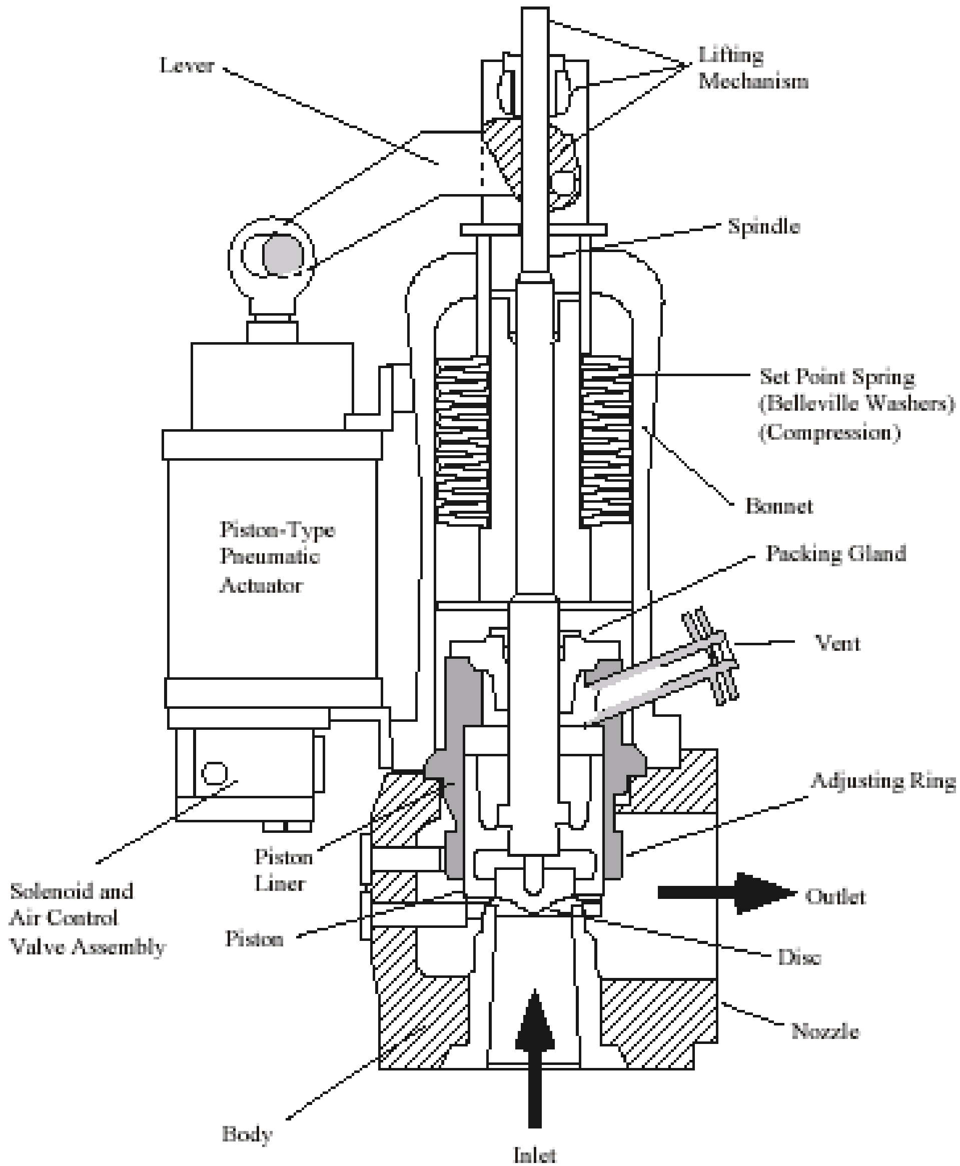
MSIV, SRV and DPV Arrangement



Depressurization Valve (DPV)



Depressurization Valve
Cross Section



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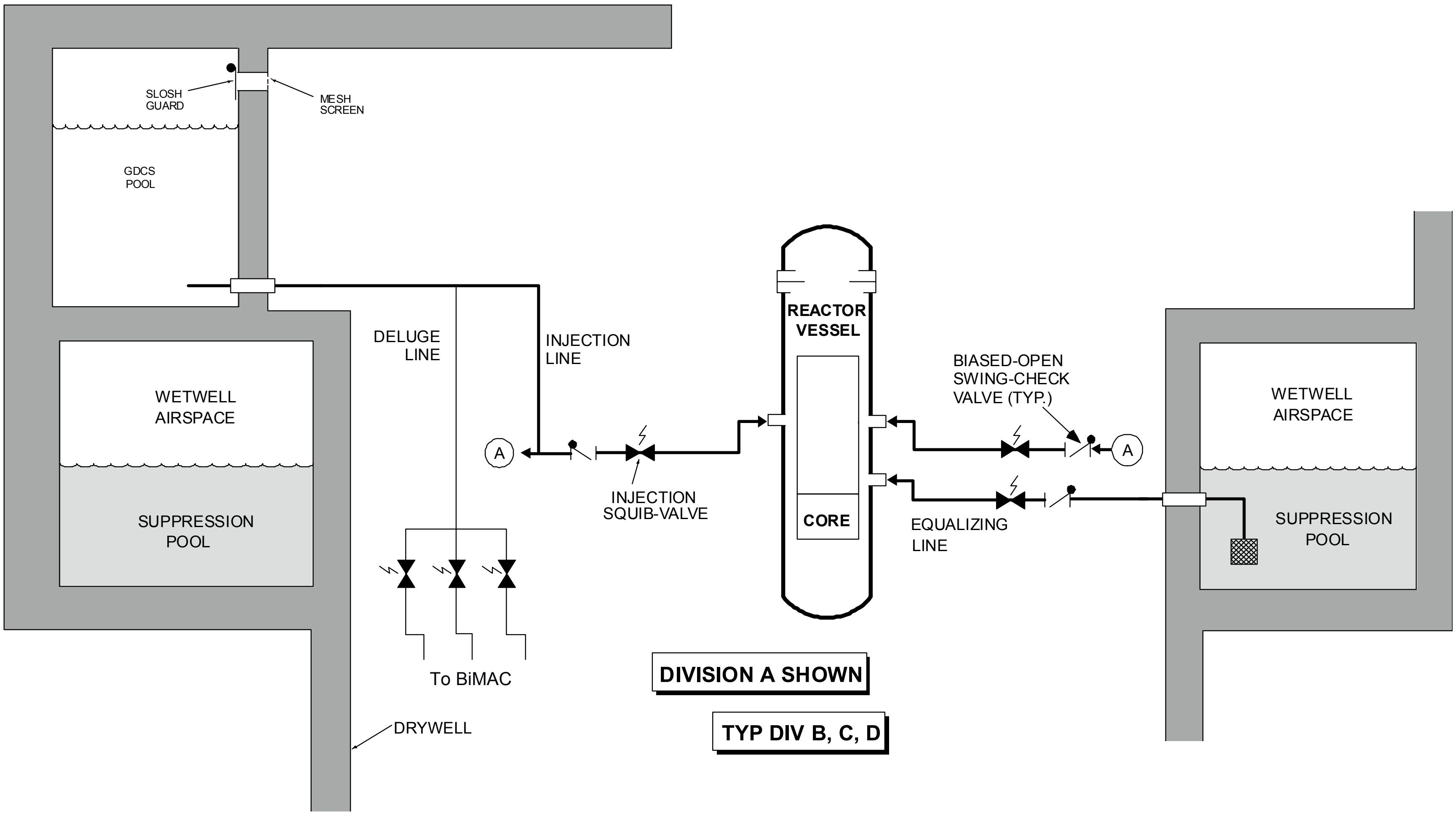

Gravity-Driven Cooling System

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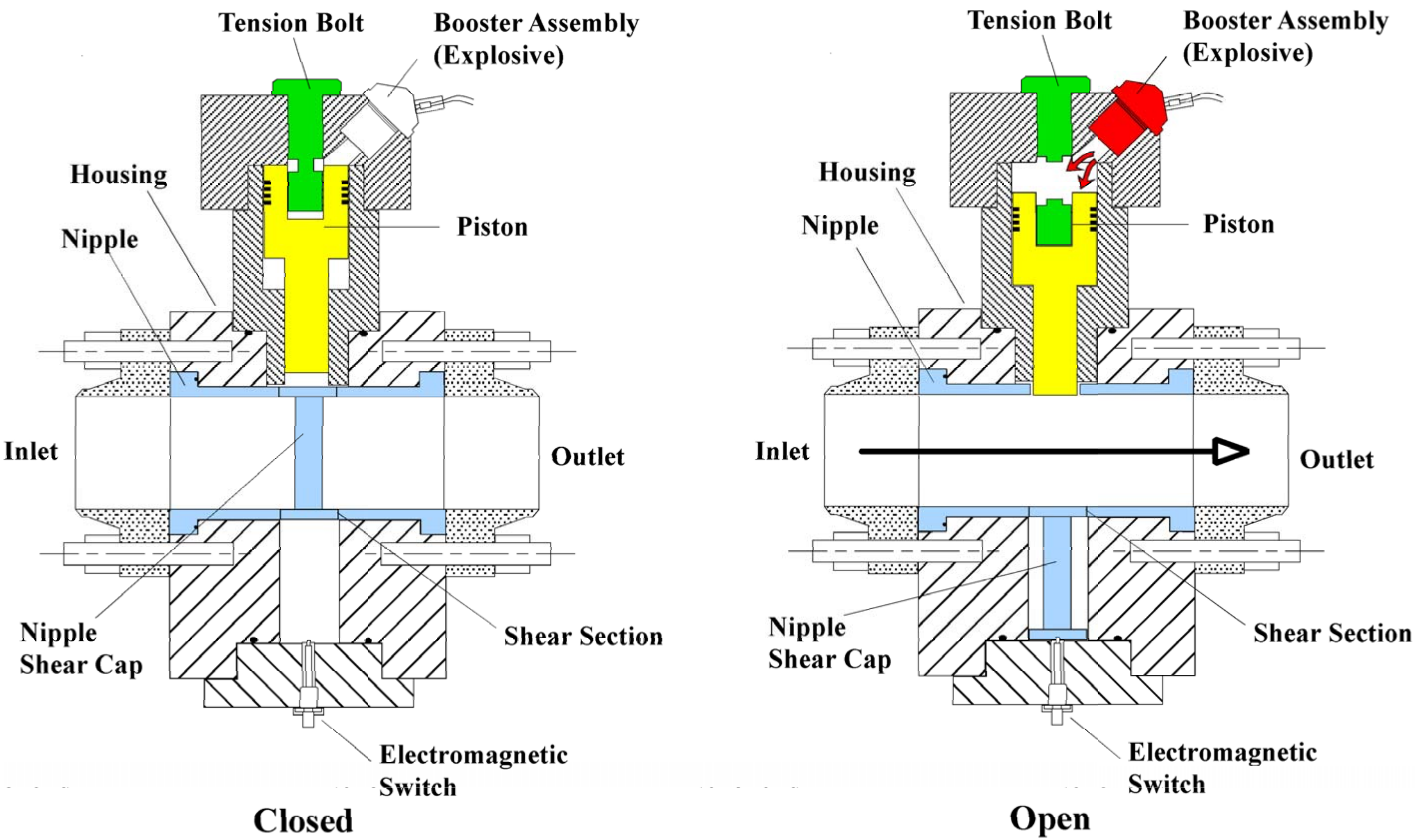

Tension Bolt
Booster Assembly (Explosive)
Housing
Nipple
Piston
Inlet
Outlet
Nipple Shear Cap
Shear Section
Electromagnetic Switch
Closed
Tension Bolt
Booster Assembly (Explosive)
Housing
Nipple
Piston
Inlet
Outlet
Nipple Shear Cap
Shear Section
Electromagnetic Switch
Open

Other Safety-Related Passive Systems

- DC Power Supplies
  - Battery banks
  - Inverters
  - Battery Chargers
- Emergency Breathing Air System
  - Main Control Room Habitability
- Standby Liquid Control (SLC)
  - Two Pressurized Tanks of Boron

Safety-Related Electrical

- Four Divisions
- DC Backed
  > Inverted power for AC loads
  > 4 Divisions with 24 hours Capability
    - Monitor
    - Control
  > 2 divisions with 72 hours Capability
    - Monitor

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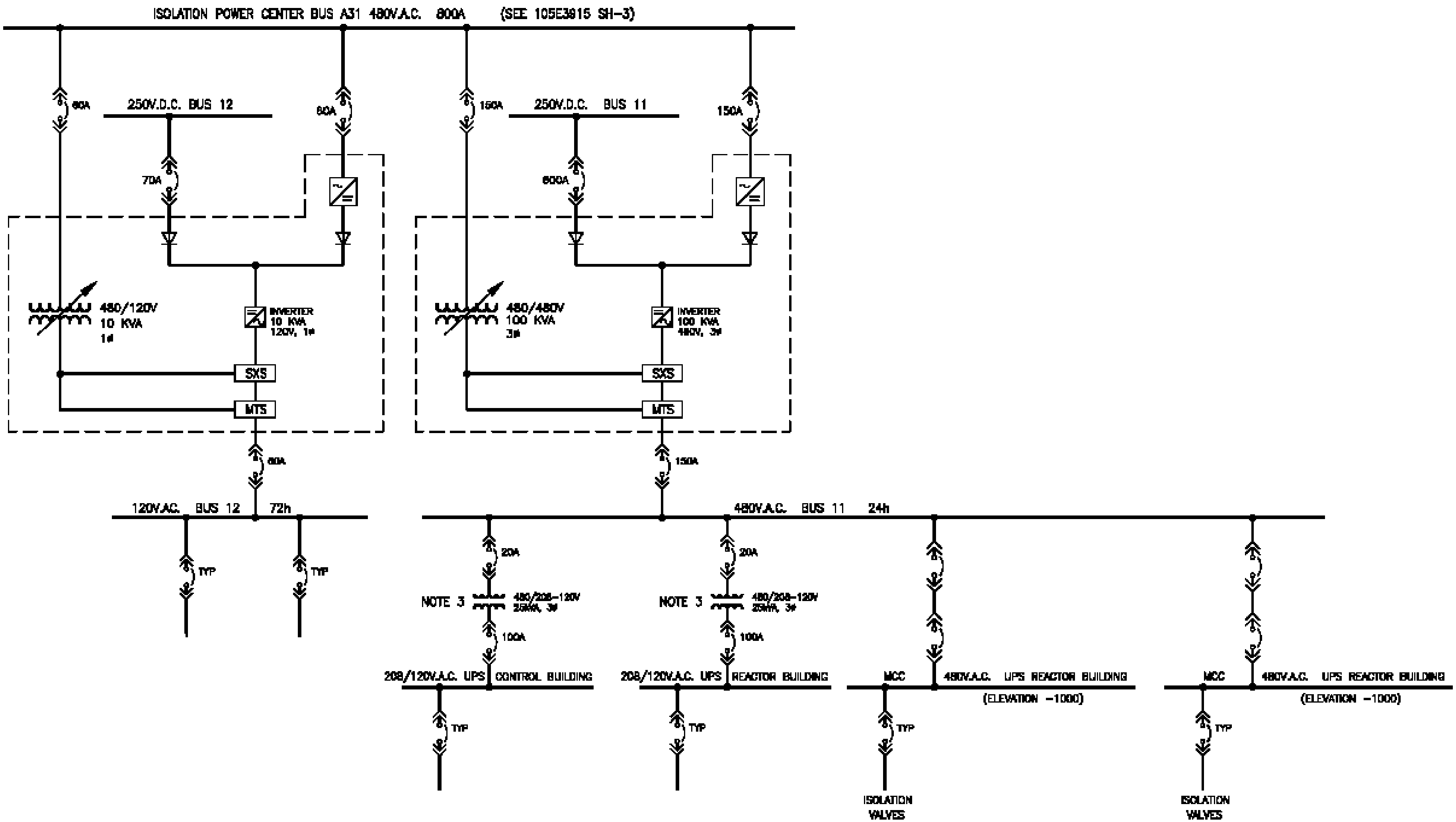
1E Electrical Arrangement



CLASS 1E UNINTERRUPTIBLE POWER SUPPLY DIVISION I

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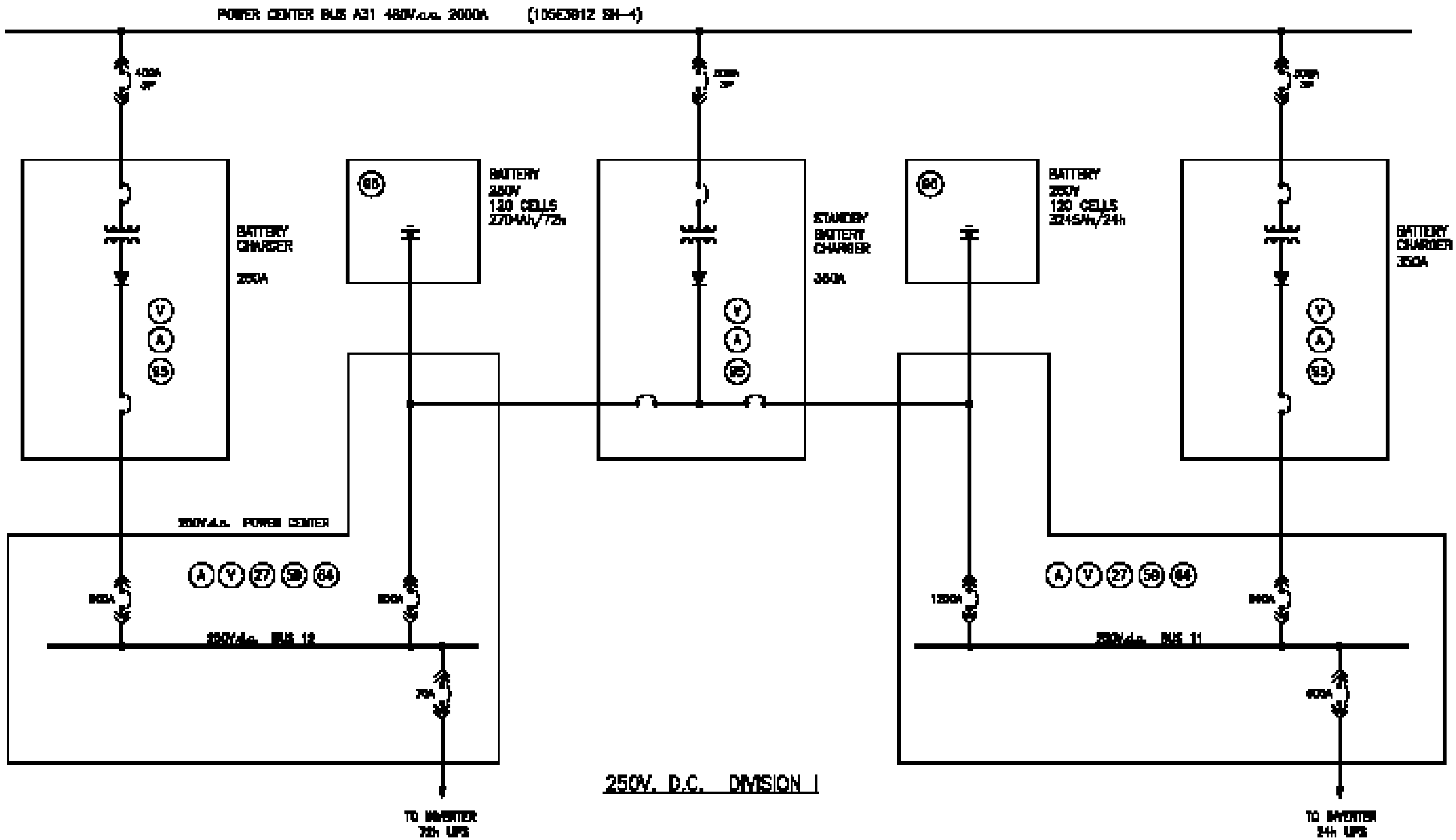
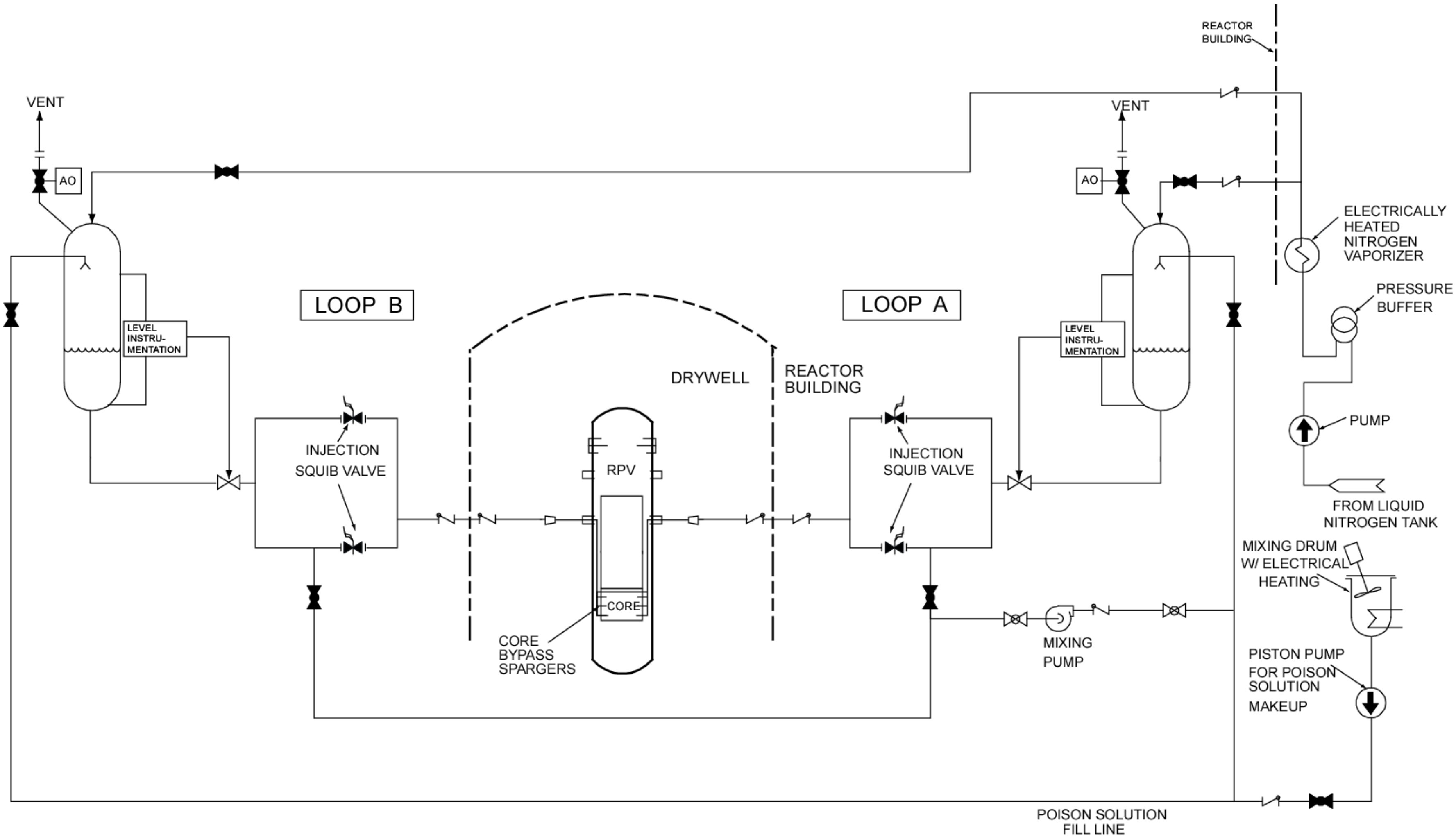
1E Electrical Arrangement (cont)

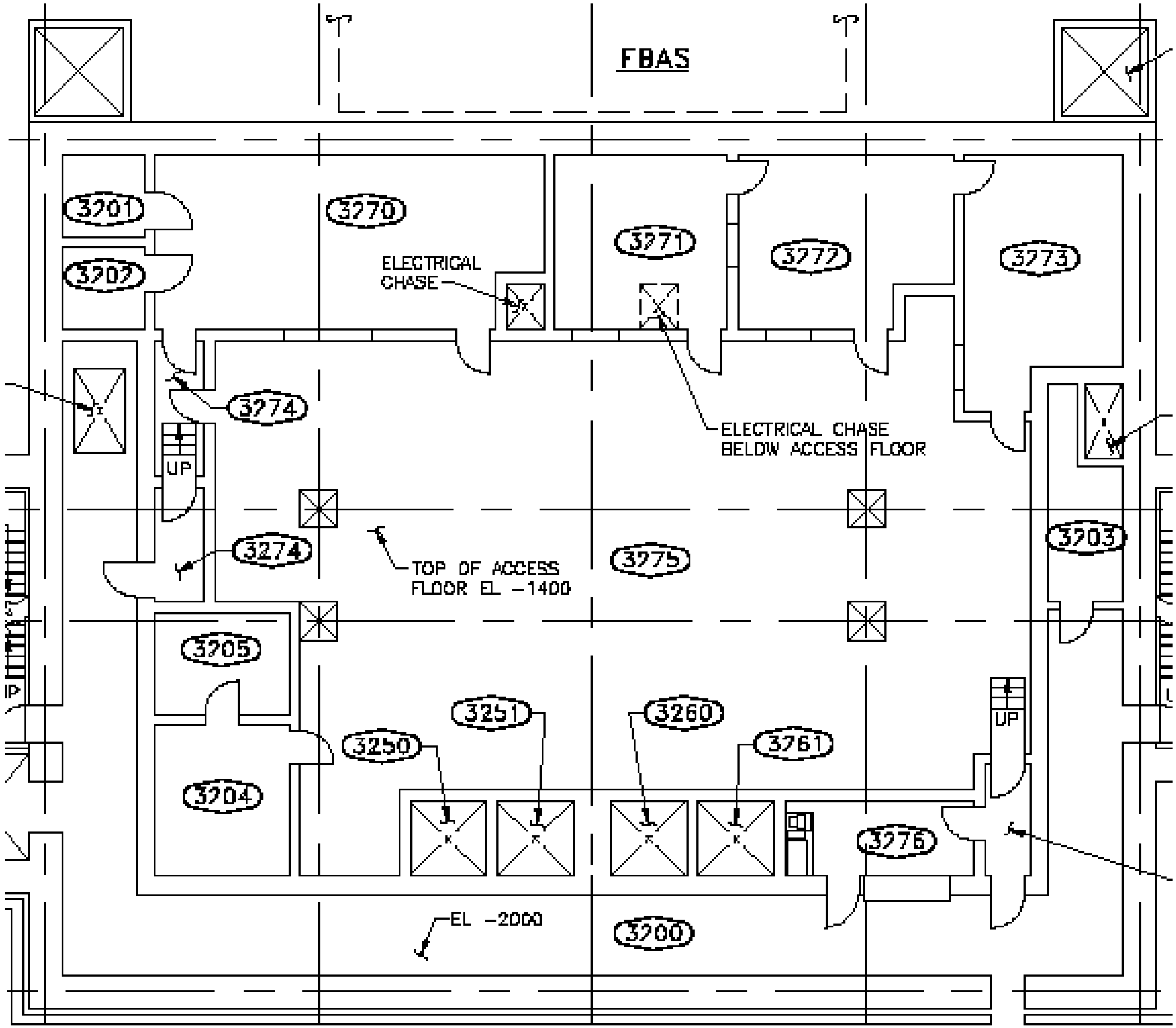


Standby Liquid Control

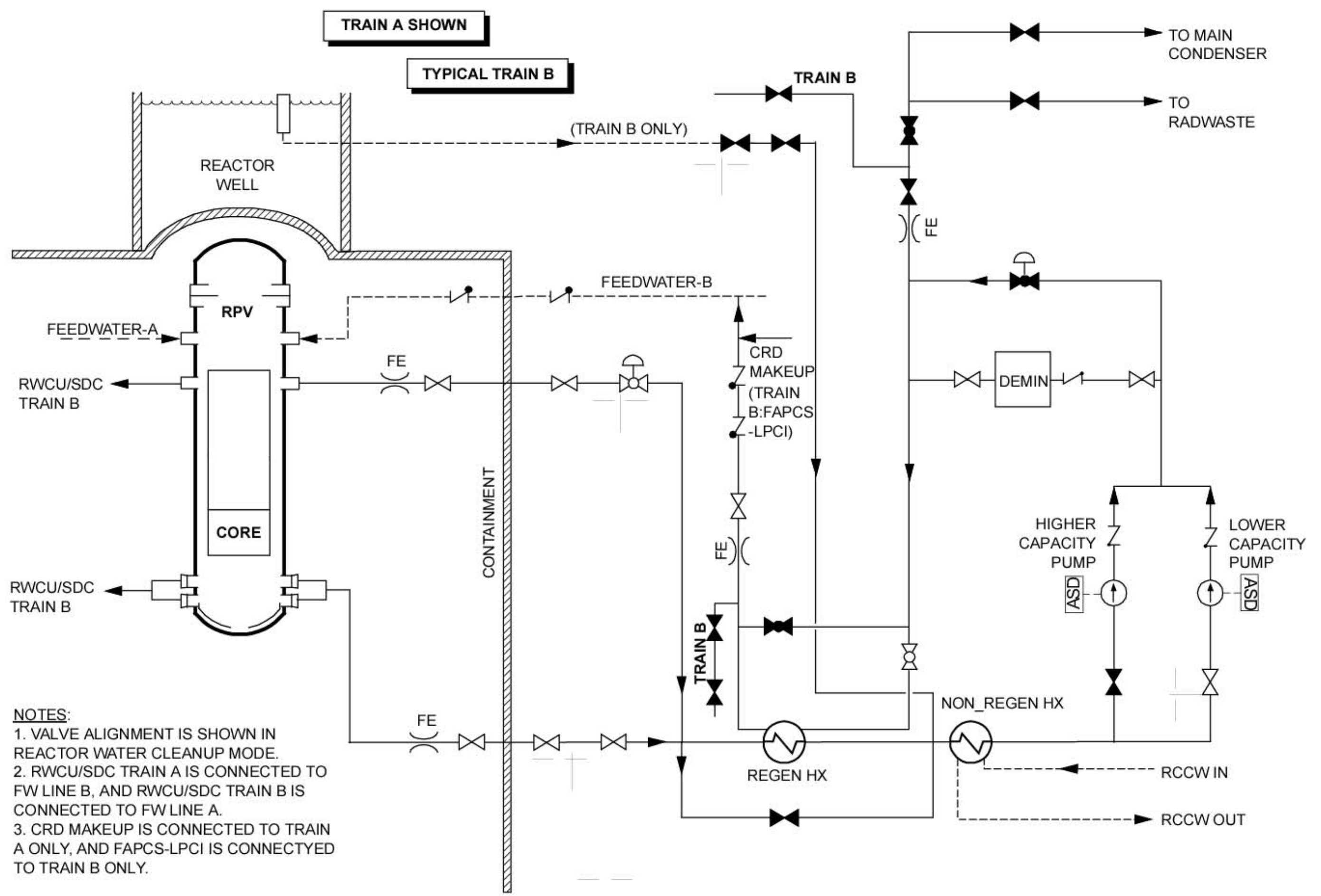


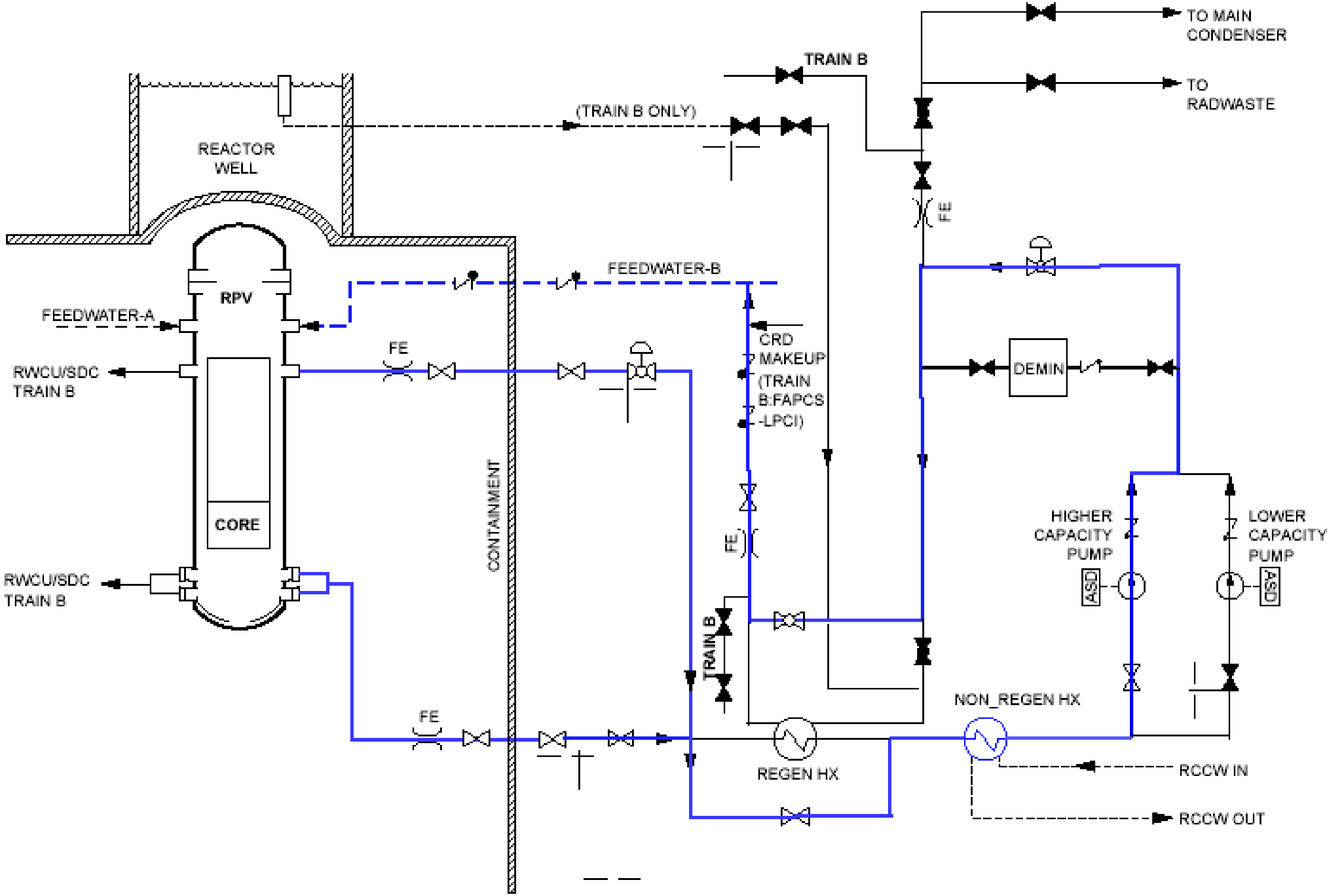
Emergency Breathing Air System

- Main Control Room Habitability
  - > Pressurized space 1/8 inch water gauge
  - > EBAS safety-related
    - – Single Failure Proof
    - – 72 hour passive capability
  - > MCR HVAC non-safety related
    - – With AC power availble
    - – 2 x 100% trains
    - – HEPA and Charcoal filtration


FBAS
3201
3270
3202
ELECTRICAL
CHASE
3271
3272
3273
3274
ELECTRICAL CHASE
BELOW ACCESS FLOOR
UP
3274
TOP OF ACCESS
FLOOR EL -1400
3275
3203
3205
3251
3260
3250
3261
UP
3204
3276
EL -2000
3200

Reactor Water Cleanup (RWCU)





Shutdown Cooling

Fine Motion Control Rod Drives (FMCRD)

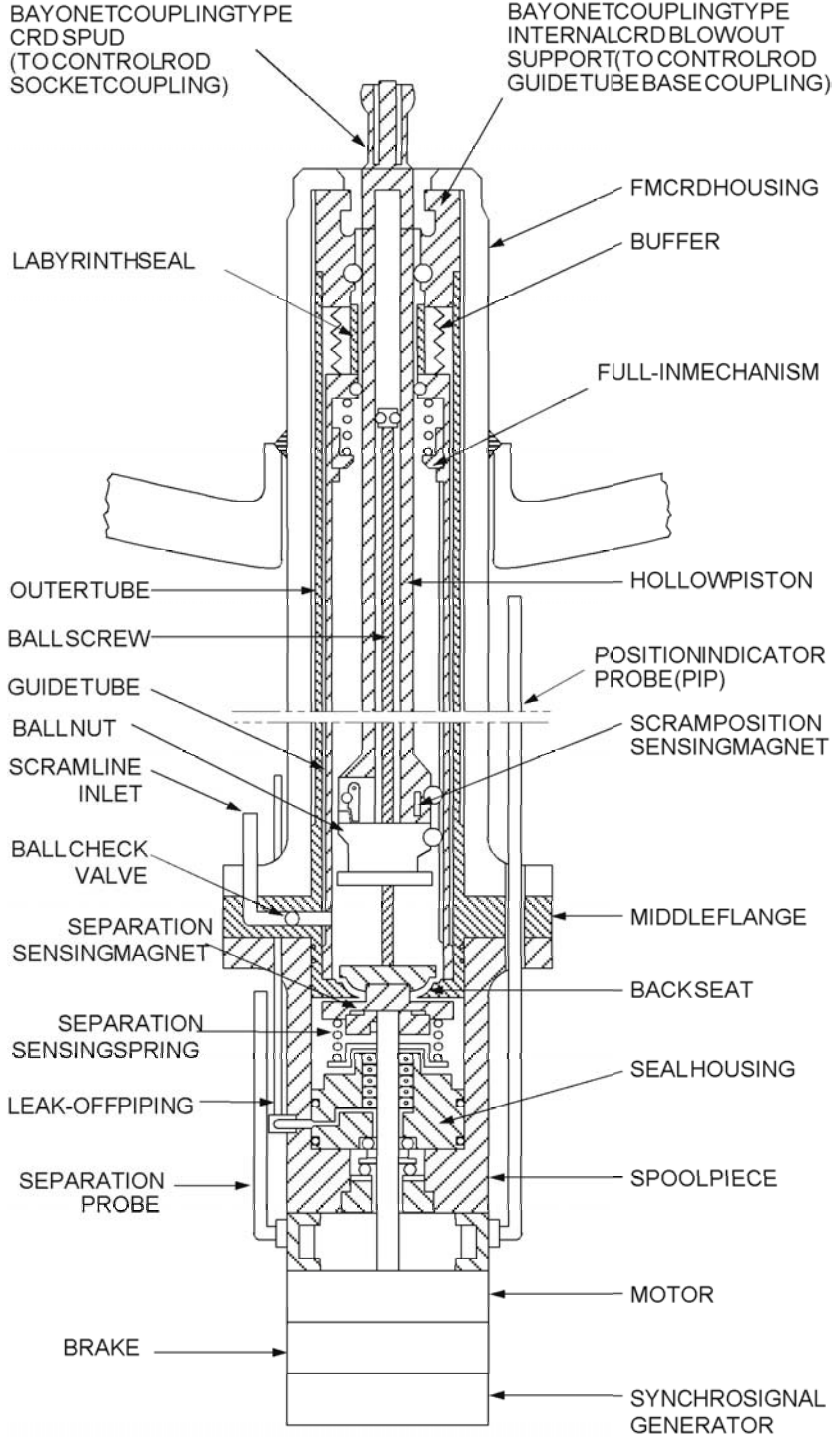
- 269 Control Rods
- Hydraulic Scram
  - > 1 HCU for 2 FMCRDs
  - > FMCRDs for 1 HCU are separated in core
  - > No Scram Discharge Volume
  - > Rapid Insertion
    - – ~1.1 seconds full out to full in
  - > Reduced maintenance
- Shoot-out Steel is eliminated

FMCRD (cont)

- Insertion and Withdrawl by Electric Motor
  - No overshoot
  - Can be ganged in groups as large as 26
  - Positioning Increments of ~3 inches
  - Rod Control and Information System (RCIS)
- Rod Drop Accident is no longer Credible
  - Detection of blade failure to follow drive
  - Check of blade to drive coupling integrity

FMCRD (cont)

- Power adjustments are made with rod movement
  - Select Control Rod Rapid Insertion (SCRRI), provides a means for rapid power reduction
- Maintenance
  - Hydraulic portions surveillance primarily
  - Electrical requires no break of pressure boundary



imagination at work

Appendix H: GA-HTGR Technical Summary

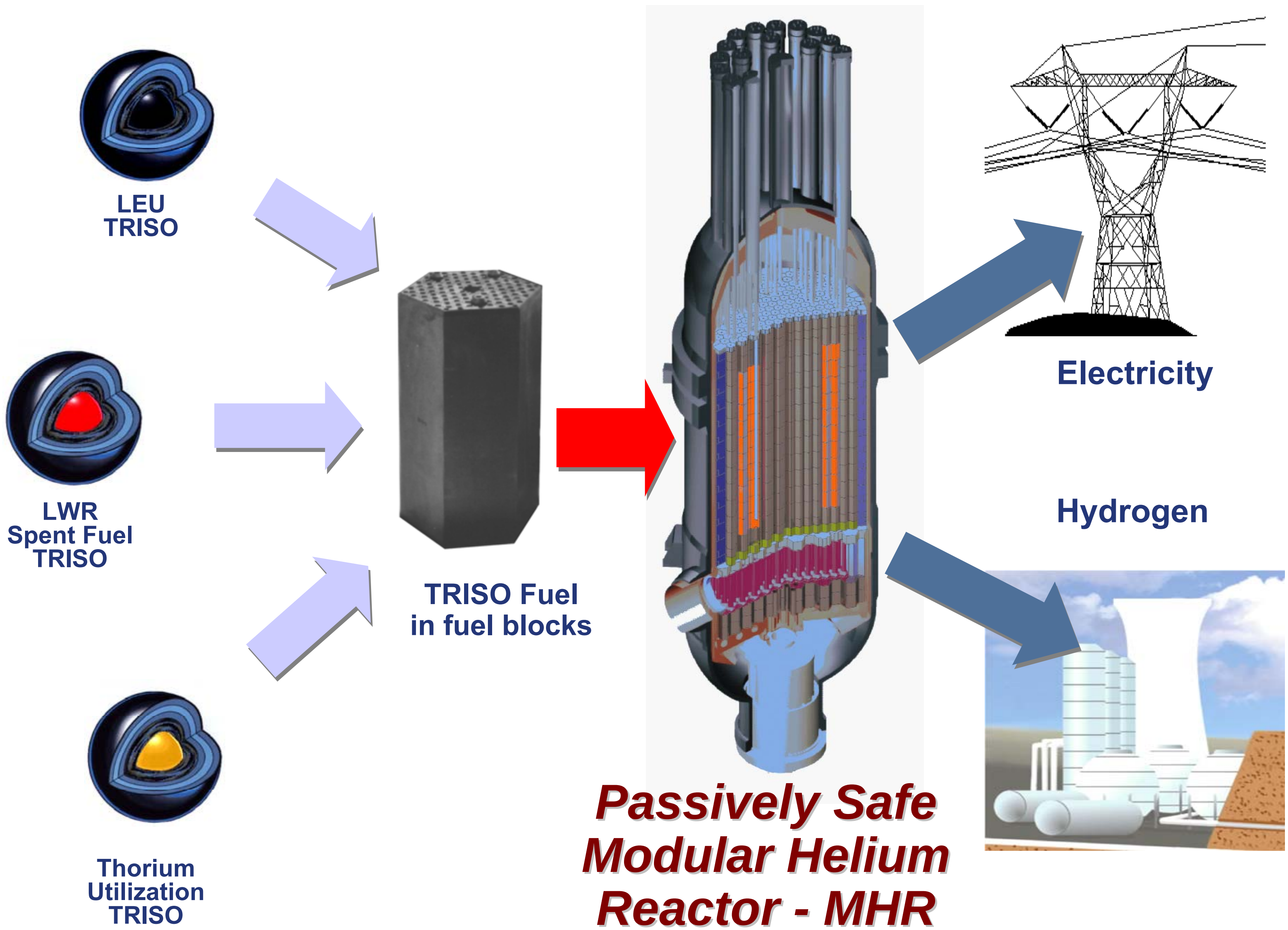
An overview and technical summary of the General Atomics GA-HTGR reactor design is provided in the following pages.

OVERVIEW OF
MODULAR HELIUM REACTOR
NUCLEAR POWER PLANTS

FOR THE SUPPLY OF
SAFE, CLEAN, ECONOMIC ENERGY

GENERAL ATOMICS

SINGLE REACTOR DESIGN HAS MULTIPLE APPLICATIONS



U.S. AND EUROPEAN TECHNOLOGY PROVIDE PROVEN BASES FOR PASSIVELY SAFE MHR

BROAD FOUNDATION OF HELIUM REACTOR TECHNOLOGY

EXPERIMENTAL REACTORS

DRAGON
(U.K.)
1963 - 76

AVR
(FRG)
1967 - 1988

DEMONSTRATION OF BASIC HTGR TECHNOLOGY

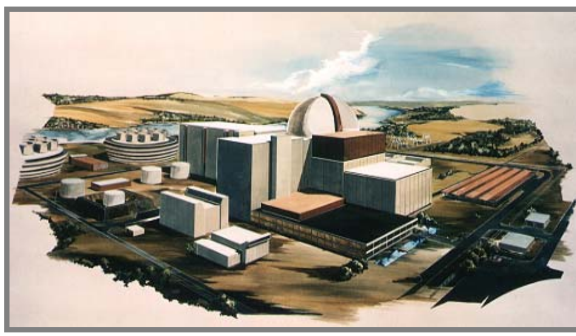
PEACH BOTTOM 1
(U.S.A.)
1967 - 1974

FORT ST. VRAIN
(U.S.A.)
1976 - 1989

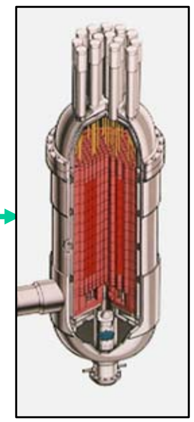
THTR
(FRG)
1986 - 1989

HTGR TECHNOLOGY PROGRAM

- **MATERIALS**
- **COMPONENTS**
- **FUEL**
- **CORE**
- **PLANT TECHNOLOGY**



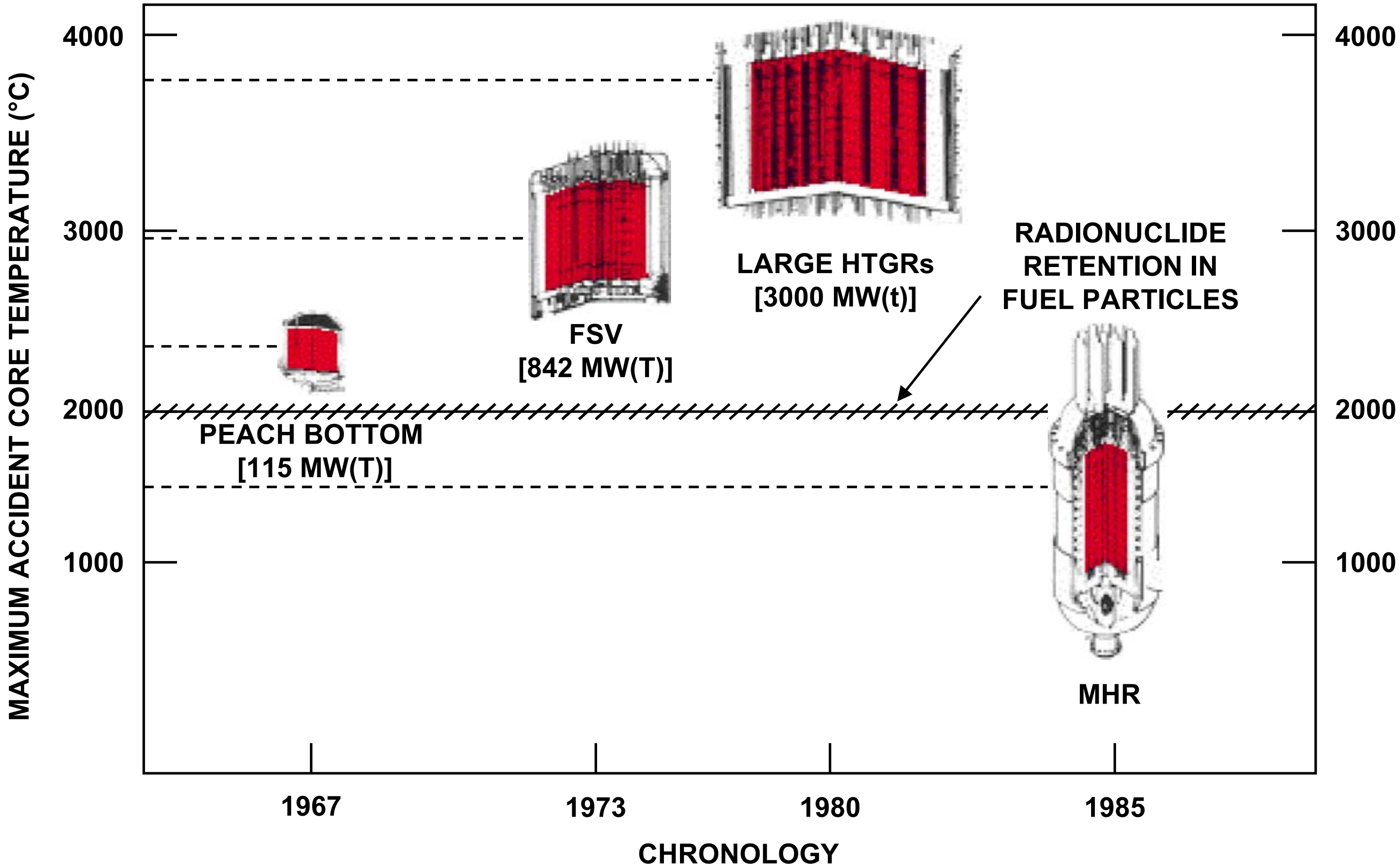
LARGE HTGR PLANTS



MHR

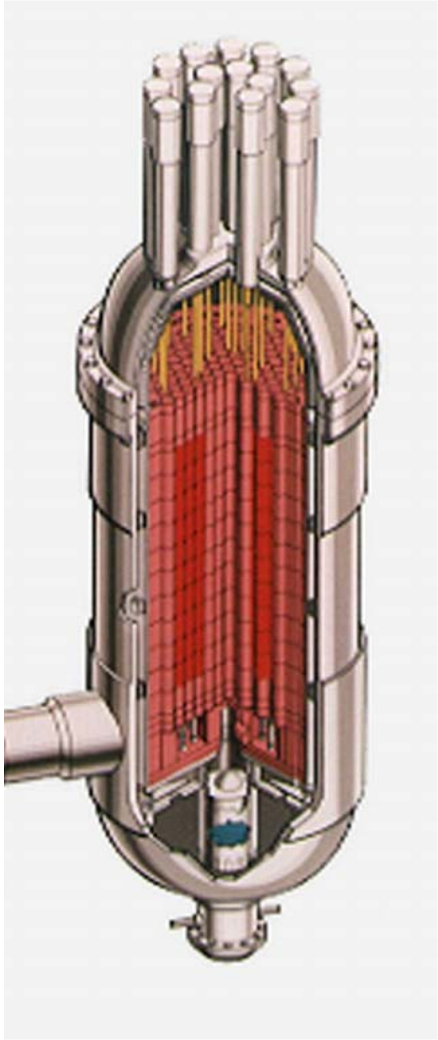
GENERAL ATOMICS

MHR REPRESENTS A FUNDAMENTAL CHANGE IN REACTOR DESIGN AND SAFETY PHILOSOPHY



...SIZED AND CONFIGURED TO TOLERATE EVEN A SEVERE ACCIDENT

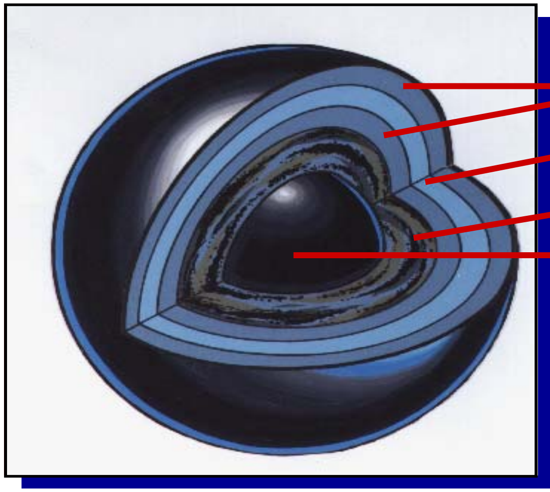
INHERENT REACTOR CHARACTERISTICS PROVIDE HIGH SAFETY



- **Helium gas coolant (inert)**
- **Refractory fuel (high temperature capability)**
- **Graphite reactor core (high temperature stability)**
- **Low power density (order of magnitude lower than LWRs)**
- **Demonstrated technologies**

. . . EFFICIENT, RELIABLE PERFORMANCE WITH INHERENT SAFETY

CERAMIC FUEL RETAINS ITS INTEGRITY UNDER SEVERE ACCIDENT CONDITIONS



TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right).

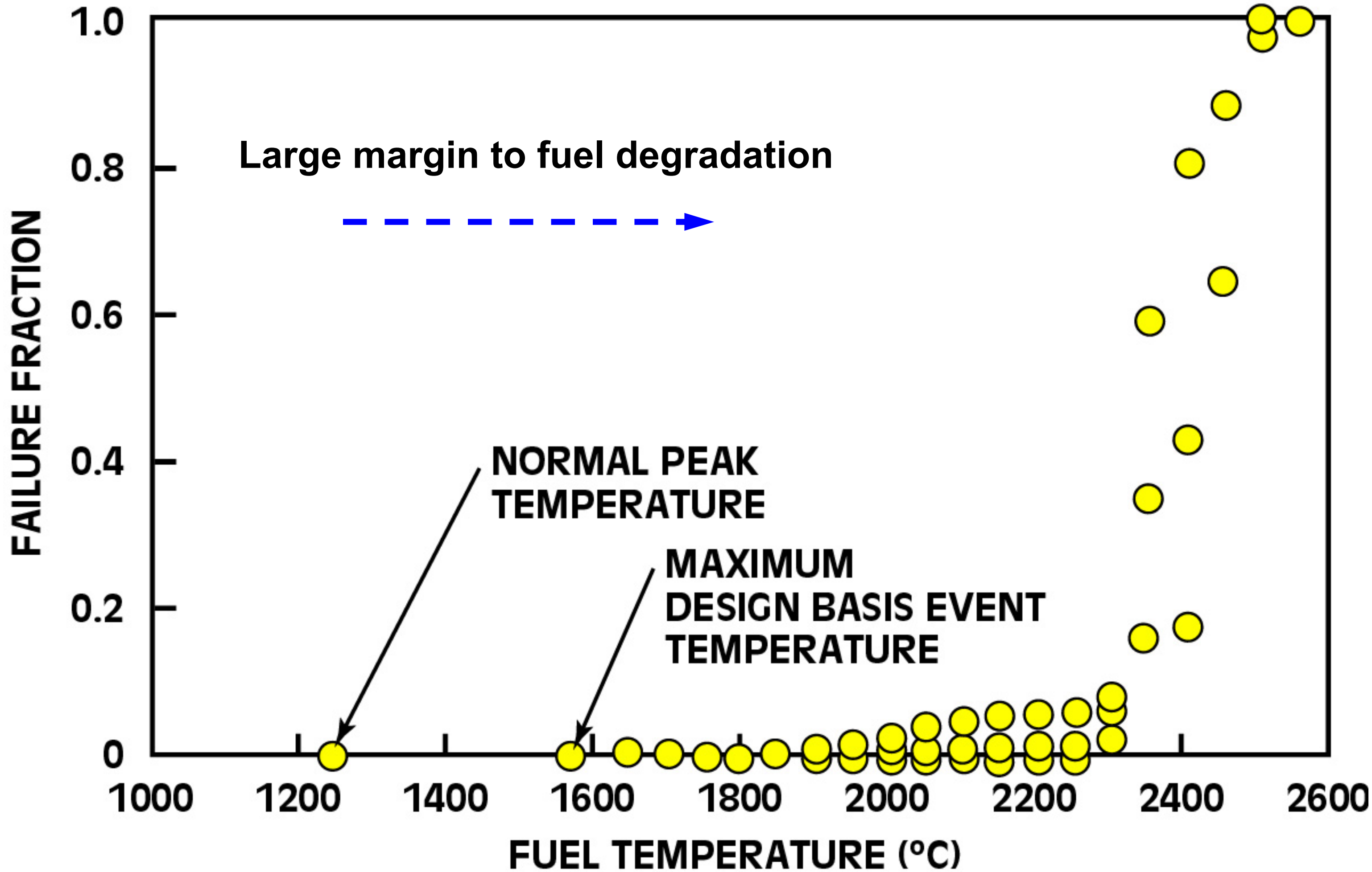
PARTICLES

COMPACTS

FUEL ELEMENTS

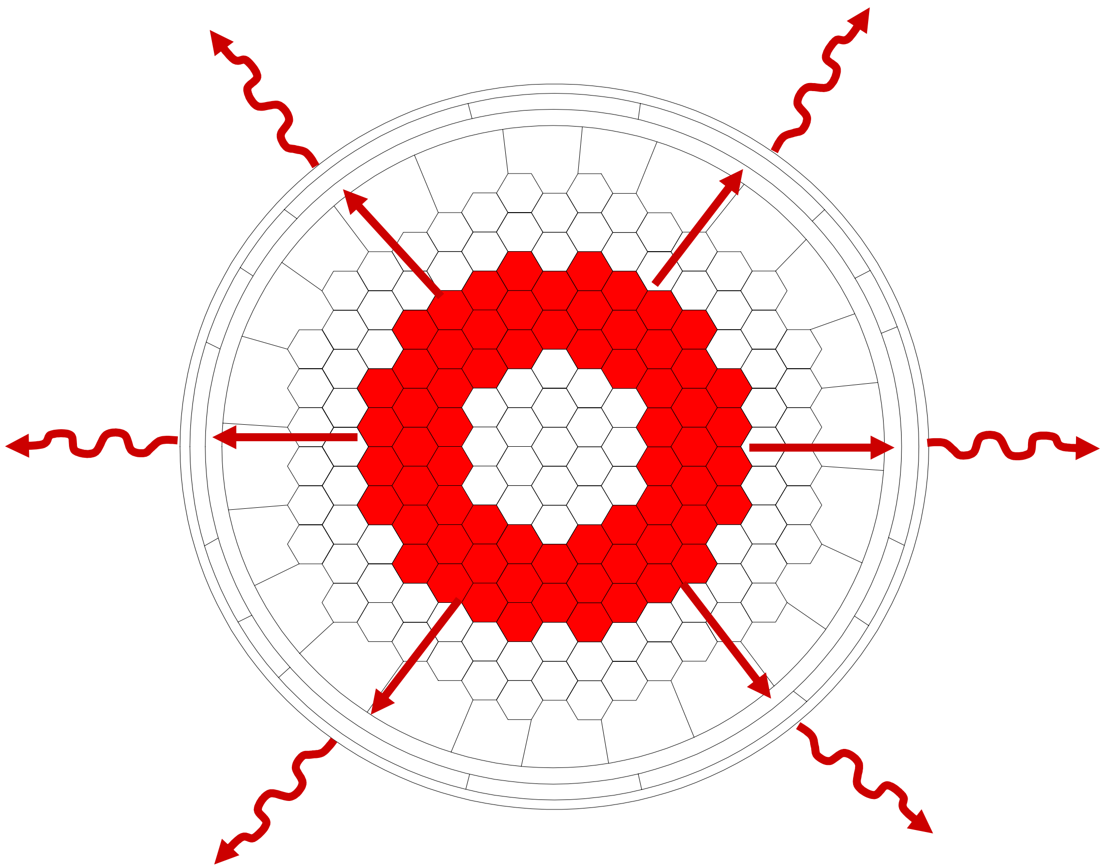
GENERAL ATOMICS

COATED PARTICLES STABLE TO BEYOND MAXIMUM ACCIDENT TEMPERATURES



GENERAL ATOMICS

ANNULAR REACTOR CORE LIMITS FUEL TEMPERATURE DURING ACCIDENTS

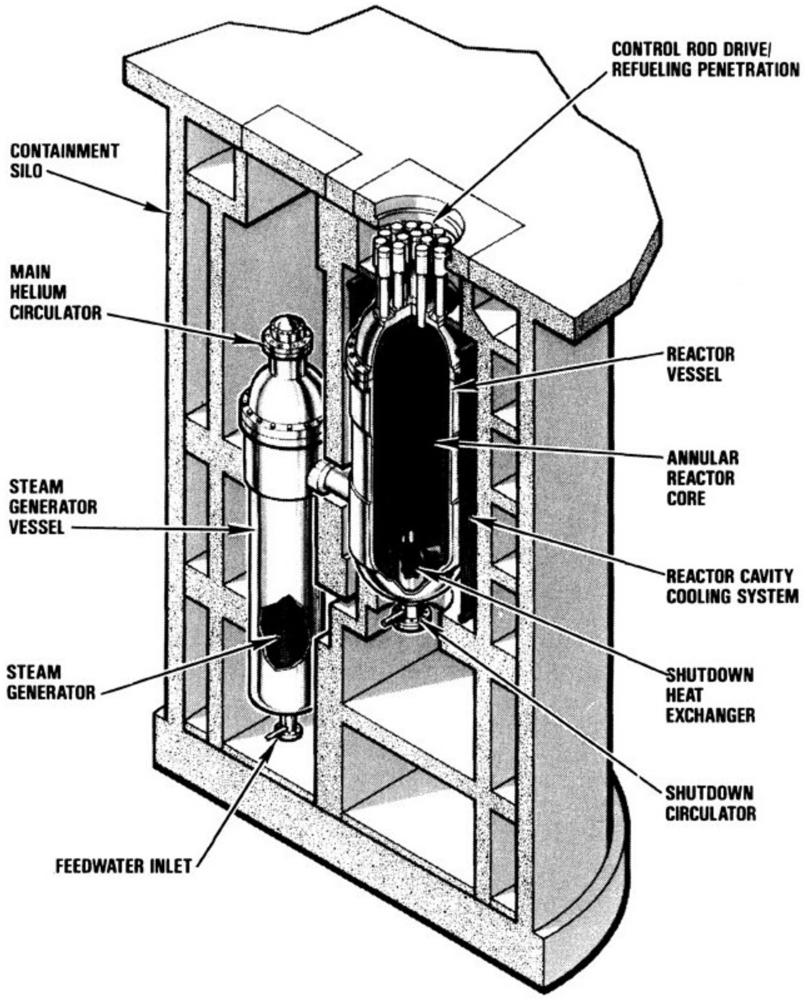


350 MWt
66 Columns
660 Elements

- Decay heat conducts radially outward to steel pressure vessel boundary
- Steel pressure vessel radiates heat into reactor cavity

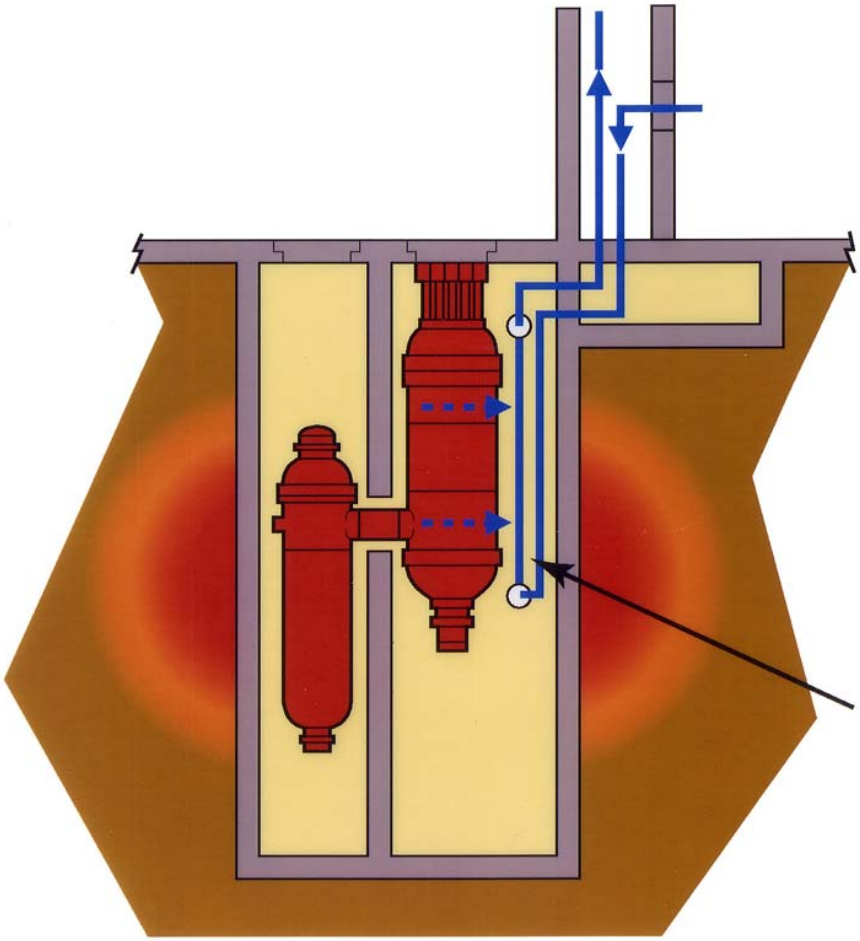
GENERAL ATOMICS

MHR MODULES LOCATED IN BELOW GRADE SILOS



- **Protection against natural disasters, missiles, terrorists**
- **Reduces seismic effects**
- **Cost-effective construction method by reduction of above grade structures**

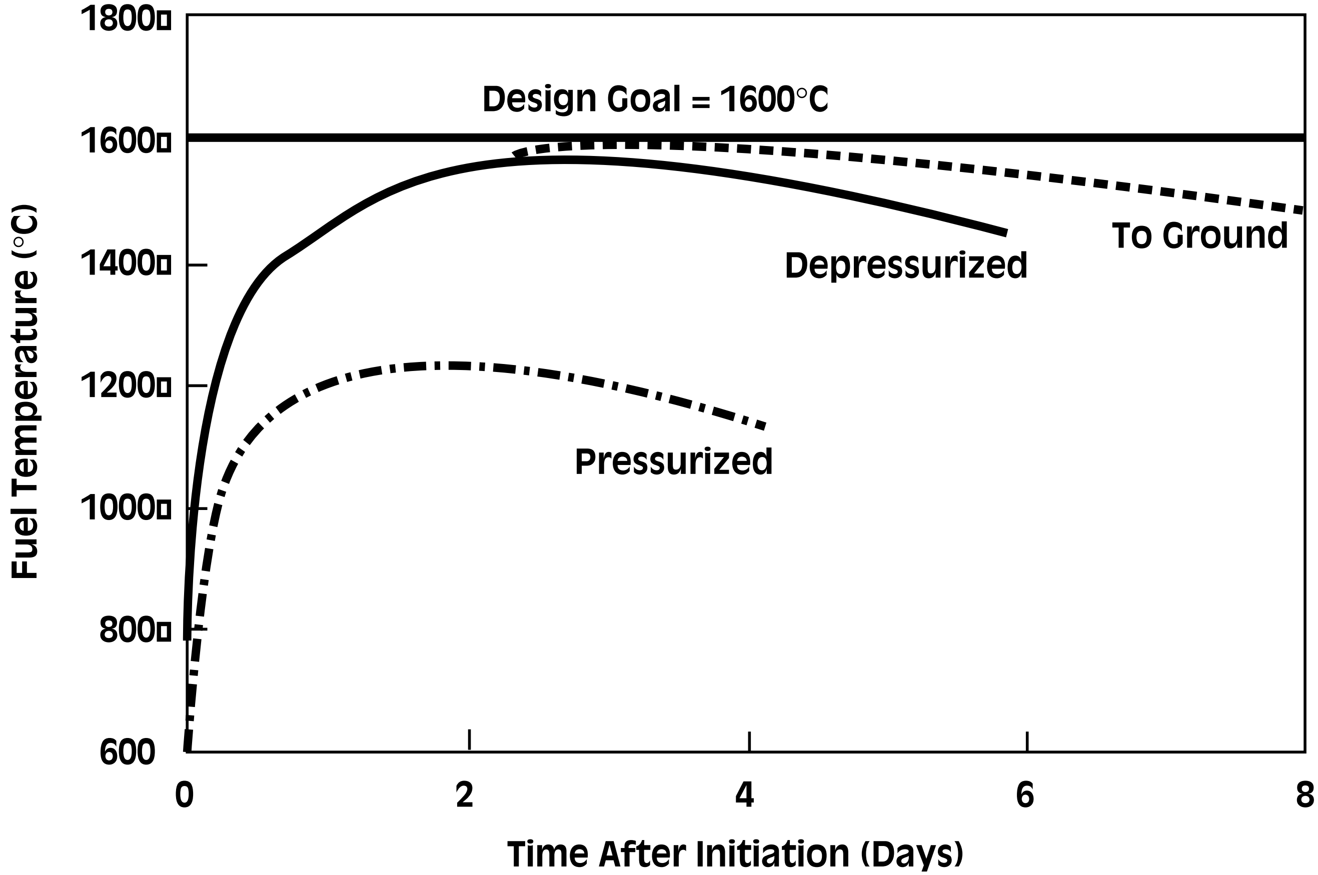
PASSIVE REACTOR CAVITY COOLING SYSTEM REMOVES CORE DECAY HEAT FROM CAVITY



REACTOR CAVITY
COOLING SYSTEM
PANELS

- Decay heat radiates from vessel to natural draft air cooling system
- No pumps or fans required
- Heat also conducts into ground

FUEL TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING LOSS OF COOLING EVENTS



. . . PASSIVE DESIGN FEATURES ENSURE FUEL REMAINS BELOW 1600°C

GENERAL ATOMICS

L-340(3)
11-16-94

MHR PROVIDES PASSIVE SAFETY BY DESIGN

- **Fission Products Retained in Coated Particles**
  - ***High temperature stability materials***
  - ***Refractory coated fuel***
  - ***Graphite moderator***

- **Worst case fuel temperature limited by design features**
  - ***Low power density***
  - ***Low thermal rating per module***
  - ***Annular Core***
  - ***Passive heat removal*** ***….CORE CAN'T MELT***

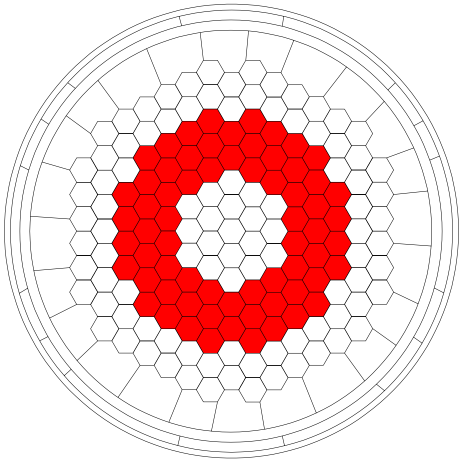
- **Core Shuts Down Without Rod Motion**

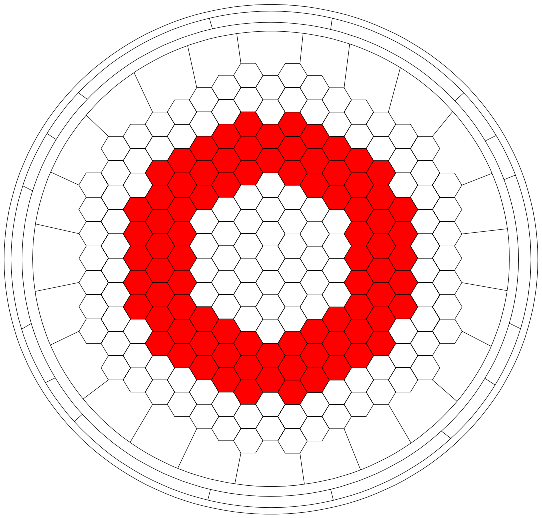
GENERAL ATOMICS

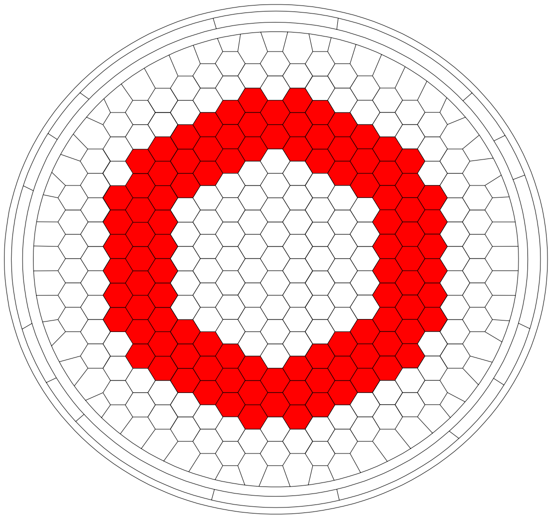
GENERAL ATOMICS

PASSIVELY SAFE MHR TECHNOLOGY FLEXIBLE IN SIZE TO MEET DIFFERENT NEEDS

350 MW(t)	450 MW(t)	600 MW(t)

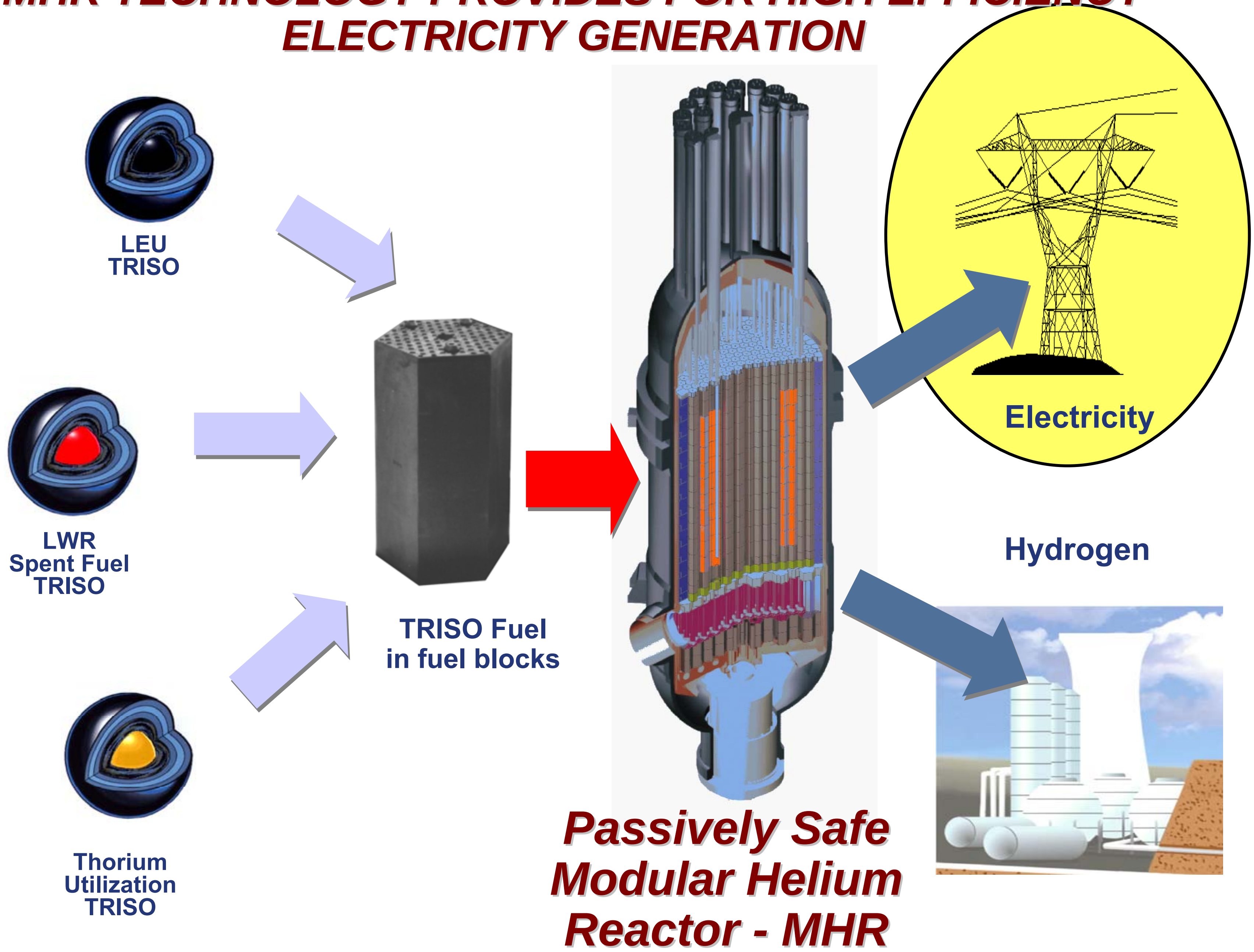






66 Columns 660 Elements	84 Columns 840 Elements	102 Columns 1020 Elements

MHR TECHNOLOGY PROVIDES FOR HIGH EFFICIENCY ELECTRICITY GENERATION



SPECTRUM OF PASSIVELY SAFE MHR PLANTS DEVELOPED FOR ELECTRICITY GENERATION

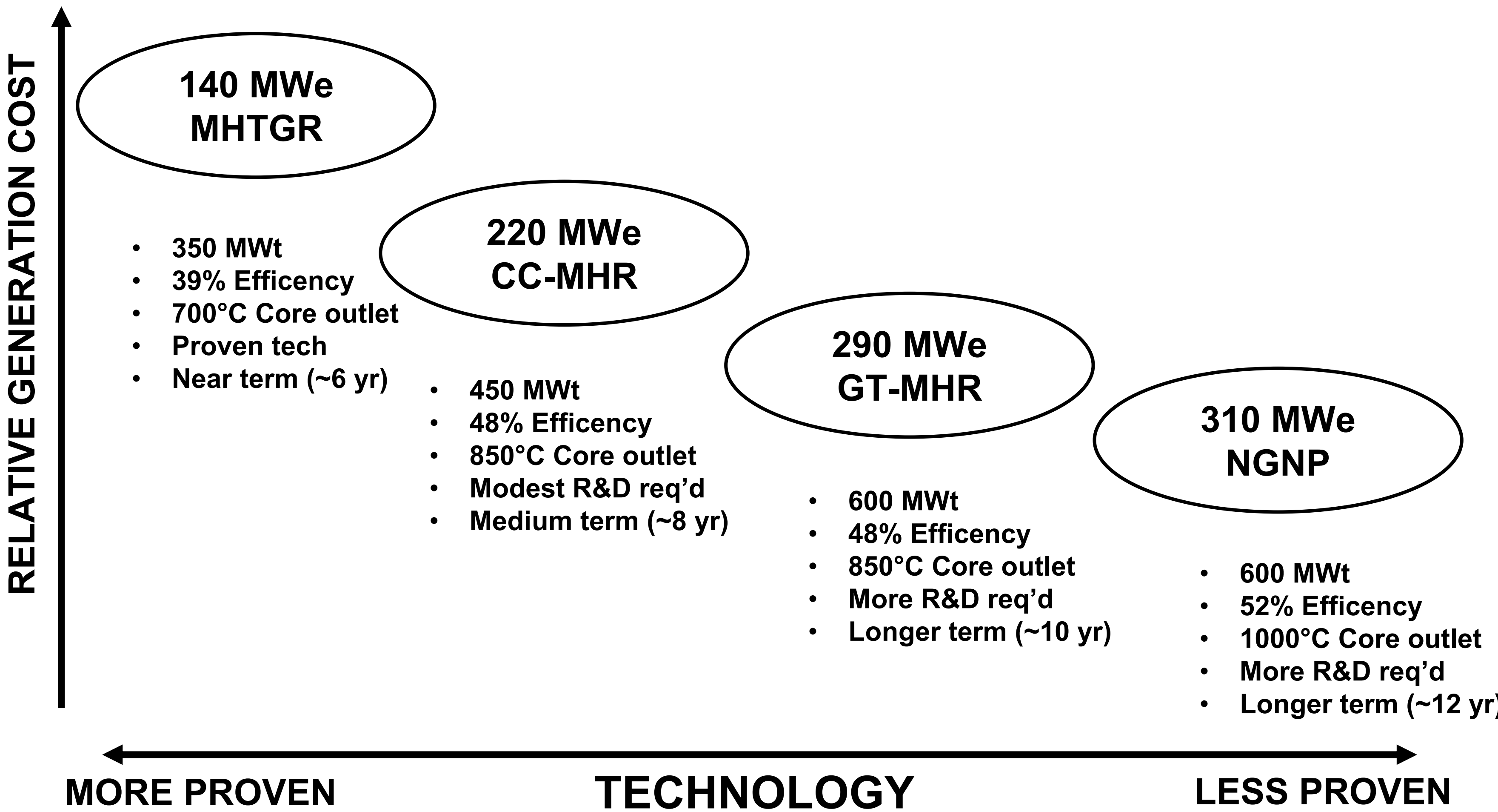
1. **140 MWe Steam Cycle 350 MWt Modular High Temperature Gas Reactor (MHTGR) - 1st passively safe MHR developed**

2. **220 MWe Combined Cycle 450 MWt MHR (CC-MHR) - Extrapolation of 350 MWt MHR to higher temp & coupled with modified combined cycle plant for higher efficency**

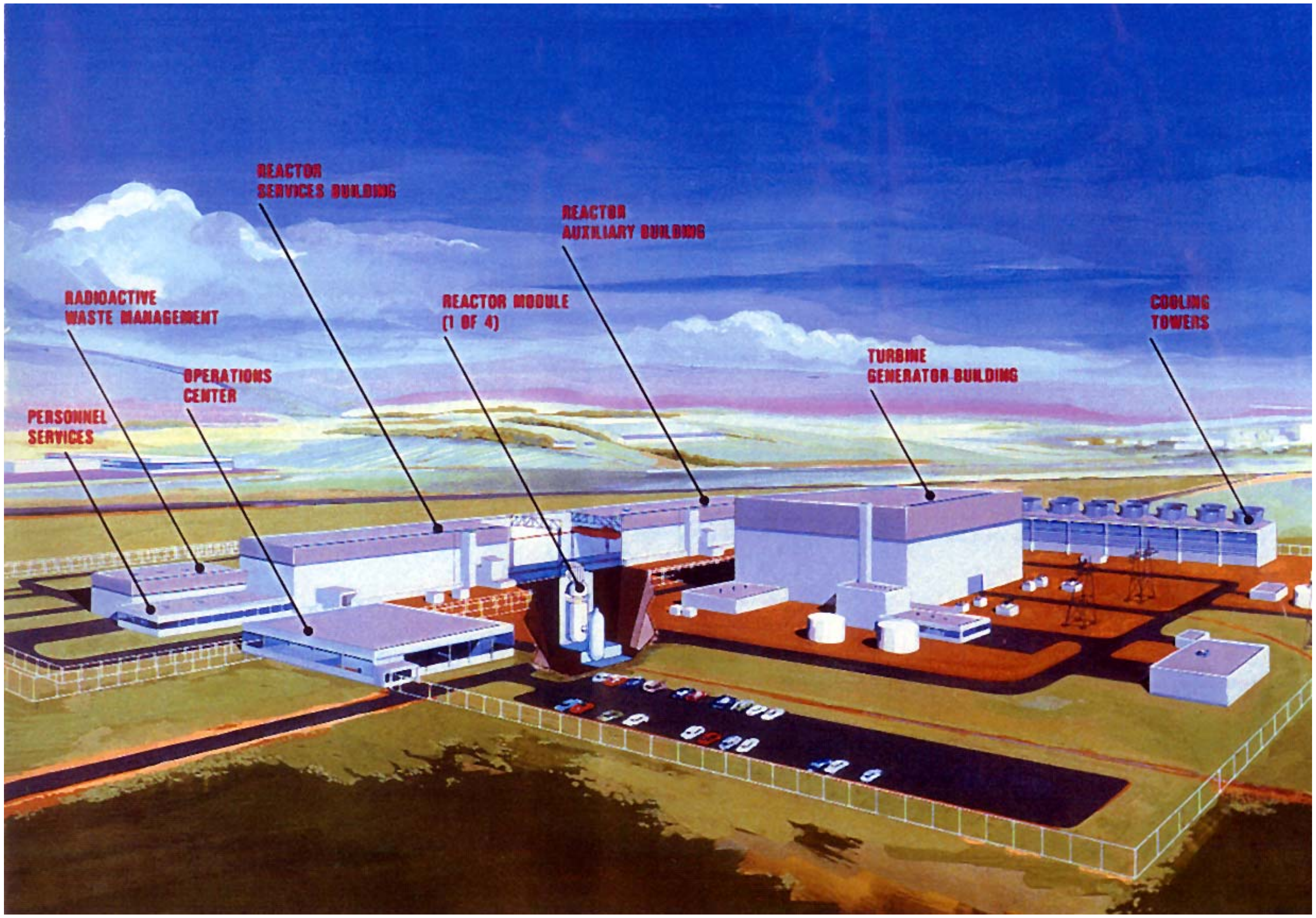
3. **290 MWe Gas Turbine - Modular Helium Reactor (GT-MHR) - 600 MWt MHR with direct Brayton cycle**

4. **310 MWe GT-MHR with 1000°C core outlet temperature Brayton cycle - Next Generation Nuclear Plant (NGNP)**

MHR ELECTRIC GENERATION PLANTS RANGE FROM NEAR TERM TO LONGER TERM

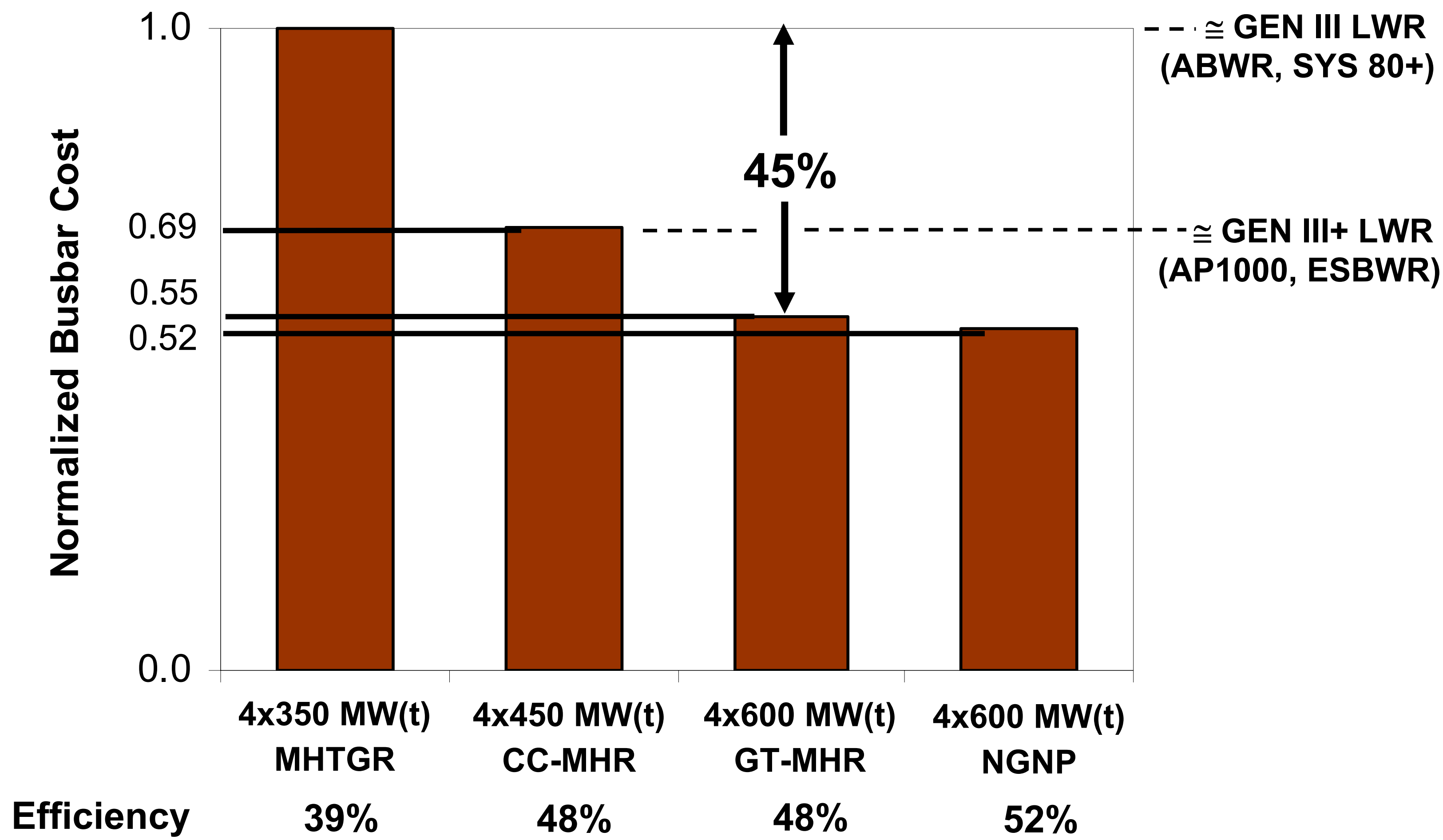


REFERENCE PLANT DESIGNS COMPRISE FOUR MODULES



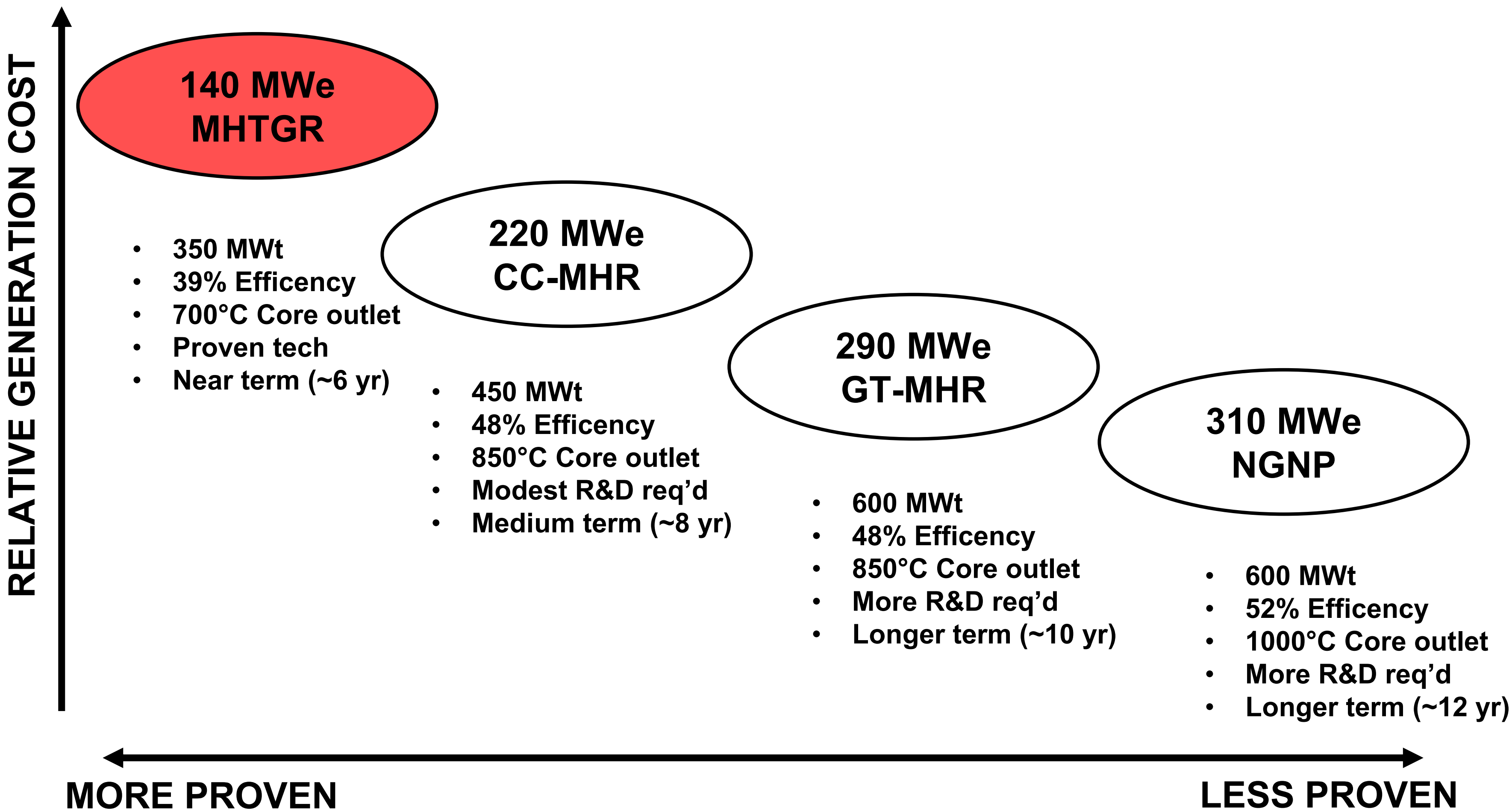
LARGER MHR SIZES & ADVANCED CONVERSION TECHNOLOGIES REDUCE POWER GENERATION COSTS

…But, 350 MWt MHTGR based on most proven technology



GENERAL ATOMICS

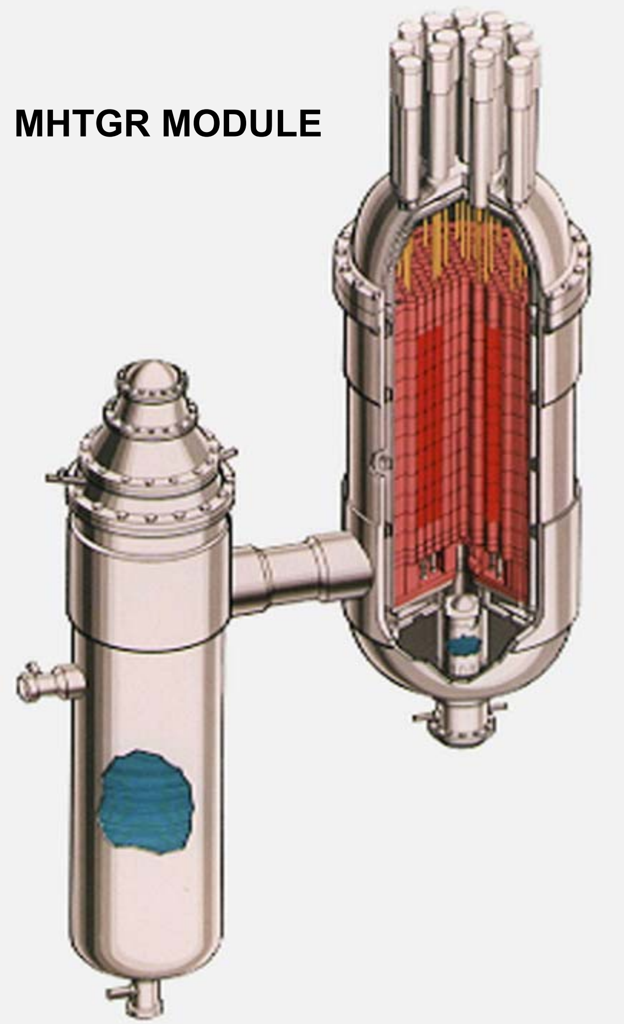
MHTGR FIRST MHR ELECTRIC GENERATION PLANT DEVELOPED



MHTGR MODULE COMBINES MELTDOWN-PROOF REACTOR & HIGH TEMPERATURE STEAM SUPPLY FOR HIGH EFFICENCY ELECTRICITY GENERATION

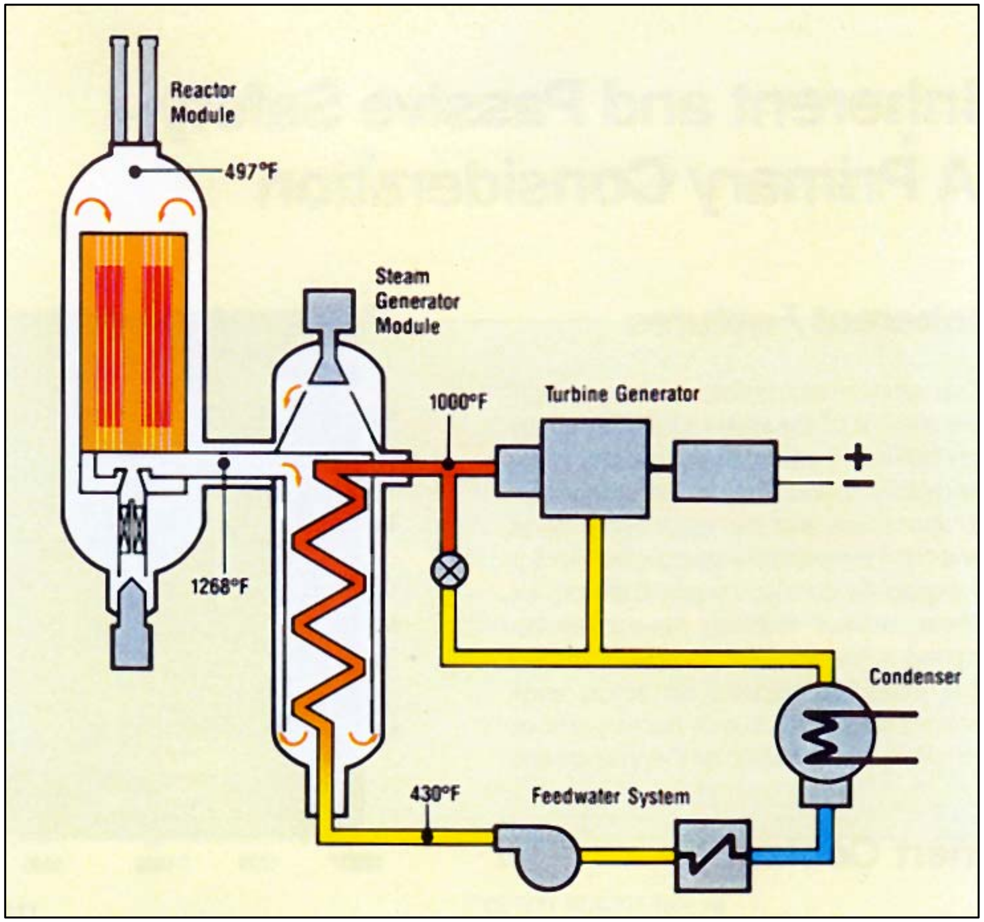
POWER LEVEL
350 MWt; 140 MWe

MHTGR MODULE



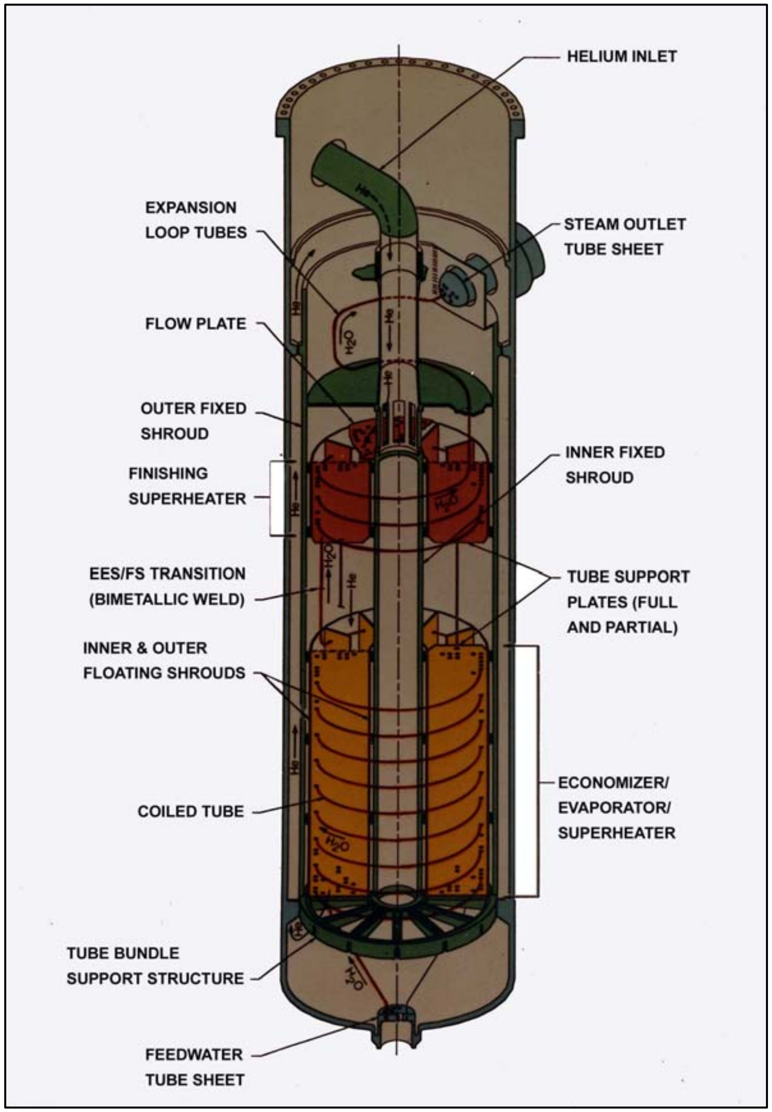
GENERAL ATOMICS

MHTGR GENERATES STEAM AT 1000°F (540°C) AND 2500 PSI (17 Mpa)



….steam quality equivalent to modern fossil-fired steam power plants

MHTGR STEAM GENERATOR IS CLOSELY RELATED TO FORT ST. VRAIN & THTR



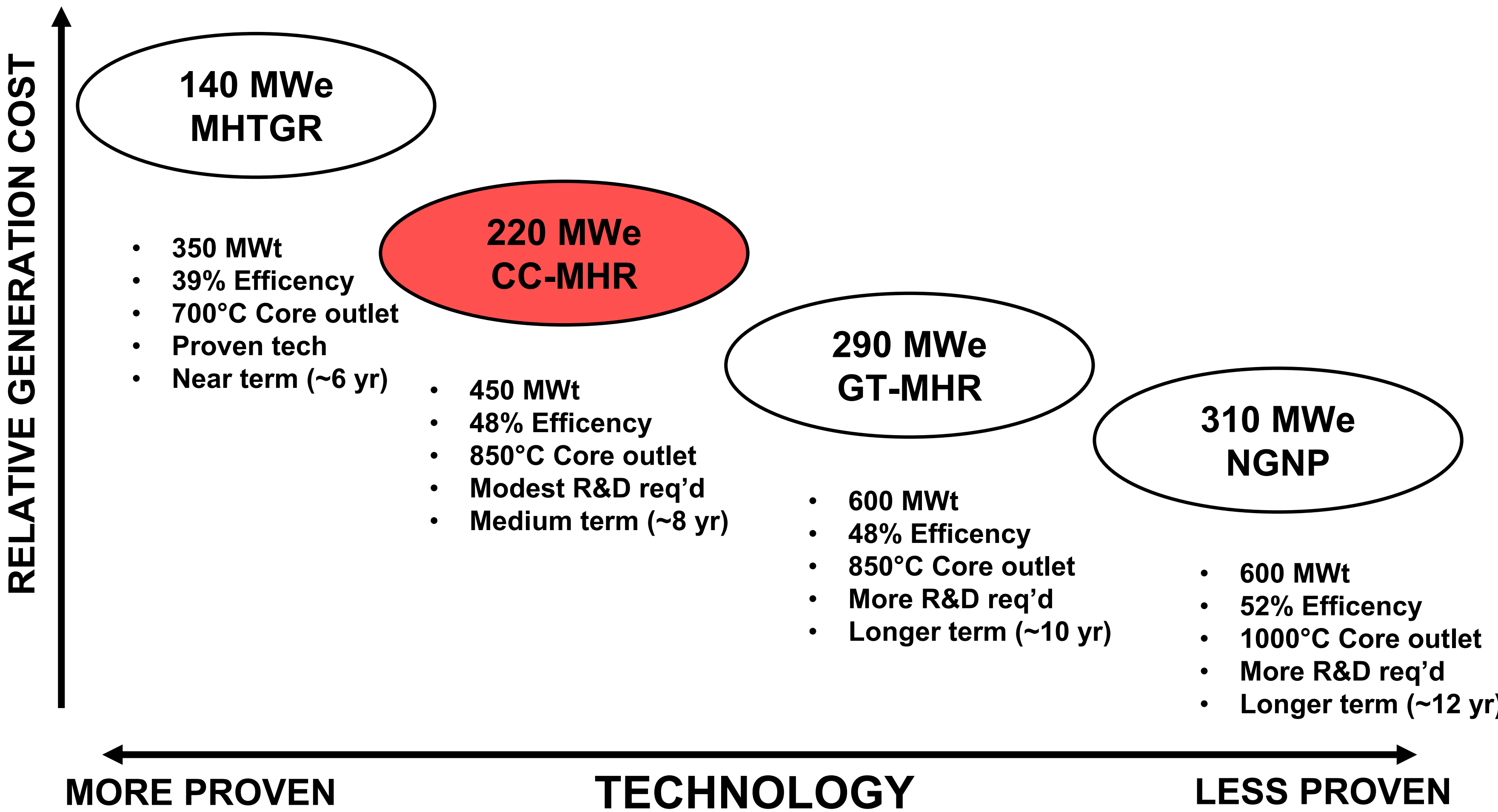
- Converts reactor heat to superheated steam
- Helically coiled once-thru boiler design, boiling inside tubes
- Tubes are part of primary pressure boundary
- Size consistent with nuclear component experience
- Design simplified relative to prior HTGR designs
- Service conditions comparable to prior gas-cooled and fossil-fired experience
- Code approved materials

GENERAL ATOMICS

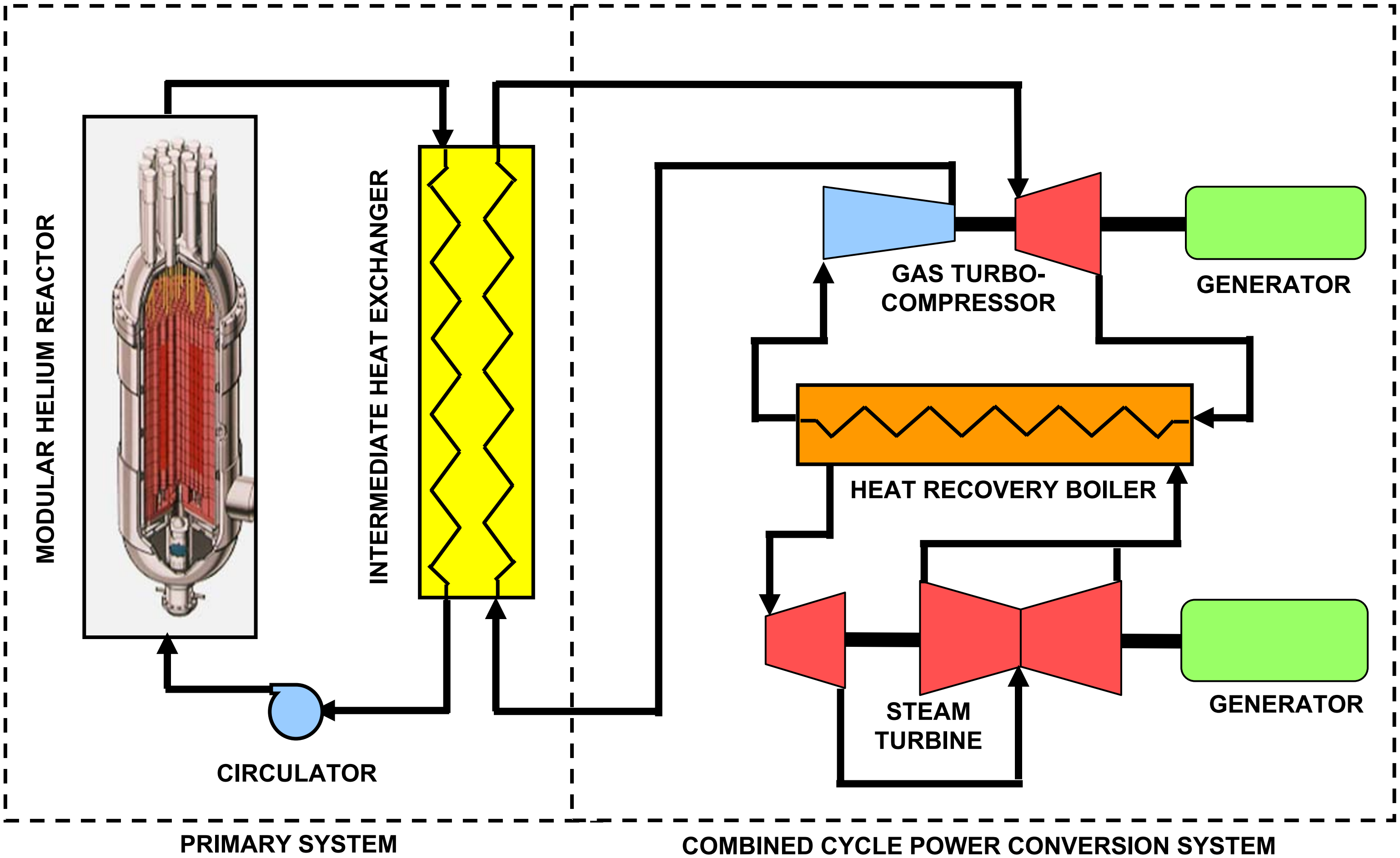
MHTGR IS A NEAR TERM ADVANCED NUCLEAR POWER SYSTEM

- MHTGR is based on proven technology
  - No R&D required
  - Detail preliminary design completed including a preliminary safety review by the US NRC
  - Only detail engineering for construction remains to be done
- MHTGR supplies high grade steam equivalent to modern fossil fired boiler plants for high efficiency electricity generation
- MHTGR passively safe by design
- First MHTGR could be deployed in about 6 years
- MHTGR plants can be configured to use one or more modules
- Module size 350 MWt or 450 MWt

COMBINED CYCLE MHR BUILDS ON RECENT TECHNOLOGY DEVELOPMENTS

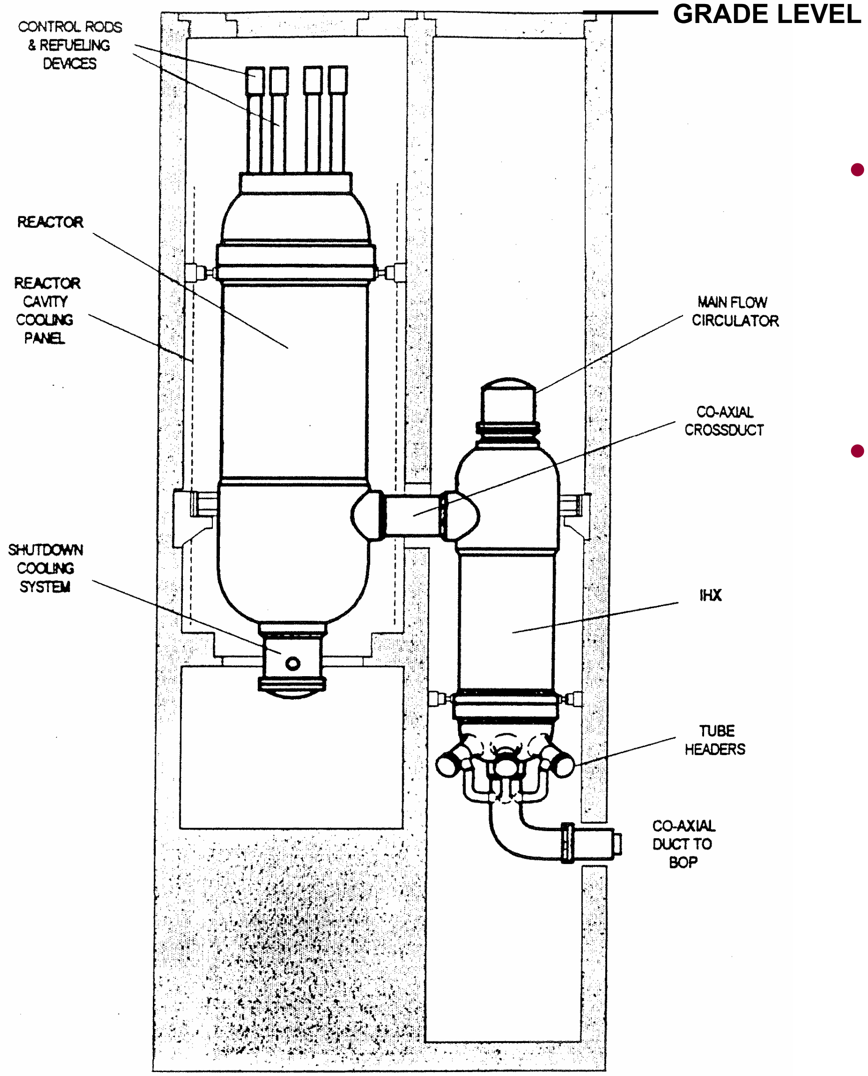


CC-MHR PLANT COUPLES AN MHR WITH A COMBINED CYCLE POWER CONVERSION SYSTEM



GENERAL ATOMICS

CC-MHR PRIMARY SYSTEM LOCATED IN BELOW GRADE SILO, SAME AS MHTGR



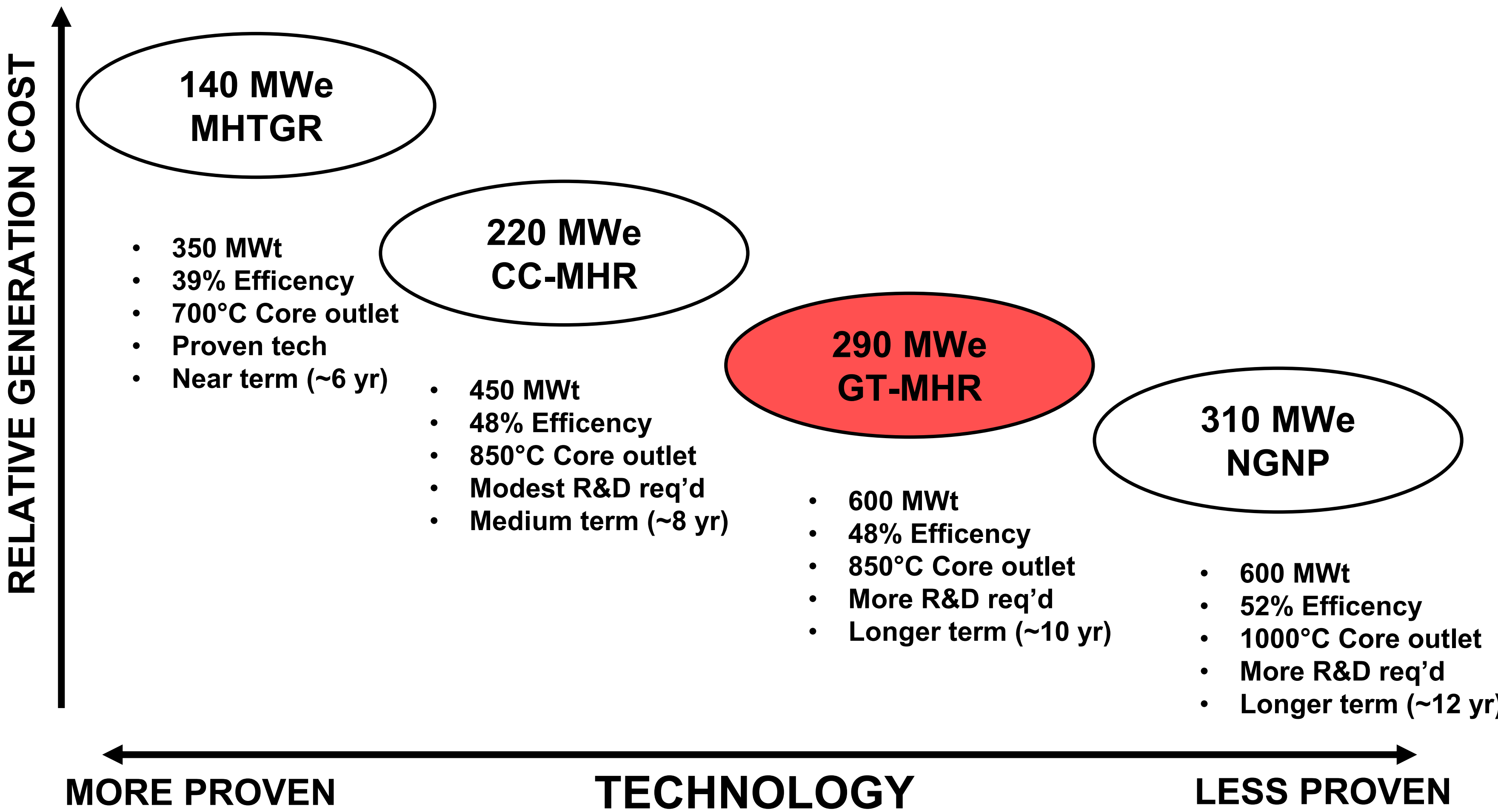
- **CC-MHR retains same passive safety characteristics as MHTGR**
- **Natural circulation reactor cavity cooling system incorporated same as MHTGR**

GENERAL ATOMICS

CC-MHR IS AN ADVANCED MHR PLANT THAT COULD BE DEPLOYED IN THE MID TERM

- **CC-MHR has substantial proven technology bases**
  - **Limited R&D required on IHX and gas turbine**
  - **Much of the MHTGR detail preliminary design applicable, including the preliminary safety review by the US NRC**
  - **Detail engineering for construction remains to be done**
- **No new R&D for MHR for increased core outlet temperature**
  - **Within envelop proven by HTTR**
- **CC-MHR makes use of the proven combined cycle power conversion system for high conversion efficiency**
- **CC-MHR passively safe by design**
- **First CC-MHR could be deployed in about 8 years**

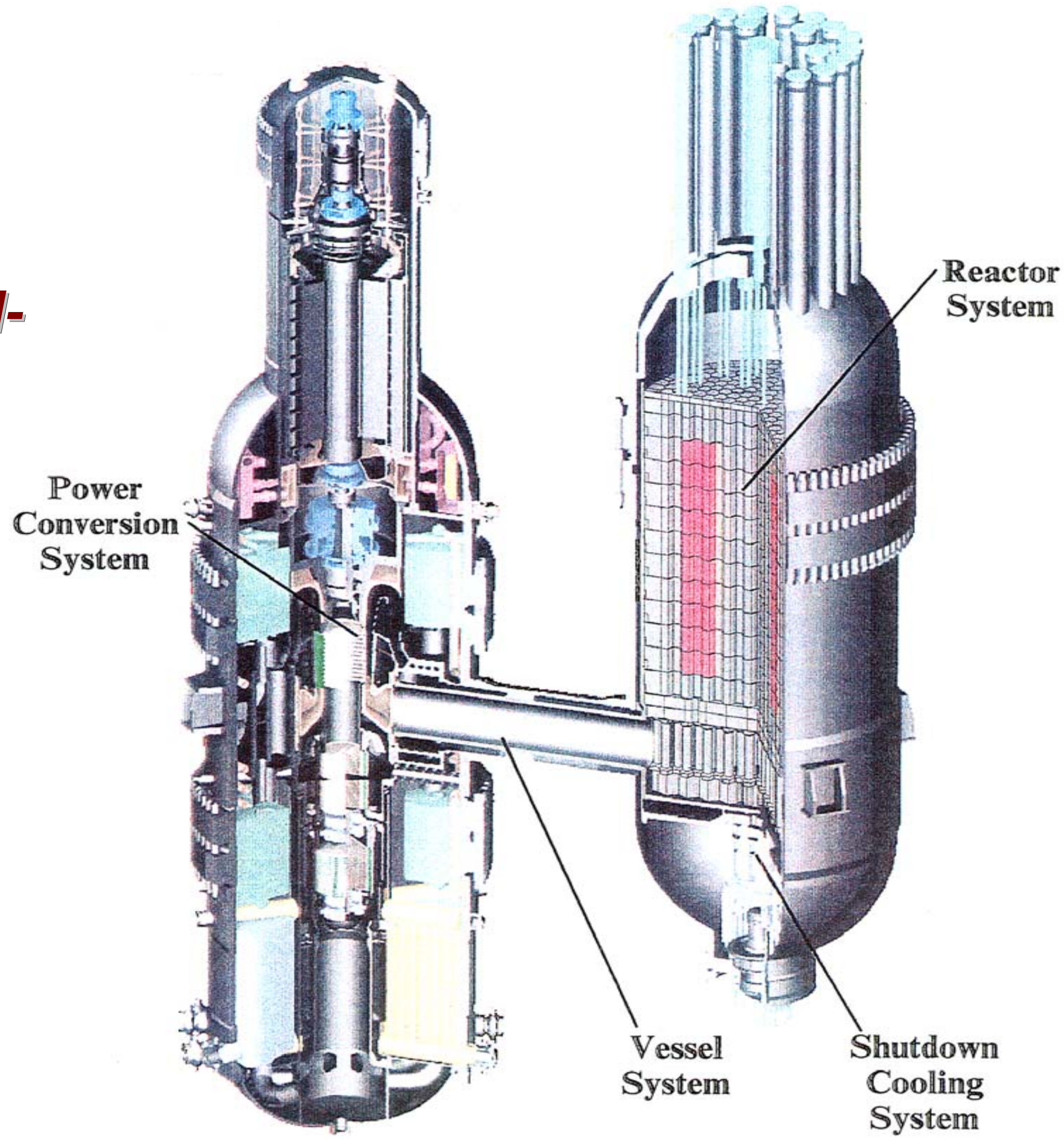
GAS TURBINE MHR DEVELOPED FOR IMPROVED ECONOMICS



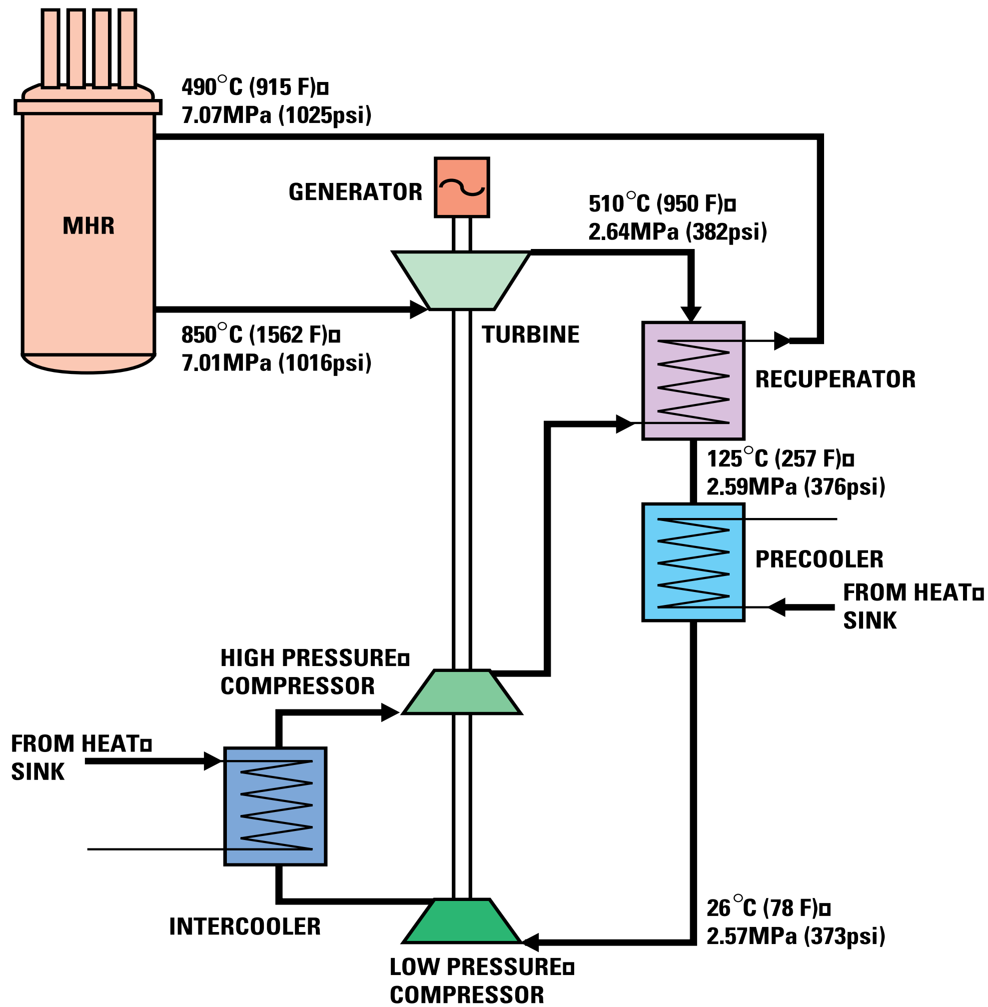
GENERAL ATOMICS

GT-MHR MODULE COMBINES MELTDOWN-PROOF ADVANCED REACTOR & HIGH EFFICENCY GAS TURBINE POWER CONVERSION SYSTEM

POWER LEVEL 600 MWt; 290 MWe

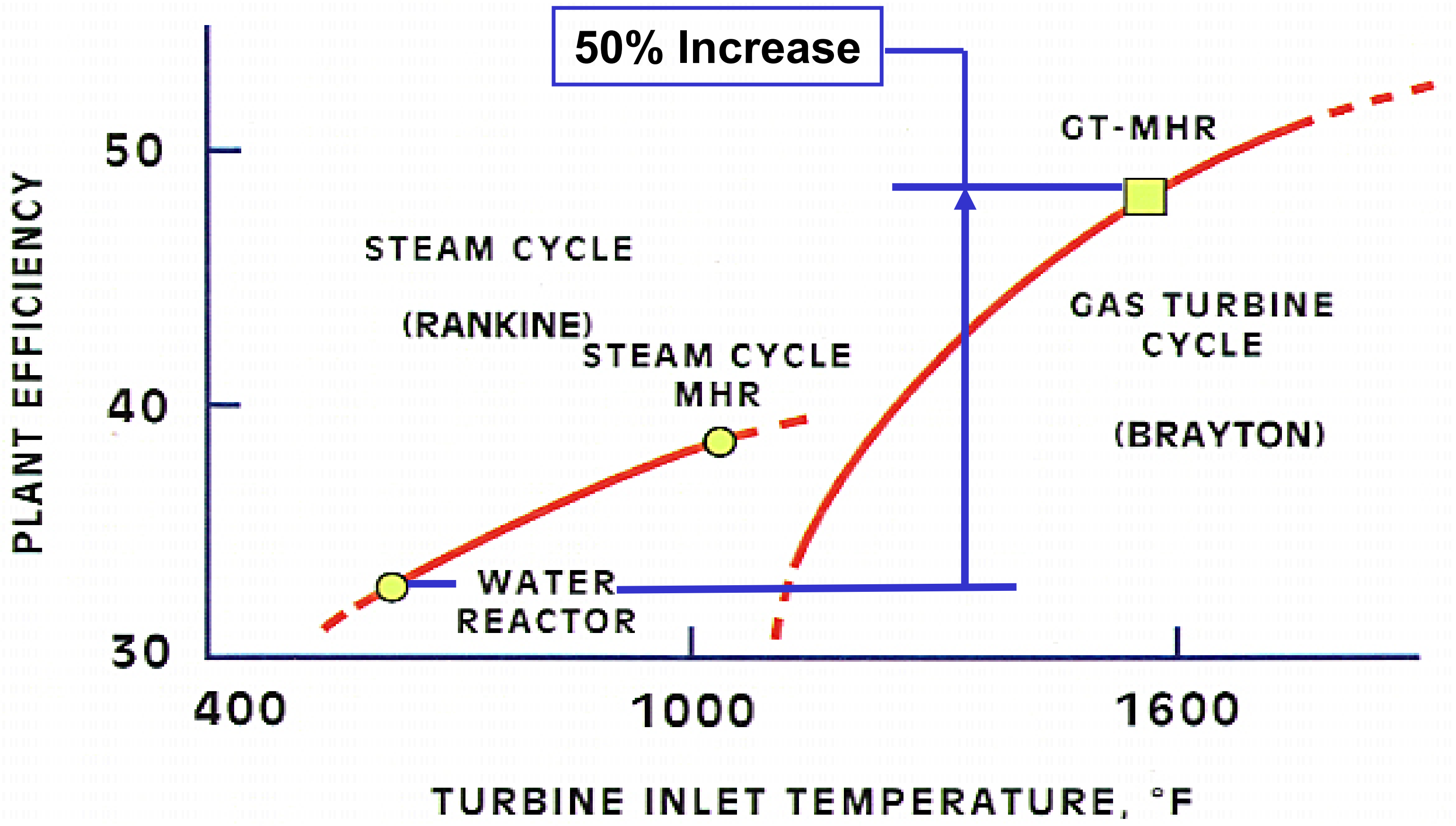


GT-MHR USES DIRECT BRAYTON CYCLE POWER CONVERSION SYSTEM



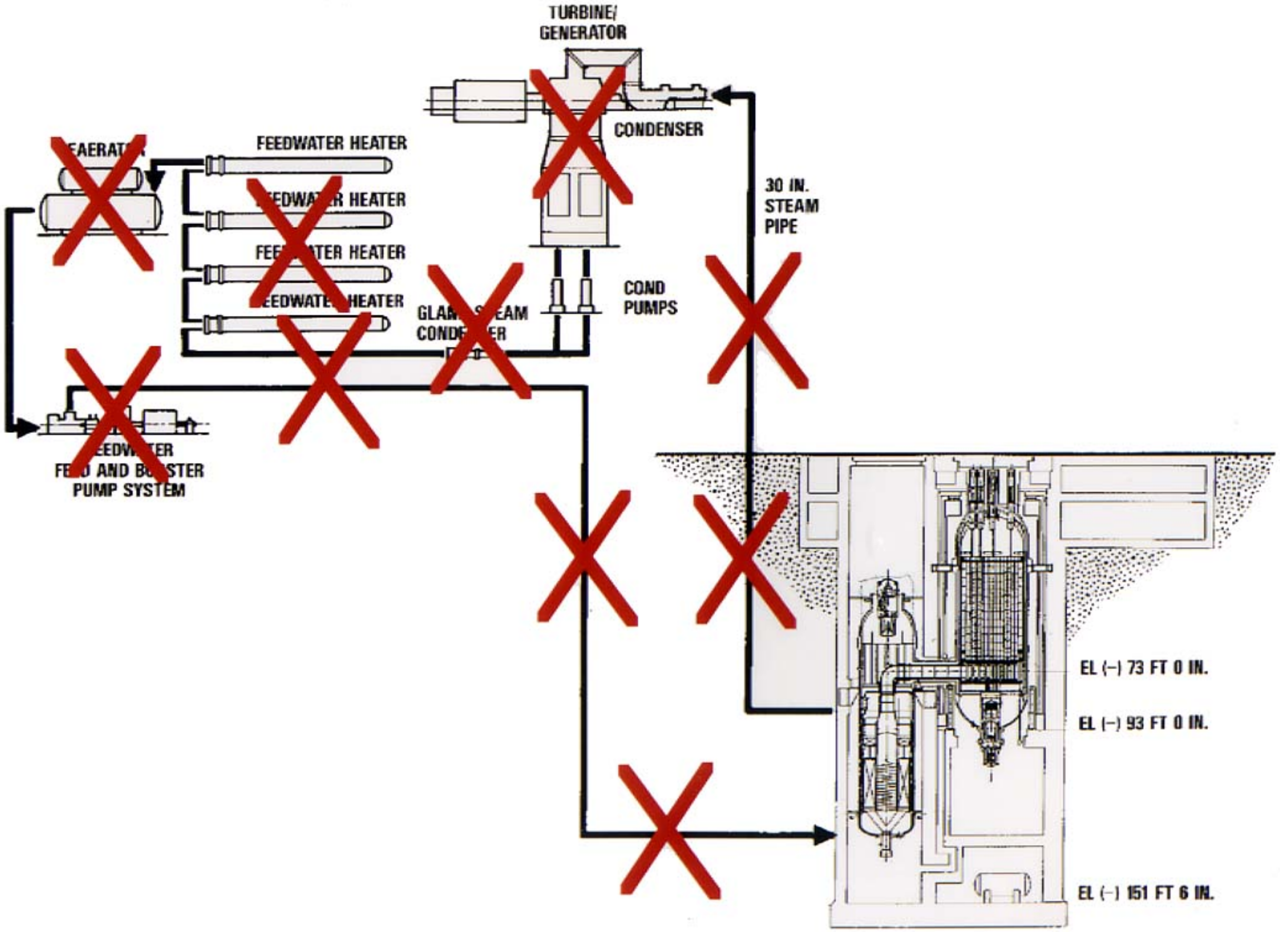
GENERAL ATOMICS

HIGH TEMPERATURE GAS REACTORS HAVE UNIQUE ABILITY TO USE BRAYTON CYCLE



GENERAL ATOMICS

DIRECT CYCLE ELIMINATES MANY COMPLICATED AND EXPENSIVE COMPONENTS



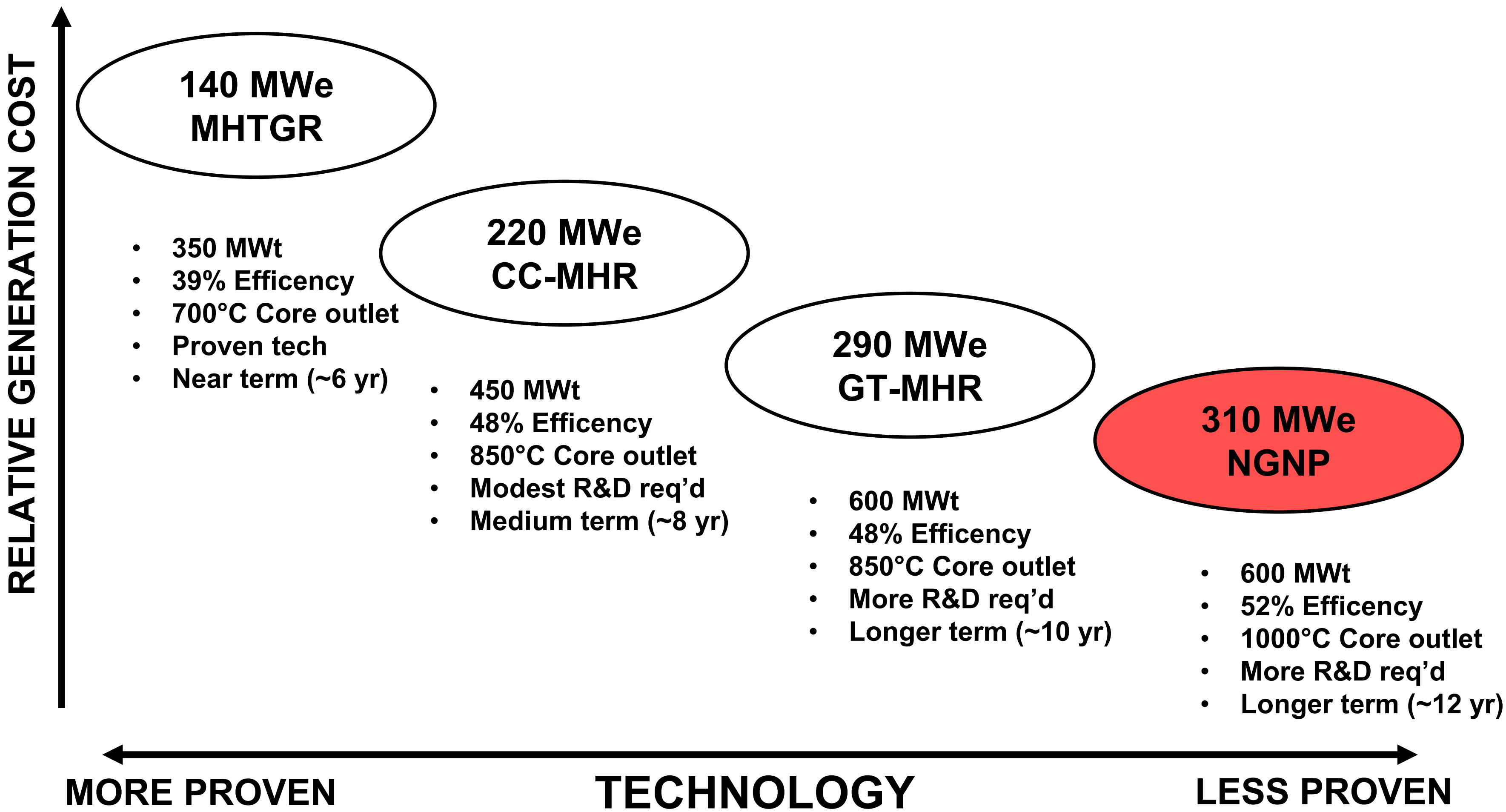
. . . **REDUCES O&M / IMPROVES PLANT AVAILABILITY**

GENERAL ATOMICS

R&D REQUIREMENTS LENGTHEN GT-MHR COMMERCIAL DEPLOYMENT SCHEDULE

- **Preliminary design of reactor module has been completed in Russia (to Russian codes & stds)**
- **Integrated power conversion unit (PCU) is longest term & most costly development item**
  - **Full scale turbomachine test planned**
  - **Tests of several PCU sub-components in process**
- **Second most critical path item is regulatory review and licensing (not yet started)**
- **First commercial GT-MHR plant deployable in about 10 years (based on four year construction schedule for 1st module) and assuming a prototype not required**

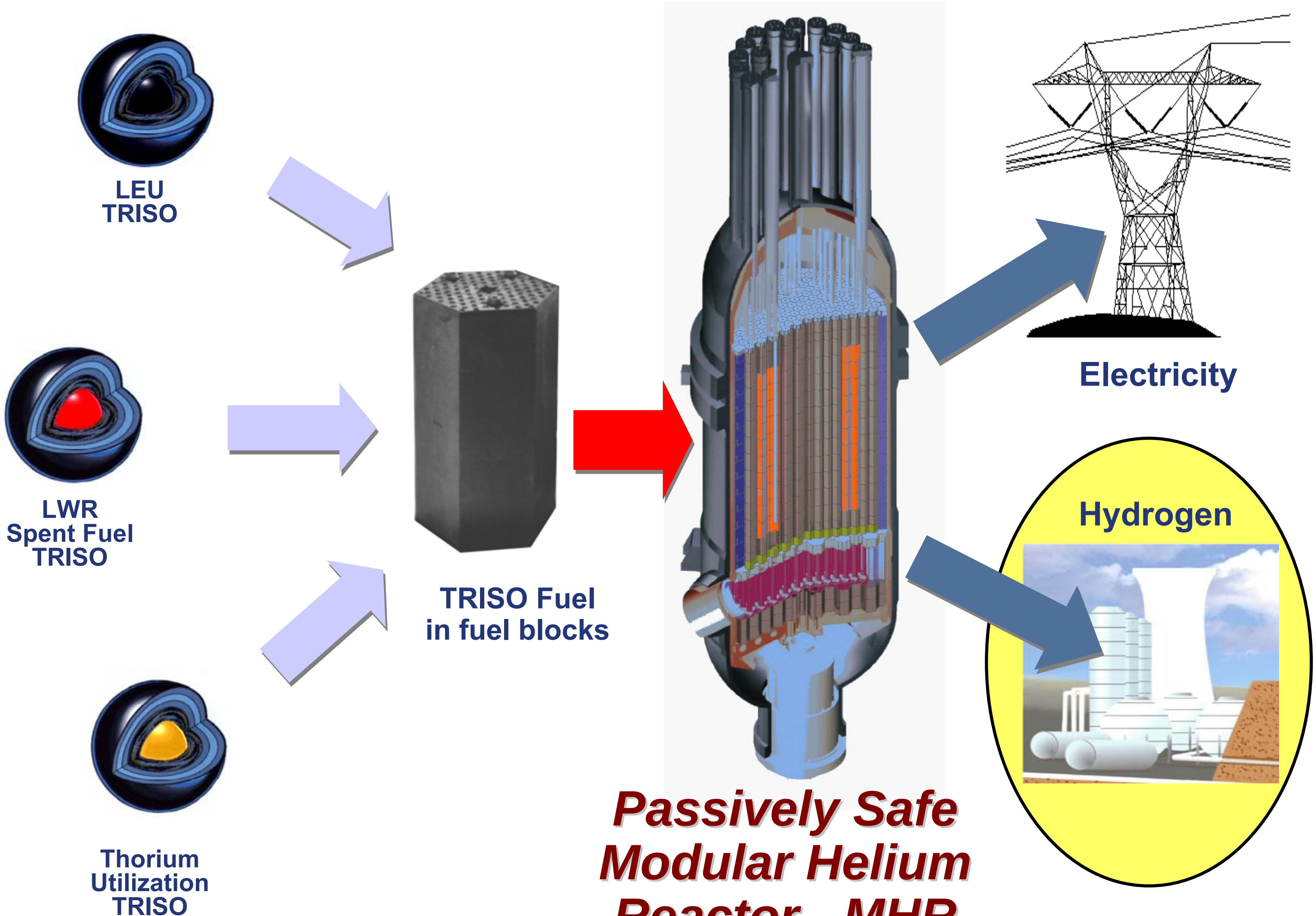
NGNP IS FIRST GENERATION IV PLANT TO BE DEMONSTRATED BY US DOE



NGNP MISSION OBJECTIVES IDENTIFIED BY US DOE

- **Demonstrate a full-scale prototype NGNP by about 2017**
- **Demonstrate high-temperature Brayton Cycle electric power production at full scale**
- **Demonstrate nuclear-assisted production of hydrogen (using about 10 % of the heat)**
- **Demonstrate by test the exceptional safety capabilities of the advanced gas cooled reactors**
- **Obtain an NRC License to construct and operate the NGNP, to provide a basis for future performance-based, risk-informed licensing**
- **Support the development, testing, and prototyping of hydrogen infrastructures**

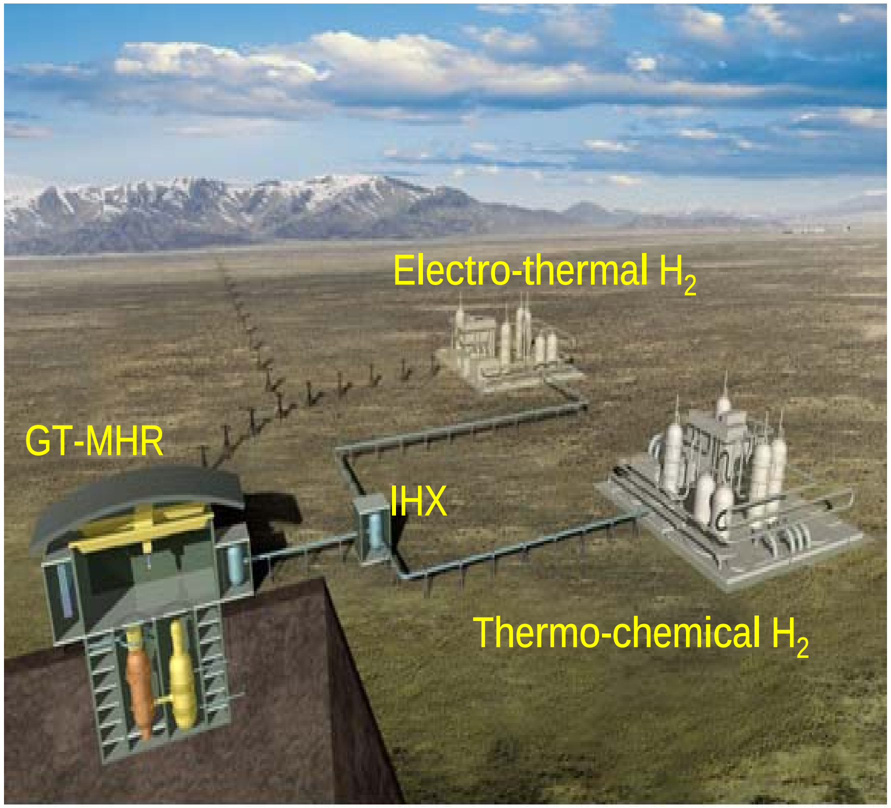
HYDROGEN PRODUCTION IS KEY NGNP OBJECTIVE



SEVERAL WAYS POSSIBLE TO PRODUCE HYDROGEN USING NUCLEAR ENERGY

- Electric power generation ➔ Electrolysis
  - Overall efficiency ~24% (LWR), ~36% (Hi T Reactors) (efficiency of electric power generation x efficiency of electrolysis)
- High temperature heat ➔ Thermochemical water-splitting
  - Net plant efficiencies of up to ~50%
- Electricity + Heat ➔ High temperature electrolysis or Hybrid thermochemical cycles
  - Efficiencies up to ~ 50%

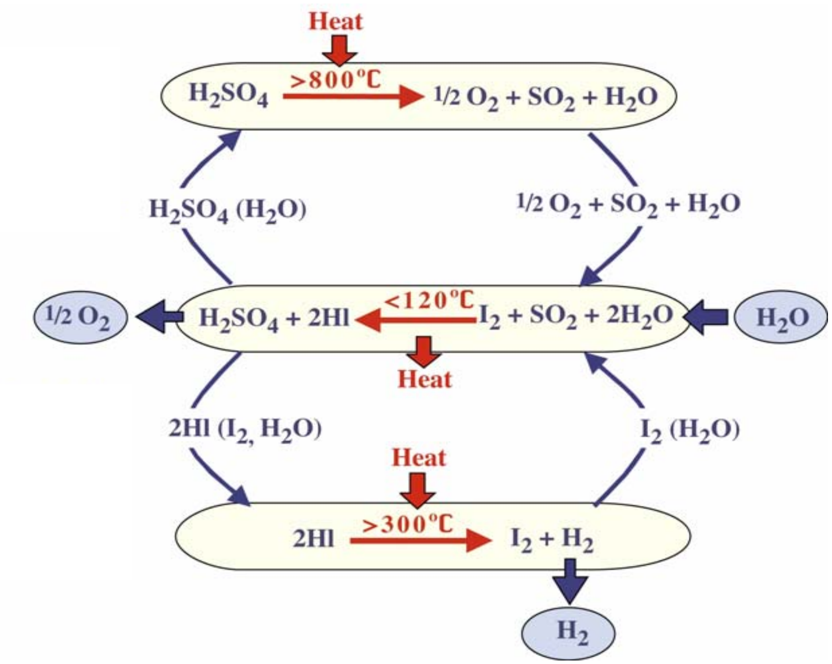
NGNP PLAN IS TO DEMONSTRATE H_2 PRODUCTION BY TWO ALTERNATIVE PROCESSES



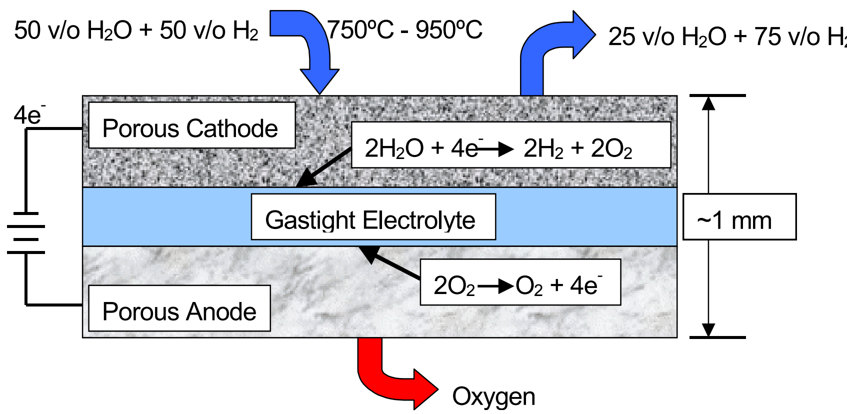
- Hydrogen production to 60 MWt
  - Allow smooth transition between 100% electricity and 90% electricity/10% hydrogen
  - Up to 20 tonnes of H_2 per day
- Hydrogen purity
  - Tritium release below NRC and EPA limits
  - Radioactivity < 10CFR20 limits
  - Meet fuel-cell standards
- Safe reactor/hydrogen interface
- Advanced fuels?

GENERAL ATOMICS

LEADING CANDIDATE HYDROGEN PRODUCTION PROCESSES ARE S-I and HTE



Sulfur-Iodine (S-I) Thermochemical Process



High Temperature Electrolysis (HTE) Process

NGNP REACTOR PLANT SIMILAR TO GT-MHR (the main difference is coolant temperature)

	GT-MHR	NGNP
• Power Level (MW)	600	600 (not optimized)
• Power Density (w/cc)	6.5	6.5
• Coolant&Pressure (Mpa/psia)	He 7.12/1032	He 7.12/1032
• Core Outlet Temp °C	850	1000
• Core Inlet Temp °C	490	490-600 (not optimized)
• Maximum Fuel Temp °C	1250	1250 (up to 1400 depending upon fuel element)
• Intermediate HX	NA	Compact Heat Exchanger

NGNP LONGER TERM ADVANCED MHR FOR ELECTRICITY AND H_2 PRODUCTION

- **Very similar to GT-MHR for electricity generation**
  - **Higher core outlet temperature requires additional R&D (fuels and materials)**
- **Production of hydrogen from high temperature MHR nuclear heat appears promising**
  - **Hydrogen production processes require significant R&D**
- **Schedule for startup projected to be 2017**
  - **First commercial deployment ~5+ years later**

Appendix I: PBMR Technical Summary

Nuclear Steam Supply from Intermediate Temperature Process Heat Plant Based on Pebble Bed Modular Reactor

The development of modern, higher temperature nuclear reactors has created the opportunity to introduce nuclear heat sources into the industrial and transport sectors by supplying process heat to produce cleaner gases, chemical products and liquid petroleum fuels. However, the nuclear heat source must meet modern reactor design standards, be economic, match process technical needs, and reliably produce the required temperatures.

South Africa's Pebble Bed Modular Reactor (PBMR) technology fits each of these requirements.

Because of its high outlet temperature (up to 950°C), heat from the PBMR can be applied to a variety of industrial process applications. Notably, the PBMR's energy can be used for the production of non-carbon derived hydrogen for transportation fuel or for upgrading coal and heavy crude oils into usable products, thereby relieving the pressure on natural gas supplies (the source of most hydrogen produced today). It can also produce emission free heat to extract bitumen from Oil Sands, and for other industrial applications where fossil fuels are currently used as the primary source of energy.

Many of these applications are under detailed investigation by PBMR, its industrial partners, and potential customers in global markets.

Figure I-1 shows a typical PBMR Process Heat Plant (PHP) configuration. The hot helium exits the bottom of the reactor and passes through helium-to-helium intermediate heat exchangers (IHXs), and gas circulators located on top of the heat exchangers pump the cooled gas back into the pressure vessel for reheating. Helium in an intermediate loop transfers the heat to the process application through the concentric pipes, as shown. Two PBMR process heat configurations currently exist: the first delivers high temperature helium at up to 950°C for thermo-chemical reactions, and the second delivers intermediate temperature helium in the 750°C range for high pressure steam production. The IHXs will be virtually identical to the recuperator intended for use in the South African Demonstration Power Plant (DPP).

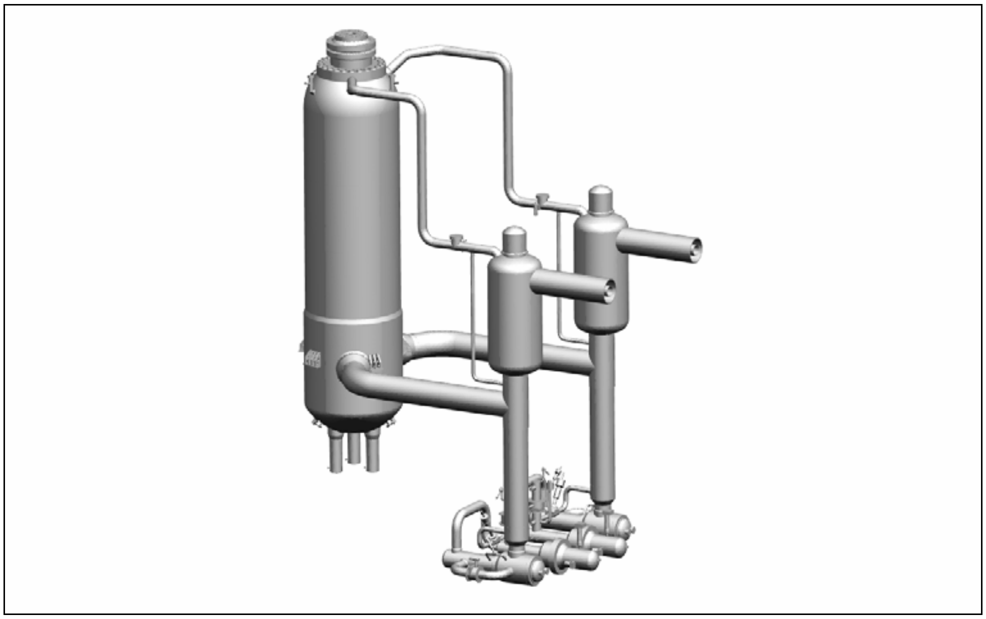


Figure I-1, Typical Process Heat Plant

Why PBMR for Process Heat?

PBMR technology has unique features which make it well-suited as heat source for process applications:

a) Access to niche, value added high-temperature process markets due to its ability to provide process temperatures up to 900°C (reactor outlet of 950°);

b) Well-matched to industrial process sizes, from 400 to 500+ MW(t);

c) Ability to co-locate with industrial process plants due to inherent safety characteristics and small exclusion zone;

d) Near-term availability, since it builds on the development and design work carried out on the South African DPP initiative;

e) Economic benefits include the displacement of premium fossil fuels, value from avoided CO_2 emissions, high reliability, improved availability due to continuous online refueling, short construction times, and reduced financing costs during construction;

f) Capable of addressing parallel markets and products such as co-generated power and desalination.

Product Range

Pebble Bed Modular Reactor (Pty) Ltd of South Africa is a multi-product reactor vendor committed to supplying the utility and process industries with emission free, high efficiency electrical power and high temperature process heat.

The South African DPP project entails the design and construction of a 165 MW(e)/400 MW(t) demonstration power plant at Koeberg near Cape Town, and a fuel plant at Pelindaba near Pretoria. The DPP will demonstrate the combination of the proven PBMR reactor design with a full-scale Brayton cycle nuclear gas turbine to provide first-of-fleet experience for the proposed multi-module electricity plant. It is the DPP development program that provides the foundation of the international Process Heat Plant (PHP) market deployment efforts.

The PHP will be based on the DPP's physical reactor design and core dimensions. The PHP is intended to operate at power levels of 400 to 500+ MW(t) with reactor outlet temperatures up to 950°C. However, the individual configuration of the PBMR based PHPs depend on the specific process heat application. Though the reactor core dimensions will remain the same for different process heat applications, the technology can be essentially differentiated into two configurations, depending on the reactor outlet temperature:

a) An Intermediate Temperature Gas cooled Reactor (ITGR), operating at reactor outlet temperatures up to 750°C;

b) A High Temperature Gas cooled Reactor (HTGR) operating at reactor outlet temperatures up to 950°C, which also meets the requirements for the generic Very High Temperature Reactor (VHTR) specification.

Process Applications & Markets

Heat from the PBMR can be used for a variety of industrial process applications. Intermediate temperatures (up to 750°C) can be used to generate process steam for co-generation applications, electricity production, in-situ Oil Sands recovery, ethanol applications, and refinery and petrochemical applications. Higher temperatures (in the 900°C range) can be used to efficiently co-generate electricity in a range of cycles, to reform methane to produce syngas (where the syngas can be used as feedstock to produce hydrogen, ammonia and methanol), and to produce hydrogen and oxygen by the thermochemical decomposition of water. Hydrogen can be sold as a merchant product, or through integration into a number of industrial operations such as coal-to-liquids, coal-to-gas, refineries, upgrading of bitumen-like products, petrochemical applications, and steel production. Lower temperature waste heat can be used to produce water through the desalination processes.

In Canada, there is interest from Oil Sands Producers (OSPs) for using the PBMR to produce the temperature and associated pressure needed for "in-situ" applications to extract bitumen from Oil Sands, displacing the intended gas fired plants that are currently used.

Furthermore, in the USA, PBMR is a partner in the Westinghouse led consortium, which has been awarded a contract by the US Department of Energy to consider the PBMR technology as the heat source for producing non-carbon emission hydrogen. This Next Generation Nuclear Plant (NGNP) project, which aims to use HTGR technology to produce hydrogen and electricity, is still in its pre-conceptual phase, but it could result in the construction of a South African designed PBMR in the US before the end of the next decade.

Work continues to allow US design certification of the PBMR, and in preparation for a pre-application review. The Nuclear Regulatory Commission (NRC) staff has held public meetings in 2006 to identify the topics that are expected to be the focus of the pre-application phase, and in 2007 to review the content and production of the required topical White Papers.

Work Ongoing & Completed

Based on collaborations with several potential users of this technology, PBMR and its partners in the nuclear and process industry have initiated and completed several initiatives including:

a) Definition of process heat delivery systems for high and intermediate temperature applications;

b) Survey of high temperature process applications and economics;

c) Initiation of pre-licensing initiatives in the US and Canada to prepare for early projects;

d) Co-operation with universities to support application and market studies, energy policy development, and to establish outreach programs;

e) Definition of first-of-fleet project and project implementation requirements;

f) Economic analysis of various applications;

g) Formation of an industry advisory group;

h) Definition of industrial nuclear co-generation and desalination plant configurations.

Value Proposition

Attractive applications for nuclear (high and intermediate temperature) process heat are driven primarily by the opportunity to displace natural gas and other premium fuels, and to respond to incentives to reduce CO_2 emissions. Even with conservatively low forecasts for growth in long term gas prices, there is a clear commercial benefit in reducing exposure to volatility and rapid increases. Economic assessments of PBMR process heat applications based on current trends have confirmed that PBMR is likely to become economically competitive in many markets, especially in markets with high premium fuel costs and CO_2 emission constraints.

PBMR Datasheet

Two Nuclear Steam Supply System variants are offered for review: a “Steam-only” as shown in Figure I-2, and a “Co-generation” as shown in Figure I-3. The “Steam Only” variant is a single 500MW(t) Pebble Bed Modular Reactor (PBMR), with the reactor delivering 750°C Helium from its Primary Heat Transfer System (PHTS) to twin IHXs, which in turn deliver 720°C Helium from its Secondary Heat Transport System (SHTS) to “tube and shell” Steam Generators (S/Gs) for Steam Assisted Gravity Drainage (SAGD) injection at the wellhead.

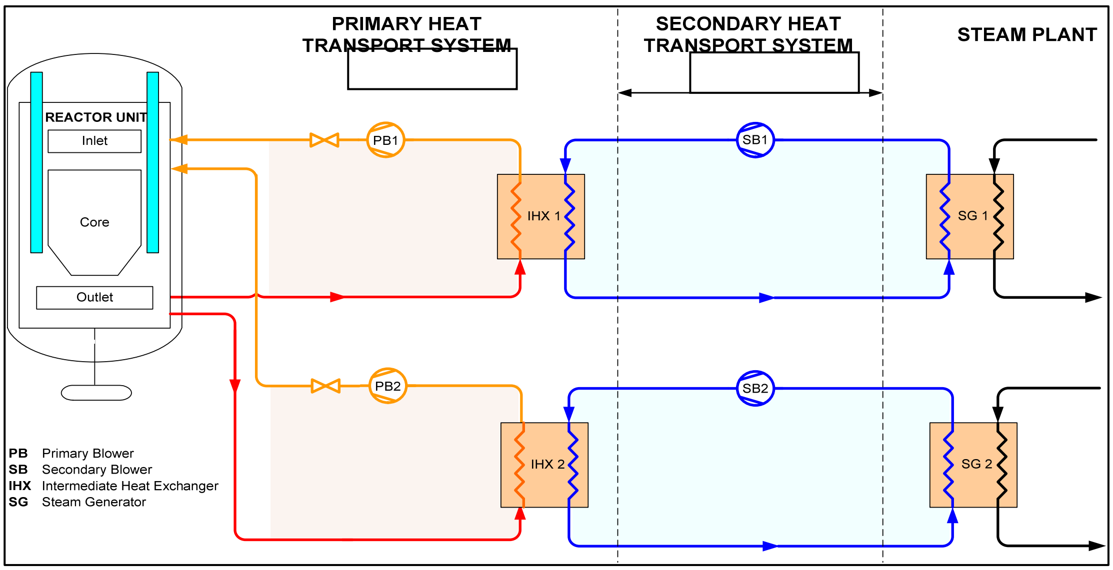


Figure I-2, Steam-Only Variant

The “Co-generation” variant is a single 500MW(t) PBMR, with the reactor delivering 750°C Helium from its PHTS to twin IHXs, which in turn deliver 720°C Helium from its SHTS to the main S/Gs for SAGD injection. The secondary turbine steam generator delivers supercritical steam to a 40MW(e) Rankin cycle turbo-alternator.

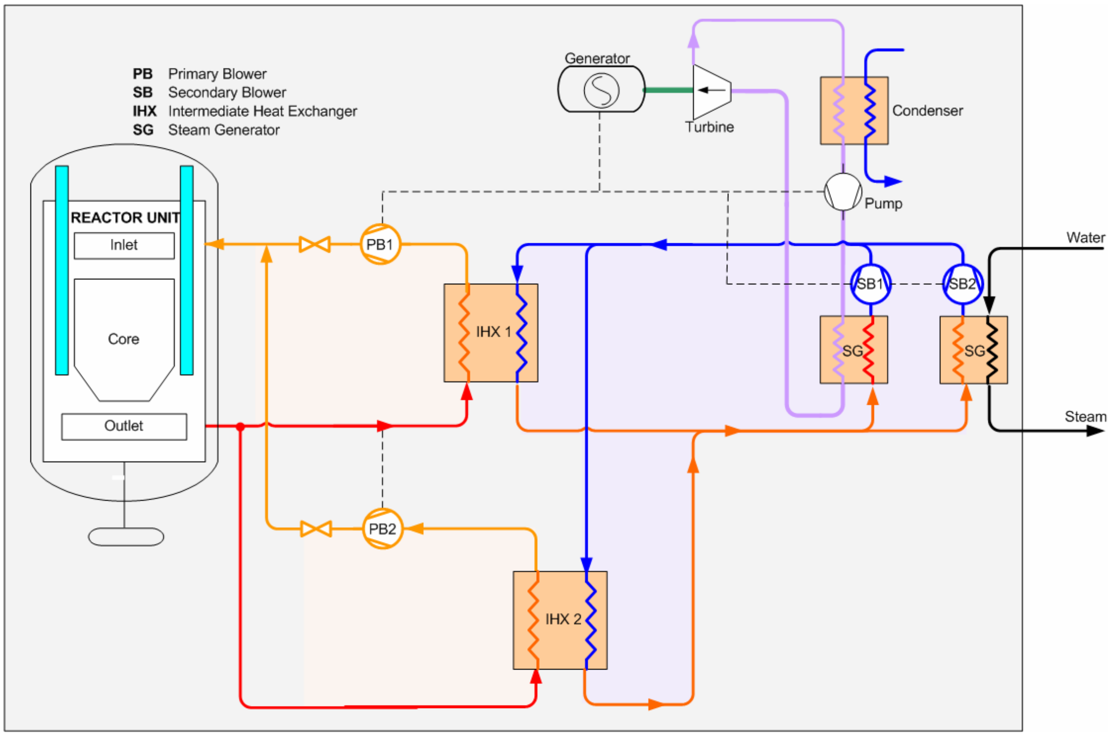


Figure I-3, Co-generation Variant

Scope of Supply

The following are parameters for a Nuclear Steam Supply System bounded by the scope of supply:

a) One (1) Process Heat for Oil Sands (PHOS) Nuclear Module building containing one Nuclear Heat Supply module;

b) One (1) PHOS Conventional Module building containing one Steam Production Plant (and Steam Turbine System for the co-generation variant) including feedwater support system comprising pre-heater and boiler steam pressure, level and temperature monitoring to feed to OSP and feedwater delivery point monitoring equipment;

c) Steam delivery point monitoring equipment;

d) Blowdown delivery point monitoring equipment;

e) Electrical power transformer(s) from the works power supply point (and Power Distribution System from the turbo-alternator to electrical power transformer(s) for the co-generation variant);

f) Emergency power diesel generator;

g) Fire protection systems (to the extent not shared with OSP);

h) Ancillary and services support system (building lighting, HVAC, compressed air, communications etc.);

i) Nuclear Module Building security support system.

Excluded from this scope of supply are:

j) Site preparation, waterproofing, foundations and civil improvements;

k) Feedwater heating and pumping system (oil separation and water treatment systems);

l) General site lighting;

m) Access roads, truck receiving, unloading and laydown area;

n) Works power supply point;

o) Steam supply connections downstream of steam delivery point;

p) Feedwater connection upstream of feedwater delivery point;

q) Blowdown supply connection downstream of blowdown delivery point, blowdown vessel and control system.

Table I-1, Nuclear Steam Supply System: Economic Parameters

Variant	Steam-Only	Cogeneration
Basis	n^{th} plant target	n^{th} plant target
Plant lifetime	30 years	30 years
Annualized planned outage	<8 days	<9 days
Annual forced outage rate	<2%	<2%
Construction licence application	2010	2010
Long lead item ordering	2011	2011
In-service date	2017	2017
Full time equivalent staff	<50	<50 (inc. security)
Annualised O&M cost	49-52	29-32 (US$M 2007)[1]
Capital cost	1,008	1,036 (US$M 2007)[2]

[1] Including insurance, fuel, decommissioning and spent fuel
[2] Including Licensing and other Owners' Costs, but excluding interest and contingencies

Table I-2, Reactor / Primary Heat Transport System: Technical Parameters

Variant	Steam-Only	Cogeneration
Reactor power	500MW(t)	500MW(t)
Primary circuit coolant	Helium	Helium
Primary circuit coolant pressure	8.2MPa	8.2MPa
Primary circuit coolant flow	205kg/s	205kg/s
Reactor outlet temperature	750°C	750°C
Reactor inlet temperature	280°C	280°C
Gas circulator power requirement	14.4MW(e)	14.4MW(e)

Table I-3, Secondary Heat Transport System: Technical Parameters

Variant	Steam-Only	Cogeneration
Secondary circuit coolant	Helium	Helium
Secondary circuit coolant pressure	8.7MPa	8.6MPa
Secondary circuit coolant flow	205kg/s	205kg/s
Main S/G inlet temperature	720°C	720°C
Main S/G outlet temperature	223°C	223°C
Main S/G delivered power	520MW(t)	414MW(t)
Turbine S/G delivered power		105MW(t)
Gas circulator power requirement	12.6MW(e)	12.5MW(e)

Table I-4, Steam Generator: Technical Parameters

Variant	Steam-Only	Cogeneration
Main steam outlet pressure	8-13MPa	4-11MPa
Main feedwater pressure	10-15MPa	6-13MPa
Main steam outlet flow	258kg/s	206kg/s
Main feedwater flow	323kg/s	258kg/s
Main feedwater inlet temperature	120-190°C	120-190°C
Main steam outlet temperature	295-330°C	250-320°C
Cogeneration turbine power		40MW(e)
Main feedwater quality	Lime softened, weak cation exchanged, de-oiled water, deoxygenated or better	

Appendix K: 1100 MWe PWR vs. 1000 MWe PBMR Cost Comparison

The cost comparison provided in Table K-1 was derived based on a comparison of systems in the 1100 MWe PWR, and a six (6) unit PBMR station with a total output of approximately 1000 MWe. Each PBMR module has a net electrical output of 165 MW. Since the cost comparison utilized proprietary PWR cost information, the detailed costs are not included. The quantitative information presented suggests that the PBMR station will have an overnight specific capital cost of approximately three (3) times the PWR1100. The cost difference is reduced if fewer PBMRs, and each with a higher output are utilized (e.g., five 500 MWth units with a 220 MWe output), which is the most feasible with the PBMR concept. A further cost reduction is available to the GA-HTGR due to design simplifications and economy of scale (600 MWth).

The above analysis indicates that the capital cost information provided by the HTGR vendors is significantly low.

Table K-1, First Order Cost Comparison: 1100 MWe PWR & Six Module 1000 MWe PBMR Station

Ref	Item	PBMR 1000 MW	PWR 1100 MW	Notes
1	Operating Units	6	1	
2	Pressure Vessels	6	1	The volume of each PBMR PV is approximately 55% greater than the PWR-1100 PV, and cannot take advantage of forged ring technology.
3	Coolant	He	Water	Helium is expensive and difficult to contain.
4	Moderator	Graphite	Water	A very large graphite volume is required by PBMR, which is expensive (>$75/kg).
5	Fuel	TRISO	Rods	The PBMR fuel cost per MWe is approximately 4 times that of the PWR-1100 (i.e., graphite, enrichment level, fabrication).
6	TG sets	6	4	The PWR-1100 turbine has one HP and 3 LPs.
7	Coolers/Condensors	12	1	
8	Recuperators/SGs	12	2	The weight of a recuperator is 40% of the PWR-1100 Steam Generator weight (cost differences are not accounted for).
9	Coolant circulation pumps/compressors	12	4	Canned RCS pumps are assumed for the PWR-1100. Helium compressors of the PBMR are relatively expensive.
10	Fuel Handling Systems	6	1	The PWR-1100 is refueled off-power using a very simple system. The PBMR system is complex and expensive (cost differences are not fully accounted for).
11	P&I Control Systems	6	1	The PBMR system is costly due to the large He tanks and complex valve configuration (cost differences are not accounted for).
12	Shutdown Systems (SDS)	12	2	The need for a second active PBMR system is not apparent. A second PWR-1100 SDS is low cost.
13	Residual Heat Removal Systems	1	6	The PBMR residual heat removal system appears to be substantially more complicated than a PWR's.
14	Purification Systems	2	1	This assumes one purification system serving the 6 PBMR modules, and accounts for complications of a multi-unit configuration. The PBMR system is approximately 3 times more expensive than the PWR-1100 system.
15	Diesel Generators (DGs)	4	2	This assumes that all DGs serve the 6 PBMR modules, otherwise the total could be 24. No safety grade diesels are required for the PWR-1000.
16	Confinement/ Containment	6	1	The current PBMR confinement system is very large and expensive.
17	Concrete	4	1	The current PBMR module concrete volume is approximately 60% of a PWR-1100 (each module).
18	C&I	4	1	Each PBMR module has approximately 55% of the C&I systems of a PWR-1100.
19	Building Volume	4	1	PBMR building volume is approximately 60% of a PWR-1100.
20	Post Accident Monitoring	1	1	This assumes that one PAM facility serves 6 PBMR modules, and accounts for complications of a multi-unit configuration.
21	General			Dry gas seals/gear boxes, etc., all add to PBMR costs.
22	Overnight Cost/MWe	$8100	$2600	For the Nth plant, first-of-a-kind (FOAK) engineering is not included.

Background

1) The output of 6½ PBMR modules is approximately equal to one (1) 1100 PWR;
2) A six (6) module plant is used as the basis of this comparison;
3) Costs are for the Nth plant (5th PWR and 5th 6-unit PBMR);
4) Costs do not include the Owner's cost;
5) Economies of sequential multi-unit construction are credited to PBMR;
6) The overnight capital cost for a six (6) module PBMR station would likely be in the order of three (3) times the cost of a single unit 1100 MWe PWR.

Appendix L: Decommissioning Nuclear Facilities

The following, identified as Briefing Note 13, was issued by the Australian Uranium Association in December of 2007, and has not been edited. A summary of references is provided at the end of the document.

Introduction

To date, 100 mines, 90 commercial power reactors, over 250 research reactors and a number of fuel cycle facilities, have been retired from operation. Some of these have been fully dismantled.

Most parts of a nuclear power plant do not become radioactive, or are contaminated at very low levels. Most metal can be recycled.

Proven techniques and equipment are available to dismantle nuclear facilities safely, and these have now been well demonstrated in several parts of the world.

Decommissioning costs for nuclear power plants, including disposal of associated wastes, are reducing and contribute only a small fraction of the total cost of electricity generation.

All power plants, coal, gas and nuclear, have a finite life beyond which it is not economically feasible to operate them. Generally speaking, early nuclear plants were designed for a life of about 30 years, though some have proved capable of continuing well beyond this. Newer plants are designed for a 40 to 60 year operating life. At the end of the life of any power plant, it needs to be decommissioned, decontaminated and demolished so that the site is made available for other uses. For nuclear plants, the term decommissioning includes all clean-up of radioactivity and progressive dismantling of the plant.

At the end of 2005, IAEA reported that eight (8) power plants had been completely decommissioned and dismantled, with the sites released for unconditional use. A further 17 had been partly dismantled and safely enclosed, 31 were being dismantled prior to eventual site release, and 30 were undergoing minimum dismantling prior to long-term enclosure.

Decommissioning Options

The International Atomic Energy Agency has defined three options for decommissioning, the definitions of which have been internationally adopted:

Immediate Dismantling (or Early Site Release/Decon in the US): This option allows for the facility to be removed from regulatory control relatively soon after shutdown or termination of regulated activities. Usually, the final dismantling or decontamination activities begin within a few months or years, depending on the facility. Following removal from regulatory control, the site is then available for re-use.

Safe Enclosure (or Safestor): This option postpones the final removal of controls for a longer period, usually in the order of 40 to 60 years. The facility is placed into a safe storage configuration until the eventual dismantling and decontamination activities occur.

Entombment: This option entails placing the facility into a condition that will allow the remaining on-site radioactive material to remain on-site without the requirement of ever removing it totally. This option usually involves reducing the size of the area where the radioactive material is located, and then encasing the facility in a long-lived structure such as concrete that will last for a period of time to ensure the remaining radioactivity is no longer of concern.

There is no right or wrong approach, each having its benefits and disadvantages. National policy determines which approach is adopted. In the case of immediate dismantling (or early site release), responsibility for the decommissioning is not transferred to future generations. The experience and skills of operating staff can also be utilized during the decommissioning program. Alternatively, Safe Enclosure (or Safestor) allows significant reduction in residual radioactivity, thus reducing radiation hazard during the eventual dismantling. The expected improvements in mechanical techniques should also lead to a reduction in the hazard and also costs.

In the case of nuclear reactors, about 99% of the radioactivity is associated with the fuel which is removed following permanent shutdown. Apart from any surface contamination of the plant, the remaining radioactivity comes from "activation products" such as steel components that have long been exposed to neutron irradiation. Their atoms are changed into different isotopes such as iron-55, cobalt-60, nickel-63 and carbon-14. The first two are highly radioactive, emitting gamma rays. However, their half life is such that after 50 years from closedown their radioactivity is much diminished and the risk to workers largely gone.

Decommissioning Experience

Over the past 40 years, considerable experience has been gained in decommissioning various types of nuclear facilities. Some 100 commercial power reactors, as well as over 250 research reactors and a number of fuel cycle facilities, have been retired from operation.

European Reactors: To decommission its retired gas cooled reactors at the Chinon, Bugey and St Laurent nuclear power stations, Electricité de France chose partial dismantling and postponed final dismantling and demolition for 50 years. As other reactors will continue to operate at those sites, monitoring and surveillance do not add to the cost.

The French are building, at Marcoule, a recycling plant for steel from dismantled nuclear facilities. This metal will contain some activation products, but it can be recycled for other nuclear plants.

Decommissioning has begun at 25 UK reactors. One of the first was the Berkeley nuclear power station (2 x 138 MWe, MAGNOX reactors), closed for economic reasons in 1989 after 27 years of operation, where defueling was completed in 1992. The cooling ponds were then drained, cleaned and filled in, and the turbine hall was dismantled and demolished. The reactor buildings are in the final stages of preparation for an extended period of care and maintenance in the Safestor phase. Ultimately, they too will be dismantled, leaving the site to be leveled and landscaped. The same pattern is being followed at other UK reactor sites.

Spain's Vandellos-1, a 480 MWe gas-graphite reactor, was closed down in 1990 after 18 years of operation due to a turbine fire, which made the plant uneconomic to repair. In 2003, ENRESA concluded phase 2 of the reactor decommissioning and dismantling project, which allows much of the site to be released. After 30 years in Safestor, when activity levels have diminished by 95%, the remainder of the plant will be removed. The cost of the 63-month project was EUR 93 million.

Japan's Tokai-1 reactor, a UK MAGNOX design, is being decommissioned after 30 years service to 1998. After 5-10 years in storage, the unit will be dismantled and the site released for other uses. Total cost is expected to be about 25 billion Yen.

Germany chose immediate dismantling over safe enclosure for the closed Greifswald nuclear power station in the former East Germany, where five reactors had been operating. Similarly, the site of the 100 MWe Niederaichbach nuclear power plant in Bavaria was declared fit for unrestricted agricultural use in mid 1995. Following removal of all nuclear systems, the radiation shield and some activated materials, the remainder of the plant was below accepted limits for radioactivity, and the state government approved final demolition and clearance of the site.

The 250 MWe Gundremmingen-A unit was Germany's first commercial nuclear reactor, operating from 1966-77. Decommissioning work started in 1983, and moved to the more contaminated parts in 1990, using underwater cutting techniques. This project demonstrated that decommissioning could be undertaken safely and economically without long delays, and recycling most of the metal.

US Reactors: Experience in the USA has varied, but 14 power reactors are using the Safestor approach, while 10 are using or have used Decommissioning. Procedures are set by the Nuclear Regulatory Commission (NRC), and considerable experience has now been gained. A total of 31 power reactors have been closed and decommissioned. Site release often excepts the on-site used fuel storage in an ISFSI (independent spent fuel storage installation), which usually must await the Department of Energy taking away the used fuel (over which it has title) to a national repository sometime in the future.

Rancho Seco (single 913 MWe PWR) was closed in 1989, and in 1995 NRC approved a Safestor plan for it. However, the utility subsequently decided upon incremental dismantling and this is well under way. With expected completion of this at the end of 2008, only the waste storage building will remain, and most of the site can be de-licenced and open for other uses.

At multi-unit nuclear power stations, the choice has been to place the first closed unit into storage until the others end their operating lives, so that all can be decommissioned in sequence. This will optimize the use of staff and the specialized equipment required for cutting and remote operations, and to achieve cost benefits.

Thus, after 14 years of comprehensive clean-up activities, including the removal of fuel, debris and water from the 1979 accident, Three Mile Island 2 was placed in Post-Defueling Monitored Storage (Safestor) until the operating licence of unit 1 expires in 2014 so that both units are decommissioned together. Safestor was also being used for San Onofre 1, which closed in 1992, until licences for units 2 and 3 expired in 2013. However, after NRC changes, dismantling was brought forward to 1999, so it became an active Decon project which is expected to be completed in 2008. A small amount of work will remain to be completed with decommissioning of units 2 and 3 on the site.

One US Decommissioning project was the 60 MWe Shippingport reactor, which operated commercially from 1957 to 1982. It was used to demonstrate the safe and cost-effective dismantling of a commercial scale nuclear power plant, and the early release of the site. Defueling was completed in two years, and five years later the site was released for use without any restrictions. Because of its size, the pressure vessel could be removed and disposed of intact. For larger units, such components have to be cut up.

Immediate Decon was also the option chosen for Fort St. Vrain, a 330 MWe high temperature gas cooled reactor that was also closed in 1989. This took place on a fixed-price contract for US$ 195 million (hence costing less than 1 cent/kWh despite only a 16-year operating life) and the project proceeded on schedule to clear the site and relinquish its licence early in 1997 - the first large US power reactor to achieve this.

For Trojan (1180 MWe, PWR) in Oregon the dismantling was undertaken by the utility itself. The plant closed in 1993, steam generators were removed, transported and disposed of at Hanford in 1995, and the reactor vessel (with internals) was removed and transported to Hanford in 1999. Except for the used fuel storage, the site was released for unrestricted use in 2005. The cooling tower was demolished in 2006.

Yankee Rowe (167 MWe, PWR) was shut down in 1991 after 30 years service. It was a Decon project and demolition was completed in 2006. Licence termination was in August 2007, allowing unrestricted public access, except for 2 ha for used fuel storage.

Another US Decon project was Maine Yankee, a 860 MWe PWR plant which was closed down in 1996 after 24 years of operation. The containment structure was finally demolished in 2004, and except for the 5 ha with the dry store for used fuel, the site was released for unrestricted public use in 2005 on budget and on schedule.

Connecticut Yankee (590 MWe PWR) was also shut down in 1996 after 28 years of operation. Decommissioning work began in 1998 and demolition was concluded in 2006. The site was released for unrestricted public use in 2007, apart from 2 ha for dry cask used fuel storage. Residual contamination on the land is below NRC's limit of 0.25 mSv per year for maximum radiation dose.

In 2005 the site of the small Saxton reactor which closed in 1972 was ready to be released for unrestricted use. It had been placed into Safstor in 1975 and the fuel shipped off site. Demolition began in 1986.

In 2006 the site of 72 MWe Big Rock Point nuclear power plant in Michigan, shut down in 1997 after 35 years operation, was largely returned to greenfield status. In January 2007 most of the land was released for derestricted public use, though 43 hectares still has the dry cask storage facility where used fuel is stored pending transfer to the national repository.

Other closed US plants are in Safstor. These include Zion 1 & 2, Humboldt bay, Indian Point 1, Dresden 1, Millstone 1, and Peach Bottom 1.

Further information on decommissioning in USA is available from NEI.

Other Facilities: The French Atomic Energy Commission is decommissioning the UP1 reprocessing plant at Marcoule. This plant started up in 1958 and treated 18,600 tonnes of metal fuels from gas cooled reactors (both defense and civil) to 1997. Progressive decontamination and dismantling of the plant and waste treatment will span 40 years and cost some EUR 5.6 billion, nearly half of this for treatment of the wastes stored on the site.

Many nuclear submarines have been decommissioned over the last decade. In USA, after defueling, the reactor compartments are cut out of the vessels and are transported inland to Hanford, where they are buried as low-level waste.

Costs & Finance

In most countries the operator or owner is responsible for the decommissioning costs.

The total cost of decommissioning is dependent on the sequence and timing of the various stages of the program. Deferment of a stage tends to reduce its cost, due to decreasing radioactivity, but this may be offset by increased storage and surveillance costs.

Even allowing for uncertainties in cost estimates and applicable discount rates, decommissioning contributes a small fraction of total electricity generation costs. In USA many utilities have revised their cost projections downwards in the light of experience, and estimates now average $325 million per reactor all-up (1998 $).

Financing methods vary from country to country. Among the most common are:

a) Prepayment, where money is deposited in a separate account to cover decommissioning costs even before the plant begins operation. This may be done in a number of ways but the funds cannot be withdrawn other than for decommissioning purposes.

b) External sinking fund (Nuclear Power Levy): This is built up over the years from a percentage of the electricity rates charged to consumers. Proceeds are placed in a trust fund outside the utility's control. This is the main US system, where sufficient funds are set aside during the reactor's operating lifetime to cover the cost of decommissioning.

c) Surety fund, letter of credit, or insurance purchased by the utility to guarantee that decommissioning costs will be covered even if the utility defaults.

In USA, utilities are collecting 0.1 to 0.2 cents/kWh to fund decommissioning. They must then report regularly to the NRC on the status of their decommissioning funds. As of 2001, $23.7 billion of the total estimated cost of decommissioning all US nuclear power plants had been collected, leaving a liability of about $11.6 billion to be covered over the operating lives of 104 reactors (on basis of average $320 million per unit).

An OECD survey published in 2003 reported US dollar (2001) costs by reactor type. For western PWRs, most were $200-500/kWe, for VVERs costs were around $330/kWe, for BWRs $300-550/kWe, and for CANDU $270-430/kWe. For gas cooled reactors the costs were much higher due to the greater amount of radioactive materials involved, reaching $2600/kWe for some UK MAGNOX reactors.

International Cooperation

The IAEA, the OECD's Nuclear Energy Agency and the Commission of the European Communities are among a number of organizations through which experience and knowledge about decommissioning is shared among technical communities in various countries.

In 1985, the OECD Nuclear Energy Agency launched an International Co-operative Program for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects. This international collaboration, covering 15 reactors and six fuel-cycle facilities, has produced a great deal of technical and financial information.

The important areas where experience is being gained and shared are the assessment of the radioactive inventories, decontamination methods, cutting techniques, remote operation, radioactive waste management and health and safety. The aims are to minimize the radiological hazards to workers and to optimize the dismantling sequence and timing to reduce the total decommissioning cost.

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[1] *Assumes that electricity is generated via condensing turbine

Appendix M: Utilizing Nuclear Power in Coal Liquefaction & Gasification

Introduction

Direct Coal Liquefaction (DCL) technology involves making a partially refined synthetic crude oil from coal, which is then further refined into synthetic gasoline or diesel fuel. In the DCL process, coal reacts with hydrogen and usually in the presence of a liquid solvent. One of the main challenges with coal-to-liquid technologies is increasing the hydrogen-carbon ratio. In the DCL process, this is achieved through adding gaseous H_2 to a slurry of pulverized coal, and recycled coal derived liquids in the presence of suitable catalysts to produce synthetic crude oil. Hydrogen is also needed for reducing oxygen, sulfur and nitrogen in the coal feedstock.

Although the DCL technology has been demonstrated as a viable option for producing liquid fuels, DCL is not presently proven at commercial scale. The largest scale for which there has been experience with DCL technology in the US is a Process Development Unit at the Hydrocarbon Technology, Inc. (HTI) R&D facility in Lawrenceville, New Jersey. This facility consumes 3 tonnes of coal per day. However, China is currently actively pursuing construction of commercial DCL plants, and is quickly becoming a major player in coal-to-liquid technology development.

The Indirect Coal Liquefaction (ICL) method implies a two-step process. The first step is a highly endothermic coal gasification process which results in production of syngas, which is a combination of various molecules, of which CO and H_2 are the main components. The second step is an exothermic catalytic process, whereby CO and H_2 molecules are combined to produce new compounds which can be used as fuels, either hydrocarbon fuels (synthetic gasoline, synthetic diesel) or oxygenated fuels. In the ICL process, the challenge of increasing the H/C ratio can be addressed by using the water-gas-shift reaction ($CO + H_2O = H_2 + CO_2$) and subsequently removing the CO_2 produced from the system. The three most important hydrocarbon fuels presently obtained from the ICL process are Fischer-Tropsch (F-T) liquids, methanol (CH_3OH) and dimethyl ether (DME).

The Fischer-Tropsch process results in the production of large hydrocarbon molecules. In particular, olefin-rich products such as naphta, where the number of carbon atoms is in the range (5 - 10) can be used for making synthetic gasoline and chemicals in high-temperature F-T process. Paraffin-rich products in turn, with the number of carbon atoms 12 to 19 (distillates) are well suited for making diesel and/or waxes in the low-temperature F-T processes. Technology development has currently proven that making synthetic diesel fuel is a better option than producing a raw naphtha product which would subsequently require substantial refining to obtain an acceptable gasoline. F-T technology is well established commercially and is the focus of global efforts to produce synthetic liquid fuels. The F-T process is outlined in Figure M-1 below.

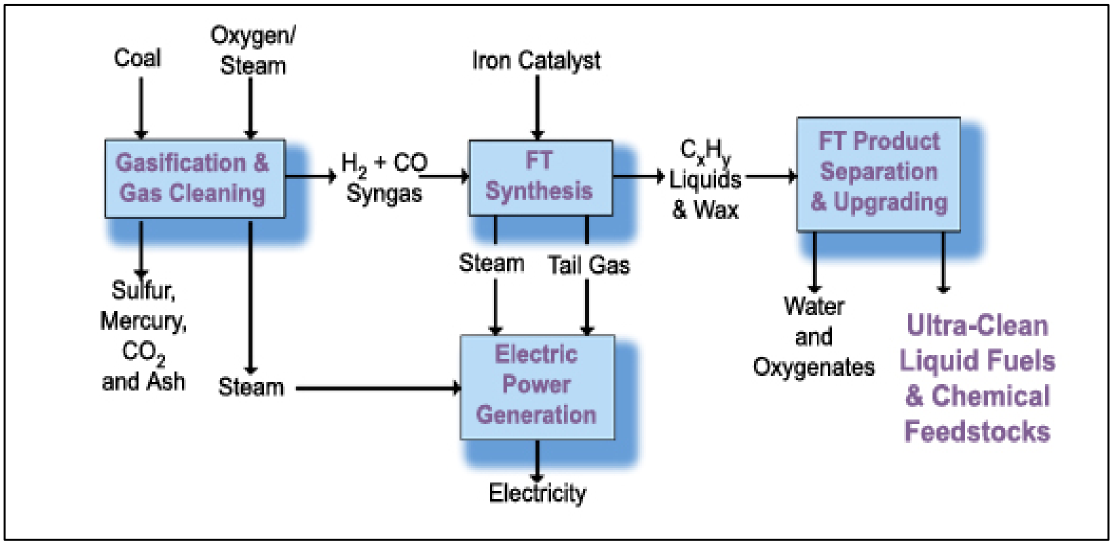


Figure M-1: Fischer-Tropsch Process Outline

As shown in Figure M-1, the main step is the synthesis process which reacts and polymerizes syngas to yield a range of products. Further processing of those products is necessary to upgrade the waxy diesel fraction, the low-octane-number gasoline fraction, and a large amount of oxygenates in the resulting water. Upgrading is important for achieving the required levels of boiling ranges, as well as the appropriate content of certain hydrocarbons so that resulting products will exhibit acceptable cold climate properties. A premium diesel fuel is manufactured from the higher-molecular-weight hydrocarbons and the wax.

The global chemical reaction equation for the F-T process is $CO + 2\ H_2 = (C\ H_2)_n + H_2O$. A chain growth mechanism is followed in the F-T reaction, where ($C\ H_2$) is the building block for the larger, chained hydrocarbons. The reaction is an exothermic process which occurs in the presence of a catalyst, primarily iron or cobalt. Due to the exothermic nature of the process, special considerations must be made to account for proper heat removal to control the temperatures.

Methanol can also be obtained from the syngas by means of the water-gas-shift reaction (above) followed by methanol synthesis reaction ($CO + 2\ H_2 = CH_3OH$). China has recently been the main producer of CH_3OH from syngas. However, under the US Department of Energy Clean Coal Technology Program, Air Products and Chemicals Inc., and the Eastman Chemical Company are currently bringing a slurry-phase reactor technology to commercial readiness for the production of methanol from coal.

Dimethyl ether can either be obtained by methanol dehydration, or by a single-step process from syngas. This type of a technology is currently being developed by Air Products and Chemicals Inc. in the US, and by the NKK Corporation in Japan.

Coal Gasification Step

As outlined in the previous section, the first step in indirect coal liquefaction is gasification to produce the synthesis gas. In fact, the gas itself can also be used as a fuel to generate electricity. Coal gasification is a highly endothermic process, and large amounts of energy are required for the process to take place. The sketch of coal gasification is shown in Figure M-2.

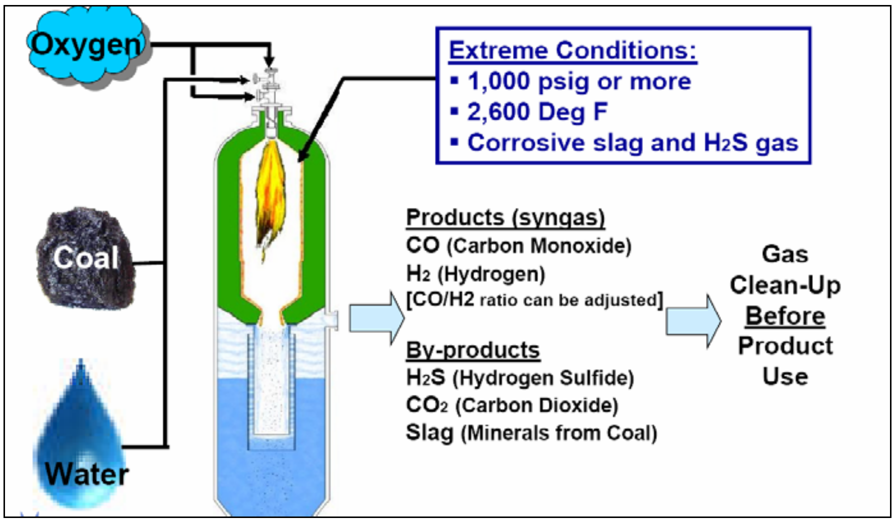


Figure M-2, Coal Gasification Process Outline

As shown in Figure M-2, in addition to the main components of resulting syngas (CO and H_2), other byproducts are present. In particular, sulfur impurities in coal are converted into hydrogen sulfide and other compounds, from which sulfur is subsequently extracted. The hydrogen sulfide can be cleared from the syngas in an integrated gasification combined cycle (IGCC). Some amount of syngas, in turn can be burned in a combustion turbine, thus providing the energy to drive the electric generator. The hot air from the turbine can be routed back to the gasifier, while the exhaust heat can be recovered to boil water to provide steam for the steam turbine-generator. This type of a combination is known as a combined cycle, and provides high power generation efficiencies.

Gasification normally occurs in the temperature range between 1000°C and 1500°C, in the presence of steam. In fact, the process temperatures may vary depending on conditions and configurations, but they may be considered of the order of 1000°C. Pressures may vary as well. In particular, relatively high pressures of 75 bar are applicable to a reactor where coal is fed in a water slurry. The heat for the process is normally provided through a partial oxidation of coal.

HTGR for Coal Gasification & Liquefaction

Using the HTHR for Gasification & Fuel Synthesis: As was described in the previous sections, our economy is geared to using liquid fuels for transportation. Producing these liquid fuels from coal is a viable technology, which is presently employed in various forms in the US, China and South Africa. Supplying hydrogen to the system is one of the key conditions for obtaining higher-grade fuels. Coal gasification is a highly endothermic process, and the required heat is currently provided through oxidation of large quantities of coal. Burning coal in order to gasify the remaining feedstock of coal raises concerns associated with CO_2 emissions, as well as utilization, since a significant amount of coal must be used up to provide heat. Therefore, using a Modular Helium Reactor to supply energy, steam, process heat, electricity and hydrogen without CO_2 production appears to be a viable option from technological, economical and environmental considerations. In particular, the very high temperatures of the order 1000 °C that are generated by GTHR appear to be an important factor in using this system for the coal gasification step described in the previous section. In addition, if the HTGR is used the oxygen would not be required in the gasification process. With respect to the Fischer-Tropsch synthesis, the HTGR can supply this process with the required hydrogen and electric power.

Schematic Design and Thermodynamic Considerations: The process flow diagram for applying HTGR to coal gasification and liquefaction is given in Figure M-3.

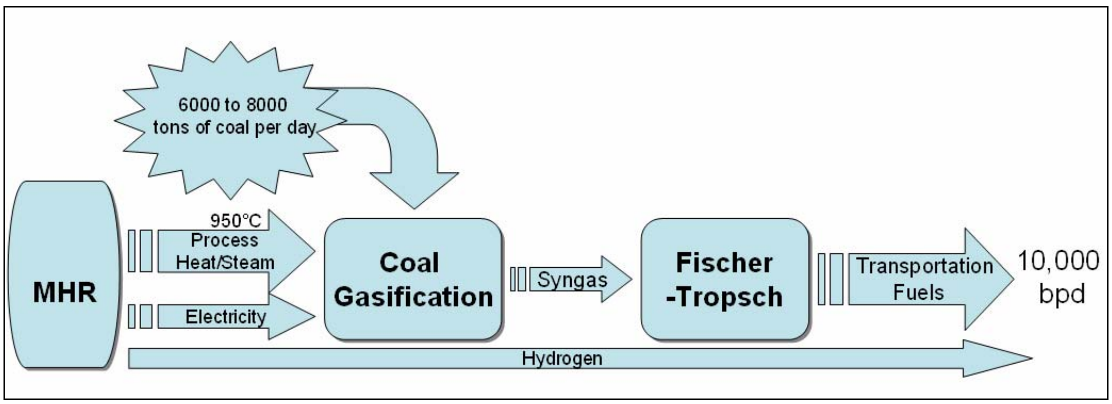


Figure M-3, Process Flow Diagram

A more specific flow diagram for the HTGR for to coal gasification combined with liquid fuel synthesis with operating temperatures is shown in Figure M-4.

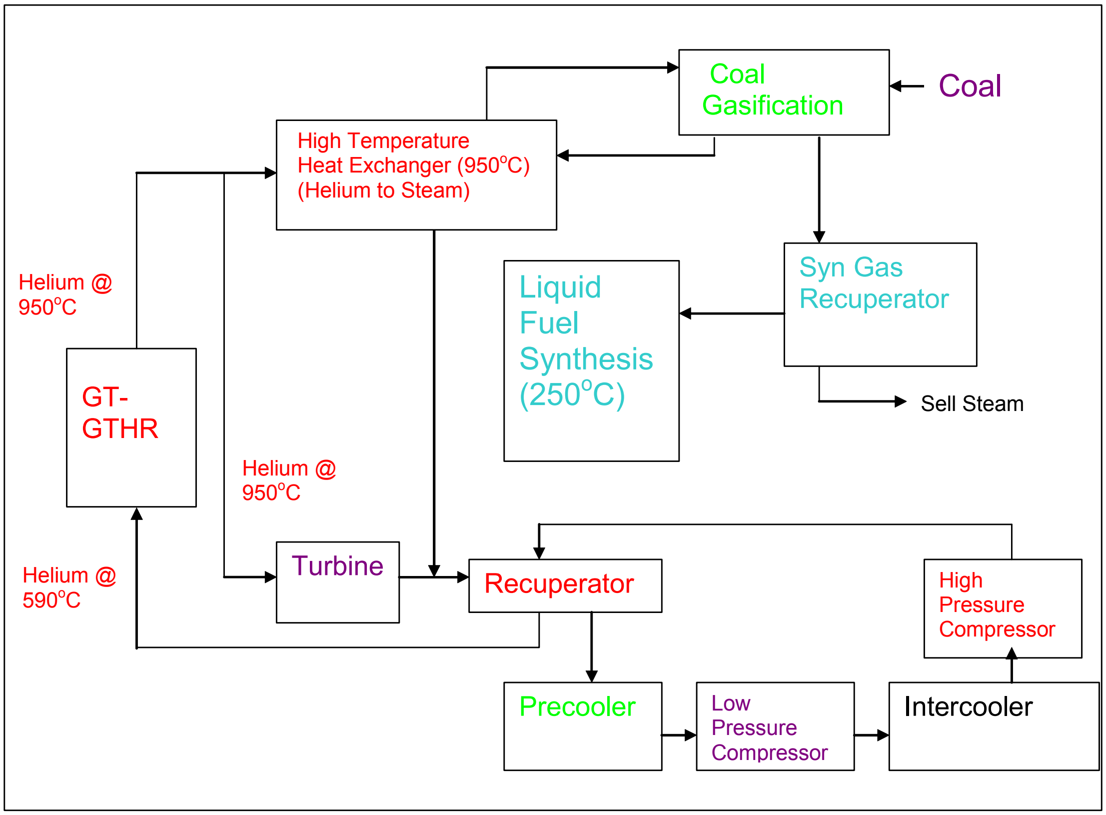


Figure M-4, Process Schematic Design

Let us assume the gasification process temperatures are in the order of 1000 °C, which can be achieved through the use of HTGR, where the helium outlet temperature can be brought up to 950 °C. Using the high temperature heat exchanger, we are going to transfer thermal energy to steam. The resulting steam temperatures coming out of the heat exchanger are expected to be in the range of 900 to 925°C. The steam is subsequently sent into the gasifier which would operate at relatively high pressures in the range of 75 bar. It is assumes that there is no condensation process, and therefore no two-phase regions. Indeed, the boiling point of water even at such pressures is below 290 °C. The liquid fuel synthesis, as described in previous sections, is an exothermic catalytic process taking place in the temperature range around 250 °C. The raw syngas will be cooled in a high-temperature syngas cooler (recuperator). As an initial estimate, most of the heat from a 600 MWth reactor is expected to go to the coal gasification process, while a certain amount in the range of 10 MW is needed for turbine/generator to provide electricity for the required processes. That value may be changed as more analysis is completed on thermodynamic efficiencies. In addition, it is in fact possible to expect that if the gas turbine electric production is just a small portion of the total heat load, the pre-cooling, inter-cooling and recuperation may not be required for efficient operation. That kind of determination will be made as more detailed analysis is conducted.

Further Notes on Coal Liquefaction & Gasification with Nuclear Input

Coal-to-Liquids: The process for converting coal to liquids (CTL) was developed in Germany in the 1920s. By World War II it became the source of 90% of that nation's liquid fuel requirements, with nine indirect and eighteen direct liquefaction plants producing 4 million metric tonnes per year. Later, as a result of the apartheid based embargoes, South Africa used technology similar to that used by the Germans and developed its own CTL industry that now produces up to 10 million metric tonnes per year meeting about 40% of the country's current liquid fuel needs. There is also a growing interest in other countries with major coal reserves (e.g. the U.S. and China) to develop processes that can exploit the large coal deposits to meet their growing petroleum requirements. For example, if the coal deposits in the U.S. were converted to liquid hydrocarbons, they would represent over 60% of the world's proven oil reserves. China is experiencing growth in coal liquefaction as a way of utilizing its coal reserves and reducing its dependence on imported oil. The South African company - Sasol is planning two CTL plants in China[3], and in the US some nine states are actively considering CTL plants. Global liquid coal production is expected to rise from 150,000 bpd today to 600,000 in 2020, and 1.8 million bpd in 2030[4].

The most advanced process for producing liquid fuels from coal is gasification, followed by the Fischer-Tropsch (F-T) process[5]. The simplified cycle is shown in Figure M-5. The process first gasifies the coal with steam to form "syngas" (a mixture of hydrogen and carbon monoxide). The sulfur is removed from this gas and the mixture is adjusted according to the desired product. The syngas is then routed to the F-T process. The F-T process assembles the hydrocarbon building blocks in the presence of a catalyst to produce high quality clean fuels. Note that the first step in the CTL process is the gasification of the coal, and so if the objective is conversion of coal to a gaseous fuel (CTG), where the same initial process step is involved.

[3] "Sasol Plans Two Coal-to-liquid Fuel Projects", published in China Daily on January 30, 2007; downloaded at http://www.china.org.cn/english/BAT/198162.htm
[4] Newsweek "Special Energy Edition", Dec 2006- Feb 2007
[5] David Gray, NRCB on Energy and Environmental Systems Workshop, October, 2005

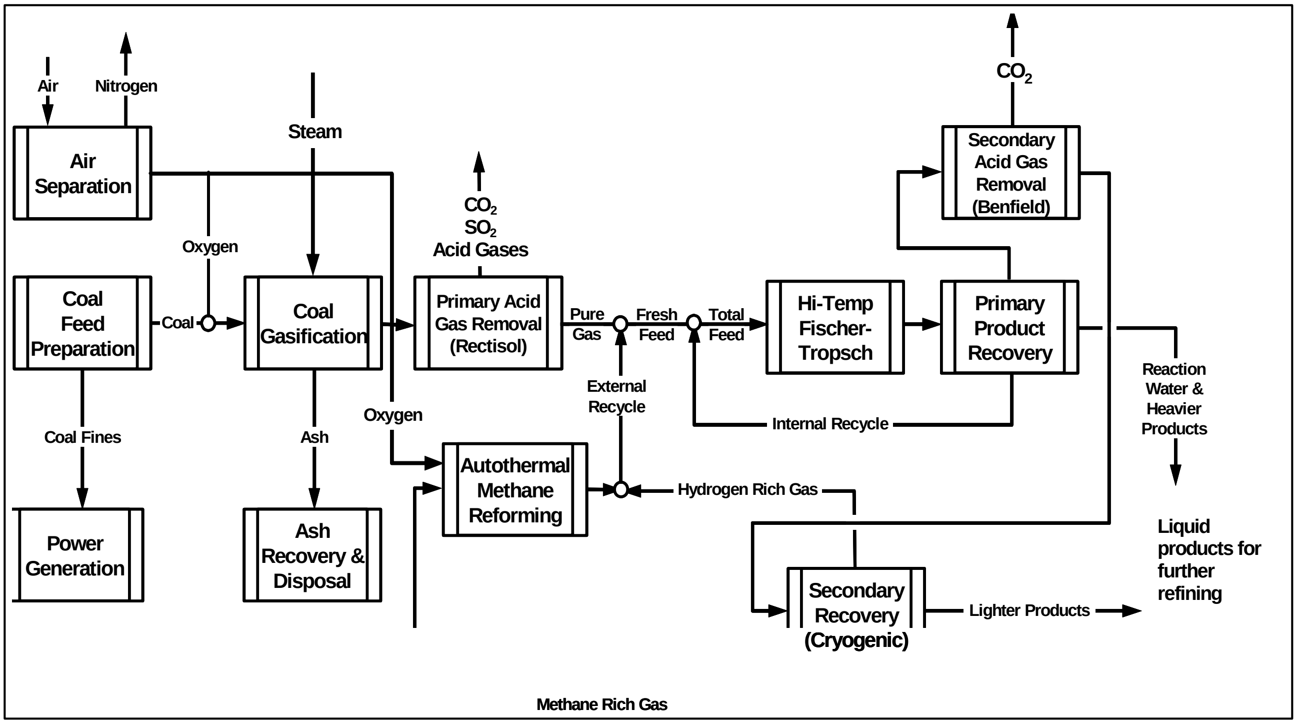


Figure M-5, Coal to Liquids Process

While this is an attractive process for converting a very abundant resource (coal) into petroleum products which are in great demand, there are several disadvantages. The process is quite energy intensive, requiring coal to fire the gasifier and electricity to power the air separation plant. In addition, the process generates a significant amount of CO_2. Estimates[6,7] of the "excess" CO_2 generated in this process over the conventional sweet crude refining process vary, depending on the degree of coal conversion, but are typically more than what is generated from burning the resultant fuel.

Coal is a hydrocarbon, which is an agglomeration of "large" molecules. A representative bituminous coal molecule is $C_{137}H_{97}O_9NS$. Liquid hydrocarbons have carbon numbers in the range of 5 to 20, with gaseous hydrocarbons ranging from 6 and below. The gasification process must break apart the coal molecules. Gaseous and liquid hydrocarbons have ratios of H to C of from 2 to 4, while coal has an approximate ratio of H to C of 0.8. The objective then is to remove C and add H. Oxygen is input to the gasification step to partially oxidize the coal. This reaction releases heat that breaks the molecular bonds, and the oxygen combines with C atoms to make CO, which along with H_2 is the feedstock for the F-T process.

The F-T process requires a syngas input with an H to C ratio of about 2. The reaction is generally shown as follows.

---

[6] Forsberg paper on transport fuels
[7] Penfield and Bolthrunis paper on Nuclear-Coal Synergism

(2n + 1) H_2 + n CO → C_nH_{2n+2} + n H_2O (Eq. 3-1)

Since the ratio of H to C in coal is ~0.8, another source of hydrogen is required. In the conventional CTL process, this additional hydrogen is obtained via the conventional "water gas shift" reaction in which part of the CO is reacted with water. This provides the required hydrogen, but is also the principal source of process-related CO_2.

CO + H_2O → CO_2 + H_2 (Eq. 3-2)

In order to illustrate the benefits of integrating an HTGR into this CTL process, we take the conventional CTL process and combine it with an HTGR driving a water-splitting plant to provide oxygen and hydrogen for the coal to liquids process. Oxygen from the water-splitting plant eliminates the air-to-oxygen plant as shown in the previous figure, and so reduces the electric power compared to the conventional CTL plant design. Input of hydrogen to the stream exiting the gasifier simplifies the syngas production step (a single box in the figure) and eliminates the water-gas shift reaction, which is the dominant source of CO_2 from the process (which in the non-nuclear case is mostly extracted from the syngas in the primary acid gas removal step). Note that there are significant savings in the CTL plant capital costs, coal costs, operations and maintenance costs. Nuclear power input to the CTL process offsets the use of coal as the hydrogen source, enhances the product yield per unit coal input, and essentially eliminates the CO_2 in the production process.

The integrated HTGR CTL process is shown in Figure M-6, with highlights of the differences from the conventional coal-powered cycle.

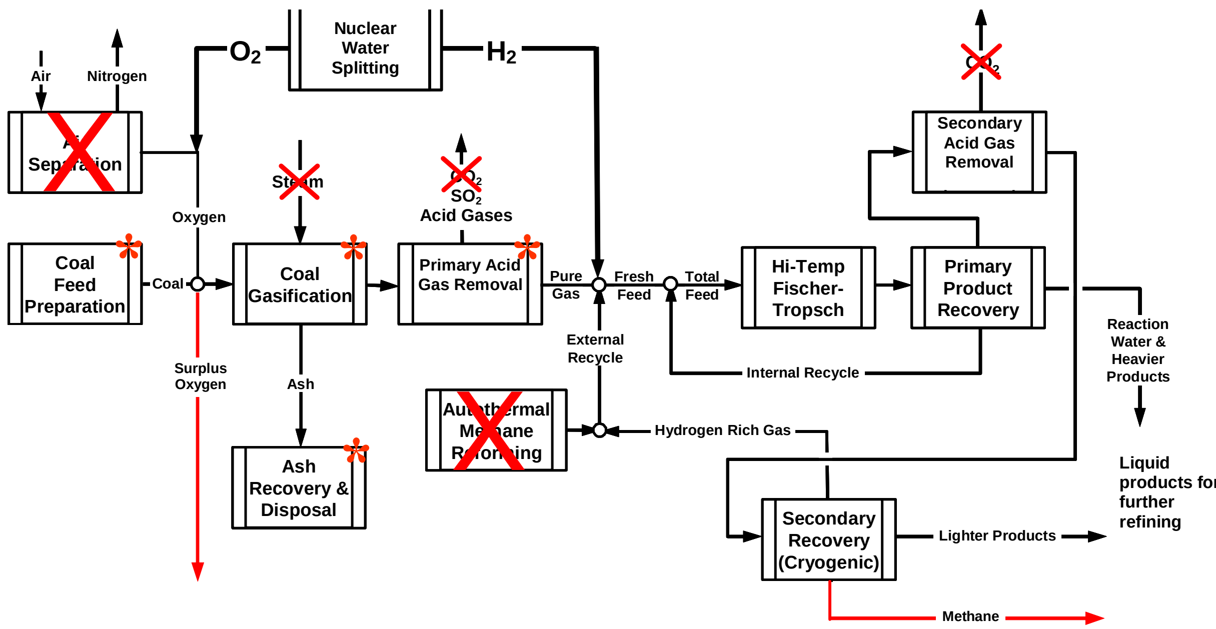


Figure M-6, Integrated HTGR Coal to Liquids Process (Showing Simplifications)

The benefits of the integrated HTGR CTL process are summarized below.

a) Overall process simplification;

b) Reduces coal use by ~40%;

c) Reduces size of coal handling and gasifiers by ~40%;

d) Eliminates the need for a separate oxygen plant;

e) Eliminates the water gas shift stage;

f) Reduces requirement for return and reforming of gas from F-T plant;

g) Environmental benefits;

h) Eliminates CO_2 production and the need for sequestration;

i) Reduces waste volumes.

An F-T plant produces a product that can be fractionated and/or refined into LPG, gasoline, diesel (highly sulfur-free), and petrochemical feedstocks.

Economic estimates to-date have focused on the cost of hydrogen and oxygen production from the various processes, and the integrated flowsheet analysis for such coal conversion processes.

In the interim, the net present values (NPV) of the displaced coal and the related CO_2 credits serve as indicators of the minimum opportunity for the HTGR and hydrogen production plants.

Coal-to-Gas: There are various technologies for coal-to-gas (CTG) conversion, where the concept is to generate a gaseous pipeline product. In the CTL processes, the purified product stream from the coal gasifier is syngas, which is a mix of H_2 and CO. Syngas is distributed by certain pipelines and used in the petroleum and petrochemical industries today, but it is made from natural gas. Syngas can be used to produce high purity hydrogen for industrial use, or for other uses such as is envisioned for the future automotive Hydrogen Economy, or it can be used as a chemical feedstock. Hydrogen has been considered for blending into natural gas pipelines to enrich the gas. If converted to methane, the product is Synthetic Natural Gas (SNG). The CTG options are shown in Figure M-7.

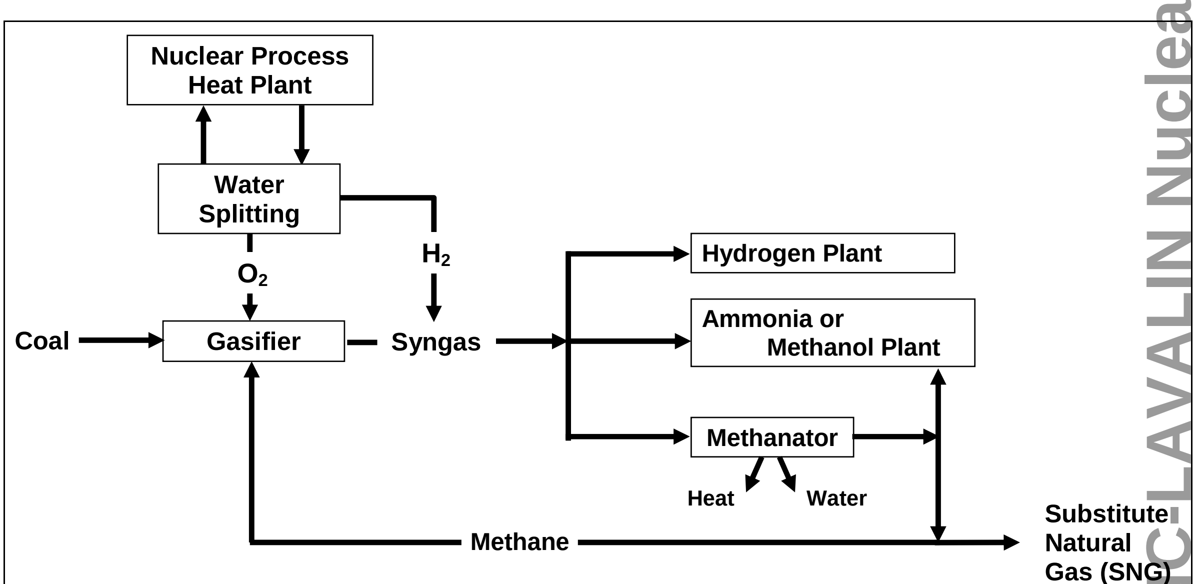


Figure M-7, Coal to Gas Options

As with the CTL, the utilization of oxygen and hydrogen from nuclear water splitting increases overall cycle efficiency, improves the yield per unit of coal, and greatly reduces CO_2 generation.

In the interim, the net present values (NPV) of the displaced coal or natural gas and the related CO_2 credits serve as indicators of the minimum opportunity for the HTGR and hydrogen production plants, as discussed in the introduction.

Additional information is included in the presentation made by General Atomics which follows.

Process Heat Applications

Coal Liquefaction and gasification from the MHR

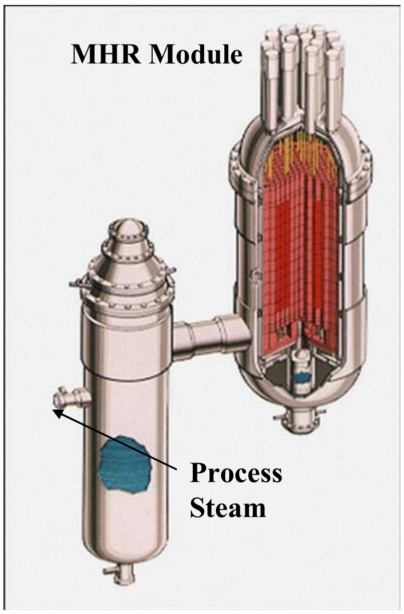
Presentation by
Alan Baxter
General Atomics

2

GENERAL ATOMICS

A first step – the MHR for Process Steam and Cogeneration

PS/C-MHR GENERATES STEAM AT 1000°F (540°C) AND 2500 PSI (17 Mpa)



PS/C-MHR Typical Plant Parameters

Thermal Power, MW(t)	600
Fuel Columns	102
Fuel Cycle	LEU/Natural U
Average Power Density, W/cm³	6.6
Primary Side Pressure, MPa (psia)	7.07 (1025)
Induced Helium Flowrate	281 kg/s
Core Inlet Temperature, °C (°F)	288(550)
Core Outlet Temperature, °C (°F)	704(1300)
Steam Temperature, °C (°F)	541(1005)
Steam Pressure, MPa (psia)	17.3(2515)
Circulator Power, MW(e)	6

Applications

- Heavy Oil Recovery
- Oil from Tar Sands
- Coal liquefaction
- Coal Gasification
- Oil from Oil Shale

3

GENERAL ATOMICS

Synthetic Fuels

- **Definition:**
  - hydrocarbon resources that require considerable chemical upgrading for use as fuel
- **Main processing requirement is adding energy and hydrogen:**
  - Starting resource can be tar sands, shale, coal.
  - Product is useable high-energy synthetic gas or liquid.
- **Basic concept:**
  - Extract hydrogen from water.
  - Followed by hydrogen enrichment of feedstock through various chemical process.

4

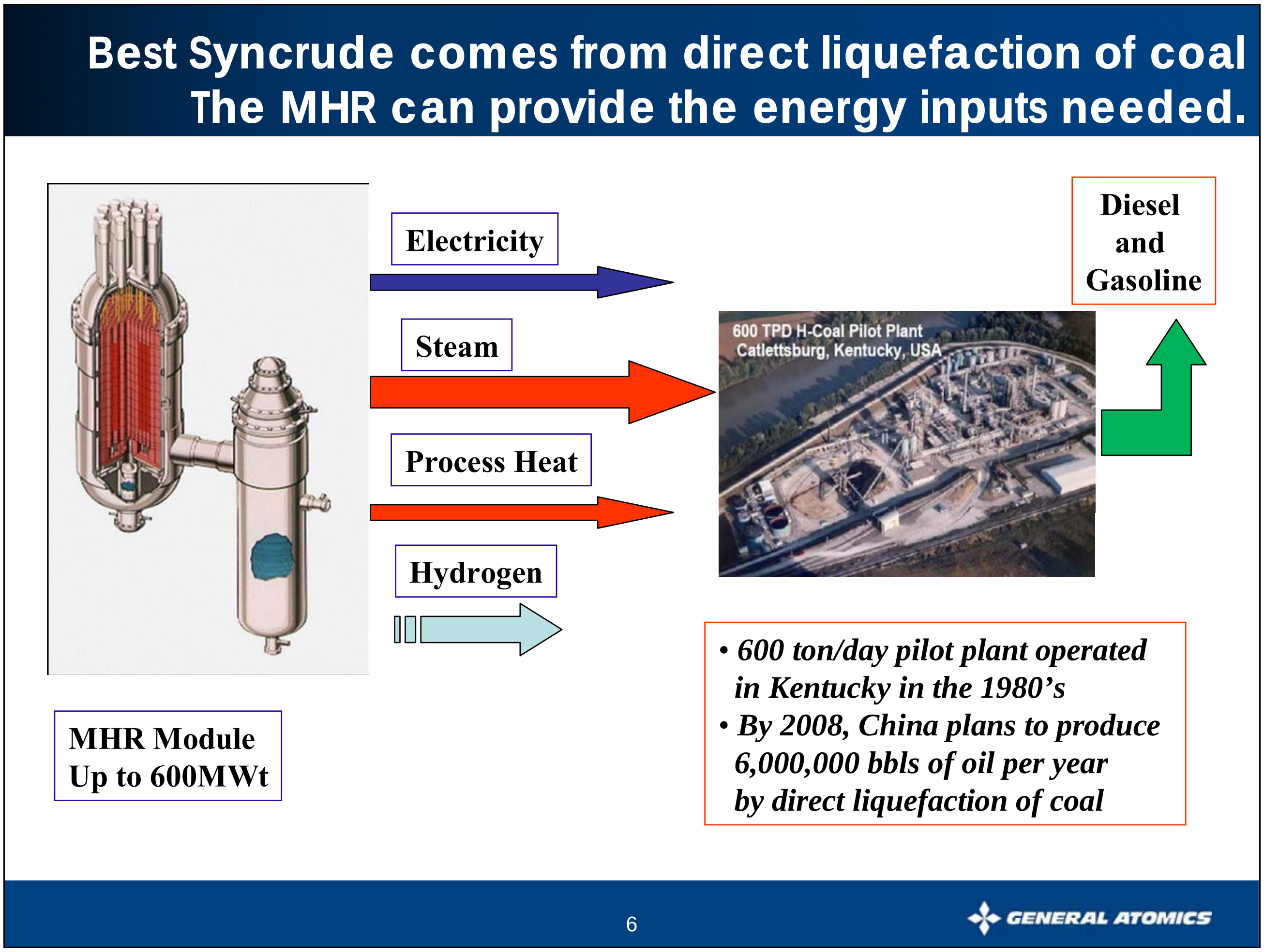
GENERAL ATOMICS

Synthetic Fuels – from Coal

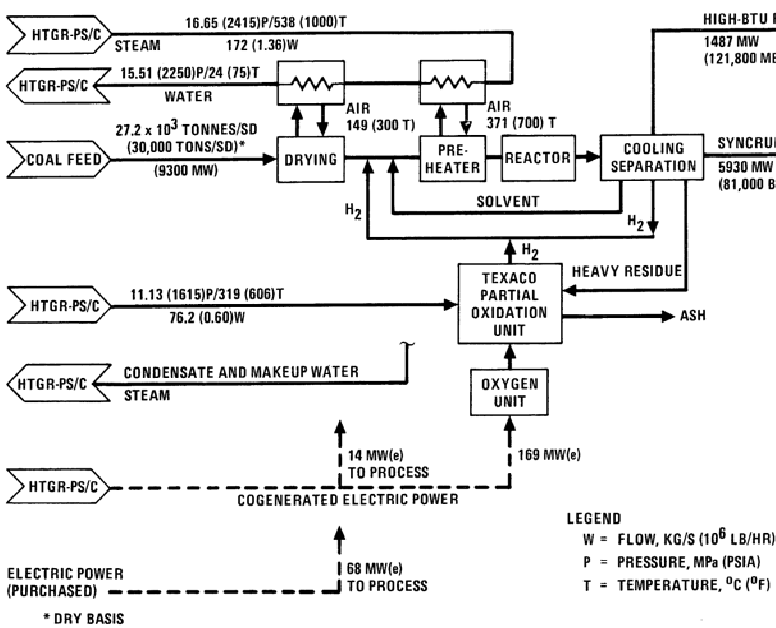
- **The US has enormous reserves of coal:**
  - Almost 500 billion tons.
- **Our economy is geared to using liquid (oil) or gaseous (natural gas) energy sources:**
  - Equipment and infrastructure.
- **Thus one near term solution to our energy problem is to convert coal to useful liquids or gases in an environmentally acceptable fashion:**
  - Liquefaction, indirect (Fischer-Tropsch) or direct using hydrogen.
  - Coal gas or hydrogasification.
  - The MHR can supply the energy, steam, process heat, electricity, and hydrogen required, without CO_2 production.

5

GENERAL ATOMICS


Best Syncrude comes from direct liquefaction of coal
The MHR can provide the energy inputs needed.
Electricity
Steam
Process Heat
Hydrogen
Diesel and Gasoline
600 TPD H-Coal Pilot Plant
Catlettsburg, Kentucky, USA
MHR Module
Up to 600MWt
• 600 ton/day pilot plant operated in Kentucky in the 1980's
• By 2008, China plans to produce 6,000,000 bbls of oil per year by direct liquefaction of coal
6
GENERAL ATOMICS

Process Flow Diagram Using the MHR for Coal Liquefaction (H-Coal Process)



- MHRs supply 1.36×10^6 lbs steam per hour.
- 135 MWt of process heat.
- 30,000 tons/dy of coal consumed

Produces:

- 81,000 bbl/dy syncrude and 121,800 mbtu/dy fuel gas

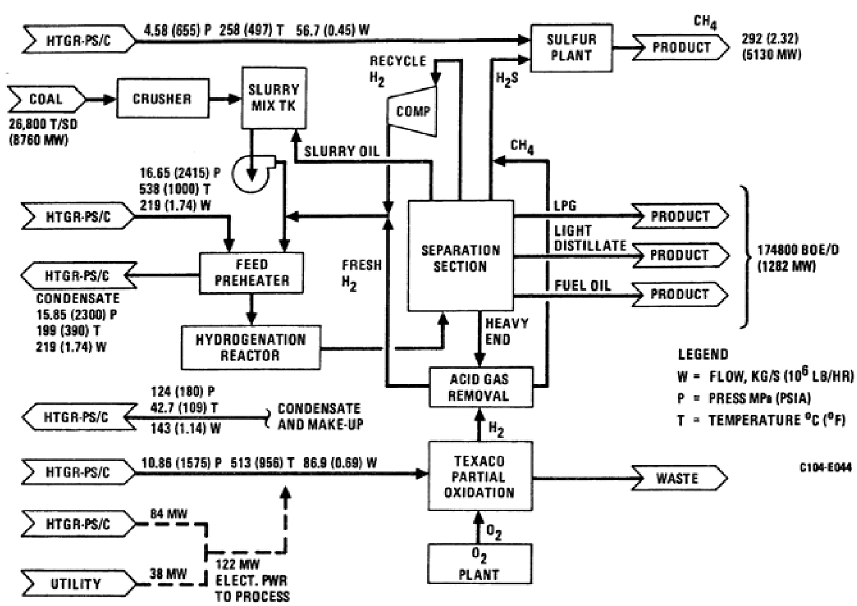
Costs should be competitive with coal burning.

Ref: "1170-MWt HTGR-PS/C Plant Application Study Report: H-Coal LiquefactionProcess Application," GA-A16091, May 1981

7

GENERAL ATOMICS

Process Flow for Synfuel Production Using the MHR and the Solvent Refined Coal (SRC-II) Process



- Two 600 MWt PS/C-MHR units.
- Delivers 1.74 million lbs/hr of 2415 psia 1000°F steam.
- Converts 27,000 tons of coal per day.
- Produces 174,800 BOE per day of fuel oils, light distillates, and gas.
- 1981 study showed nuclear Cheaper than coal or Steam production

(BOE = Barrels of Oil Equivalent)

Ref: "1170-MWt HTGR-PS/C Plant Application Study Report: SRC-II Process Application," GA-A16108, May 1981

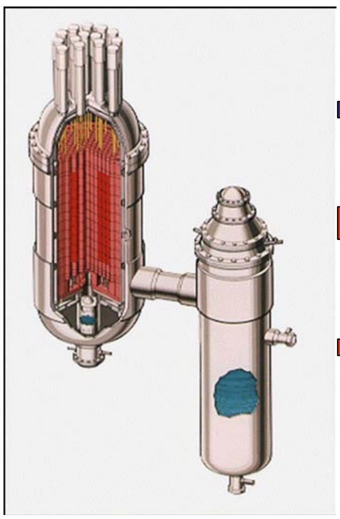
GENERAL ATOMICS

Synthetic Fuels – from Coal

- **The US has enormous reserves of coal:**
  - Almost 500 billion tons.
- **Our economy is geared to using liquid (oil) or gaseous (natural gas) energy sources:**
  - Equipment and infrastructure.
- **Thus one near term solution to our energy problem is to convert coal to useful liquids or gases in an environmentally acceptable fashion:**
  - Liquefaction, indirect (Fischer-Tropsch) or direct using hydrogen.
  - Coal gas or hydrogasification.
  - The MHR can supply the energy, steam, process heat, electricity, and hydrogen required, without CO_2 production.

9 GENERAL ATOMICS

Best Syncrude comes from direct liquefaction of coal
The MHR can provide the energy inputs needed.



Electricity

Steam

Process Heat

Hydrogen

MHR Module
Up to 600MWt

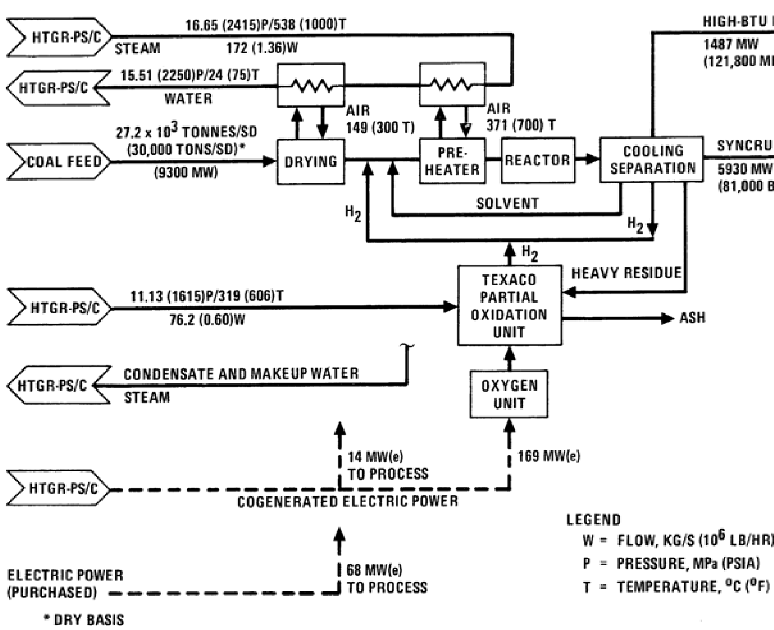
Diesel and Gasoline

- *600 ton/day pilot plant operated in Kentucky in the 1980's*
- *By 2008, China plans to produce 6,000,000 bbls of oil per year by direct liquefaction of coal*

10

GENERAL ATOMICS

Process Flow Diagram Using the MHR for Coal Liquefaction (H-Coal Process)



- MHRs supply 1.36×10^6 lbs steam per hour.
- 135 MWt of process heat.
- 30,000 tons/dy of coal consumed

Produces:

- 81,000 bbl/dy syncrude and 121,800 mbtu/dy fuel gas

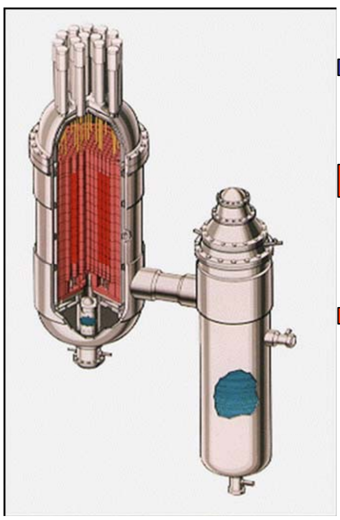
Costs should be competitive with coal burning.

Ref: "1170-MWt HTGR-PS/C Plant Application Study Report: H-Coal LiquefactionProcess Application," GA-A16091, May 1981

11

GENERAL ATOMICS

The MHR can provide the energy inputs needed for efficient, clean coal gasification.



MHR Module
Up to 600MWt

Electricity

Steam

Process Heat

Hydrogen

Natural Gas

- *2 Modules could allow the Processing of ~14,500 tons of coal per day.*
- *Produce ~43,000 BOE per day.*

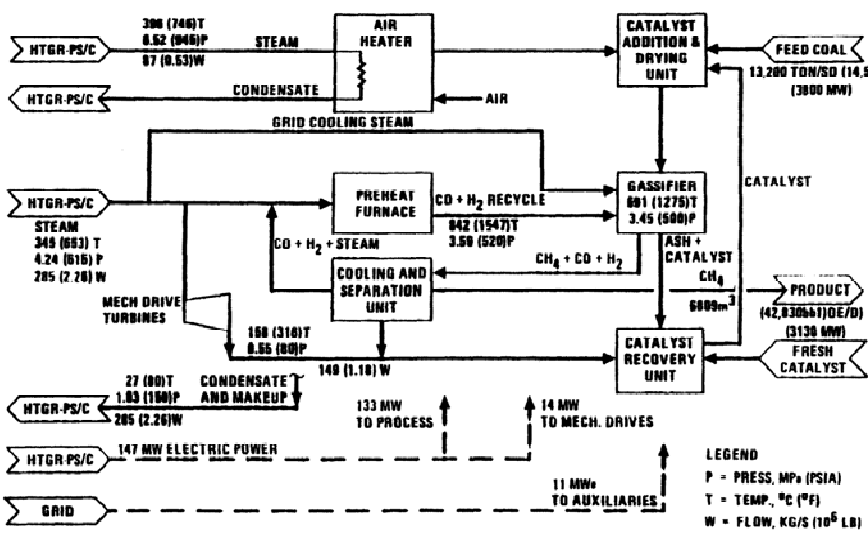
GENERAL ATOMICS

Title
NUCLEAR ENERGY OPTIONS EVALUATION REPORT

Project
Oil Sands Phase I Energy Options Feasibility Study

The MHR Coal Gasification Process Flow Diagram

2 unit PS/C-MHR plant supporting an Exxon Catalytic Coal Gasification (ECCG) Process



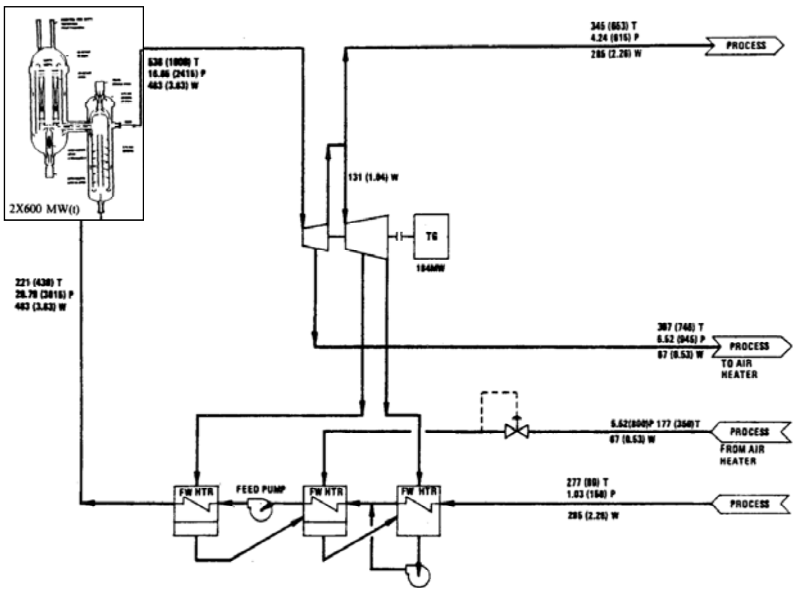
2 x 600MWt PS/C Units

- Processes ~14500 tons of coal per day.
- Produces ~43,000 BOE per day.
- Produces 147 MWe for system power needs.
- 1981 cost estimate showed lower costs than using coal.

GENERAL ATOMICS

 PROPRIETARY

The MHR Coal Gasification Cycle Diagram



Ref: "1170-MWt HTGR-PS/C Plant Application Study Report:
Exxon Catalytic Coal Gasification Process Application," GA-A16113, May 1981

14

GENERAL ATOMICS

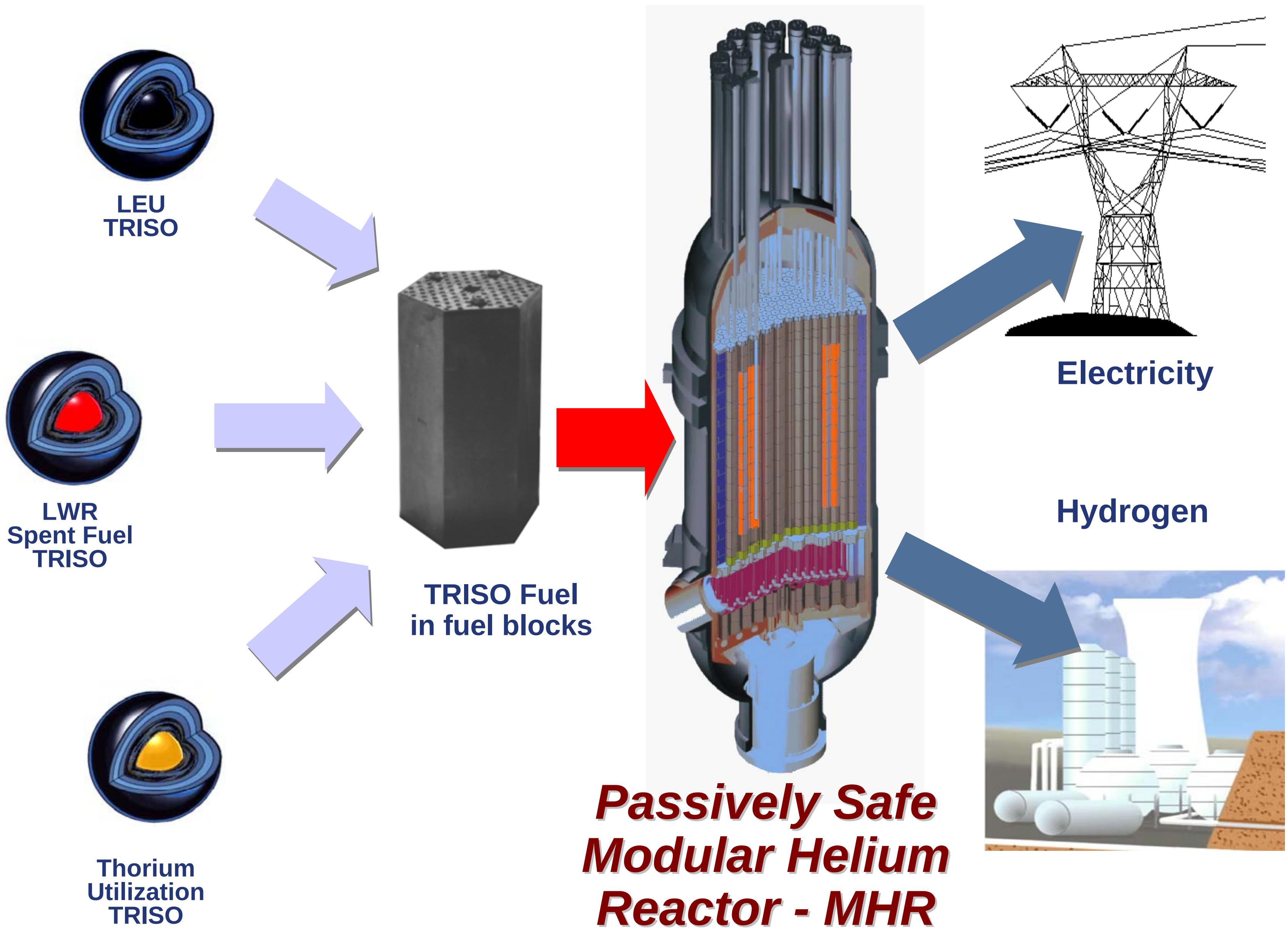
Appendix N: General Atomics Study of GA-HTGR Applicability to the Oils Sands

The General Atomics Study of GA-HTGR Applicability to the Oils Sands is provided in the following pages.

OVERVIEW OF
MODULAR HELIUM REACTOR
NUCLEAR POWER PLANTS

FOR THE SUPPLY OF
SAFE, CLEAN, ECONOMIC ENERGY

SINGLE REACTOR DESIGN HAS MULTIPLE APPLICATIONS



U.S. AND EUROPEAN TECHNOLOGY PROVIDE PROVEN BASES FOR PASSIVELY SAFE MHR

BROAD FOUNDATION OF HELIUM REACTOR TECHNOLOGY

EXPERIMENTAL REACTORS

DEMONSTRATION OF BASIC HTGR TECHNOLOGY

DRAGON
(U.K.)
1963 - 76

AVR
(FRG)
1967 - 1988

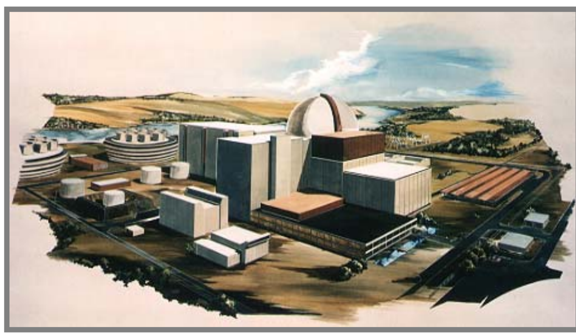
PEACH BOTTOM 1
(U.S.A.)
1967 - 1974

FORT ST. VRAIN
(U.S.A.)
1976 - 1989

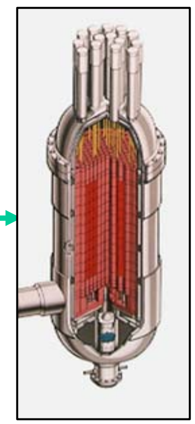
THTR
(FRG)
1986 - 1989

HTGR TECHNOLOGY PROGRAM

- **MATERIALS**
- **COMPONENTS**
- **FUEL**
- **CORE**
- **PLANT TECHNOLOGY**



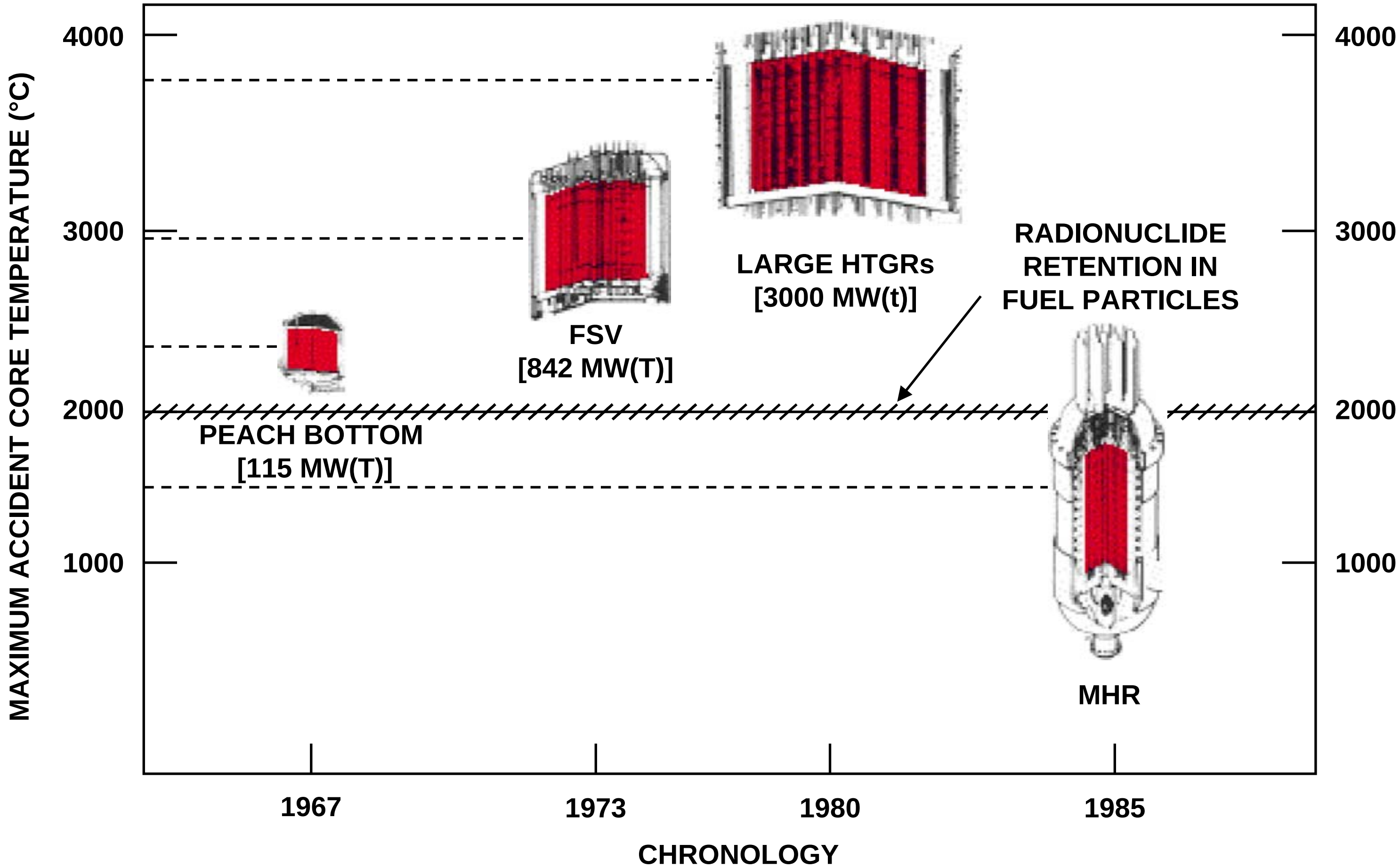
LARGE HTGR PLANTS



MHR

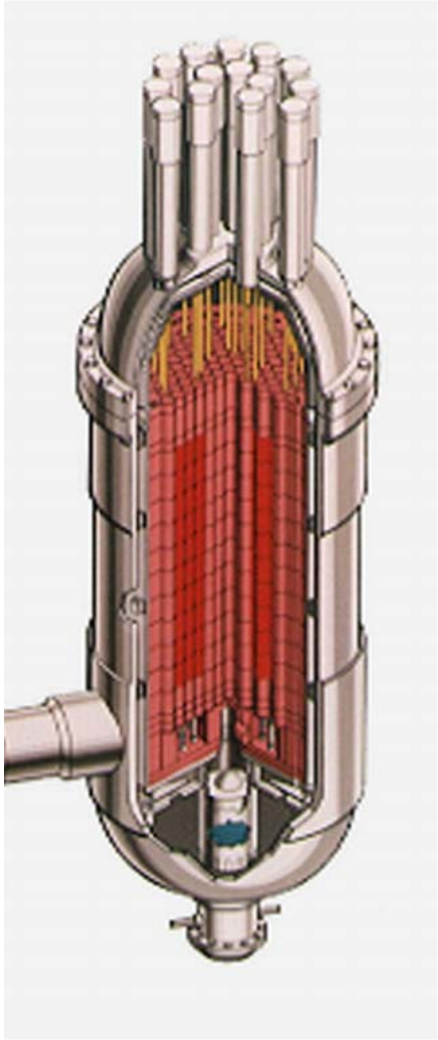
GENERAL ATOMICS

MHR REPRESENTS A FUNDAMENTAL CHANGE IN REACTOR DESIGN AND SAFETY PHILOSOPHY



...SIZED AND CONFIGURED TO TOLERATE EVEN A SEVERE ACCIDENT

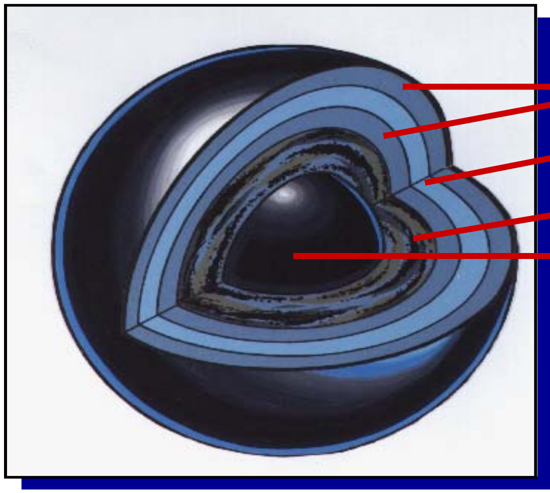
INHERENT REACTOR CHARACTERISTICS PROVIDE HIGH SAFETY



- **Helium gas coolant (inert)**
- **Refractory fuel (high temperature capability)**
- **Graphite reactor core (high temperature stability)**
- **Low power density (order of magnitude lower than LWRs)**
- **Demonstrated technologies**

. . . EFFICIENT, RELIABLE PERFORMANCE WITH INHERENT SAFETY

CERAMIC FUEL RETAINS ITS INTEGRITY UNDER SEVERE ACCIDENT CONDITIONS



TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right).

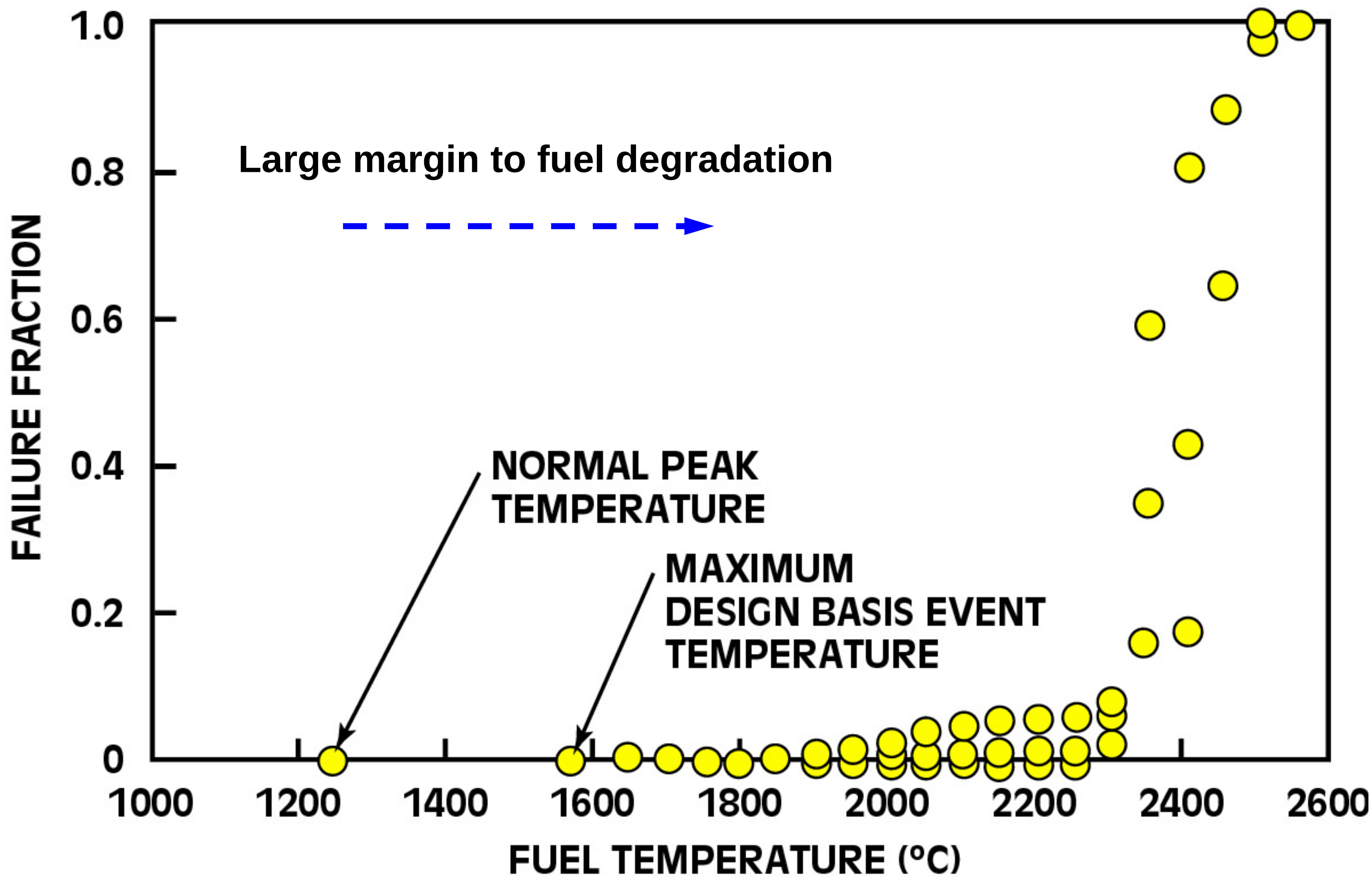
PARTICLES

COMPACTS

FUEL ELEMENTS

GENERAL ATOMICS

COATED PARTICLES STABLE TO BEYOND MAXIMUM ACCIDENT TEMPERATURES



GENERAL ATOMICS

GENERAL ATOMICS

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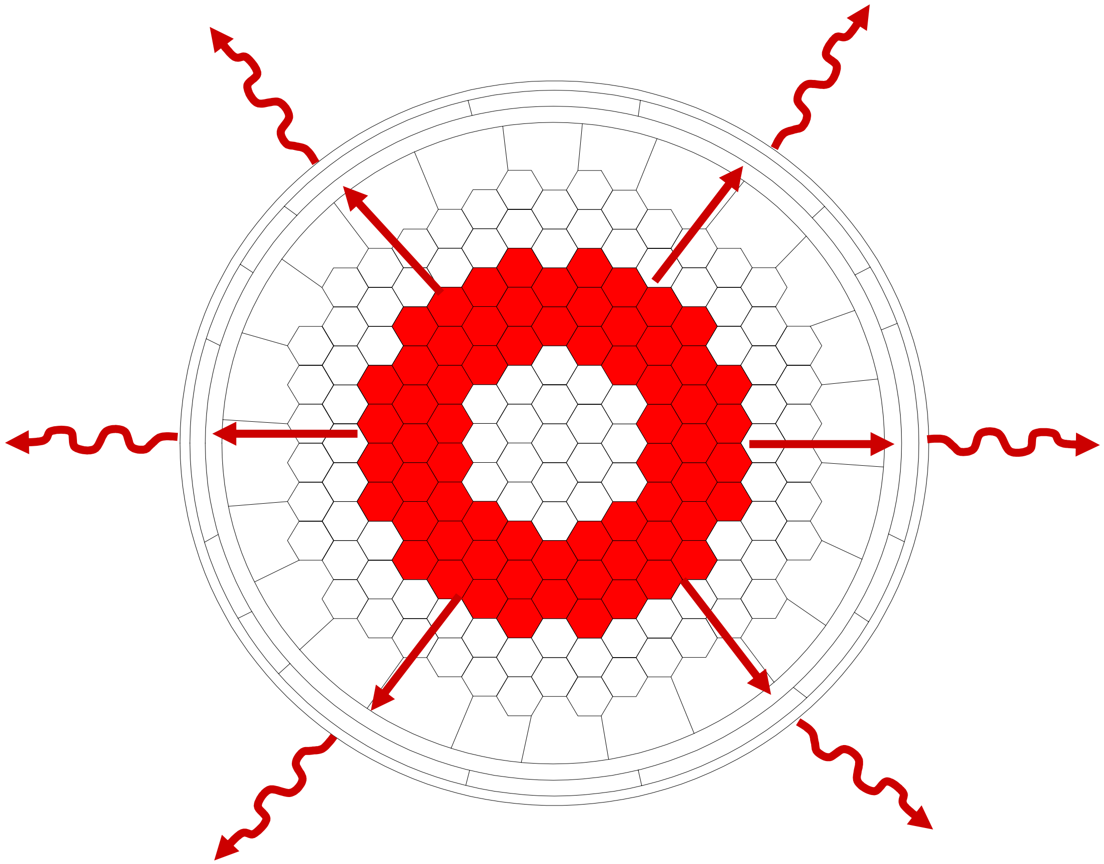
ANNULAR REACTOR CORE LIMITS FUEL TEMPERATURE DURING ACCIDENTS

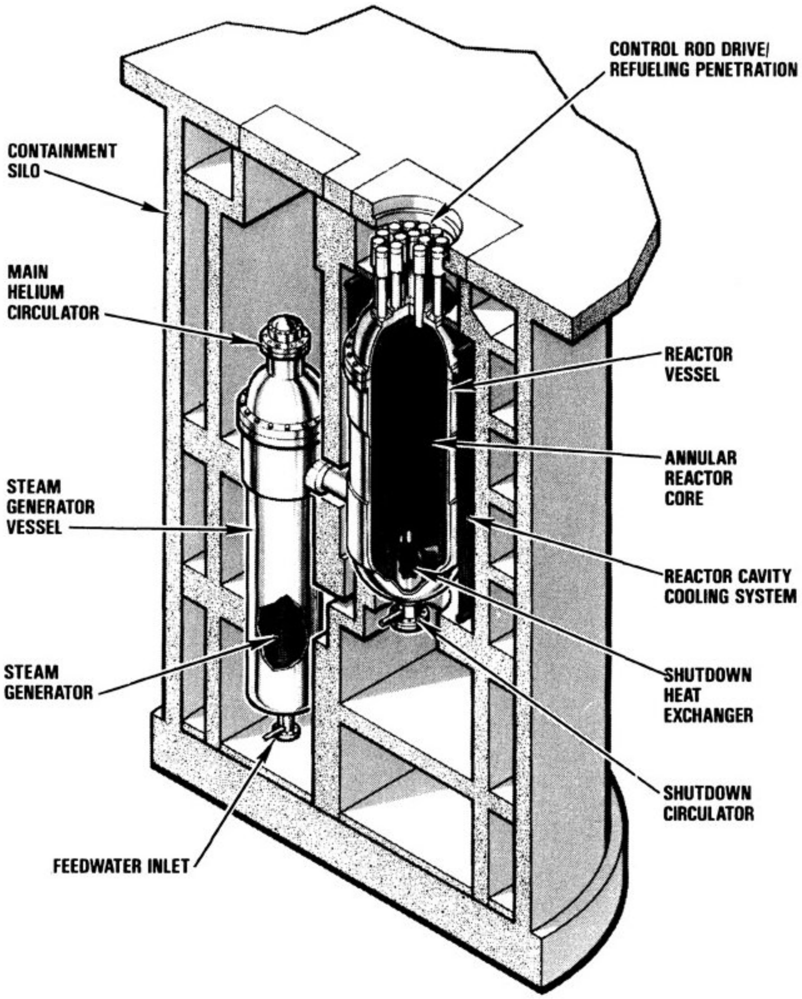


350 MWt
66 Columns
660 Elements

- **Decay heat conducts radially outward to steel pressure vessel boundary**
- **Steel pressure vessel radiates heat into reactor cavity**

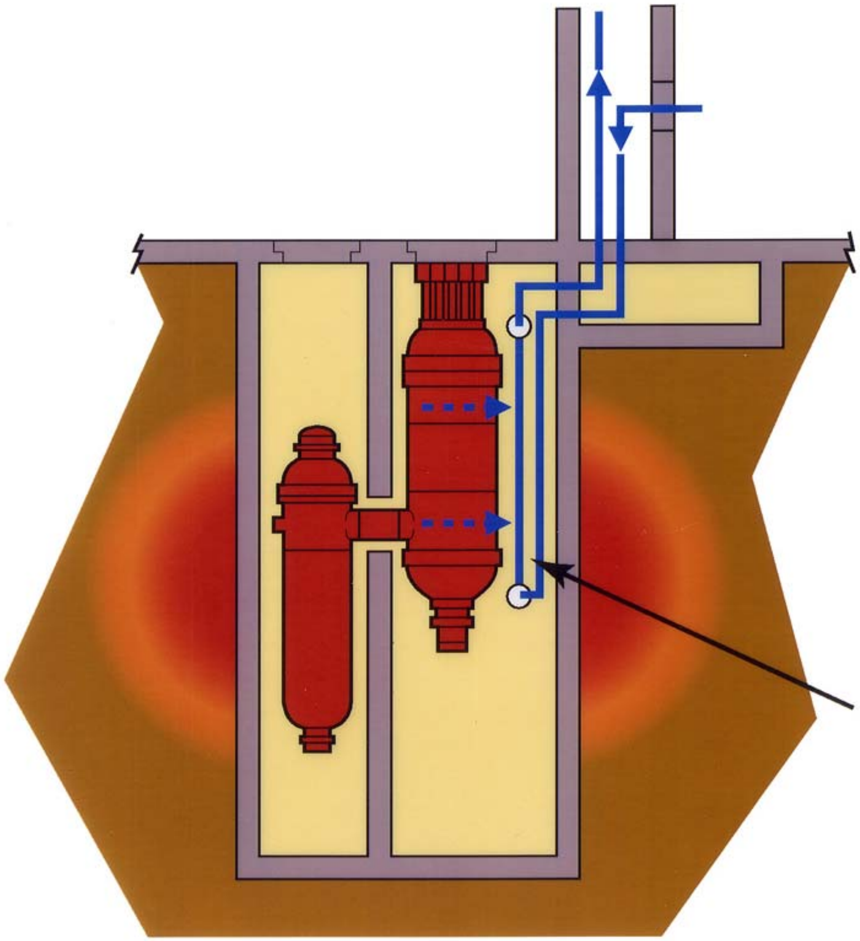
GENERAL ATOMICS

MHR MODULES LOCATED IN BELOW GRADE SILOS



- **Protection against natural disasters, missiles, terrorists**
- **Reduces seismic effects**
- **Cost-effective construction method by reduction of above grade structures**

PASSIVE REACTOR CAVITY COOLING SYSTEM REMOVES CORE DECAY HEAT FROM CAVITY

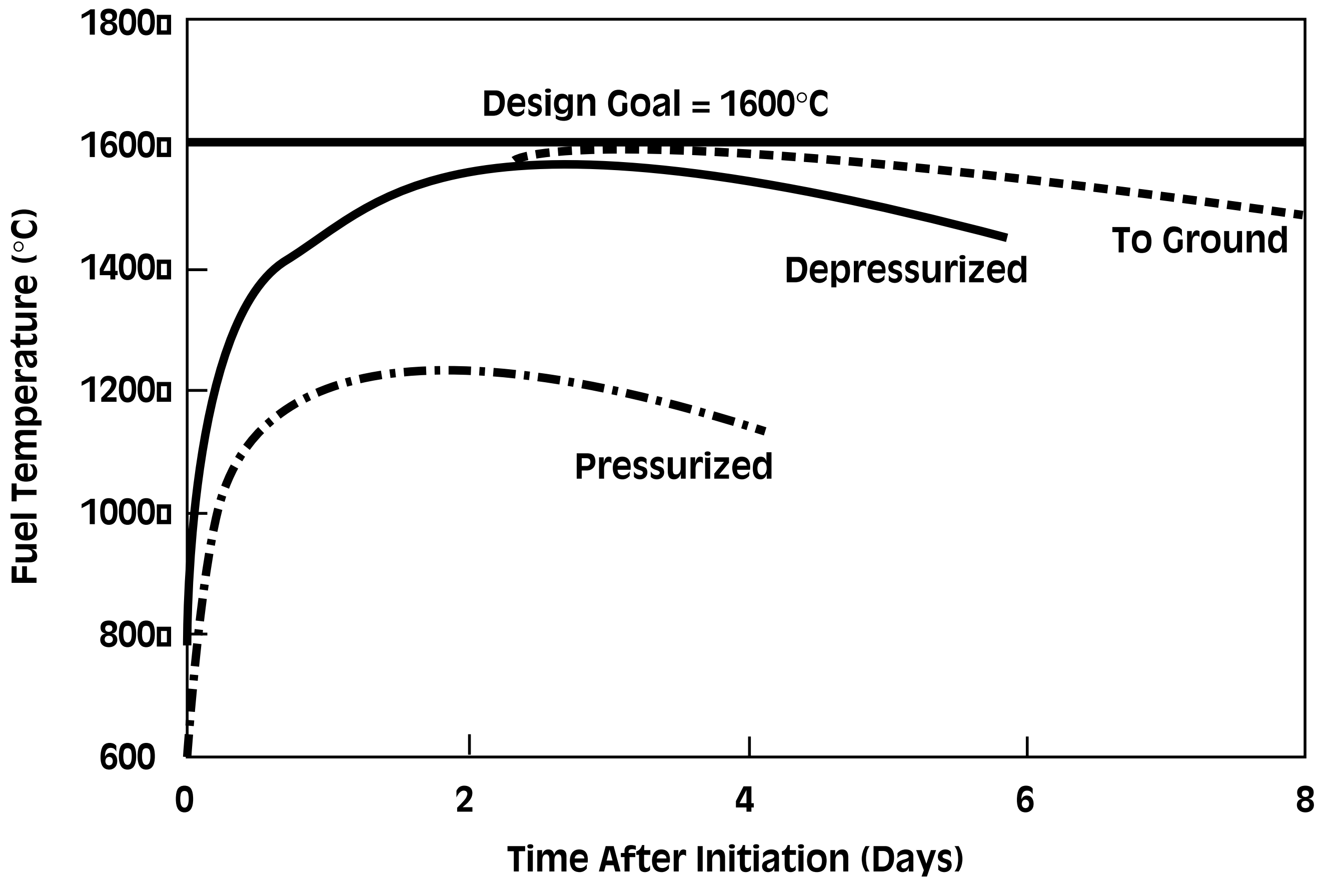


- Decay heat radiates from vessel to natural draft air cooling system
- No pumps or fans required
- Heat also conducts into ground

REACTOR CAVITY COOLING SYSTEM PANELS

GENERAL ATOMICS

FUEL TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING LOSS OF COOLING EVENTS



. . . PASSIVE DESIGN FEATURES ENSURE FUEL REMAINS BELOW 1600°C

GENERAL ATOMICS

MHR PROVIDES PASSIVE SAFETY BY DESIGN

- **Fission Products Retained in Coated Particles**
  - *High temperature stability materials*
  - *Refractory coated fuel*
  - *Graphite moderator*

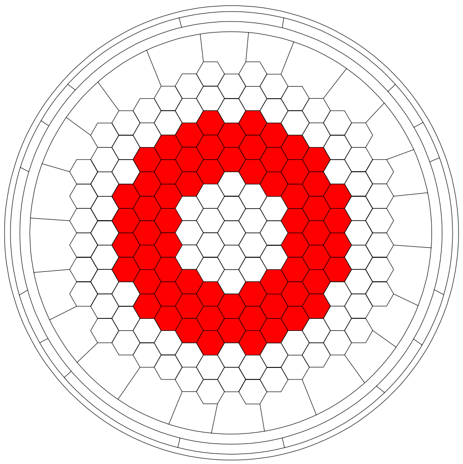
- **Worst case fuel temperature limited by design features**
  - *Low power density*
  - *Low thermal rating per module*
  - *Annular Core*
  - *Passive heat removal* ***….CORE CAN'T MELT***

- **Core Shuts Down Without Rod Motion**

GENERAL ATOMICS
GENERAL ATOMICS

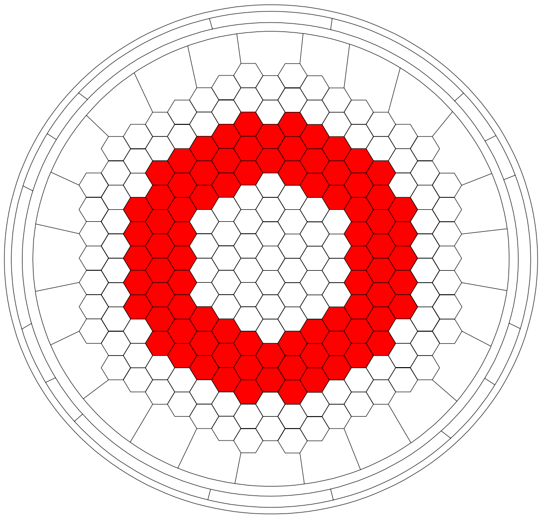
PASSIVELY SAFE MHR TECHNOLOGY
FLEXIBLE IN SIZE TO MEET DIFFERENT NEEDS

350 MW(t)



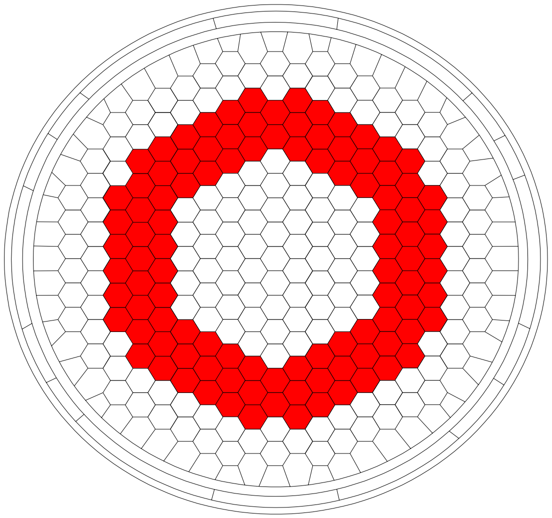
66 Columns
660 Elements

450 MW(t)



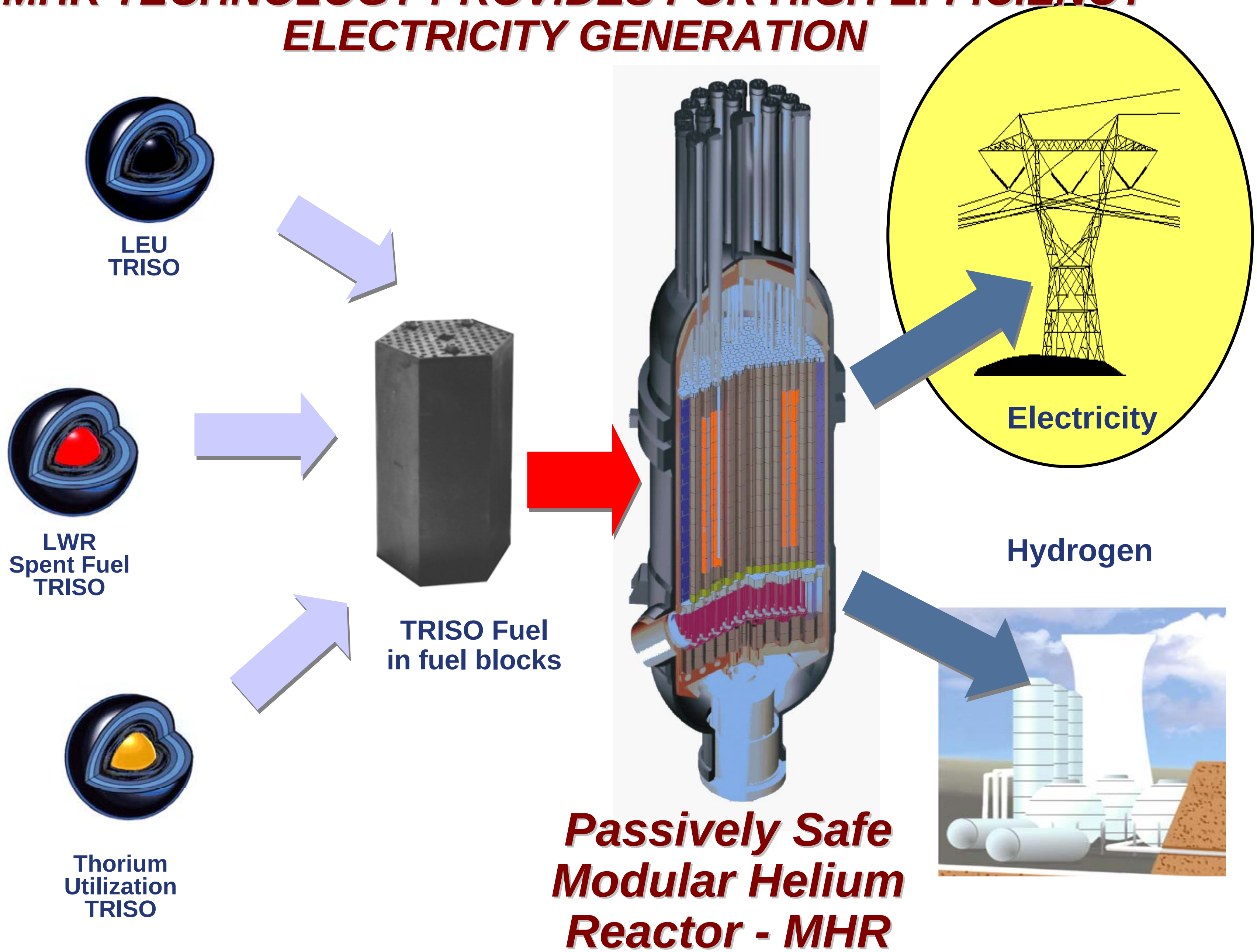
84 Columns
840 Elements

600 MW(t)



102 Columns
1020 Elements

MHR TECHNOLOGY PROVIDES FOR HIGH EFFICIENCY ELECTRICITY GENERATION



SPECTRUM OF PASSIVELY SAFE MHR PLANTS DEVELOPED FOR ELECTRICITY GENERATION

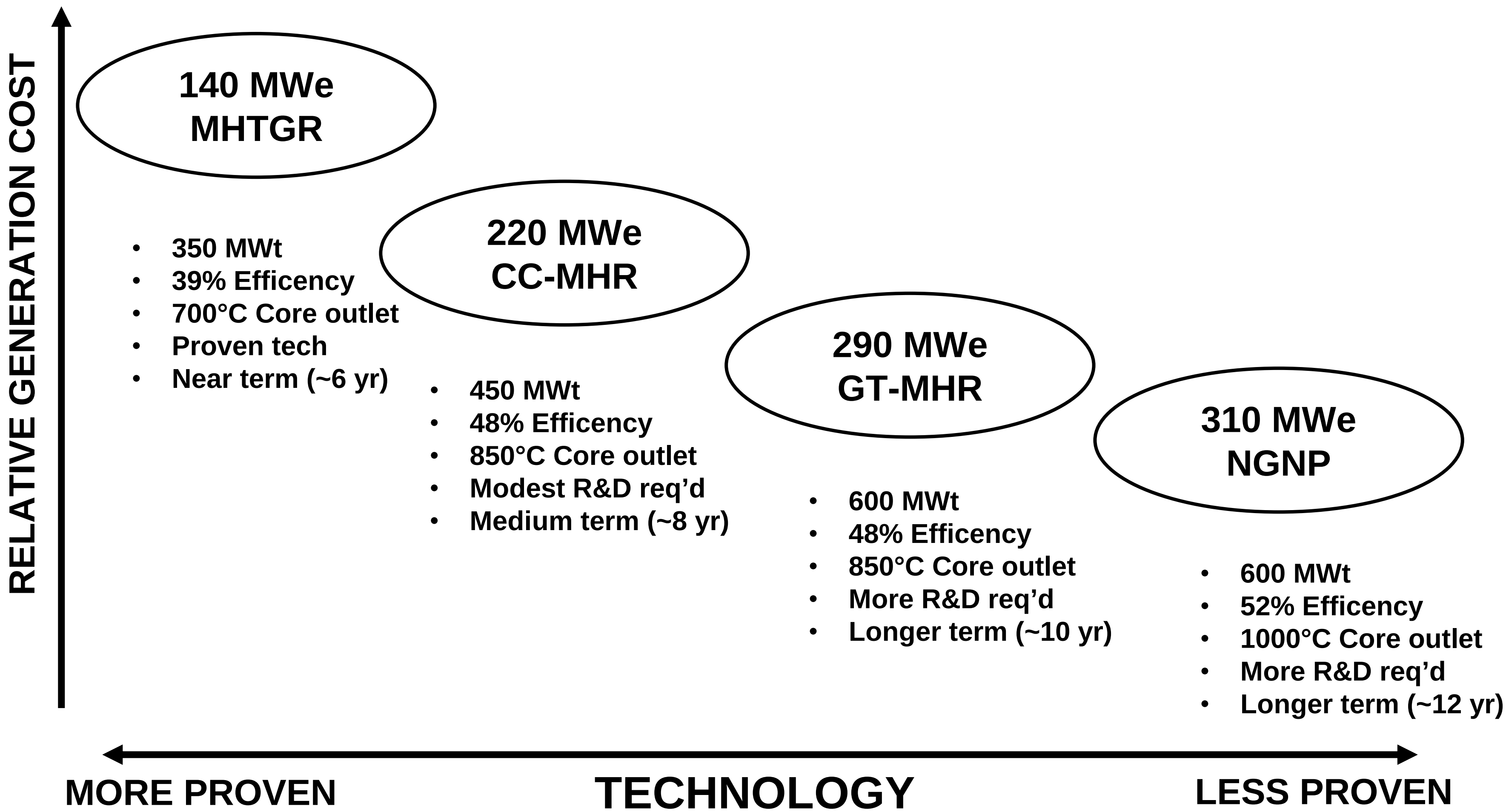
1. **140 MWe Steam Cycle 350 MWt Modular High Temperature Gas Reactor (MHTGR) - 1st passively safe MHR developed**

2. **220 MWe Combined Cycle 450 MWt MHR (CC-MHR) - Extrapolation of 350 MWt MHR to higher temp & coupled with modified combined cycle plant for higher efficency**

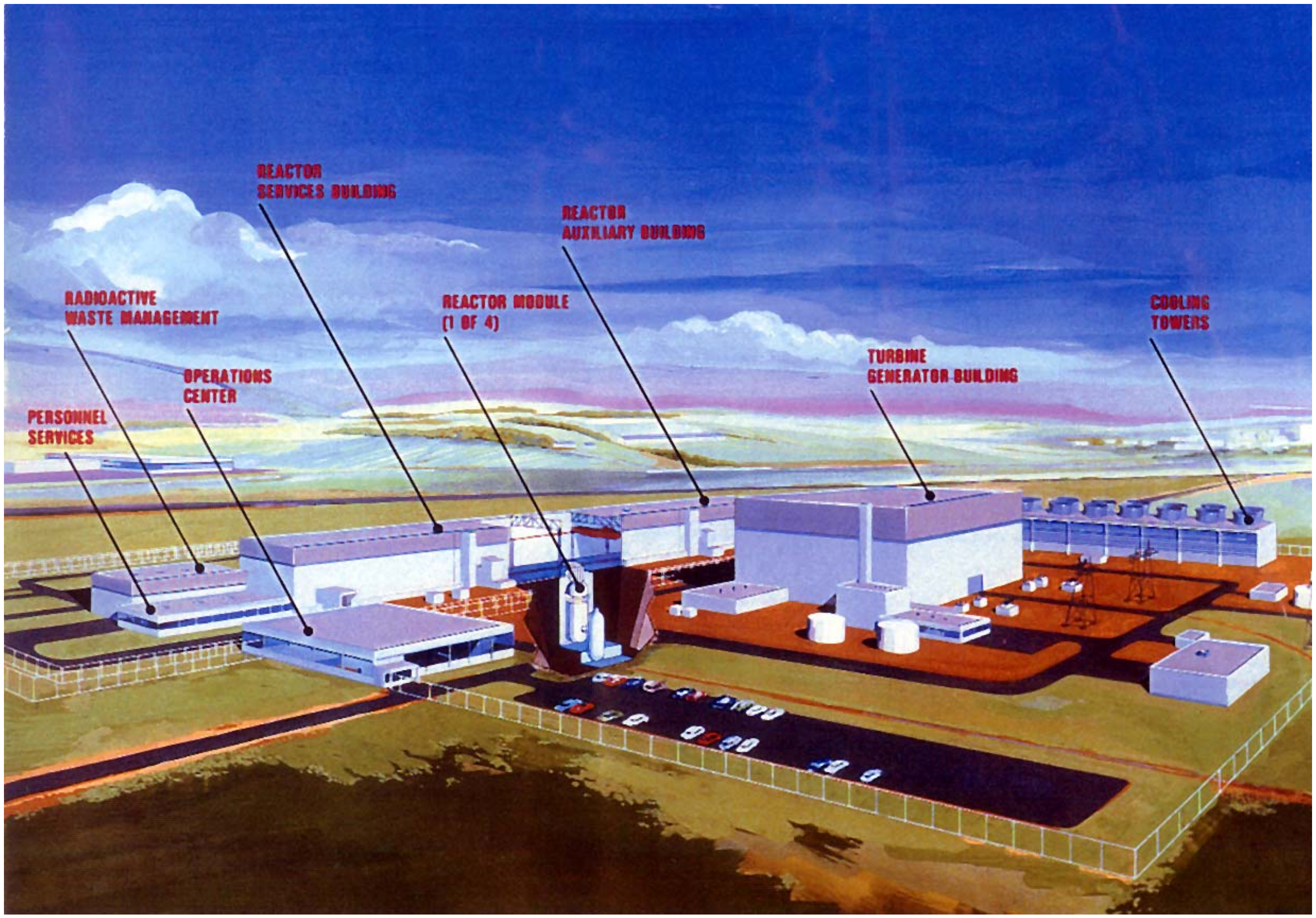
3. **290 MWe Gas Turbine - Modular Helium Reactor (GT-MHR) - 600 MWt MHR with direct Brayton cycle**

4. **310 MWe GT-MHR with 1000°C core outlet temperature Brayton cycle - Next Generation Nuclear Plant (NGNP)**

MHR ELECTRIC GENERATION PLANTS RANGE FROM NEAR TERM TO LONGER TERM

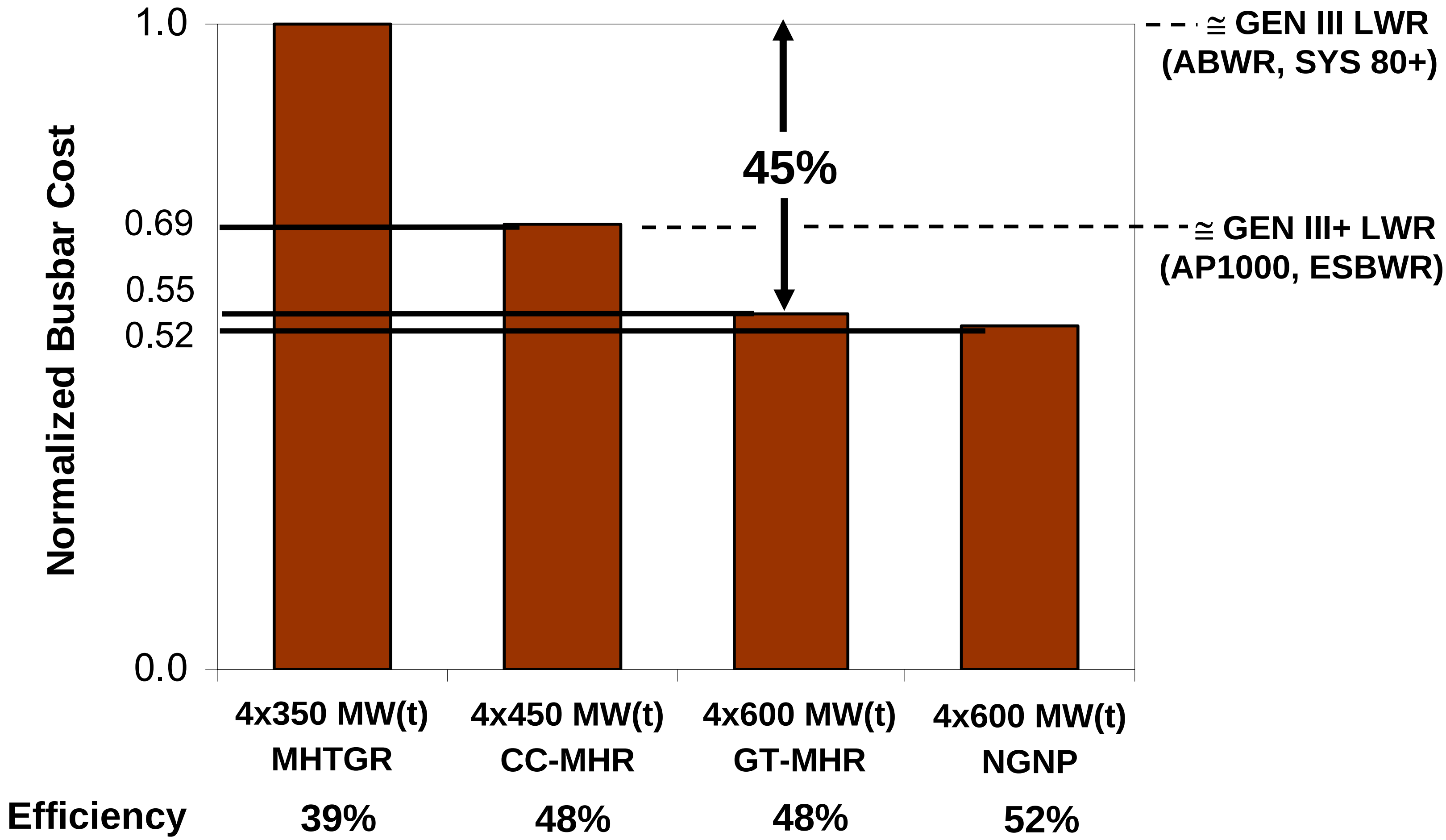


REFERENCE PLANT DESIGNS COMPRISE FOUR MODULES



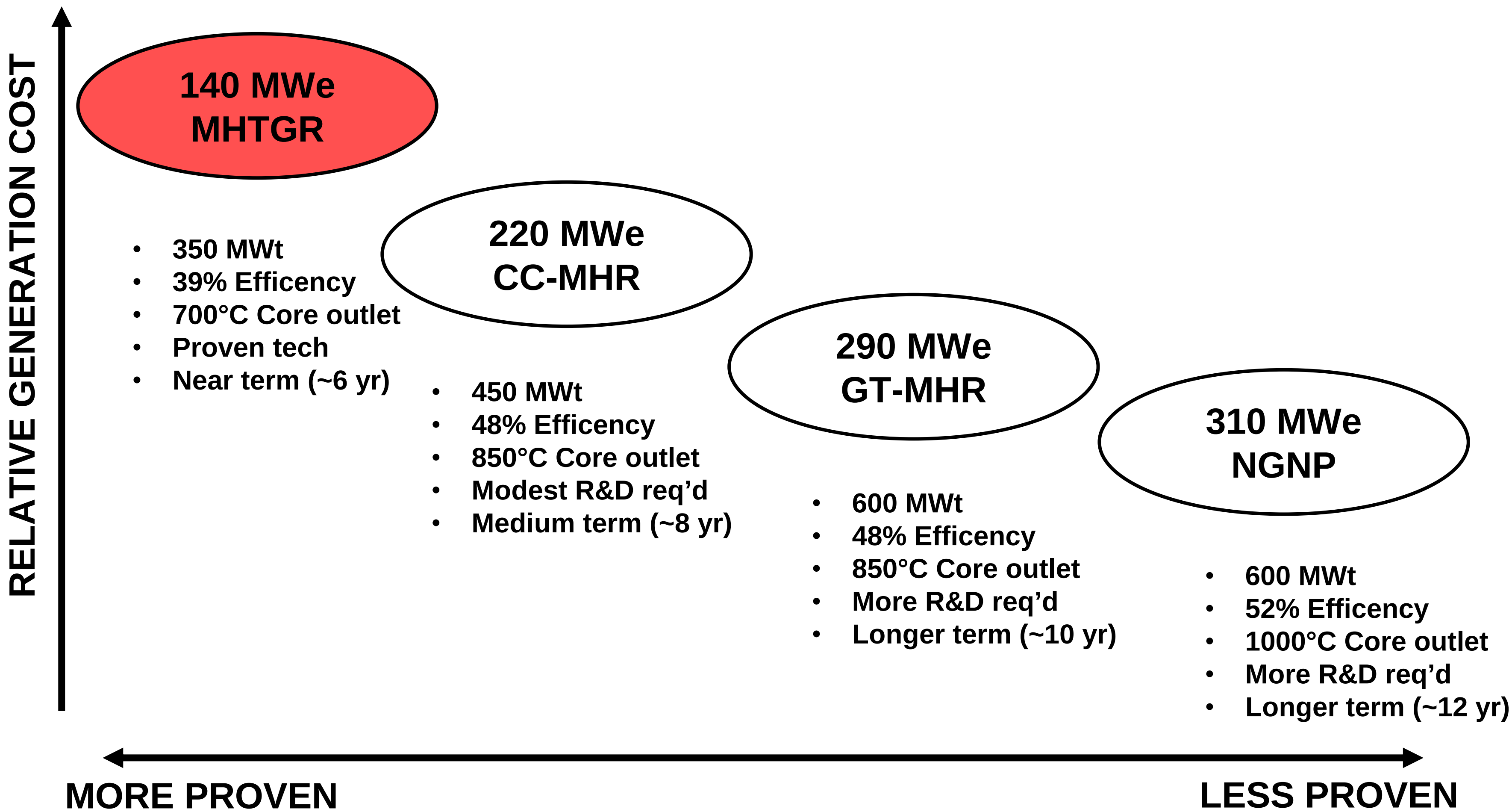
LARGER MHR SIZES & ADVANCED CONVERSION TECHNOLOGIES REDUCE POWER GENERATION COSTS

…But, 350 MWt MHTGR based on most proven technology



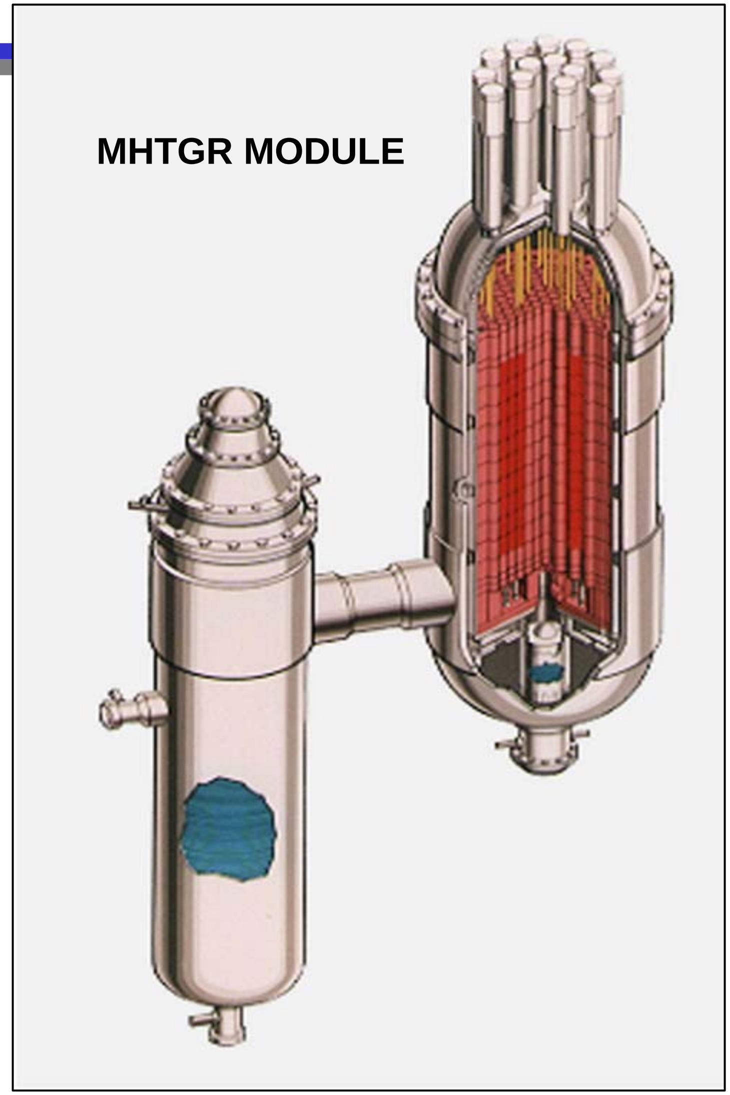
GENERAL ATOMICS

MHTGR FIRST MHR ELECTRIC GENERATION PLANT DEVELOPED

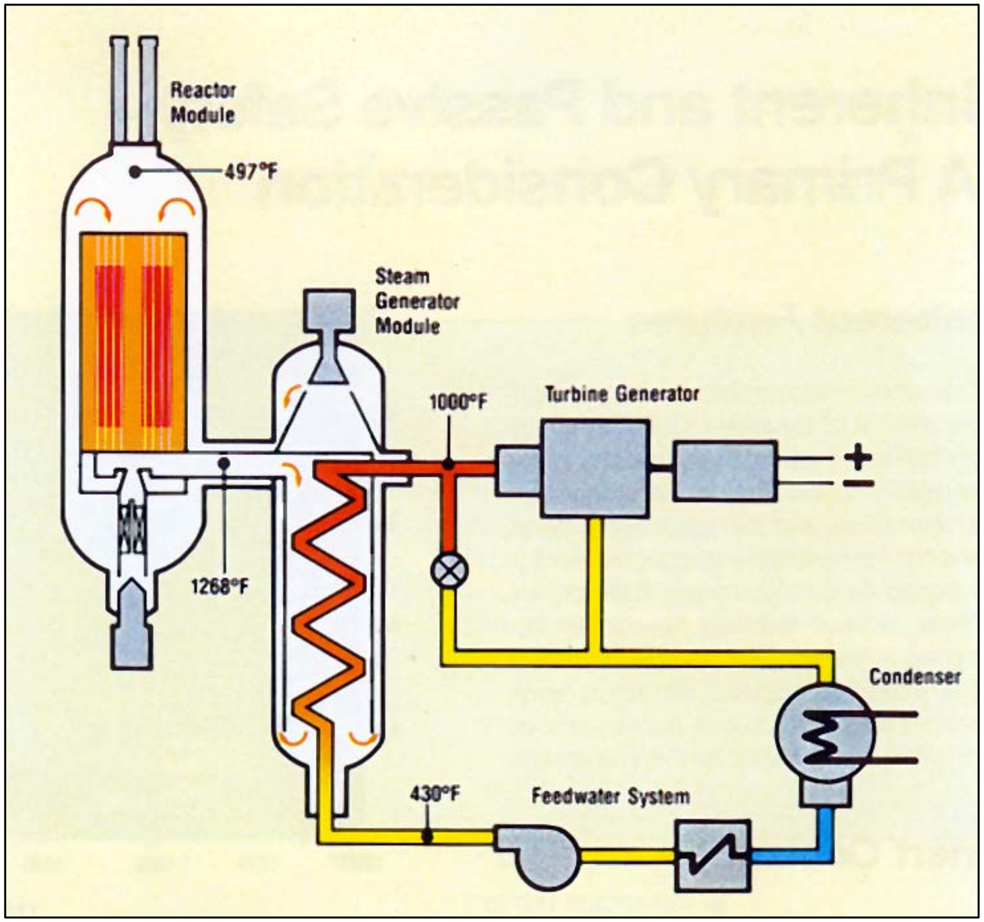


MHTGR MODULE COMBINES MELTDOWN-PROOF REACTOR & HIGH TEMPERATURE STEAM SUPPLY FOR HIGH EFFICENCY ELECTRICITY GENERATION

POWER LEVEL 350 MWt; 140 MWe



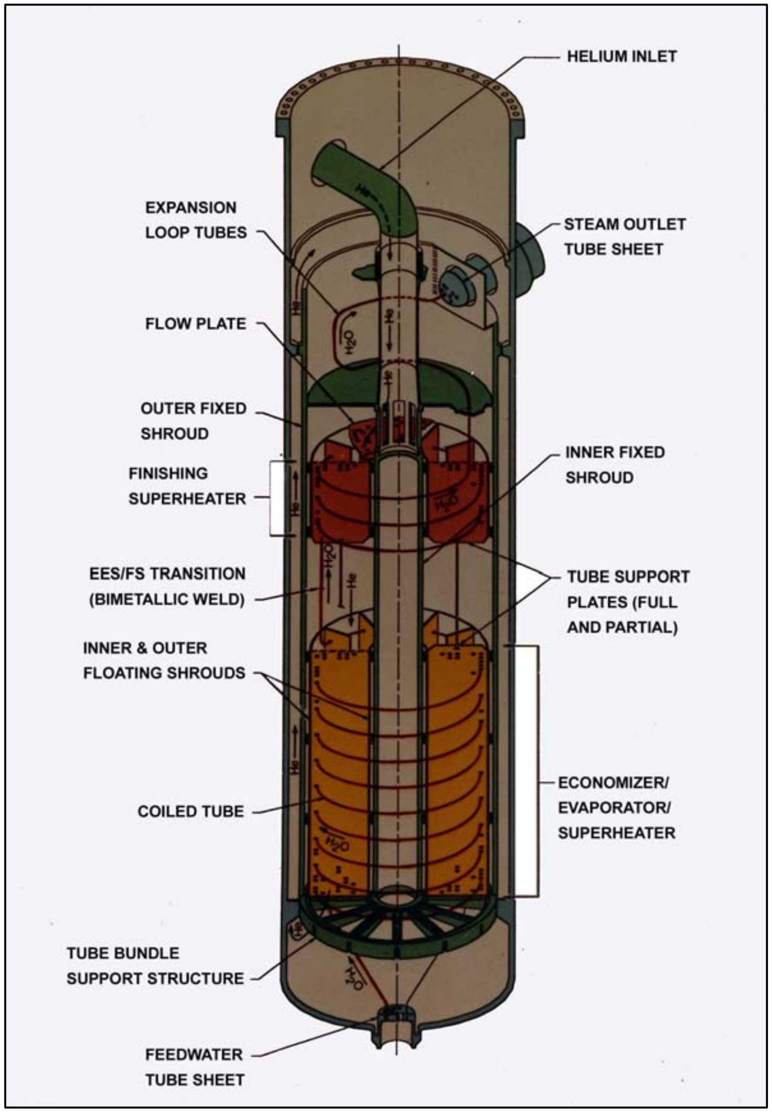
MHTGR GENERATES STEAM AT 1000°F (540°C) AND 2500 PSI (17 Mpa)



….steam quality equivalent to modern fossil-fired steam power plants

GENERAL ATOMICS

MHTGR STEAM GENERATOR IS CLOSELY RELATED TO FORT ST. VRAIN & THTR



- **Converts reactor heat to superheated steam**
- **Helically coiled once-thru boiler design, boiling inside tubes**
- **Tubes are part of primary pressure boundary**
- **Size consistent with nuclear component experience**
- **Design simplified relative to prior HTGR designs**
- **Service conditions comparable to prior gas-cooled and fossil-fired experience**
- **Code approved materials**

MHTGR IS A NEAR TERM ADVANCED NUCLEAR POWER SYSTEM

- **MHTGR is based on proven technology**
  - **No R&D required**
  - **Detail preliminary design completed including a preliminary safety review by the US NRC**
  - **Only detail engineering for construction remains to be done**
- **MHTGR supplies high grade steam equivalent to modern fossil fired boiler plants for high efficiency electricity generation**
- **MHTGR passively safe by design**
- **First MHTGR could be deployed in about 6 years**
- **MHTGR plants can be configured to use one or more modules**
- **Module size 350 MWt or 450 MWt**

COMBINED CYCLE MHR BUILDS ON RECENT TECHNOLOGY DEVELOPMENTS

RELATIVE GENERATION COST

140 MWe MHTGR

- 350 MWt
- 39% Efficency
- 700°C Core outlet
- Proven tech
- Near term (~6 yr)

220 MWe CC-MHR

- 450 MWt
- 48% Efficency
- 850°C Core outlet
- Modest R&D req'd
- Medium term (~8 yr)

290 MWe GT-MHR

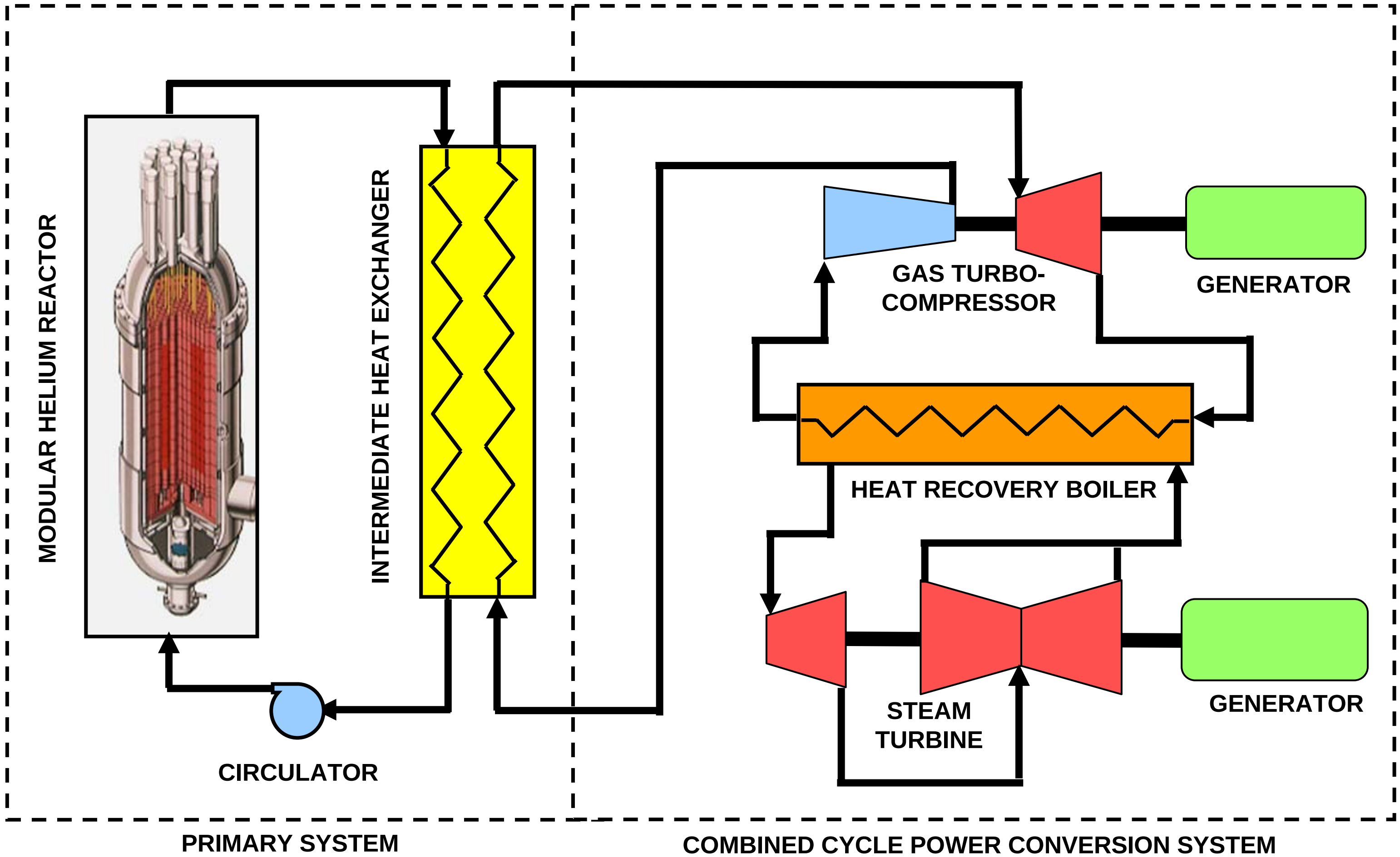
- 600 MWt
- 48% Efficency
- 850°C Core outlet
- More R&D req'd
- Longer term (~10 yr)

310 MWe NGNP

- 600 MWt
- 52% Efficency
- 1000°C Core outlet
- More R&D req'd
- Longer term (~12 yr)

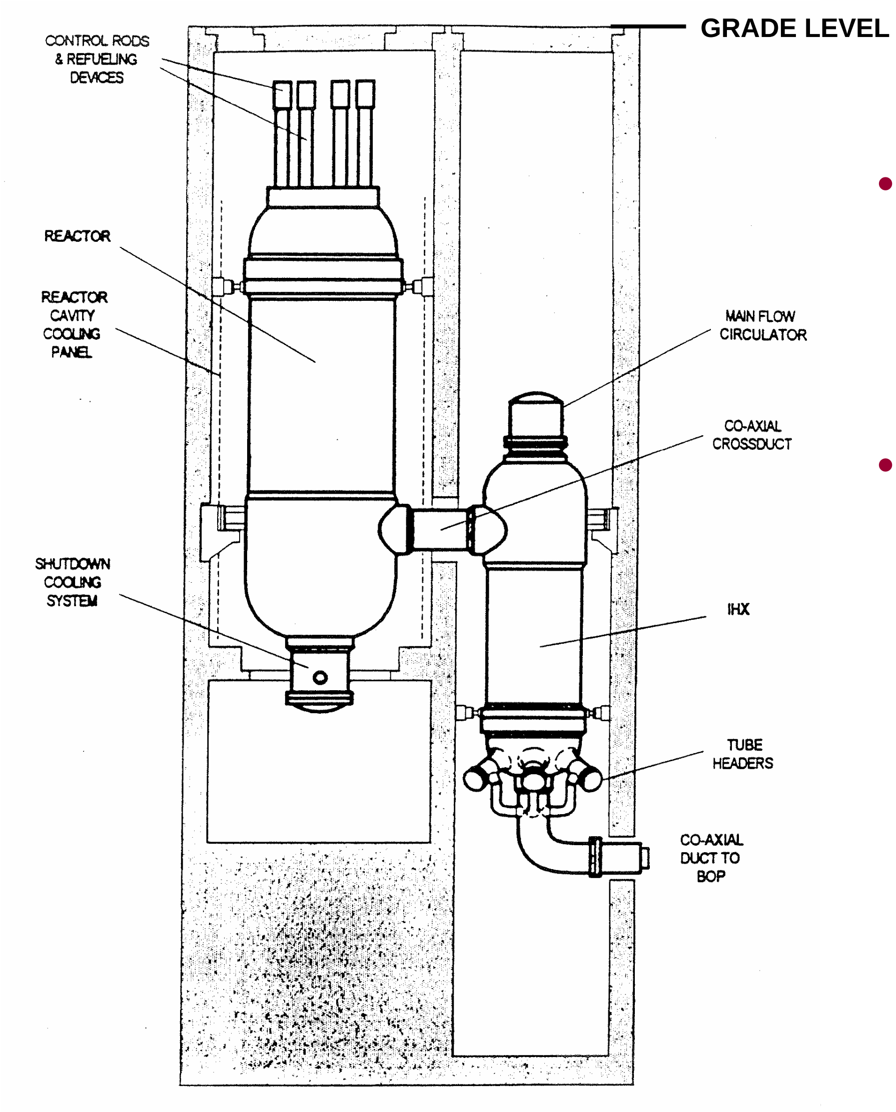
MORE PROVEN — TECHNOLOGY — LESS PROVEN

CC-MHR PLANT COUPLES AN MHR WITH A COMBINED CYCLE POWER CONVERSION SYSTEM



GENERAL ATOMICS

CC-MHR PRIMARY SYSTEM LOCATED IN BELOW GRADE SILO, SAME AS MHTGR



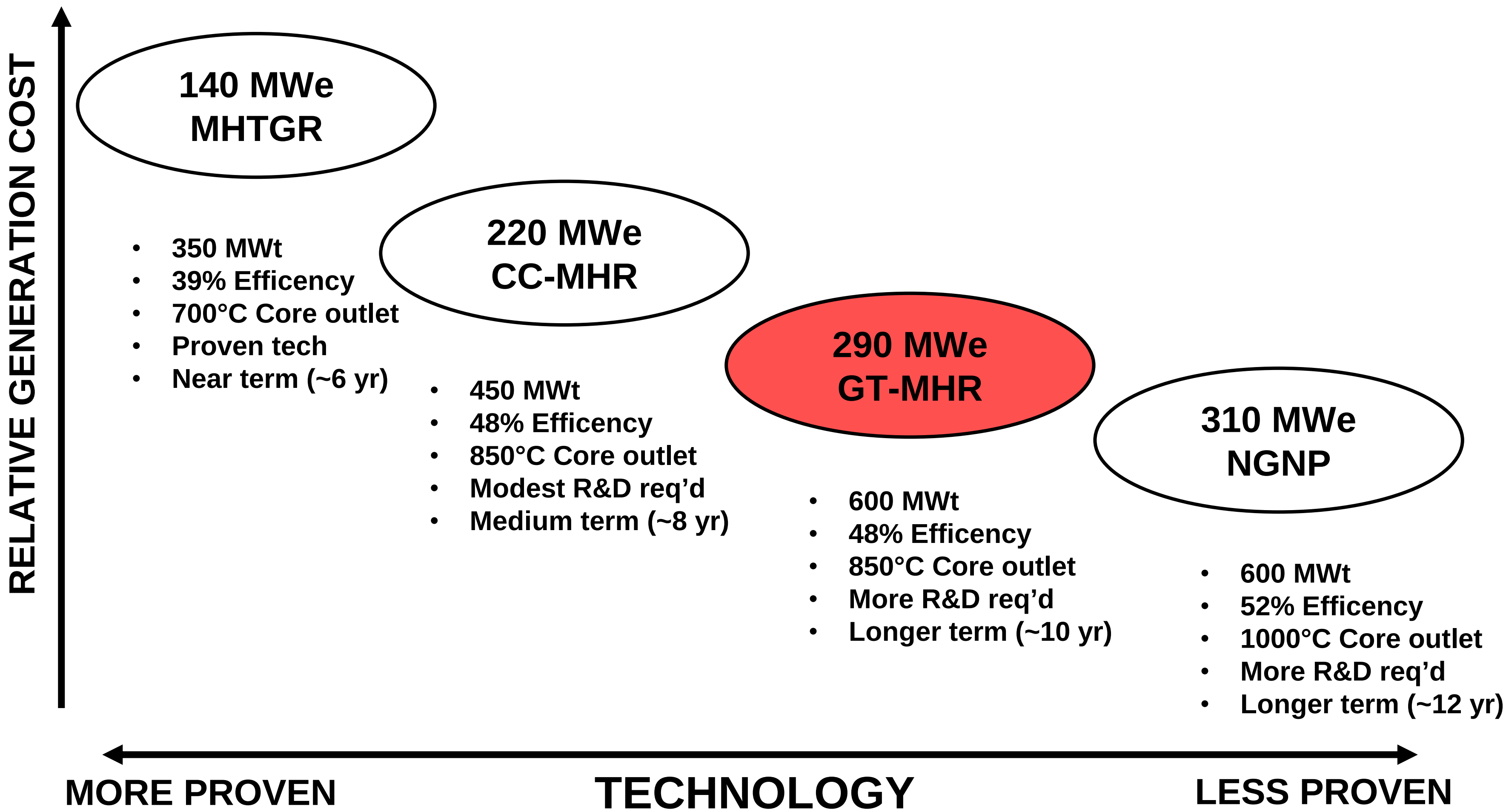
- **CC-MHR retains same passive safety characteristics as MHTGR**
- **Natural circulation reactor cavity cooling system incorporated same as MHTGR**

GENERAL ATOMICS

CC-MHR IS AN ADVANCED MHR PLANT THAT COULD BE DEPLOYED IN THE MID TERM

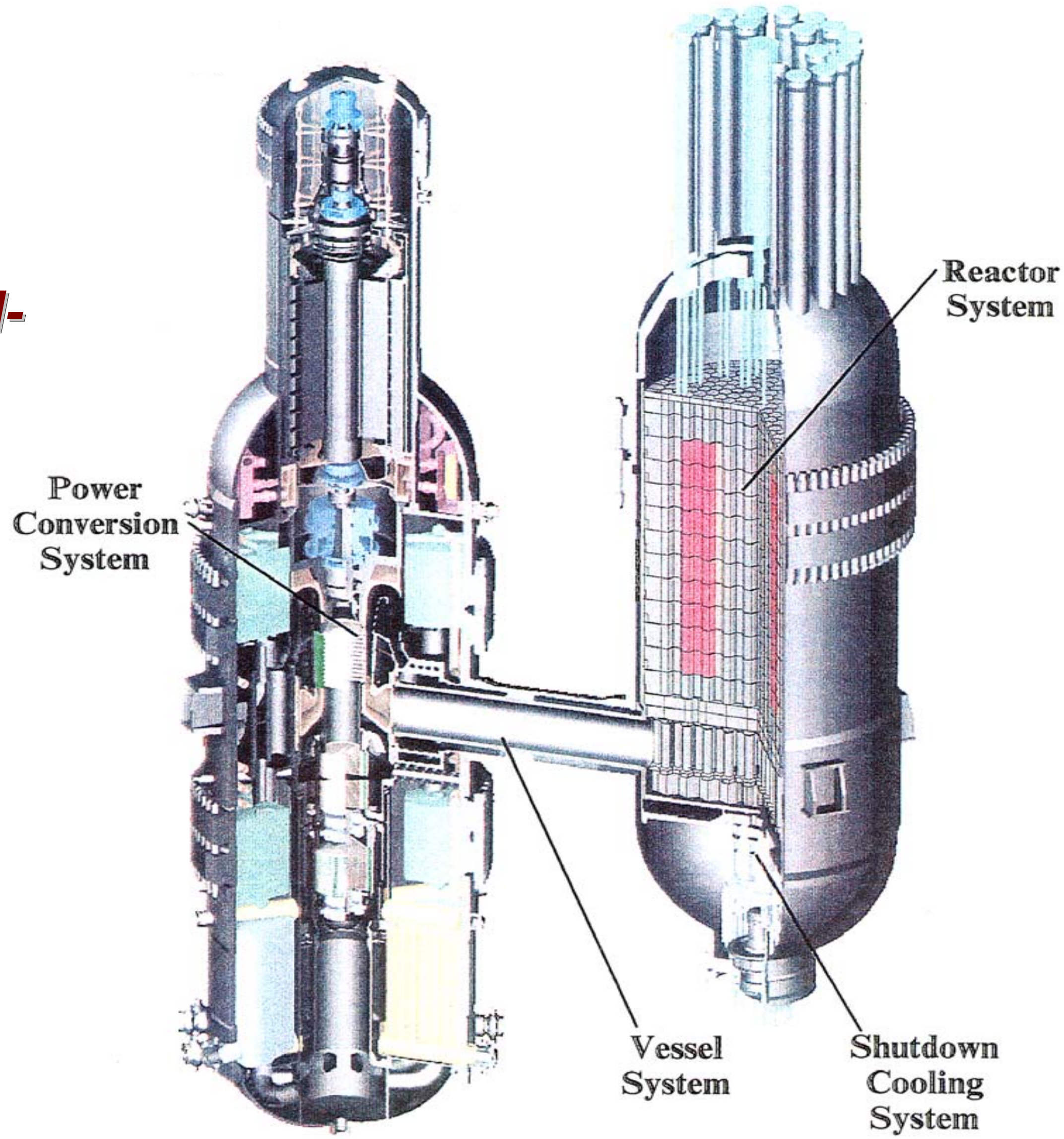
- CC-MHR has substantial proven technology bases
  - Limited R&D required on IHX and gas turbine
  - Much of the MHTGR detail preliminary design applicable, including the preliminary safety review by the US NRC
  - Detail engineering for construction remains to be done
- No new R&D for MHR for increased core outlet temperature
  - Within envelop proven by HTTR
- CC-MHR makes use of the proven combined cycle power conversion system for high conversion efficiency
- CC-MHR passively safe by design
- First CC-MHR could be deployed in about 8 years

GAS TURBINE MHR DEVELOPED FOR IMPROVED ECONOMICS

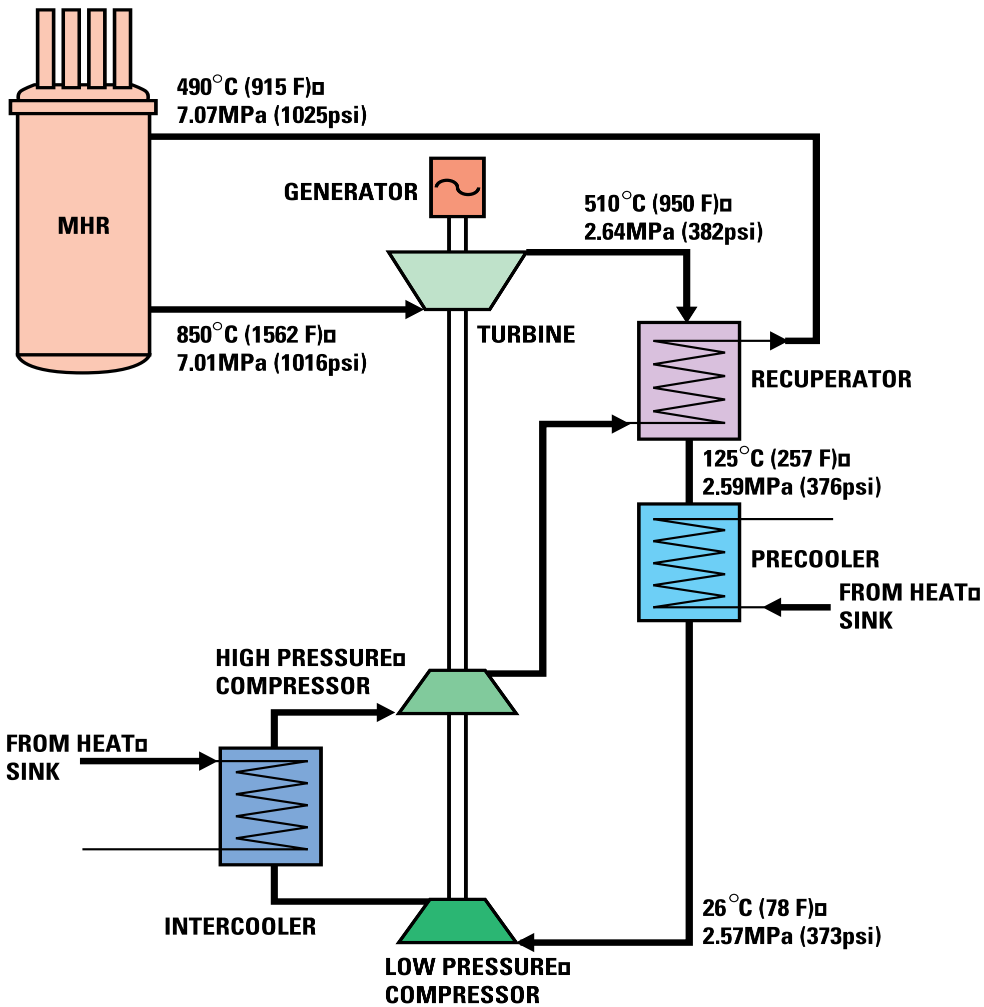


GT-MHR MODULE COMBINES MELTDOWN-PROOF ADVANCED REACTOR
&
HIGH EFFICENCY GAS TURBINE POWER CONVERSION SYSTEM

POWER LEVEL
600 MWt; 290 MWe

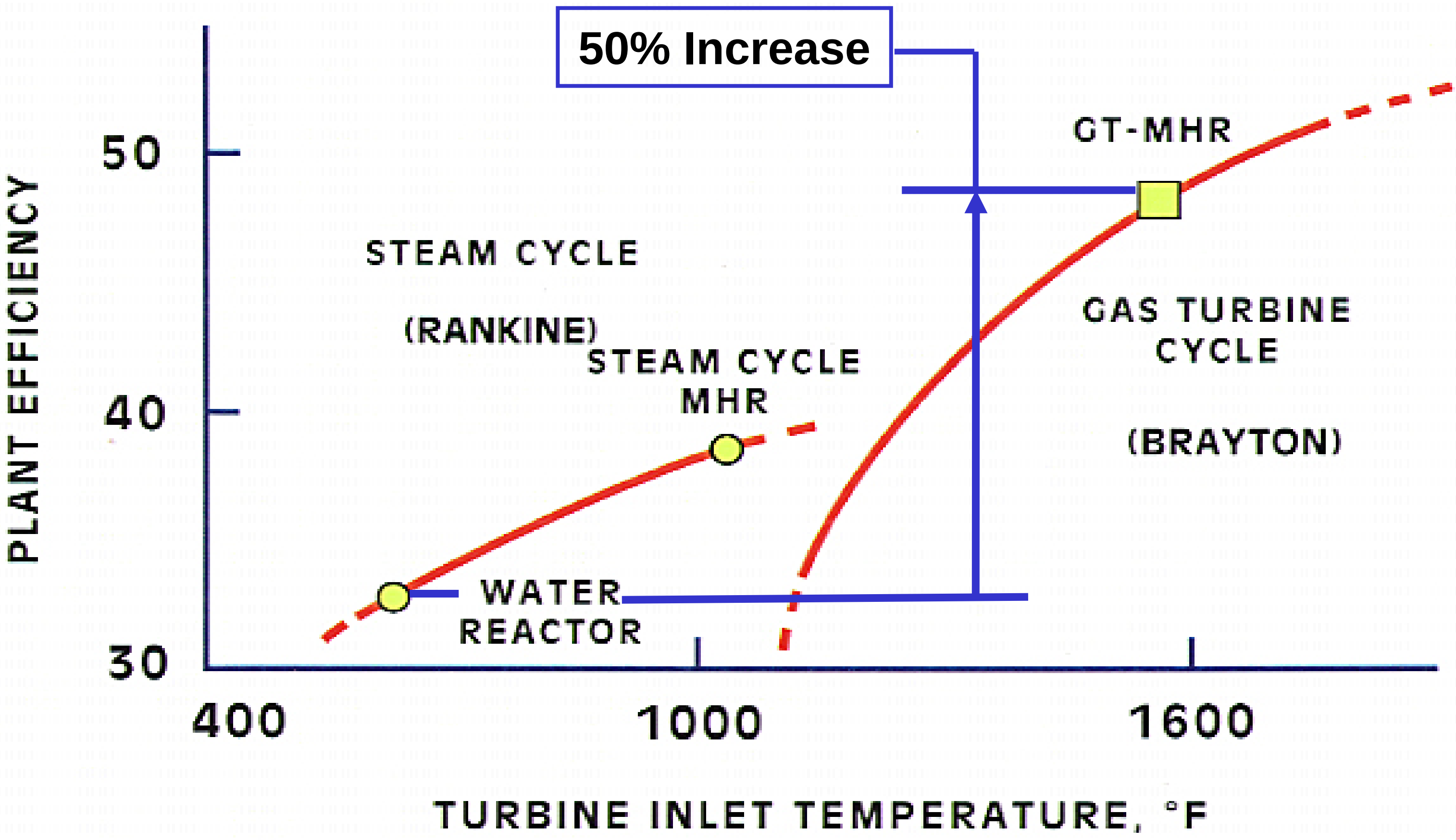


GT-MHR USES DIRECT BRAYTON CYCLE POWER CONVERSION SYSTEM



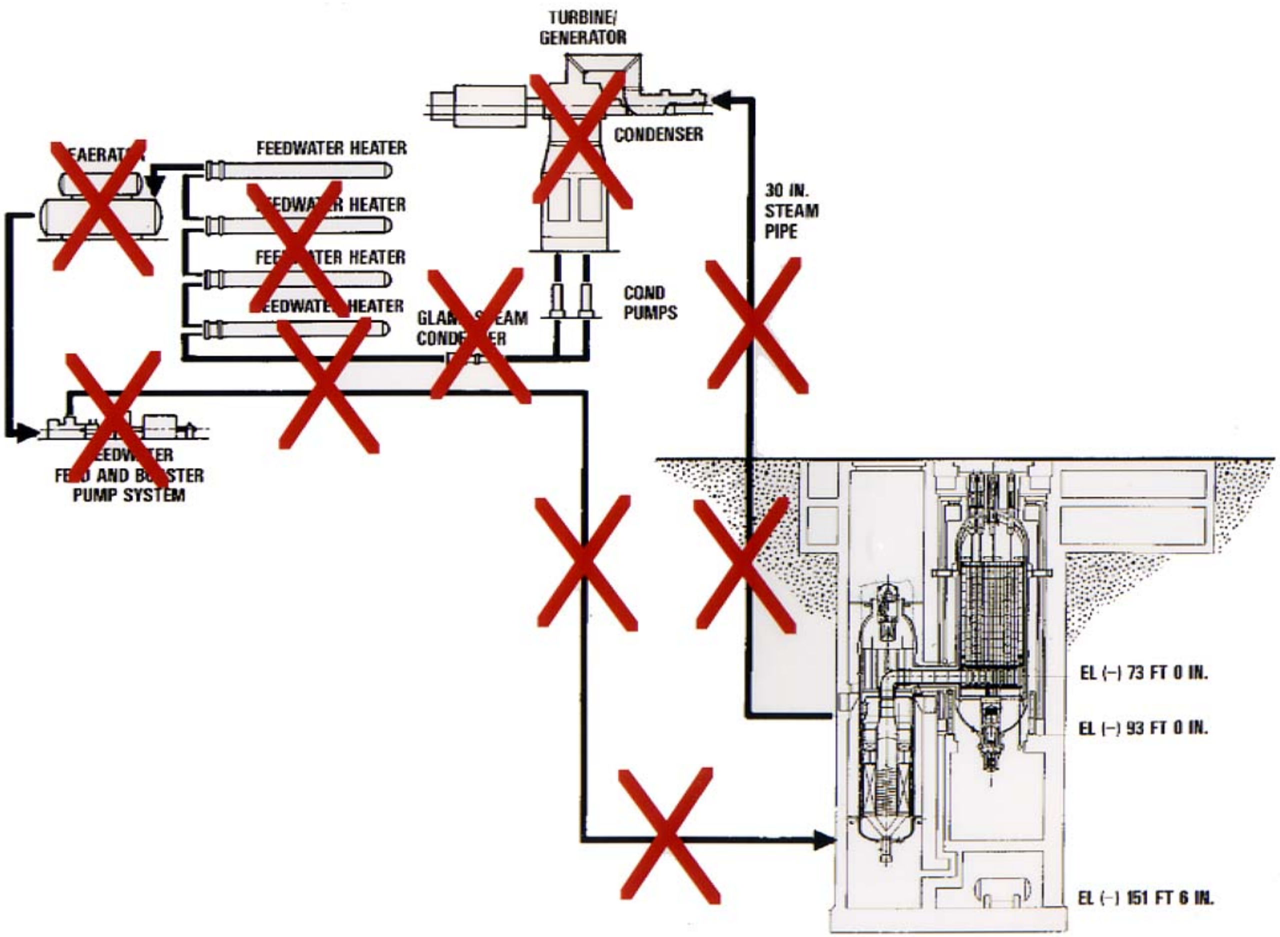
GENERAL ATOMICS

HIGH TEMPERATURE GAS REACTORS HAVE UNIQUE ABILITY TO USE BRAYTON CYCLE



GENERAL ATOMICS

DIRECT CYCLE ELIMINATES MANY COMPLICATED AND EXPENSIVE COMPONENTS



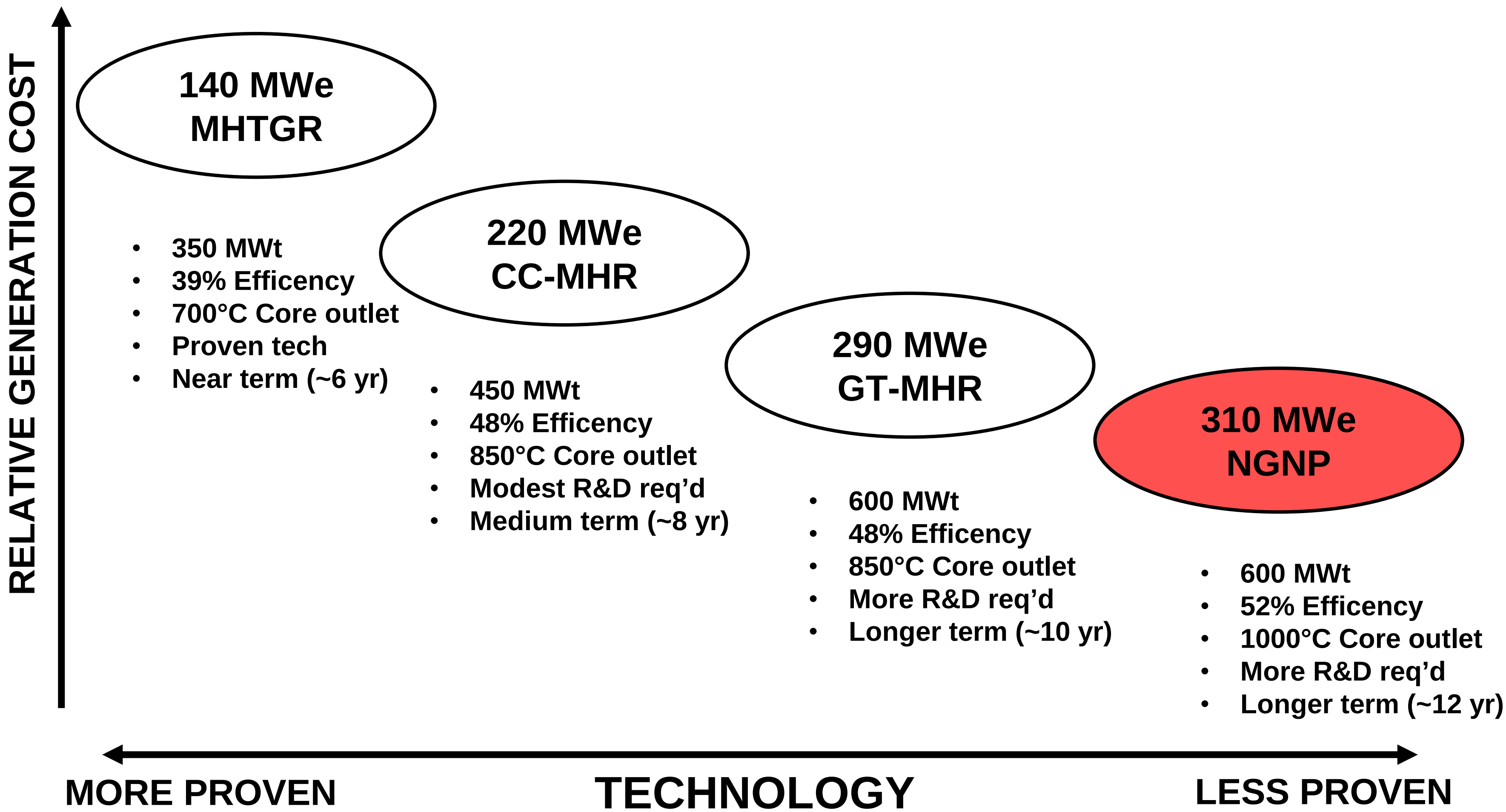
. . . REDUCES O&M / IMPROVES PLANT AVAILABILITY

GENERAL ATOMICS

R&D REQUIREMENTS LENGTHEN GT-MHR COMMERCIAL DEPLOYMENT SCHEDULE

- **Preliminary design of reactor module has been completed in Russia (to Russian codes & stds)**
- **Integrated power conversion unit (PCU) is longest term & most costly development item**
  - **Full scale turbomachine test planned**
  - **Tests of several PCU sub-components in process**
- **Second most critical path item is regulatory review and licensing (not yet started)**
- **First commercial GT-MHR plant deployable in about 10 years (based on four year construction schedule for 1st module) and assuming a prototype not required**

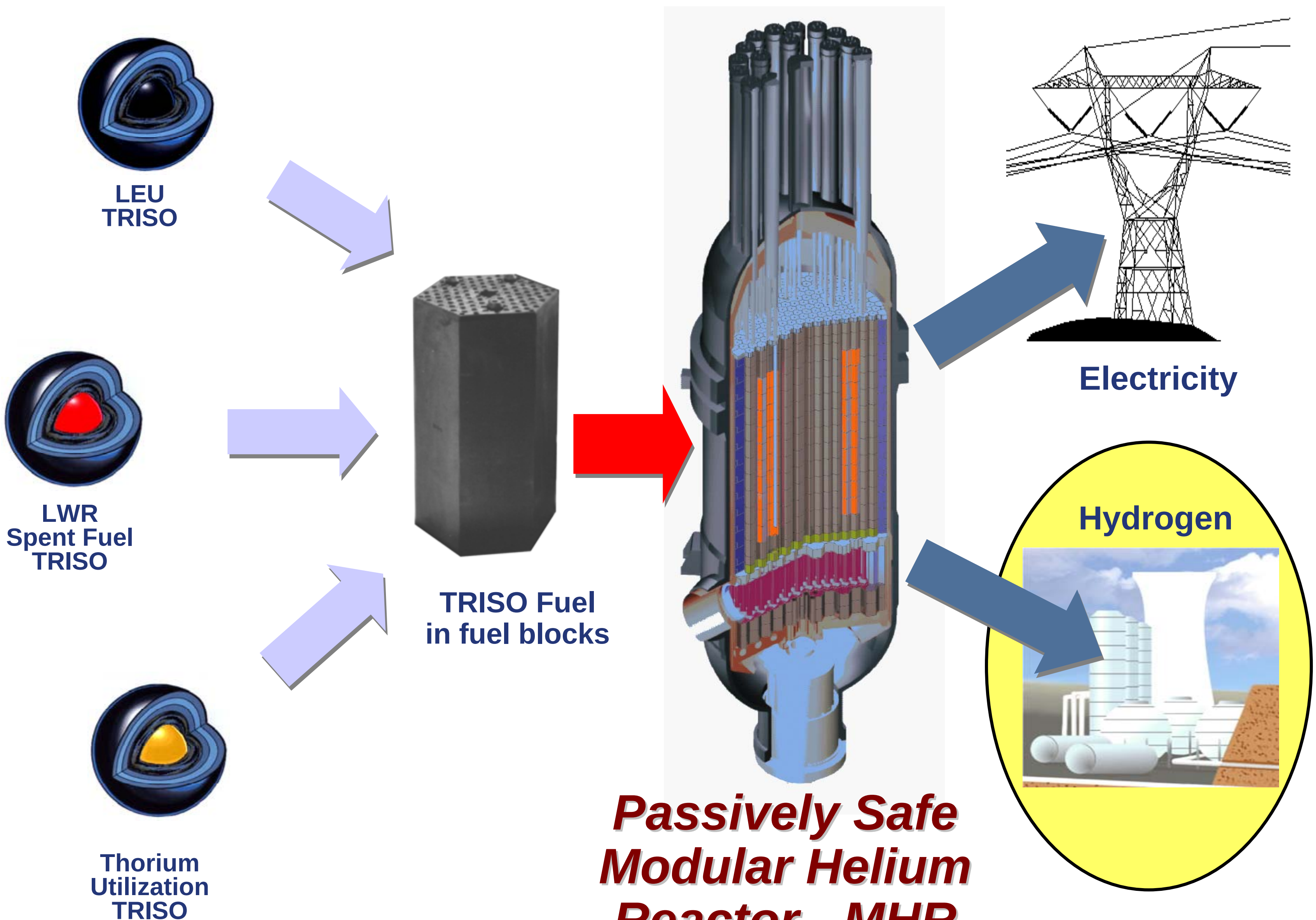
NGNP IS FIRST GENERATION IV PLANT TO BE DEMONSTRATED BY US DOE



NGNP MISSION OBJECTIVES IDENTIFIED BY US DOE

- Demonstrate a full-scale prototype NGNP by about 2017
- Demonstrate high-temperature Brayton Cycle electric power production at full scale
- Demonstrate nuclear-assisted production of hydrogen (using about 10 % of the heat)
- Demonstrate by test the exceptional safety capabilities of the advanced gas cooled reactors
- Obtain an NRC License to construct and operate the NGNP, to provide a basis for future performance-based, risk-informed licensing
- Support the development, testing, and prototyping of hydrogen infrastructures

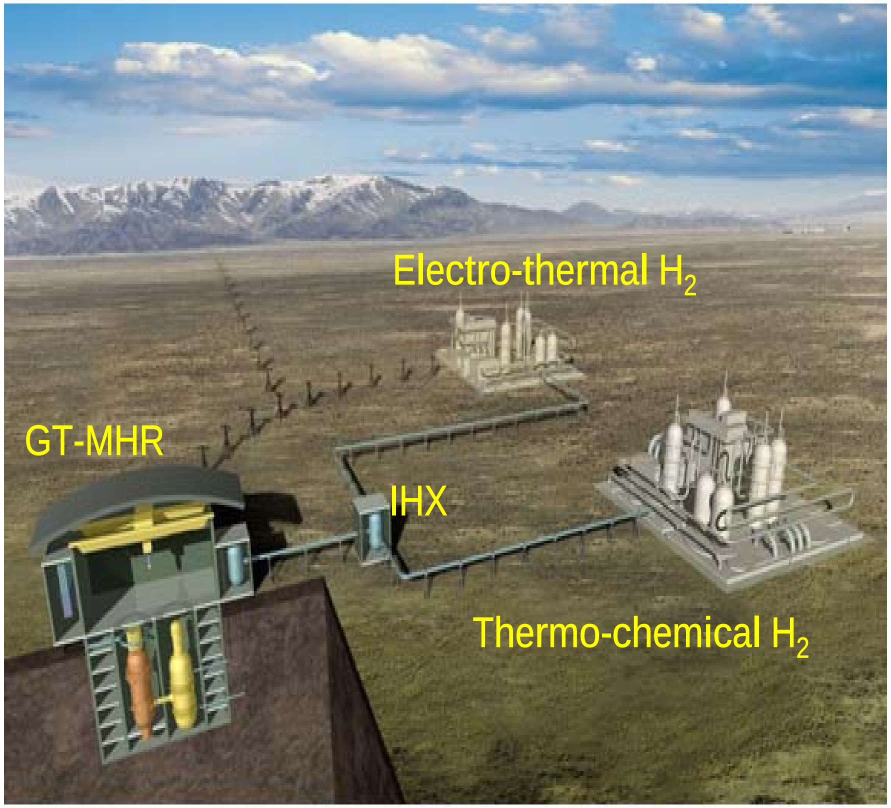
HYDROGEN PRODUCTION IS KEY NGNP OBJECTIVE



SEVERAL WAYS POSSIBLE TO PRODUCE HYDROGEN USING NUCLEAR ENERGY

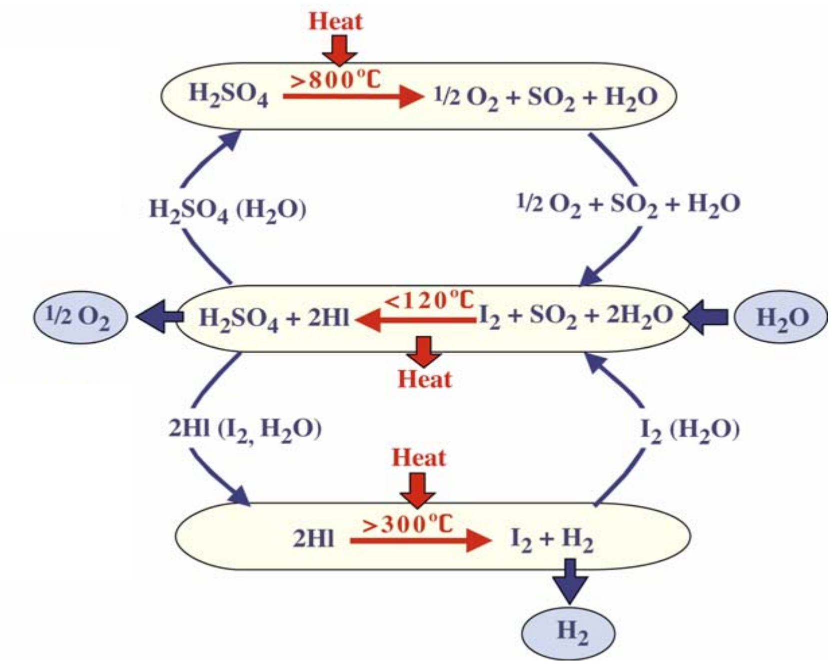
- **Electric power generation ➔ Electrolysis**
  - **Overall efficiency ~24% (LWR), ~36% (Hi T Reactors) (efficiency of electric power generation x efficiency of electrolysis)**
- **High temperature heat ➔ Thermochemical water-splitting**
  - **Net plant efficiencies of up to ~50%**
- **Electricity + Heat ➔ High temperature electrolysis or Hybrid thermochemical cycles**
  - **Efficiencies up to ~ 50%**

NGNP PLAN IS TO DEMONSTRATE H_2 PRODUCTION BY TWO ALTERNATIVE PROCESSES

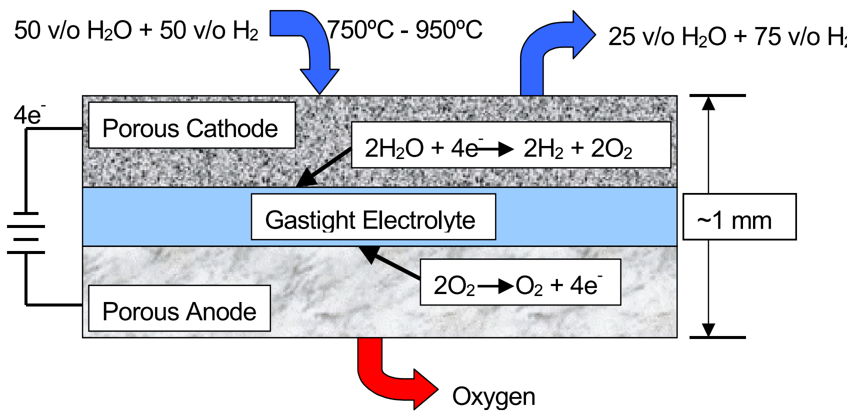


- Hydrogen production to 60 MWt
  - Allow smooth transition between 100% electricity and 90% electricity/10% hydrogen
  - Up to 20 tonnes of H_2 per day
- Hydrogen purity
  - Tritium release below NRC and EPA limits
  - Radioactivity < 10CFR20 limits
  - Meet fuel-cell standards
- Safe reactor/hydrogen interface
- Advanced fuels?

LEADING CANDIDATE HYDROGEN PRODUCTION PROCESSES ARE S-I and HTE



Sulfur-Iodine (S-I)
Thermochemical Process



High Temperature Electrolysis
(HTE) Process

GENERAL ATOMICS

NGNP REACTOR PLANT SIMILAR TO GT-MHR (the main difference is coolant temperature)

	GT-MHR	NGNP
• Power Level (MW)	600	600 (not optimized)
• Power Density (w/cc)	6.5	6.5
• Coolant&Pressure (Mpa/psia)	He 7.12/1032	He 7.12/1032
• Core Outlet Temp ºC	850	1000
• Core Inlet Temp ºC	490	490-600 (not optimized)
• Maximum Fuel Temp ºC	1250	1250 (up to 1400 depending upon fuel element)
• Intermediate HX	NA	Compact Heat Exchanger

NGNP LONGER TERM ADVANCED MHR FOR ELECTRICITY AND H_2 PRODUCTION

- **Very similar to GT-MHR for electricity generation**
  - **Higher core outlet temperature requires additional R&D (fuels and materials)**
- **Production of hydrogen from high temperature MHR nuclear heat appears promising**
  - **Hydrogen production processes require significant R&D**
- **Schedule for startup projected to be 2017**
  - **First commercial deployment ~5+ years later**

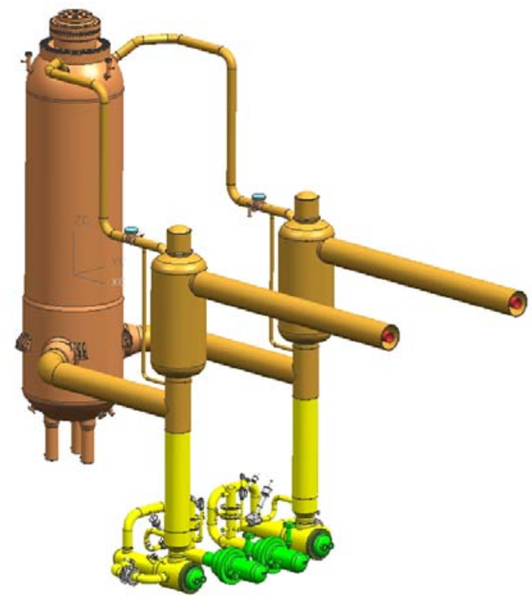
Appendix O: PTAC Briefing on PBMR Oil Sands Applications (August 2007)

The PBMR Presentation on PBMR Oil Sands applications is provided in the following pages.

Presentation to PTAC

Process Heat for Oil Sands

HTGR and ITGR Deployment



Reiner Kuhr

Tony Morris

Shaw Stone & Webster

29th August 2007

Shaw® Stone & Webster Management Consultants, Inc.

Oil Sands implementation

P B M R

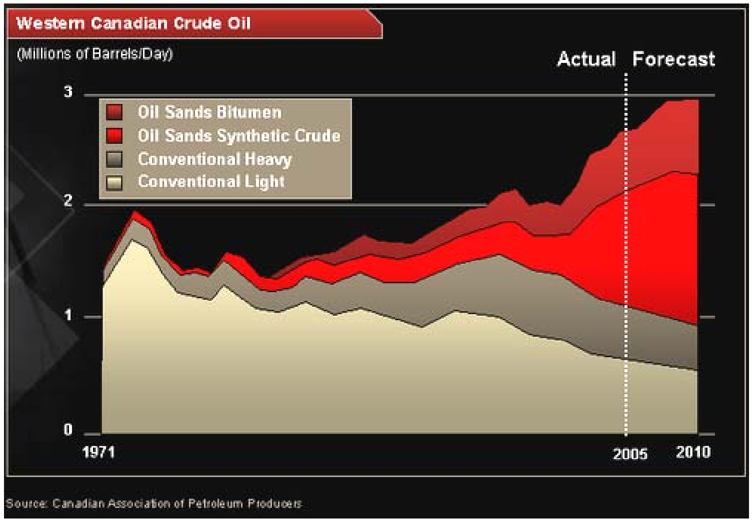
Dealing with implementation expansion

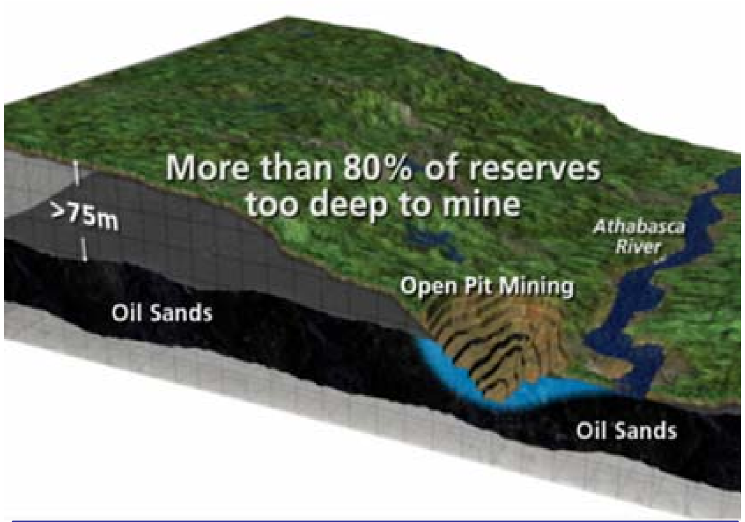
- *energy cost certainty*
- *natural gas displacement*
- *climate change emission free*
- *hydrogen production*

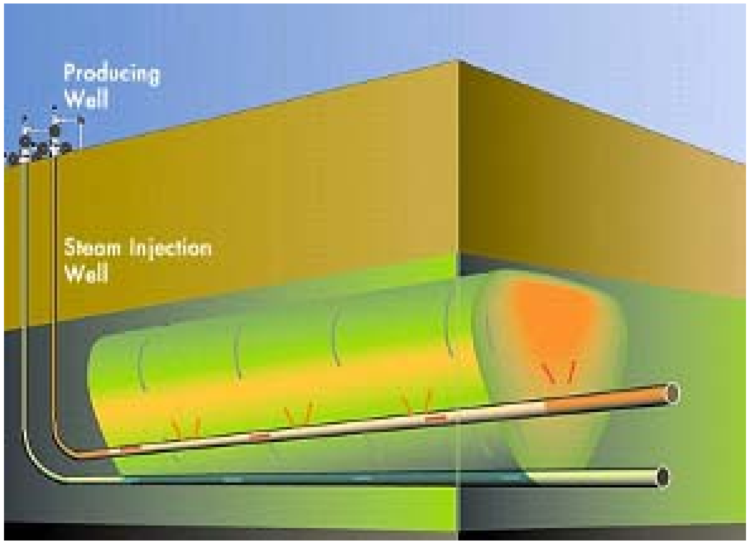
Why Pebble Bed for Oil Sands?

- South African Demonstration Power Plant (DPP) builds on three decades of German experience including a prototype and a full scale demonstration plant
  - learning embedded in the design
- Intermediate Temperature Gas Reactor (ITGR) matches Oil Sand's safety, technical and performance requirements for Steam Assisted Gravity Drainage (SAGD)
- Both Steam-only and Co-generation ITGRs are suitable for Process Heat for Oil Sands (PHOS) applications
  - modular, 100-130k bbl/CDE saturated steam per reactor
  - steam pressures up to 16MPa available
  - turndown capability meets refinery performance criteria
  - high availability matches oil plant's maintenance cycle
- High Temperature Gas Reactor (HTGR) hydrogen production developments underway
  - Steam Methane Reforming for bitumen upgrading and refinery operations
  - thermo-chemical process development offers further potential

SAGD Description





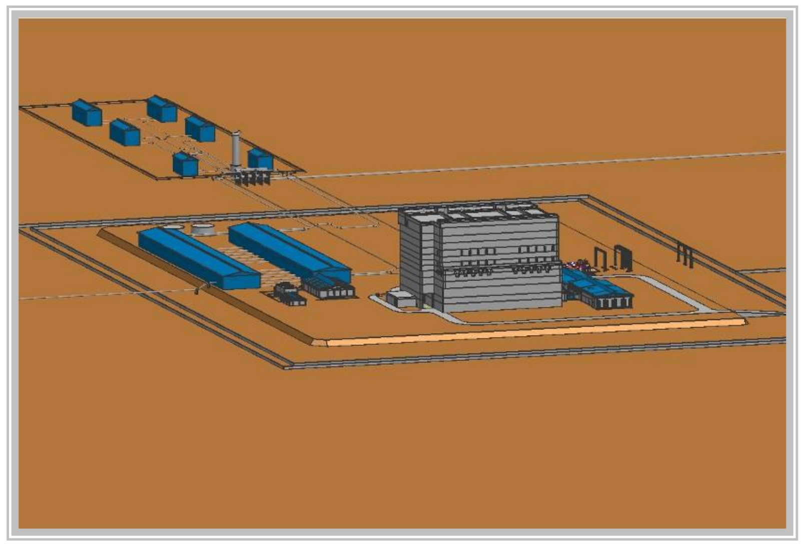


Process Heat Benefits

- Development of new SAGD areas without natural gas firing
  - increases energy cost certainty
  - reduces premium fuel availability concerns
  - climate change emission free operation
- Economic, proven alternative to clean coal
- Natural resource utilisation
- Offers a range of public benefits
  - environment
  - extends availability of premium fuel
  - high tech jobs
- Potential for international support, cooperation and industrial partnerships

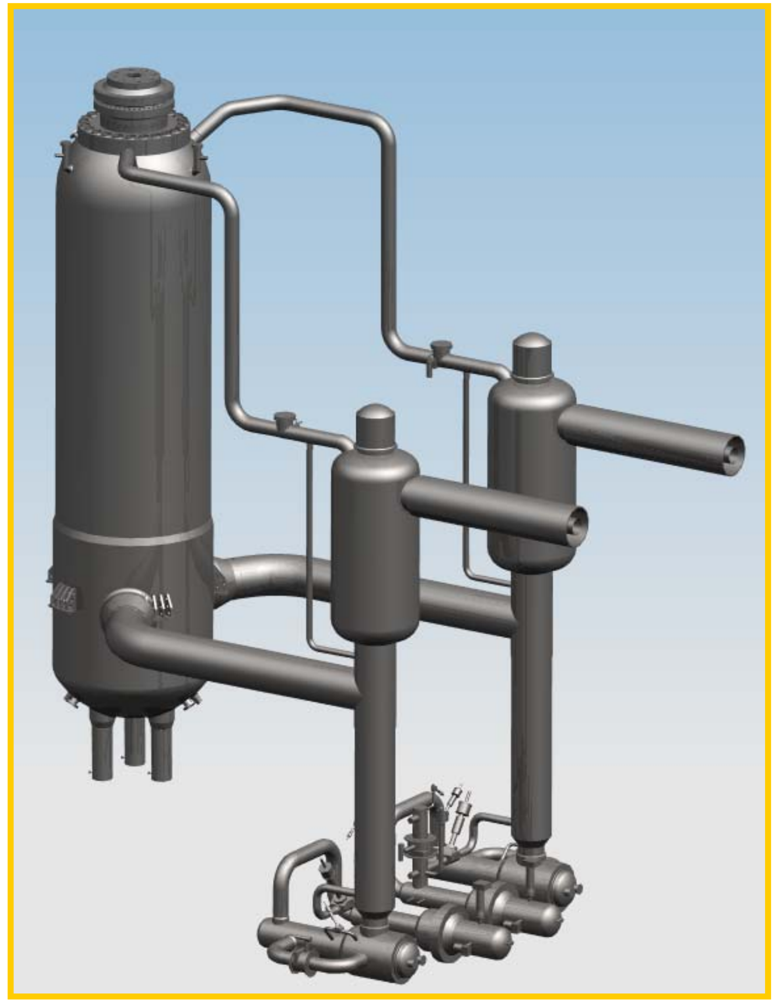
Oil Sands Requirements

- Up to 16MPa saturated steam
- Water treatment options
- Designed for reliable operation
- Remote and difficult site conditions that impact construction and operations
- Plant lifetime and steam capacity to match resource exploitation plans
- Minimises natural gas usage and CO_2 emissions



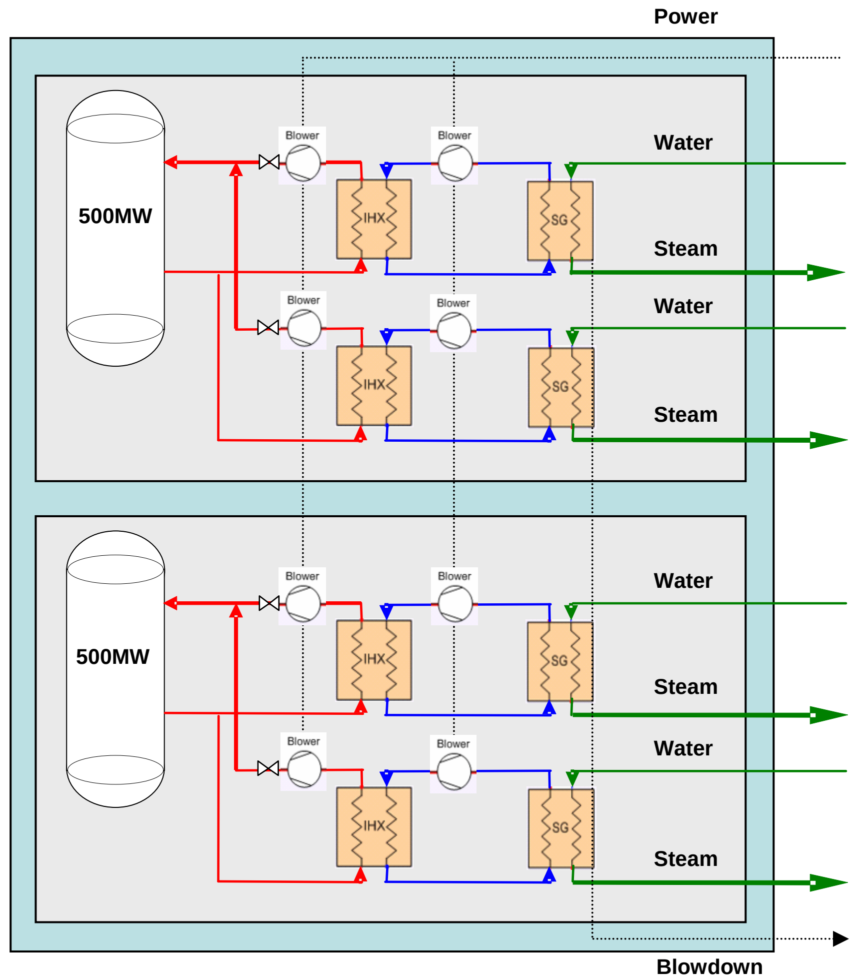
PHOS Heat Supply

- 500MW(thermal) pebble bed modular reactor produces 720°C helium to boilers through to intermediate heat exchangers (IHX)
- IHX separates reactor circuit from the shell and tube process boilers
- Each steam-only reactor supports up to 130k bbl/CDE steam
  - 36MW(e) needed for internal works power requirements
  - oil and feedwater pumping loads also need additional power
- Each cogeneration reactor supports up to 100k bbl/CDE steam
  - also meets internal works power requirements
  - power also available for oil and feedwater pumping loads



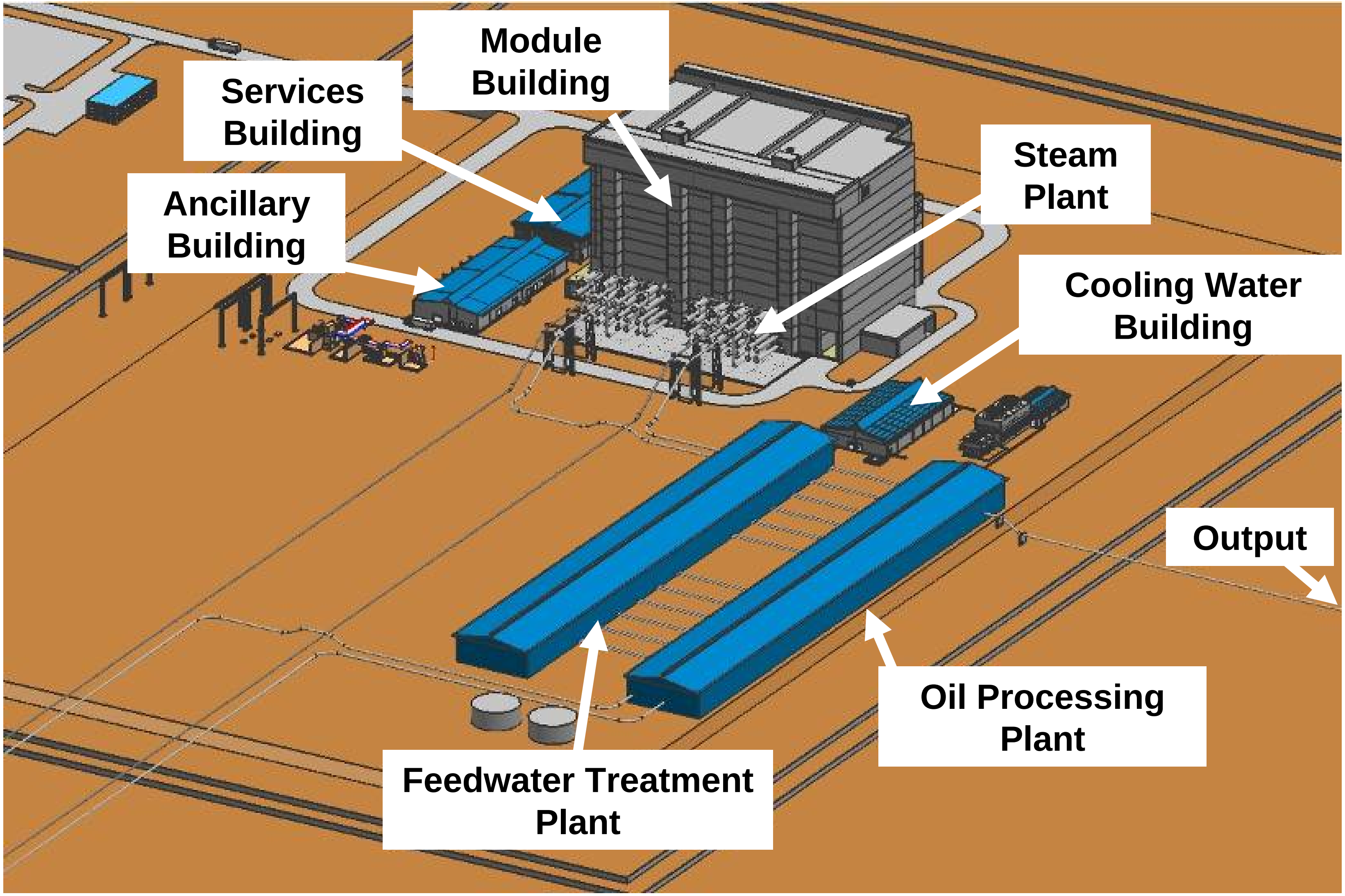
PHOS 11MPa Steam-only

- Two 500MW(t) units can provide 260k bbl/CDE steam
  - 1.8m Te of CO_2 emissions avoided annually
- Each reactor delivers heat through two parallel helium loops to modular boilers
- Boilers produce 11MPa/ 319°C saturated steam for injection
- High quality liquid blowdown returned from boilers to preheat softened feedwater to 190°C
- >96% availability with planned maintenance and boiler cleaning

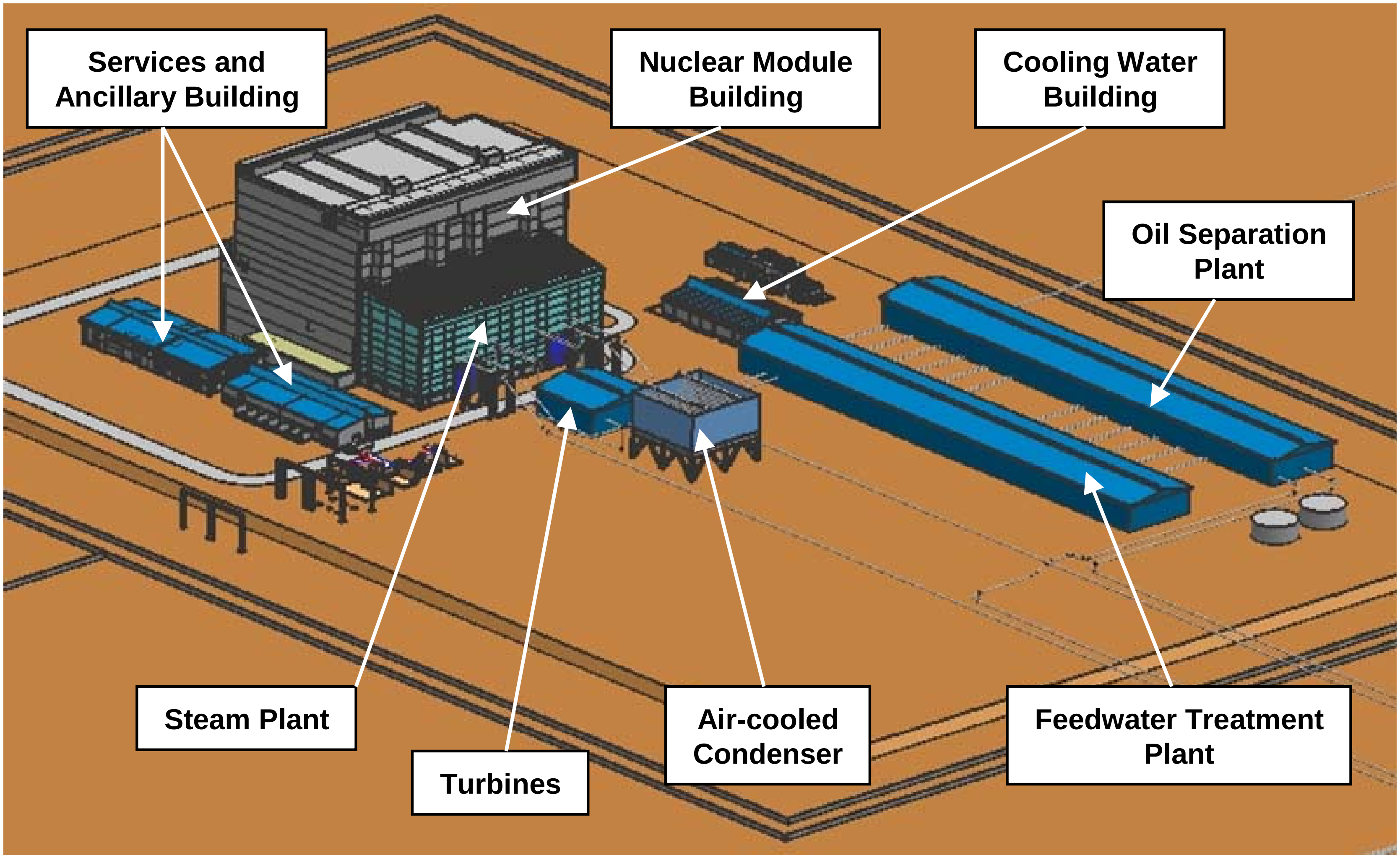


Steam-only Layout

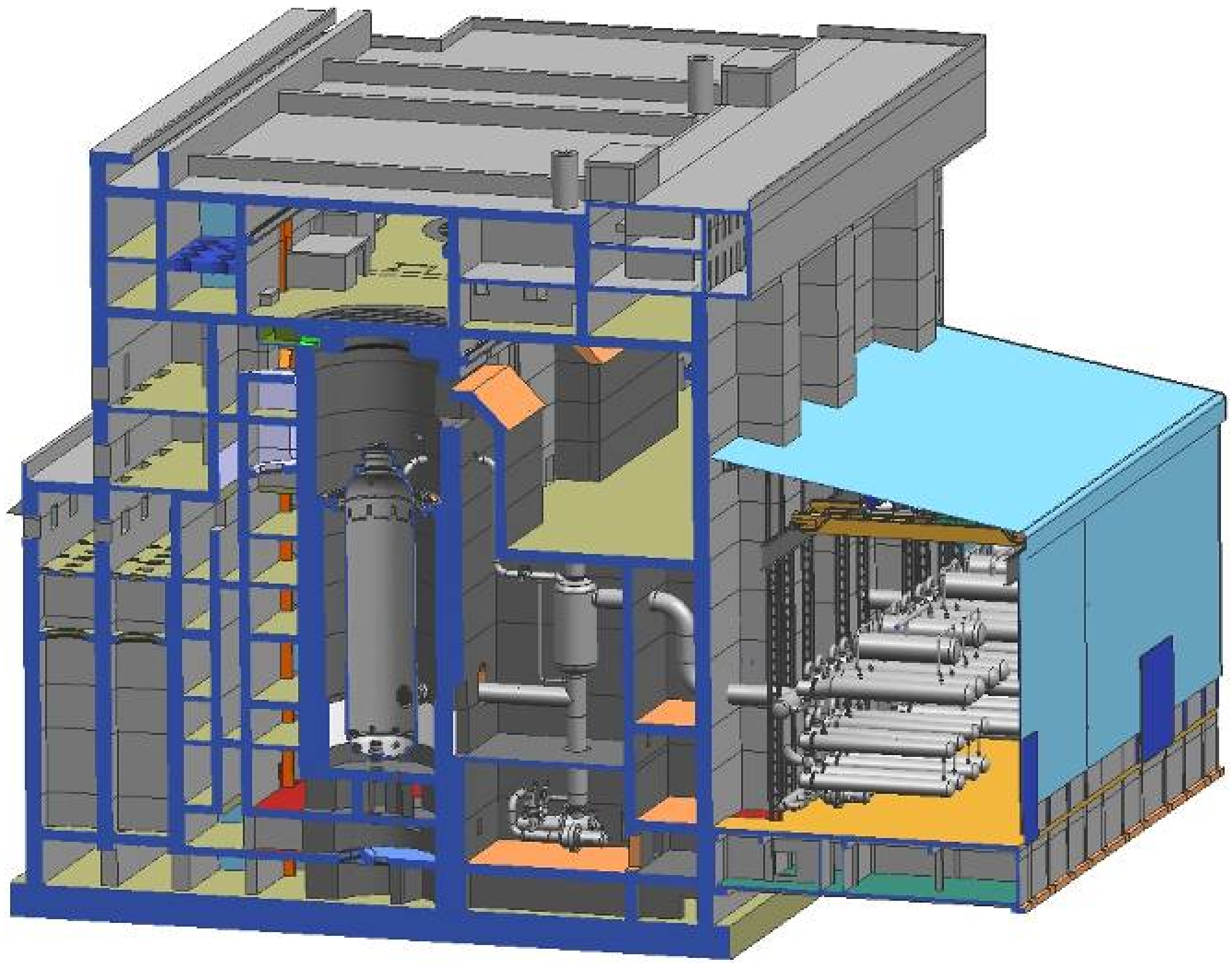
PBMR



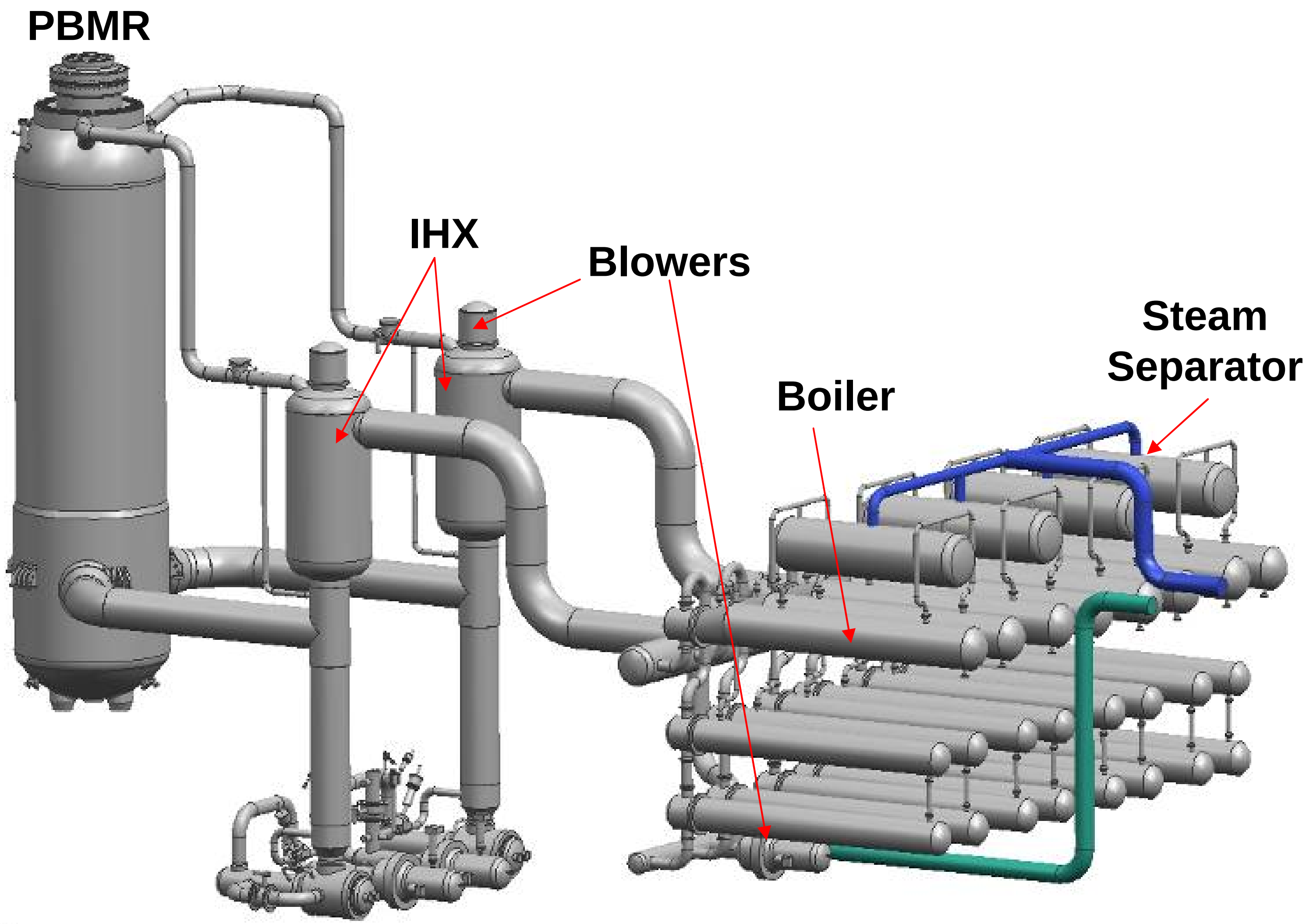
Cogeneration Layout



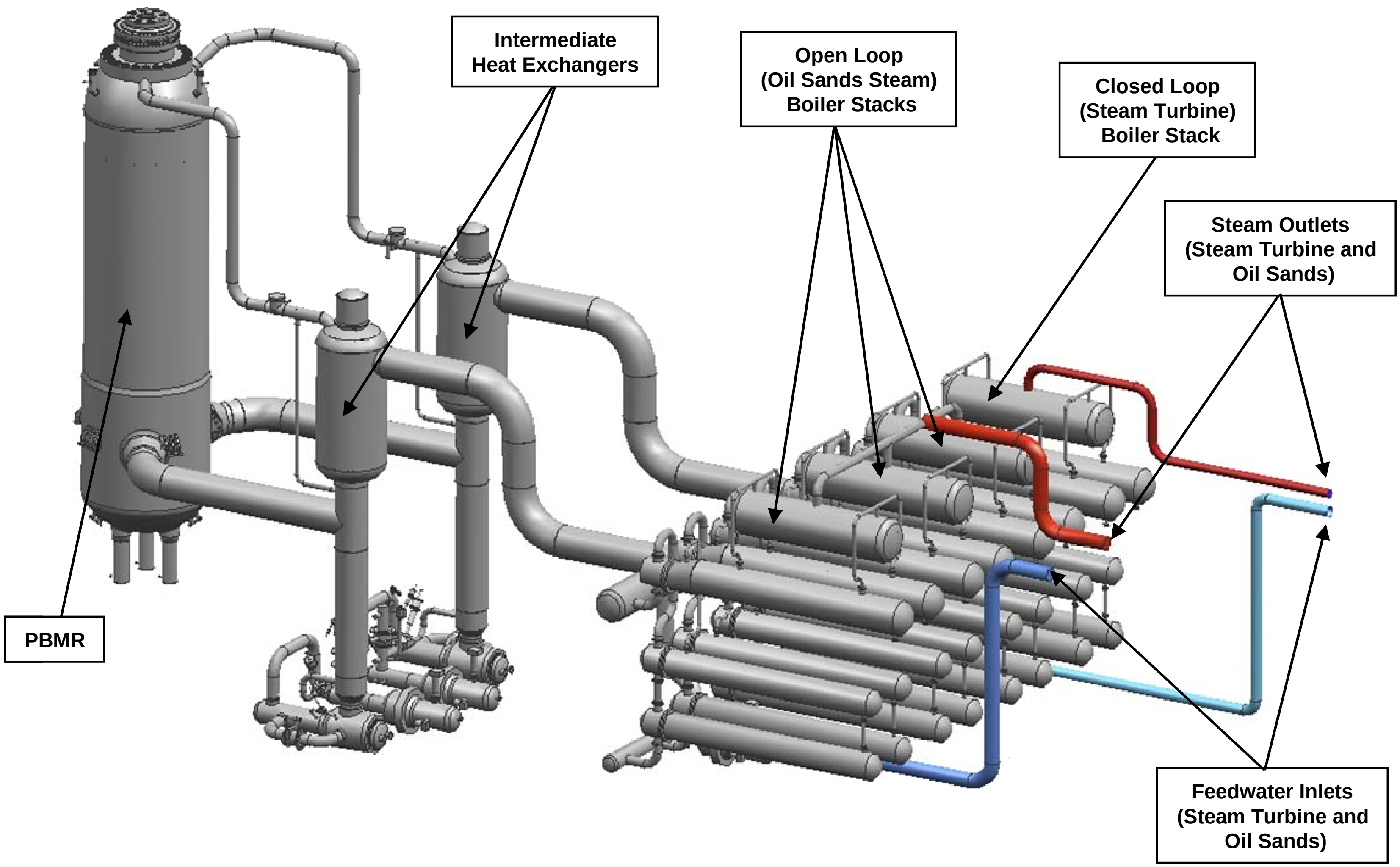
Module Internals



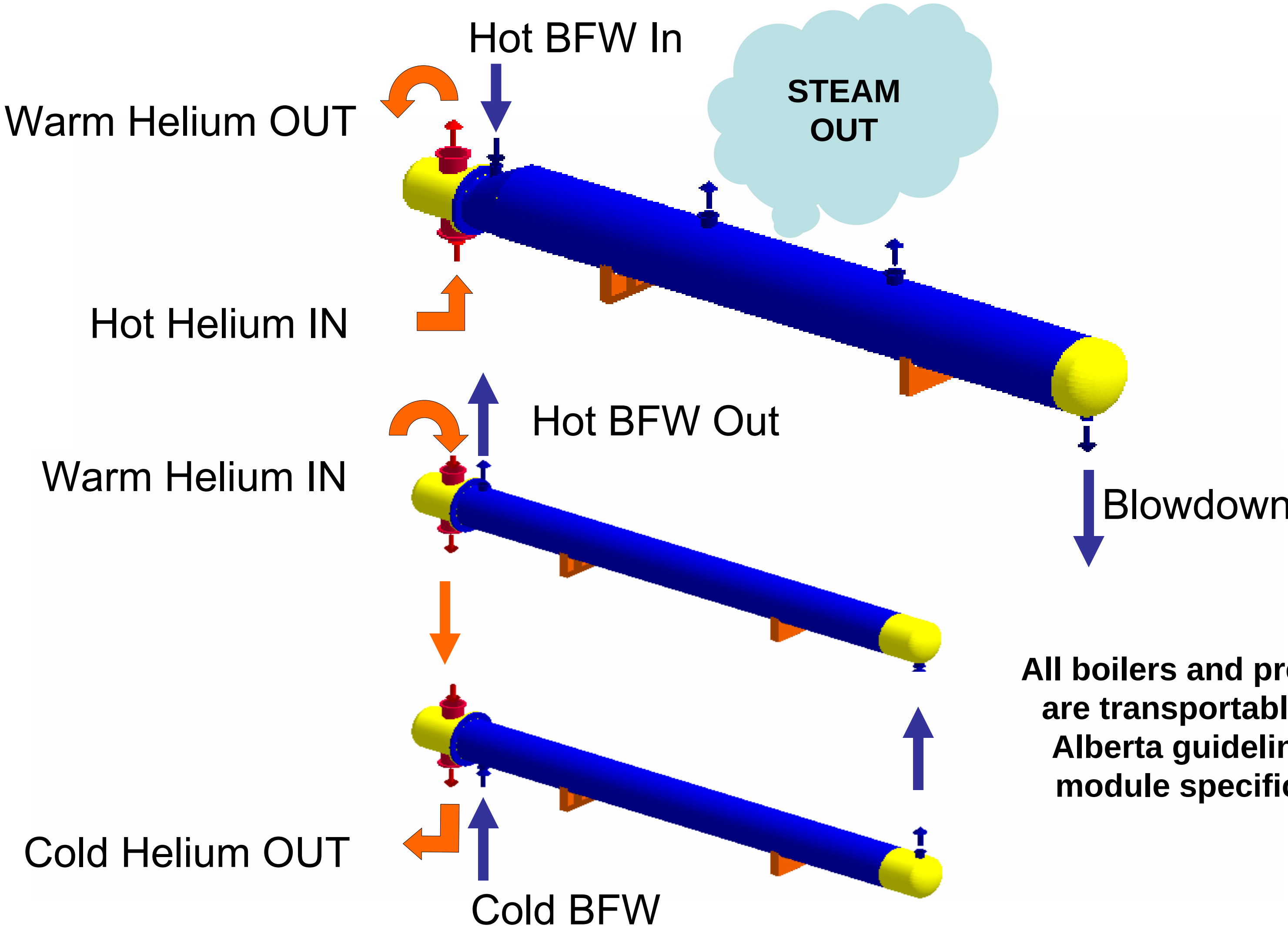
Steam-only Detail



Cogeneration Detail



Boiler Arrangement



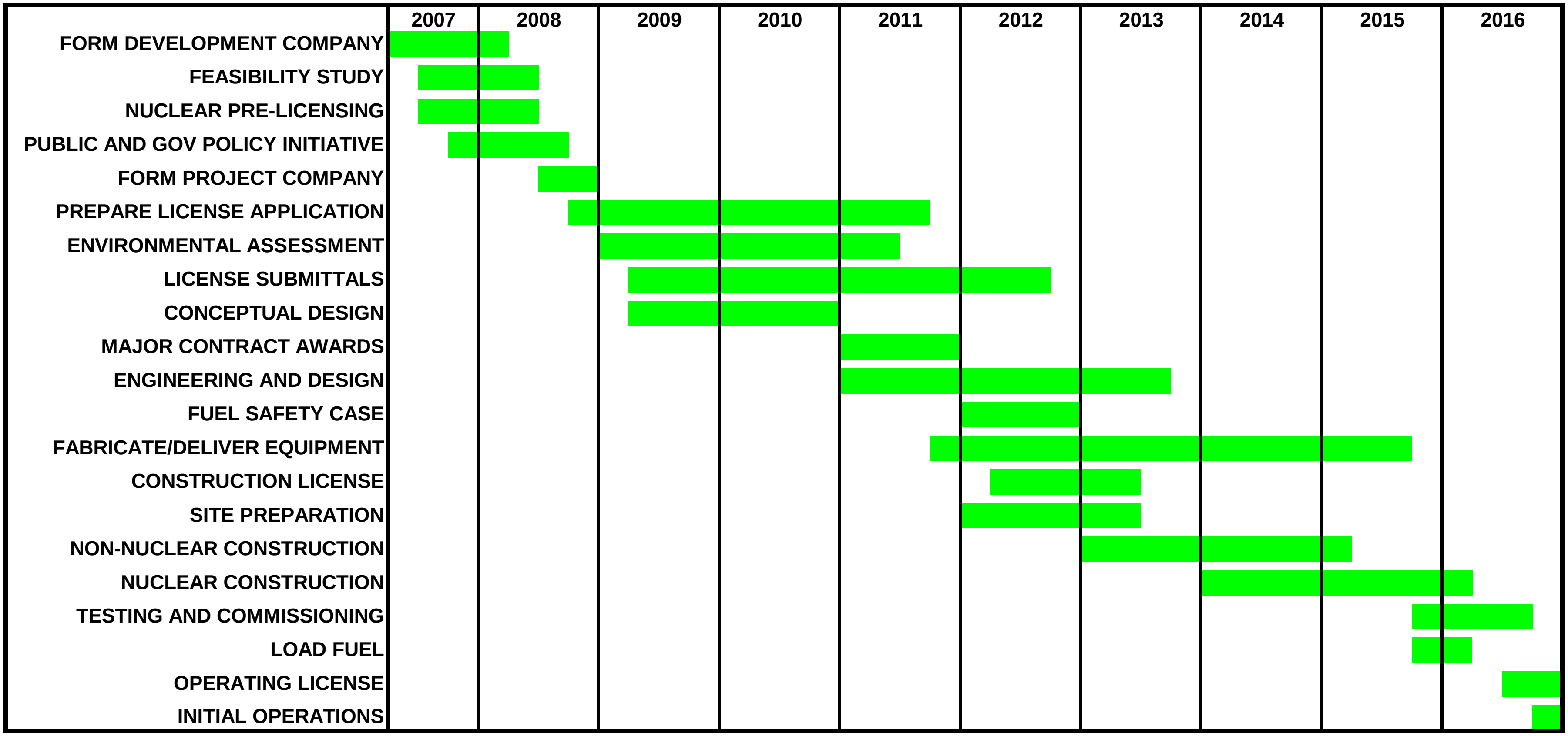
All boilers and pre-heaters are transportable within Alberta guidelines and module specifications

Next Steps

P B M R

- Feasibility Study with interested parties at agreed SAGD site
- Public outreach and consultation process
- Deploy public policy benefits
- Employment and skills studies are required
- Commence First Nation interaction
- Range of near term PHP improvement opportunities available for development in Canada
- Develop the supply chain for high value components sourced in Canada
- Support for first nuclear license application

First-of-Fleet Indicative Schedule



Appendix P: Assessment of Submarine Reactors for Civilian Applications

Nuclear reactors that are used in all US submarines and the submarines of all other countries are Pressurized Water Reactors, with the exception of a small portion of the Russian submarine fleet.

Modern submarine reactors are designed to meet extremely demanding service duties that far exceed those of commercial nuclear power plants. One recent requirement for US submarines is for the reactor to operate through a postulated depth charge explosion near the hull, which imposes a 15.0 g seismic load on the reactor systems. By comparison, commercial power reactors are designed for seismic events in the 0.3g to 0.5g range. This requirement is particularly challenging, since the reactor and its support structures must also not generate or transmit any noise to the submarine's surroundings during normal operation.

As a result of the high seismic design basis load, all submarine reactor components including the fuel, fuel supports, and control mechanisms are extremely robust and require the use of large amounts of in-core structures, which are largely fabricated from zirconium alloys to meet the structural requirements. The large, in-core metal load results in extremely high neutron absorption/loss. This, in combination with relatively long refueling cycles, requires the use of Highly Enriched Uranium (HEU) fuel. HEU fuel is defined internationally as having a U^{235} content of 20% or greater, and is by international agreement precluded for use in civilian and commercial applications. It is believed that current US and British submarine reactors operate with U^{235} enrichment in the range of 40% to 50%. Commercial PWRs operate with enrichment in the 3.5% to 4.2% range.

There are many other differences between submarine and commercial PWRs. For example, the submarine reactors operate at much higher power densities, and have much higher coolant velocities in both the reactor core and the reactor coolant piping systems than commercial reactors. Submarine reactor containment systems are also much more compact than those of commercial PWRs, and take advantage of the submarine hull structure.

As a result of the design difference between submarine and commercial PWRs that stem from the demanding submarine application, submarine reactors are much more expensive to construct and operate than commercial PWRs of the same output. A commercial version (minimum design changes) of a submarine PWR is likely to have higher capital cost by a factor of between 5 and 10, higher Operations and Maintenance costs (in a civilian environment) by a factor of between 2 and 3, and fueling costs that are likely to be higher by a factor of between 10 and 15, if fuel were available (HEU cannot be utilized in commercial applications).

The latest US submarine reactors are believed to generate power in the range of 400 MW electrical. Commercial power reactors in this size range are practical. AECL has completed the conceptual designs of the CANDU 3 (450 MWe) and the ACR-700 (650 MWe). Westinghouse has completed the conceptual design and obtained a Standard Product

Licence in the US for the AP700 (660 MWe). All of these designs were dropped in favor of larger units based on economy of scale considerations.

Rolls Royce took advantage of their submarine PWR experience to develop the SIR concept (Safe Integral Reactor) in the early 1980s, and briefly joined forces with Combustion Engineering for the commercialization of the concept. However, SIR was determined to be uneconomic and was dropped later in the 1980s. SIR had a typical commercial PWR reactor core and fuel configuration, and incorporated the steam generators into the upper section of the pressure vessel (integral).

In summary, submarine reactors are not suitable and cannot be adapted to civilian applications. However, the medium and small PWR, BWR, and CANDU plants are technically feasible, and were operated during the early years of nuclear commercialization.

Appendix Q: Oil Sands AP1000 Siting Guide

The Oil Sands AP1000 Siting Guide is provided in the following pages.

AP1000 Siting Guide: Site Information for an Early Site Permit Application

CONTENTS

1 INTRODUCTION

Part of the EPRI Early Site Permit Demonstration Program was the development of a guide for site selection criteria and procedures. "Siting Guide: Site Selection and Evaluation Criteria for an Early Site Permit Application" has been issued to serve as a roadmap and tool for applicants to use in developing detailed siting plans for their specific region of the country.

This AP1000 document (APP-0000-X1-001) can be used in conjunction with the EPRI Siting Guide: Site Selection and Evaluation Criteria for an Early Site Permit Application for evaluating the siting of an AP1000 to a potential site. It also has sufficient information to support the plant/site interface portions of a Combined License application.

1.1 Background

In November 1990, the Nuclear Power Oversight Committee (NPOC) prepared a strategic plan for building new nuclear power facilities. An essential element in the strategy (Building Block 5) consisted of initiating a project to obtain Nuclear Regulatory Commission (NRC) approval through newly issued 10 CFR Part 52 (Early Site Permits; Standard Design Certifications and Combined Licenses for Nuclear Power Reactors). The plan was designed to be implemented either through attainment of an early site permit (ESP) or through the submission of, and NRC approval of, a combined construction and operating license (COL) application for a design certified ALWR under the NRC standardization rule. In 1990 Sandia National Laboratory issued a Request for Quotation to test the ESP process in a demonstration program. In early 1991, the Joint Contractors were formed and selected by the DOE through SNL to implement the Early Site Permit Demonstration Program. The Joint Contractors were assisted by the Electric Power Research Institute (EPRI) and the Nuclear Management and Resources Council (NUMARC) and developed a phased approach to the preparation, review, and application to NRC for acceptance of an early site permit. An output of this effort was the EPRI Siting Guide.

1.2 Purpose and Goals

The EPRI Siting Guide has been designed to be responsive to 10 CFR 52, 10 CFR 100, and related regulations and guidance, and form a framework or roadmap for an applicant to use in developing a detailed siting plan for a specific region of the country. The purpose and scope of this AP1000 siting information document (APP-0000-X1-001) is to provide specific AP1000 information relating directly to the Siting Guide. It is based upon providing information for a single AP1000. If siting a twin unit, values should be doubled except for the acreage required. To determine the amount of land area required for a twin station a site specific plot plan should be developed.

1.3 Report Structure

This section provides an overview of the balance of the report. Section 2.0 presents AP1000 design information in the same order and format as criteria are presented in Section 3 of the EPRI Siting Guide. The discussion of the bases for criteria and the use of design information is contained in the Siting Guide and not repeated here. Note that all data in this AP1000 document is reference in that the data is controlled in some other AP1000 design document. Section 3 of this AP1000 Siting Guide contains detailed site interface information not addressed in the EPRI Siting Guide. Section 4 contains other information identified as Plant Parameter Envelopes that are not covered in the balance of this document. Section 5 is an addition to the information presented in the EPRI siting Guide. The section contains a listing of the site related Combined License (COL) information items identified in the AP1000 Design Control Document. These COL information items are not necessarily required for an Early Site Permit, but they are required to be part of a COL for an AP1000. As such, this information will ultimately be required by NRC and should be considered in the planning for site licensing activities.

2 DETAILED DISCUSSION OF AP1000 SITING INFORMATION

This section provides detailed AP1000 siting information for each siting criterion of the EPRI Siting Guide. This information is presented so that it can be applied to an ESP or COL application anywhere in the continental United States. Accordingly, some "customization" of utility functions may be appropriate for specific regions; and some information may not be applicable for some siting applications.

Each applicant should also conduct a review of the materials in this document; the state siting, emergency planning, and environmental regulations applicable to the region of interest; and the physical characteristics of the region of interest.

Plant Parameters Envelopes (PPEs) define the envelope of the AP1000/site interface conditions that, if not satisfied by the site, may preclude locating AP1000 on the selected site. An ESP or COL applicant can utilize PPEs to represent a bound on whether an AP1000 can be considered for the site without further analysis and justification to NRC.

2.1 Health and Safety Criteria

2.1.1 Accident Cause-Related Criteria

2.1.1.1 Geology/Seismology

Current NRC regulations identify three geologic, seismologic, and soil parameters that must be evaluated to determine the suitability of prospective sites. First, the Safe Shutdown Earthquake (SSE) must be determined to establish a vibratory ground motion design basis, and detailed information regarding capable tectonic structures and sources are needed to determine the SSE. Second, the occurrence of, or potential for, surface faulting or deformation must be identified and evaluated to permit evaluation of site conditions with respect to standard facility designs. Third, other geologic conditions (e.g., geologic hazards and soil characteristics) that could affect the safety of a facility must also be evaluated.

The following site parameter criteria are intended to provide applicants with specific values included in the AP1000 Design Certification for use in ESP and COL application. The criteria discussed in the following geology/seismology sections provide a set of conditions within which an AP1000 can be sited without additional licensing.

2.1.1.1.1 *Vibratory Ground Motion*

See Section 4, Table Item 1.5.

2.1.1.1.2 *Capable Tectonic Structures or Sources*

The AP1000 Design Certification provides for no fault displacement potential within the investigative area.

2.1.1.1.3 *Surface Faulting and Deformation*

With regard to surface faulting and deformation, no absolute exclusionary criteria have been identified for AP1000 other than the fault displacement criteria addressed in 2.1.1.1.2.

2.1.1.1.4 *Geologic Hazards*

With regard to geologic hazards, no absolute exclusionary criteria have been identified for AP1000. Therefore, geologic hazards should be addressed as an avoidance criterion. The following geologic and related man-made conditions should be avoided in locating a facility:

- Areas of active (and dormant) volcanic activity;
- Subsidence areas caused by withdrawal of subsurface fluids such as oil or groundwater, including areas which may be effected by future withdrawals;
- Potential unstable slope areas, including areas demonstrating paleolandslide characteristics;
- Areas of potential collapse (e.g., karstic areas in limestone, salt, or other soluble formations);
- Mined areas, such as near-surface coal mined-out areas, as well as areas where resources are present and may be exploited in the future;
- Areas subject to seismic and other induced water waves and floods.

2.1.1.1.5 *Soil Stability*

With regard to soil stability, the AP1000 structural design is based on the AP600 design. AP600 has an average allowable static soil bearing capacity requirement of 8000 pounds per square inch or greater and a shear wave velocity requirement of 1000 ft/sec or greater. The current AP1000 Design Certification is based upon a rock foundation with the average allowable soil bearing capacity to be greater than or equal to 8400 lb/ft^2 over the footprint of the nuclear island at its excavation depth. The shear wave velocity shall be greater than or equal to 3500 ft/sec based upon low-strain, best-estimate soil properties over the footprint of the nuclear island at its excavation depth. There are no constraints on soils surrounding the nuclear island. No liquefaction potential is assumed. We expect to expand the licensed soil stability requirements for AP1000 to be at least those of AP600 at the time of Combined License application or before.

2.1.1.2 Cooling System Requirements

Since AP1000 is a passive nuclear plant, it requires **no** safety-related heat sink to reach safe shutdown other than the water contained in its passive cooling system tank located atop the reactor building. Thus a safety-related ultimate heat sink system similar to traditional nuclear plants is not required. The ultimate heat sink for a passive plant is air, which is motivated by natural convection over the containment vessel.

The AP1000 has two nonsafety-related systems for discharging waste heat from the plant. These are a conventional circulating water system to remove the waste heat related to power production and a smaller service water system. The service water system in AP1000 has its own cooling tower, which is separate from the circulating water system. The circulating water system pump discharge lines connect to a common header which connects to the inlet water boxes of the condenser as well as supplies cooling water to the Turbine Cooling System (TCS) and condenser vacuum pump seal water heat exchangers.

AP1000 circulating water requirements can vary greatly depending on site specific conditions and limitations. The AP1000 requires no more or no less circulating water than any other similarly sized nuclear plant. Essentially the plant needs to reject approximately 2/3 of 3415 MWt or about 2270 MWt. If the plant uses a cooling tower, site ambient air temperature and humidity conditions, and the design rise across the cooling tower / condenser are needed to estimate the required flow rate. (A very rough estimate is that the required flow rate is somewhere between 450,000 gpm to 850,000 gpm). The AP1000 design used as a reference for Design Certification assumes a circulating water system with a cooling tower, a flow rate of 600,000 gpm, and a 25.2 °F range.

Make-up for a circulating water system that utilizes a cooling tower can be estimated to be up to 4% of the circulating water flow rate.

The service water system consists of two 100-percent-capacity service water pumps, automatic backwash strainers, a two-cell cooling tower with a divided basin, and associated piping, valves, controls, and instrumentation.

The service water pumps, located in the turbine building, take suction from piping which connects to the basin of the service water cooling tower. Service water is pumped through strainers to the component cooling water heat exchangers for removal of heat. Heated service water from the heat exchangers then returns through piping to a mechanical draft cooling tower where the system heat is rejected to the atmosphere. Cool water, collected in the tower basin, flows through fixed screens to the pump suction piping for recirculation through the system.

NOMINAL SERVICE WATER FLOWS AND HEAT LOADS AT DIFFERENT OPERATING MODES

	Component Cooling Water Pumps and Heat Exchangers	SWS Pumps and Cooling Tower Cells (Number Normally is Service)	Flow (gpm)	Heat Transferred (Btu/hr)
Normal Operation (Full Load)	1	1	8,000	83×10^6
Cooldown	2	2	16,000	296×10^6 (148×10^6 per cell)
Refueling (Full Core Offload)	2	2	16,000	74×10^6
Plant Startup	2	2	16,000	96×10^6
Minimum to Support Shutdown Cooling and Spent Fuel Cooling	2	2	14,400	240×10^6 (120×10^6 per cell)

A small portion of the service water flow is normally diverted to the circulating water system (CWS) basin. This blowdown is used to control levels of solids concentration in the SWS. [An alternate blowdown flow path is provided to the waste water system (WWS) for times when the CWS is not operating.] This design affords a single blowdown interface from the CWS to the site.

Make-up for the service water cooling tower is estimated to be 80 gpm nominally. Potable water and sanitary drain requirements can be estimated based on the assumption that there may be up to 300 operating personnel required for the first single unit and up to 420 operating personnel required for the first twin unit. The AP1000 design for these systems is based upon 1000 persons on site and 100 gallons/day/person.

2.1.1.2.1 Cooling Water Supply

PPE Section	Requirement	AP1000 Value
2.7.15	Makeup Flow Rate (Closed Cycle Systems)	See Section 4, Table Item 2.7.15
2.7.16 2.8.15 2.10.11	Maximum Consumption of Raw Water (Closed Cycle System)	See Section 4, Table Items 2.7.16, 2.8.15 and 2.10.11

PPE Section	Requirement	AP1000 Value
2.7.17 2.8.16 2.10.12	Monthly Average Consumption of Raw Water (Closed Cycle Systems)	See Section 4, Table Items 2.7.17, 2.8.16 and 2.10.12
2.9.2	Cooling Water Flow Rate (Cooling Tower)	See Section 4, Table Item 2.9.2

2.1.1.2.2 Ambient Temperature Requirements

PPE Section	Requirement	AP1000 Value
2.1.1	Normal Maximum Ambient Temperature with 1% Exceedance	See Section 4, Table Item 2.1.1
2.1.2	Normal Maximum Wet Bulb Temperature with 1% Exceedance	See Section 4, Table Item 2.1.2
2.1.3	Normal Minimum Ambient Temperature with 1% Exceedance	See Section 4, Table Item 2.1.3
2.1.5	Maximum Safety Ambient Temperature with 0% Exceedance	See Section 4, Table Item 2.1.5
2.1.6	Maximum Safety Wet Bulb Temperature with 0% Exceedance	See Section 4, Table Item 2.1.6
2.1.7	Minimum Safety Ambient Temperature with 0% Exceedance	See Section 4, Table Item 2.1.7
2.7.3 2.8.2	Approach Temperature	See Section 4, Table Items 2.7.3 and 2.8.2

2.1.1.3 Flooding

The maximum flood level assumed for AP1000 is the plant design grade elevation. The standard grid coordinate system for AP1000 labels plant grade as plant elevation 100 ft. Structural analyses have assumed grade to be at 100 ft. Actual grade will be a few inches lower to prevent surface water from entering doorways.

Adverse effects of flooding due to high water or ice effects do not have to be considered for site-specific non-safety-related structures and water sources outside the scope of the certified AP1000 design. Flooding of intake structures, cooling canals, or reservoirs or channel diversions would not prevent safe operation of the plant.

2.1.1.4 Nearby Hazardous Land Uses

AP1000 has no specific requirements or restrictions on nearby land use over and above those generally imposed by NRC for plants of this type. There are design provisions for detection of aerosols that may be toxic to the main control room staff and there are combined license applicant action items requiring identification of nearby hazardous land use.

2.1.1.5 Extreme Weather Conditions

See Section 4, Table Item 1.

2.1.1.5.1 Winds

The design wind is specified as a basic wind speed of 145 mph with an annual probability of occurrence of 0.02 based on the most severe location identified in American Society of Civil Engineers," Minimum Design Loads for Buildings and Other Structures," ASCE 7-98. This wind speed is the 3 second gust speed at 33 feet above the ground in open terrain (ASCE 7-98, exposure C). This basic wind speed of 145 mph is the 3 second gust speed that has become the basis of wind design codes since 1995. It corresponds to the 110 mph fastest mile wind used as the basis for the AP600 design in accordance with the 1988 edition of ASCE 7-98. Higher winds with a probability of occurrence of 0.01 are used in the design of seismic Category I structures by using an importance factor of 1.15.

2.1.1.5.2 Precipitation

There are no additional AP1000 requirements or restrictions.

2.1.2 Accident Effects-Related

2.1.2.1 Population

There are no additional or specific AP1000 requirements or restrictions related to population concentration or distribution. See Section 4, Table Item 9.6.6.

2.1.2.2 Emergency Planning

There are no additional or specific AP1000 requirements or restrictions related to emergency planning.

2.1.2.3 Atmospheric Dispersion

See Section 4, Table Item 9.1.

2.1.3 Operational Effects-Related

2.1.3.1 Surface Water – Radionuclide Pathway

See Section 4, Table Item 10.1.

There are no additional or specific AP1000 requirements or restrictions related to radionuclide pathways.

2.1.3.1.1 Dilution Capacity

There are no additional or specific AP1000 requirements or restrictions related to dilution capacity.

2.1.3.1.2 Baseline Loadings

There are no additional or specific AP1000 requirements or restrictions related to baseline loadings.

2.1.3.1.3 Proximity to Consumptive Users

There are no additional or specific AP1000 requirements or restrictions related to proximity of consumptive users.

2.1.3.2 Groundwater Radionuclide Pathway

There are no additional or specific AP1000 requirements or restrictions related to the groundwater radionuclide pathway.

2.1.3.3 Air Radionuclide Pathway

There are no additional or specific AP1000 requirements or restrictions related to air radionuclide pathway.

2.1.3.3.1 Topographic Effects

There are no additional or specific AP1000 requirements or restrictions related to the site topography as it relates to air radionuclide pathway.

2.1.3.3.2 Atmospheric Dispersion

See Section 4, Table Item 9.2.

2.1.3.4 Air-Food Ingestion Pathway

There are no additional or specific AP1000 requirements or restrictions related to the air-food ingestion pathway.

2.1.3.5 Surface Water – Food Radionuclide Pathway

There are no additional or specific AP1000 requirements or restrictions related to the use of irrigation waters in downstream areas is a potential pathway for radionuclides.

2.1.3.6 Transportation Safety

There are no additional or specific AP1000 requirements or restrictions related to potential impacts from facility operations on transportation safety that could occur as a result of increased hazards such as fog and ice from the operation of cooling systems (e.g., cooling towers and cooling reservoirs).

2.2 Environmental Criteria

2.2.1 Construction-Related Effects on Aquatic Ecology

2.2.1.1 Disruption of Important Species/Habitats

There are no additional or specific AP1000 requirements or restrictions related to the disruption of important species or habitats.

2.2.1.2 Bottom Sediment Disruption Effects

There are no additional or specific AP1000 requirements or restrictions related to bottom sediment disruption effects. The nature and extent of construction and cooling water related disruption is site specific.

2.2.1.2.1 Contamination

There are no additional or specific AP1000 requirements or restrictions related to contamination.

2.2.1.2.2 Grain Size

There are no additional or specific AP1000 requirements or restrictions related to grain size.

2.2.2 Construction-Related Effects on Terrestrial Ecology

2.2.2.1 Disruption of Important Species/Habitats and Wetlands

There are no additional or specific AP1000 requirements or restrictions related to construction-related effects on terrestrial ecology.

2.2.2.1.1 Important Species/Habitats

There are no additional or specific AP1000 requirements or restrictions related to construction-related effects on important species or their habitats.

2.2.2.1.2 Groundcover/Habitat

There are no additional or specific AP1000 requirements or restrictions related to construction related effects on groundcover.

2.2.2.1.3 Wetlands

There are no additional or specific AP1000 requirements or restrictions related to construction-related effects on wetlands.

2.2.2.2 Dewatering Effects on Adjacent Wetlands

During construction, dewatering is required for AP1000 to the depth of 40 feet below the working grade elevation for the excavation of the Nuclear Island. The footprint of this excavation is an irregular rectangle about 260 feet by 160 feet. In addition, dewatering will be required for the site specific circulating water system. At a minimum this excavation will include the condenser waterbox sump under the turbine building, the circulating water pipe trench and the pump house or cooling tower sump. After plant completion, dewatering is not required.

2.2.2.2.1 Depth to Water Table

See Section 4, Table Item 1.8.2.

2.2.2.2.2 Proximal Wetlands

There are no additional or specific AP1000 requirements or restrictions related to the proximity of wetlands.

2.2.3 Operational-Related Effects on Aquatic Ecology

2.2.3.1 Thermal Discharge Effects

2.2.3.1.1 Migratory Species Effects

There are no additional or specific AP1000 requirements or restrictions related to potential effects on migratory species water and land use during construction.

2.2.3.1.2 Disruption of Important Species/Habitats

There are no additional or specific AP1000 requirements or restrictions related to the disruption of important species or their habitats during plant operation.

2.2.3.1.3 Water Quality

Most of the values presented below for AP1000 are estimates for use in preliminary site investigations. AP1000 is designed to be adaptable to a variety of cooling water sources. Details of blowdown rates, constituents and concentrations will be site specific. They are a function of the type of cooling (cooling tower or once through), the inlet water quality and the cycles of concentration. Once-through discharge temperature and temperature rise will most likely be dictated by inlet temperature, inlet flow rate and local environmental regulations. The values presented should envelop most sites in the United States. They are as follows:

PPE Section	Requirement	AP1000 Value
2.7.4 2.8.3 2.10.2	Blowdown Constituents and Concentrations	See " Blowdown Constituents and Concentrations" table directly below this table
2.7.5 2.10.3	Blowdown Flow Rate (Mechanical Draft & Pond)	See Section 4, Table Items 2.7.5 and 2.10.3
2.8.4	Blowdown Flow Rate (Natural Draft)	See Section 4, Table Item 2.8.4
2.7.6 2.8.5 2.10.4	Blowdown Temperature (Closed Cycle)	See Section 4, Table Items 2.7.6, 2.8.5 and 2.10.4
2.7.9 2.8.8 2.10.7	Cycles of Concentration (Closed Cycle)	See Section 4, Table Items 2.7.9, 2.8.8 and 2.10.7
2.9.1	Cooling Water Discharge Temp (Once-	See Section 4, Table Item 2.9.1

PPE Section	Requirement	AP1000 Value
	through)	
2.9.3	Cooling Water Temperature Rise (Once-through)	See Section 4, Table Item 2.9.3
2.9.5	Heat Rejection Rate (Once-through)	See Section 4, Table Item 2.9.5

Blowdown Constituents and Concentrations

Constituent	Concentration (ppm)[1]		
	River Source	Well/Treated Water	Envelope
Chlorine demand	10.1	--	10.1
Free available chlorine	0.5	--	0.5
Chromium	--	--	--
Copper	--	6	6
Iron	0.9	3.5	3.5
Zinc	--	0.6	0.6
Phosphate	--	7.2	7.2
Sulfate	599	3,500	3,500
Oil and grease	--	--	--
Total dissolved solids	--	17,000(1)	17,000(1)
Total suspended solids	49.5	150	150
BOD, 5-day	--	--	--

(1) Assumed cycles of concentration equals 4

These parameters define the thermal and water quality impacts that cooling system blowdown effluents will have on the receiving water body for the various cooling system configurations.

2.2.3.2 Entrainment/Impingement Effects

2.2.3.2.1 Entrainable Organisms

There are no additional or specific AP1000 requirements or restrictions related to entrainable organisms.

2.2.3.3 Dredging/Disposal Effects

2.2.3.3.1 Upstream Contamination Sources

There are no additional or specific AP1000 requirements or restrictions related to potential upstream contamination sources.

2.2.3.3.2 Sedimentation Rates

There are no additional or specific AP1000 requirements or restrictions related to sedimentation rates.

2.2.4 Operational-Related Effects on Terrestrial Ecology

2.2.4.1 Drift Effects on Surrounding Areas

2.2.4.1.1 Important Species Habitat Areas

There are no additional or specific AP1000 requirements or restrictions related to the plants operational drift effects on important species habitat areas.

2.2.4.1.2 Source Water Suitability

There are no additional or specific AP1000 requirements or restrictions related to the drift effects of site source water including evaporation rate and concentrations of dissolved solids.

2.3 Socioeconomics Criteria

The siting, construction and operation of a nuclear power station can place stresses on the local labor supply, transportation facilities, and community services. An evaluation of suitability of nuclear power station sites should include an assessment of impacts of construction and operation, including transmission and transportation corridors, and potential problems relating to community services (e.g., schools, police and fire protection, water and sewage, and health facilities).

Incompatible land uses, referred to as "nearby hazardous land uses," are discussed in Section 3.1.1.4. The following sections discuss the socioeconomic and environmental justice criteria associated with construction and operation of a nuclear power facility.

2.3.1 Socioeconomic - Construction Related Effects

See Section 4, Table Item 29.4.

There are no additional or specific AP1000 requirements or restrictions related to construction workforce or other construction related socioeconomic effects.

2.3.2 Socioeconomics – Operation

The operation of a single AP1000 requires a labor force of about 300 skilled workers (including security personnel and an allowance for attrition) for the first plant and about 200 each for follow plants. If twins are paced on one site the first twin requires about 420 skilled workers (including security personnel and an allowance for attrition) and follow twins require about 320.

2.3.3 Environmental Justice

There are no additional or specific AP1000 requirements or restrictions related to environmental justice

2.3.4 Land Use

There are no additional or specific AP1000 requirements or restrictions related to land use. Land uses that are incompatible with nuclear power facilities because of the hazards they pose to safe operation are categorized as "nearby hazardous land uses;" these are discussed in Section 2.1.1.4.

2.4 Engineering and Cost-Related Criteria

This section addresses those criteria that are cost-sensitive. Consideration of these criteria allows important site-related cost differentials to be considered in the site selection process. Because of the amount of detailed design work incorporated into the AP1000 design, cost estimates for it should be considered relatively reliable. This is due to the amount of reusable design created for AP600 and the resulting detailed bill of material developed during the design phase.

Cost estimates specified in these criteria should be developed in constant-year dollars, taking into account timing of each expense and a consistent discount rate. For example, a "present value" for operational costs such as water pumping and transmission losses should be developed so these costs can be directly compared with construction costs. All costs should be discounted to a single year.

2.4.1 Health and Safety Related Criteria

A number of these issues are also addressed in Section 3.1 and from a site suitability perspective, it may be helpful to revisit these evaluations as part of the development of the Engineering and Cost-Related criteria. Correlation with the health and safety utility functions may be helpful in evaluating cost.

2.4.1.1 Water Supply

There are no additional or specific AP1000 requirements or restrictions related to the cost of water supply. The analysis in this section addresses the costs associated with supplying the facility water requirements, in light of future, competitive, non-facility consumption rates.

2.4.1.2 Pumping Distance

There are no additional or specific AP1000 requirements or restrictions related to the cost of constructing pumping stations and infrastructure developments necessary to transport water from the source to the site.

2.4.1.3 Flooding

Flooding was initially treated in Section 2.1.1.3. The site storm drain system should be adequate to remove expected precipitation without flooding. There are no additional or specific AP1000 requirements or restrictions related to the cost of flooding protection.

2.4.1.4 Vibratory Ground Motion

For the AP1000, site cost increments that are a function of Peak Ground Acceleration do not exist as a result of standardization. There may a cost associated with site soil preparation for foundations of non-safety-related buildings or construction load paths.

2.4.1.5 Soil Stability

Soil stability was initially treated in Section 2.1.1.1.4 from the standpoint of soil properties and their relationship to the suitability of foundation conditions. For this criterion, the applicant should estimate the cost of site-specific foundation design features and associated construction requirements that might arise from soil conditions (e.g., slope stability).

2.4.1.6 Industrial Site Remediation

The purpose of this criterion is to capture costs associated with any environmental cleanup activities, that may be required at industrial sites before they can be developed for a nuclear power facility. There are no additional or specific AP1000 requirements or restrictions related to the cost of remediation.

2.4.2 Transportation or Transmission-Related Criteria

AP1000 has been designed to allow shipment by rail. It is preferable to ship larger units (assembled from the rail shippable units) by barge. An access and transportation plan will be required for each site to optimize the balance between offsite fabrication, shipping and onsite assembly. See Section 4, Table Item 29.1.

2.4.2.1 Railroad Access

See 2.4.2 above. An adequate railroad spur is recommended, but not required.

2.4.2.2 Highway Access

There are no additional or specific AP1000 requirements or restrictions related to highway access.

2.4.2.3 Barge Access

See 2.4.2 above. Adequate barge and load handling facilities will be required if barge delivery is appropriate for the site in question.

2.4.2.4 Transmission Cost and Market Price Differentials

2.4.2.4.1 Transmission Construction

AP1000 has no requirement for redundant connections to transmission grids. There are no additional or specific AP1000 requirements or restrictions related to transmission.

2.4.2.4.2 Electricity Market Price Differentials

There are no additional or specific AP1000 requirements or restrictions related to electricity market price differentials.

2.4.3 Criteria Related to Land Use and Site Preparation

2.4.3.1 Topography

The standard AP1000 design is based upon a relatively level site. Site plot plans for a variety of circulating water supply options are shown on AP1000 drawings APP-0000-X2-010 through APP-0000-X2-022. The standard AP1000 plot plans showing construction laydown, access and assembly areas are AP1000 drawings APP-0000-X2-810 through APP-0000-X2-822. The costs associated with any topographic features that would translate into site-specific differences in site preparation costs. For example, extensive cutting and filling, grading, and blasting could be factors that differentiate among sites.

2.4.3.2 Land Rights

There are no additional or specific AP1000 requirements or restrictions related to land rights.

2.4.3.3 Labor Rates

A significant portion of AP1000 can be fabricated in a shop or shipyard. This reduces the expected amount of site labor for a plant of this type and size. The impact of this construction approach may require negotiations with impacted labor unions both at the site and at the fabrication factories.

3
ADDITIONAL DETAIL SITE INTERFACES

3.1 Security Criteria

The AP1000 Design Certification is based upon the existence of an adequate site boundary security system. There are no additional or specific AP1000 requirements or restrictions related to land rights.

3.2 Grounding and Lightning Criteria

The AP1000 Design Certification is based upon the existence of an adequate station grounding system and a connection between it and the lightning protection system. There are no additional or specific AP1000 requirements or restrictions.

3.3 Raw Water Criteria

The AP1000 raw water treatment system will be based upon an adequate supply of surface water, clear well water or municipal water.

3.4 Detail Site Interface Dimensions

These AP1000 documents define detailed site interface dimensions.

AP1000 Document Number	Document Title
APP-0000-X2-010	AP1000 Single Unit Site Plot Plan Plant with Pumphouse
APP-0000-X2-011	AP1000 Single Unit Site Plot Plan Plant with Cooling Tower
APP-0000-X2-020	AP1000 Twin Unit Site Plot Plan with Separate Pumphouses
APP-0000-X2-021	AP1000 Twin Unit Site Plot Plan with Common Pumphouse
APP-0000-X2-022	AP1000 Twin Unit Site Plot Plan with Cooling Tower
APP-0000-X2-810	AP1000 Single Unit Construction Plot Plan Plant with Pumphouse

APP-0000-X2-811	AP1000 Single Unit Construction Plot Plan Plant with Cooling Tower
APP-0000-X2-820	AP1000 Twin Unit Construction Plot Plan with Separate Pumphouses
APP-0000-X2-821	AP1000 Twin Unit Construction Plot Plan with Common Pumphouse
APP-0000-X2-822	AP1000 Twin Unit Construction Plot Plan with Cooling Towers
APP-0000-X4-901	AP1000 Plant Grid Coordinates & Column Line Identification View A-A
APP-0000-X4-902	AP1000 Plant Grid Coordinates & Column Line Identification Views B-B
APP-0000-X4-903	AP1000 Plant Grid Coordinates & Column Line Identification Views C-C
APP-0030-X4-001	AP1000 Plant Grid Coordinates & Column Line Identification Plan
APP-0031-X4-001	Yard Arrangement Fuel Tank Storage/Transfer Facility
APP-0031-X4-002	Plant Grid Coordinates for Fuel Tank Storage/Transfer Facility Plan
APP-0035-X4-001	Yard Arrangement CWS Cooling Tower
APP-00350-X4-001	Yard Arrangement CWS Cooling Tower Area
APP-0036-X4-001	Yard Arrangement Hydrogen Storage Tank Area
APP-00360-X4-001	Yard Arrangement Hydrogen Storage Tank Area
APP-0070-X4-001	AP1000 Plant Grid Coordinates & Roof Plan

3.5 Detail Fuel and Waste Shipping Information

3.5.1 Information on Annual Fuel Requirements

3.5.1.1 Standard Technical Configuration

Reactor Power	3400 MW_t
Plant Power	1117 – 1150 MW_e
Number of Plants per Unit	1

3.5.1.2 Expected Fuel Loading

Initial Core Fuel Loading	84.5 MTU

Annual Average Fuel Loading 24.4 MTU

3.5.1.3 Average Fuel Enrichment (initial load)

Region 1	2.35 weight % U-235
Region 2	3.40 weight % U-235
Region 3	4.45 weight % U-235

3.5.1.4 Fuel Form

Total mass	1730 lb/assembly
Uranium mass	0.5383 MTU/assembly
Volume (FA envelope)	13404.3 in^3
Outside Dimensions	8.426x8.426x188.8 in
Number of Assemblies (Initial)	157
Number of Assemblies (Reload)	68 on 18 month cycle

3.5.1.5 Fuel Materials

Fuel	211,588 lb UO_2
Structure and Cladding	43,105 lb Zircaloy or ZIRLO™
	270 lb Alloy 718 (top & bottom Grids for 157 assemblies)

3.5.1.6 Expected Typical Transport Truck

3.5.1.7 Fresh Fuel Transport Containers

Capacity	2 assemblies per container
Shipping	6 containers per truck

3.5.1.8 Fuel reload data:

Cycle Length	18 months - 520 EFPD @ 3400 MWT
Capacity Factor	95% including refueling outage
Reload fuel requirement	68 Fuel Assemblies
Average Enrichment	4.51 w/o U235

3.5.1.9 Spent fuel data:

At 5 years decay, the average spent fuel assembly curie content:

Actinides	8.506E+04 curies
Fission Products	4.450E+05 curies
Total	5.301E+05 curies

3.5.1.10 Spent fuel data:

At 5 years decay, the average spent fuel assembly curie content:

Actinides	8.506E+04 curies
Fission Products	4.450E+05 curies
Total	5.301E+05 curies

3.5.1.11 Spent Fuel Shipping Information

Quantity of spent fuel (MTU):	
Truck Cask	To be provided later
Rail Car Cask	To be provided later

3.2.1.12 Average Fuel Burnup

Expected	21000 MWD/MTU (3400 MWt x 520 efpd / 84.5 MTU)
Design	60000 MWD/MTU

3.2.1.13 Estimate of Decay Heat in watts per MTU after 5 years of decay

While we use ORIGEN, we have not used it for decay heat calculation for AP1000. We therefore have estimated decay heat based on ANS 1979 standards, with 0 sigma margin, at five years to be 1.127E-4 watts/watt. With core power of 3400 MW and core loading of 84.5 MTU, the estimated specific decay heat for AP1000 is 4530 watts/MTU.

3.5.1.14 Estimates of spent fuel inventories and radioactivity

ORIGEN results for spent fuel inventories and radioactivity are addressed by AP1000 document APP-SSAR-GS2-496. This is based on one burned AP1000 assembly, decayed to 5 years. (Note that ORIGEN was run assuming a core loading of 83.6 MTU.) The 5 year decay data is in the last column (as label indicates). Also note that the inventory units are total Curies (based on 532337.6 grams for an assembly).

3.5.1 Information on Expected Low Level Waste Production

3.5.2.1 LLW Production

Volume	1964 cubic feet per year (average, as shipped)
Activity	1830 curies per year (average, as shipped)

3.5.2.2 LLW from Decommissioning

No AP1000 specific estimate has been made. Information from Sizewell indicates 6200 cubic meters of LLW from decommissioning. The AP1000 value should be significantly less (maybe half) considering the design differences.

4

OTHER PLANT PARAMETER ENVELOPES

Structure, System, Component			(Value)
1.	Structures		
	1.1	Foundation Embedment	39' 6" to bottom of Basemat from Plant Grade
	1.2	Height	234' 0"
	1.4	Precipitation (for Roof Design)	
		1.4.1 Maximum Rainfall Rate	19.4 in/hr (6.3 in/5 min)
		1.4.2 Snow Load	75 lbs/sq ft on ground with exposure factor of 1.0 and importance factor of 1.2 (safety) and 1.0 (non-safety)
	1.5	Safe Shutdown Earthquake (SSE)	
		1.5.1 Design Response Spectra	modified Regulatory Guide 1.60
		1.5.2 Peak Ground Acceleration	0.30g at base of foundation or at grade
		1.5.3 Time History	Envelope SSE Resp Spectra
		1.5.4 Fault Displacement Potential	None
	1.8	Site Water Level (Allowable)	
		1.8.1 *Maximum Flood (or Tsunami)*	*Plant grade or plant elevation 100 feet.* See Section 2.1.1.3
		1.8.2 Maximum Ground Water	Less than 98 feet with plant grade defined at 100 feet.
	1.9	Soil Properties Design Bases	
		1.9.1 Liquefaction	None. See Section 2.1.1.1.5
		1.9.2 Minimum Bearing Capacity (Static)	Greater than or equal to 8,000 pounds per square foot over the footprint of the nuclear island at its excavation depth. See Section 2.1.1.1.5
		1.9.3 Minimum Shear Wave Velocity	Greater than or equal to 1000 ft/sec based on low strain best estimate soil properties. See Section 2.1.1.1.5
	1.11	Tornado (Design Bases)	
		1.11.1 Maximum Pressure Drop	2.0 PSID
		1.11.2 Maximum Rotational Speed	240 MPH
		1.11.3 Maximum Translational Speed	60 MPH
		1.11.4 Maximum Wind Speed	300 MPH

Structure, System, Component			(Value)
	1.11.5	Missile Spectra	A 4000 pound automobile at 105 mph horizontal and 74 mph vertical, a 275 pound 8 inch shell at 105 mph horizontal and 74 mph vertical, and a 1 inch diameter steel ball at 105 mph horizontal and 105 mph vertical.
	1.11.6	Radius of Maximum Rotational Speed	150 ft
	1.11.7	Rate of Pressure Drop	1.2 psi/sec
1.12	Wind		
	1.12.1	Basic Wind Speed	145 MPH. See Section 2.1.1.5.1
	1.12.2	Importance Factors	See Section 2.1.1.5.1
2.	*Normal Plant Heat Sink*		*Also see discussion in Section* 2.1.1.2
2.1	Ambient Air Requirements		
	2.1.1	Normal Shutdown Max Ambient Temp (1% Exceedance)	100 °F db/77 °F wb coincident
	2.1.2	Normal Shutdown Max Wet Bulb Temp (1% Exceedance)	80 °F wb non-coincident
	2.1.3	*Normal Shutdown Min Ambient* Temp (1% Exceedance)	-10 °F
	2.1.5	Rx Thermal Power Max Ambient Temp (0% Exceedance)	115 °F db/80 °F wb coincident
	2.1.6	Rx Thermal Power Max Wet Bulb Temp (0% Exceedance)	81 °F wb non-coincident
	2.1.7	Rx Thermal Power Min Ambient Temp (0% Exceedance)	-40 °F
2.2	Blowdown Pond Acreage		24 hr blowdown
2.3	Condenser/Heat Exchanger Duty		7.54E9 Btu/hr
2.6	Maximum Inlet Temp Condenser/Heat Exchanger		91 °F
2.7	Mech Draft Cooling Towers		Also see discussion in Section 2.1.1.2
	2.7.1	Acreage	25 acres
	2.7.3	Approach Temperature	10 °F
	2.7.4	Blowdown Constituents and Concentrations	See Section 2.2.3.1.3
	2.7.5	Blowdown Flow Rate (Circ and Service Water)	6000 (24,500 max) gpm
	2.7.6	Blowdown Temperature (Circ and Service Water)	100 °F
	2.7.7	Cooling Tower Temperature Range	25.2 °F
	2.7.8	Cooling Water Flow Rate	600,000 gpm (nominal)
	2.7.9	Cycles of Concentration	4

Structure, System, Component			(Value)
	2.7.10	Evaporation Rate (Circulating and Service Water)	15,000 gpm
	2.7.12	Heat Rejection Rate	7.54E9 Btu/hr
	2.7.13	Height	60 ft
	2.7.15	Makeup Flow Rate (Circulating and Service Water)	21,000 gpm
	2.7.16	Maximum Consumption of Raw Water (Circulating and Service Water)	30,000 gpm
	2.7.17	Monthly Average Consumption of Raw Water (Circulating and Service Water)	21,000 gpm
	2.7.18	Noise	55 dba at 1000 ft
	2.7.22	Stored Water Volume	7,000,000 gal
2.8	Natural Draft Cooling Towers		Also see discussion in Section 2.1.1.2
	2.8.1	Acreage	2.3 acres without basin
	2.8.2	Approach Temperature	10 °F
	2.8.3	Blowdown Constituents and Concentrations	See Section 2.2.3.1.3
	2.8.4	Blowdown Flow Rate (Circ and Service Water)	6,000 (24,500 max) gpm
	2.8.5	Blowdown Temperature (Circ and Service Water)	100 °F
	2.8.6	Cooling Tower Temperature Range	25.2 °F
	2.8.7	Cooling Water Flow Rate	600,000 gpm
	2.8.8	Cycles of Concentration	4
	2.8.9	Evaporation Rate (Circulating and Service Water)	15,000 gpm
	2.8.11	Heat Rejection Rate	7.54E9 Btu/hr
	2.8.12	Height	500 ft
	2.8.14	Makeup Flow Rate (Circulating and Service Water)	21,000 gpm
	2.8.15	Maximum Consumption of Raw Water (Circulating and Service Water)	30,000 gpm
	2.8.16	Monthly Average Consumption of Raw Water (Circulating and Service Water)	21,000 gpm
	2.8.17	Noise	55 dba at 1000 ft
	2.8.20	Stored Water Volume	5,500,000 gal

Structure, System, Component			(Value)
2.9	Once-Through Cooling		Also see discussion in Section 2.1.1.2
	2.9.1	Cooling Water Discharge Temperature	88 °F
	2.9.2	Cooling Water Flow Rate	850,000 gpm
	2.9.3	Cooling Water Temperature Rise	18 °F
	2.9.4	Evaporation Rate	14,500 gpm
	2.9.5	Heat Rejection Rate	7.76E9 Btu/hr. See Sections 2.1.1.2.
2.10	Ponds		Also see discussion in Section 2.1.1.2
	2.10.1	Acreage	Site Specific
	2.10.2	Blowdown Constituents and Concentrations	See Section 2.2.3.1.3
	2.10.3	Blowdown Flow Rate	Site Specific
	2.10.4	Blowdown Temperature	Site Specific
	2.10.5	Cooling Pond Temperature Range	Site Specific
	2.10.6	Cooling Water Flow Rate	Site Specific
	2.10.7	Cycles of Concentration	Site Specific
	2.10.8	Evaporation Rate	Site Specific
	2.10.9	Heat Rejection Rate	7.54E9 Btu/hr
	2.10.10	Makeup Flow Rate	Site Specific
	2.10.11	Maximum Consumption of Raw Water	Site Specific
	2.10.12	Monthly Average Consumption of Raw Water	Site Specific
	2.10.13	Stored Water Volume	Site Specific
3.	Ultimate Heat Sink		None. See Section 2.1.1.2
4.	Containment Heat Removal System (Post-Accident)		
4.1	Ambient Air Requirements		
	4.1.1	Maximum Ambient Air Temperature (0% Exceedance)	115 °F db/80 °F wb
	4.1.2	Minimum Ambient Temperature (0% Exceedance)	-40 °F
5.	Potable Water/Sanitary Waste System		
5.2	Discharge to Site Water Bodies		
	5.2.1	Flow Rate	30,000 gal/day normal (single unit) 42,000 gal/day normal (twin unit) 100,000 gal/day (max)
5.4	Raw Water Requirements		
	5.4.1	Maximum Use	100,000 gal/day
	5.4.2	Monthly Average Use	30,000 gal/day normal (single unit) 42,000 gal/day normal (twin unit)

Structure, System, Component			(Value)
6.	Demineralized Water System		
	6.2	Discharge to Site Water Bodies	25 expected (70 max) gpm
		6.2.1 Flow Rate	
	6.4	Raw Water Requirements	
		6.4.1 Maximum Use	200 gpm
		6.4.2 Monthly Average Use	75 gpm
7.	Fire Protection System		
	7.1	Raw Water Requirements	
		7.1.1 Maximum Use	625 gpm
		7.1.2 Monthly Average Use	225,000 gal/mo (5 gpm)
		7.1.4 Stored Water Volume	775,000 gallons
8.	Miscellaneous Drain		
	8.2	Discharge to Site Water Bodies	
		8.2.1 Flow Rate	25 (50) gpm
9.	Unit Vent/Airborne Effluent Release Point		
	9.1	Atmospheric Dispersion (CHI/Q) (Accident)	
		9.1.1 0.5 mile, 0-2 hr	0.61E-3 sec/m^3
		9.1.2 2 mile, 0-8 hr	1.35E-4 sec/m^3
		9.1.5 2 mile, 8-24 hour	1.0E-4 sec/m^3
		9.1.3 2 mile, 1-4 day	5.4E-5 sec/m^3
		9.1.4 2 mile, 4-30 day	2.2E-5 sec/m^3
	9.2	Atmospheric Dispersion (CHI/Q) (Annual Average)	Site Boundary 2.0E-5 sec/m^3
	9.3	Containment Leakage Rate	0.5%/day (+35 scfh/ms line BWR only)
	9.5	Dose Consequences	
		9.5.1 Normal	10CFR20, 10CFR50 APP I
		9.5.2 Post-Accident	10CFR -20, -50 APP I, -100
		9.5.3 Severe Accidents	25 rem wb in 24 hr @ 0.5 mi <1E-6/rx-yr
	9.6	Release Point	
		9.6.1 Configuration (Horiz vs Vert)	Vertical
		9.6.3 Elevation (Normal)	160'
		9.6.4 Elevation (Post Accident)	Ground Level
		9.6.6 Minimum Distance to Site Boundary	0.5 mile
		9.6.7 Temperature	50-120 °F (estimate)
		9.6.8 Volumetric Flow Rate	171, 500 SCFM (Norm)
	9.7	Source Term	
		9.7.1 Gaseous (Normal)	See Table 4
		9.7.2 Gaseous (Post-Accident)	See Chap 15 Tables - Reg Guide 1.70
		9.7.4 Tritium	350 ci/yr

Structure, System, Component				(Value)
10.	Liquid Radwaste System			
	10.1	Dose Consequences		
		10.1.1	Normal	10 CFR 50, Appendix I 10 CFR 20
		10.1.2	Post-Accident	10 CFR 20 10 CFR 100
	10.2	Release Point		
		10.2.1	Flow Rate	1.4 gpm average
	10.3	Source Term		
		10.3.1	Liquid	0.26 ci/yr, see Table 5
		10.3.2	Tritium	1010 ci/yr
11.	*Gaseous Radwaste System*			
12.	Solid Radwaste System			
	12.1	Acreage		
		12.1.1	Low Level Radwaste Storage	2 years @ expected generation rate 1 year @ maximum generation rate
	12.2	Solid Radwaste		
		12.2.1	Activity	1830 ci/yr
		12.2.2	Principal Radionuclides	See Table 1
		12.2.3	Volume	1964 cu ft/yr avg expected shipped
13.	Reactor Coolant System			
14.	RCS Cleanup System			
15.	CVCS Letdown Subsystem			
16.	CVCS Purification Subsystem			
17.	CVCS Shim/Bleed Subsystem			
18.	Spent Fuel Storage			
	18.3	*Spent Fuel Dry Storage*		
		18.3.1	Acreage	15 acres
		18.3.2	Minimum Distance to Nearest Residence	3500 ft
		18.3.3	Minimum Distance to Power Block	1500-2200 ft
		18.3.4	Storage Capacity	60 years dry storage
19.	Steam Generator Blowdown System			
20.	Standby-Gas Treatment System			
21.	Auxiliary Boiler System			
	21.1	Exhaust Elevation		150 ft above plant grade
	21.2	*Flue Gas Effluents*		*See Table 2*
	21.3	Fuel		
		21.3.2	Type	No. 2
	21.4	Heat Input Rate (Btu/hr)		156,000,000 Btu/hr
22.	Condensate Cleanup System			

Structure, System, Component			(Value)
23. Gas Storage System			
24. Heating, Ventilation and Air Conditioning System			
24.1	Ambient Air Requirements		
	24.1.2	Non-safety HVAC max ambient temp (1% Exceedance)	100 °F db/77 °F wb coincident
	24.1.3	Non-safety HVAC min ambient temp (1% Exceedance)	-10 °F
	24.1.4	Safety HVAC max ambient temp (0% Exceedance)	115 °F db/80 °F wb coincident
	24.1.5	Safety HVAC min ambient temp (0% Exceedance)	-40 °F
	24.1.6	Vent System max ambient temp (5% Exceedance)	95 °F dry bulb/77 °F coincident wet bulb
		(1% Exceedance)	100 °F db/77 °F wb coincident
	24.1.7	Vent System min ambient temp (5% Exceedance)	-5 °F
		(1% Exceedance)	-10 °F
25. Onsite/Offsite Electrical Power System			
25.1	Acreage		
	25.1.1	Switchyard	12 acres
25.3	Duty Cycles		35 peak-to-peak per day
26. Standby Power System			
26.1	Diesel Capacity (kW)		2 x 4000 kW
26.2	Diesel Exhaust Elevation		50 ft
26.3	Diesel Flue Gas Effluents		See Table 3
26.4	Diesel Fuel		
	26.4.1	Resupply Time	7 days
	26.4.2	Type	No. 2 Oil Per ASTM D 975
26.5	Diesel Noise		55 dba at 1000 ft
26.6	Gas-Turbine Capacity (kW)		None
26.7	Gas-Turbine Exhaust Elevation		None
26.8	Gas-Turbine Flue Gas Effluents		None
26.9	Gas-Turbine Fuel		None
	26.9.2	Type	
26.10	Gas-Turbine Noise		None
27. Severe Accident Features			
28. Plant Characteristics			
28.1	Access Routes		
	28.1.3	Heavy Haul Routes	4 acres
	28.1.5	Spent Fuel Cask Weight	100 tons
28.2	Acreage		27 acres

Structure, System, Component			(Value)
	28.4	Megawatts - Thermal	3415 MWt
	28.5	Plant Design Life	60 years
	28.6	Plant Population	
		28.6.1 Operation	About 300. See Section 2.3.2
		28.6.2 Refueling	*1000 people*
	28.9	Station Capacity Factor	93%
29.	Construction		
	29.1	Access Routes	
		29.1.1 *Construction Module Dimensions*	
		Shipping Dimensions (ft)	
		Reactor Vessel	22 (Dia) x 34 (L)
		Steam Generator	20 (Dia) x 80 (L)
		Turbine Rotor	*18 (Dia)* x 29 (L)
		Generator Stator	18 (Dia) x 40 (L)
		Modules by Rail	12(H) x 12(W) x 80(L)
		Modules by Barge	90(H) x 82(W) x 93(L) or
			130(Dia) x 51(H)
		29.1.2 Heaviest Construction Shipment	
		Heaviest Shipment Weight	
		Reactor Vessel	652,000 lbs
		Steam Generator	1,464,000 lbs
		Turbine Rotor	350,000 lbs
		Generator Stator	1,020,000 lbs
		Modules by Rail	160,000 lbs.
		Modules by Barge	1,900,000 lbs.
	29.2	Acreage	
		29.2.1 Laydown Area	10 acres
		29.2.2 Temporary Construction Facilities	2.36 acres
	29.3	*Construction*	
		29.3.6 Noise	76-101 db @ 50 ft
	29.4	Plant Population	
		29.4.1 Construction	1200 monthly maximum
	29.5	Site Preparation Duration	18 months with construction and test of 4 to 5 years

Table 1 Principal Radionuclides in Solid Radwaste[1]

Radionuclide	PWR (Ci/yr)
Fe-55	311.488
Fe-59	--
Co-60	287.256
Mn-54	22.428
Cr-51	0.29151
C0-58	62.289
Ni-63	316.386
H-3	1.6057
C-14	0.285
Nb-95	0.3233
Ag-110m	0.04604
Zr-95	0.07163
Ba-140	0.08725
Pu-241	0.114027
La-140	0.04011
Other	29.982
Total (rounded to nearest hundred)	1100

Notes:
(1) See PPE Section 12.2.2

Table 2 Yearly Emissions Auxiliary Boilers[1]

Pollutant Discharged[2]	AP600 Quantity (lbs)
Particulates	17,250
Sulfur oxides	51,750
Carbon monoxide	-
Hydrocarbons	50,100
Nitrogen oxides	

Notes:
(1) See PPE Section 21.2.
(2) Emissions are based on 30 days/year operation for each of the generators.

Table 3 Yearly Emissions From Diesel Generators (DG)[1]

Pollutant Discharged[2]	Two 4000 kW Standby DGs Quantity[2] (lbs)	Two 35 kW Ancillary DGs Quantity[2] (lbs)
Particulates	<800	<10
Sulfur Oxides	<2,500	<5
Carbon Monoxide	<1,000	<30
Hydrocarbons	<600	<11
Nitrogen oxides	<12,000	<140

Notes:
(1) See PPE Section 26.3.
(2) Emissions are based on 4 hrs/month operation for each of the generators.

Table 4
EXPECTED ANNUAL AVERAGE RELEASE OF AIRBORNE RADIONUCLIDES AS DETERMINED BY THE PWR-GALE CODE, REVISION 1 (RELEASE RATES IN Ci/yr)

Noble Gases(1)	Waste Gas System	Building/Area Ventilation			Condenser Air Removal System	Total
		Cont.	Auxiliary Building	Turbine Building		
Kr-85m	0.	3.0E+01	4.0E+00	0.	2.0E+00	3.6E+01
Kr-85	1.65E+02	2.4E+03	2.9E+01	0.	1.4E+01	4.1E+03
Kr-87	0.	9.0E+00	4.0E+00	0.	2.0E+00	1.5E+01
Kr-88	0.	3.4E+01	8.0E+00	0.	4.0E+00	4.6E+01
Xe-131m	1.42E+02	1.6E+03	2.3E+01	0.	1.1E+01	1.8E+03
Xe-133m	0.	8.5E+01	2.0E+00	0.	0.	8.7E+01
Xe-133	3.0E+01	4.5E+03	7.6E+01	0.	3.6E+01	4.6E+03
Xe-135m	0.	2.0E+00	3.0E+00	0.	2.0E+00	7.0E+00
Xe-135	0.	3.0E+02	2.3E+01	0.	1.1E+01	3.3E+02
Xe-138	0.	1.0E+00.	3.0E+00	0.	2.0E+00	6.0E+00
					Total	1.1E+04
Additionally:						
H-3 released via gaseous pathway						350
C-14 released via gaseous pathway						7.3
Ar-41 released via containment vent						34

Iodines(1)	Fuel Handling Area(2)	Building/Area Ventilation			Condenser Air Removal System	Total
		Cont.	Auxiliary Building	Turbine Building		
I-131	4.5E-03	2.3E-03	1.1E-01	0.	0.	1.2E-01
I-133	1.6E-02	5.5E-03	3.8E-01	2.0E-04	0.	4.0E-01

Radionuclide(1)	Waste Gas System	Building/Area Ventilation			Total
		Cont.	Auxiliary Building	Fuel Handling Area(2)	
Cr-51	1.4E-05	9.2E-05	3.2E-04	1.8E-04	6.1E-04
Mn-54	2.1E-06	5.3E-05	7.8E-05	3.0E-04	4.3E-04

Co-57	0.	8.2E-06	0.	0.	8.2E-06
Co-58	8.7E-06	2.5E-04	1.9E-03	2.1E-02	2.3E-02
C0-60	1.4E-05	2.6E-05	5.1E-04	8.2E-03	8.7E-03
Fe-59	1.8E-06	2.7E-05	5.0E-05	0.	7.9E-05
Sr-89	4.4E-05	1.3E-04	7.5E-04	2.1E-03	3.0E-03
Sr-90	1.7E-05	5.2E-05	2.9E-04	8.0E-04	1.2E-03
Zr-95	4.8E-06	0.	1.0E-03	3.6E-06	1.0E-03
Nb-95	3.7E-06	1.8E-05	3.0E-05	2.4E-03	2.5E-03
Ru-103	3.2E-06	1.6E-05	2.3E-05	3.8E-05	8.0E-05
Ru-106	2.7E-06	0.	6.0E-06	6.9E-05	7.8E-05
Sb-125	0.	0.	3.9E-06	5.7E-05	6.1E-05
Cs-134	3.3E-05	2.5E-05	5.4E-04	1.7E-03	2.3E-03
Cs-136	5.3E-06	3.2E-05	4.8E-05	0.	8.5E-05
Cs-137	7.7E-05	5.5E-05	7.2E-04	2.7E-03	3.6E-03
Ba-140	2.3E-05	0.	4.0E-04	0.	4.2E-04
Ce-141	2.2E-06	1.3E-05	2.6E-05	4.4E-07	4.2E-05

Notes:

1. The appearance of 0. in the table indicates less than 1.0 Ci/yr for noble gas or less than 0.0001 Ci/yr for iodine. For particulates, release is not observed and assumed less than 1 percent of the total particulate releases.
2. The fuel handling area is within the auxiliary building but is considered separately.

Table 5
RELEASES TO DISCHARGE CANAL (CI/YR) CALCULATED BY GALE CODE

Nuclide	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Total Releases[1]
Corrosion and Activation Products					
Na-24	0.00053	0.0[2]	0.00008	0.00061	0.00163
Cr-51	0.00068	0.0	0.0	0.00070	0.00185
Mn-54	0.00048	0.0	0.0	0.00049	0.00130
Fe-55	0.00037	0.0	0.0	0.00037	0.00100
Fe-59	0.00008	0.0	0.0	0.00008	0.00020
Co-58	0.00125	0.0	0.00001	0.00126	0.00336
Co-60	0.00016	0.0	0.0	0.00017	0.00044
Zn-65	0.00015	0.0	0.0	0.00015	0.00041
W-187	0.00004	0.0	0.0	0.00005	0.00013
Np-239	0.00008	0.0	0.0	0.00009	0.00024
Fission Products					
Br-84	0.00001	0.0	0.0	0.00001	0.00002
Rb-88	0.00010	0.0	0.0	0.00010	0.00027
Sr-89	0.00004	0.0	0.0	0.00004	0.00010
Sr-90	0.0	0.0	0.0	0.0	0.00001
Sr-91	0.00001	0.0	0.0	0.00001	0.00002
Y-91m	0.0	0.0	0.0	0.00001	0.00001
Y-93	0.00003	0.0	0.0	0.00002	0.00009
Zr-95	0.00010	0.0	0.0	0.00005	0.00023
Nb-95	0.00009	0.0	0.0	0.00005	0.00021
Mo-99	0.00028	0.0	0.00001	0.00013	0.00057
Tc-99m	0.00027	0.0	0.00001	0.00013	0.00055
Ru-103	0.00183	0.00001	0.00002	0.00185	0.00493
Rh-103m	0.00183	0.00001	0.00002	0.00185	0.00493
Ru-106	0.02729	0.00011	0.00021	0.02761	0.07352
Rh-106	0.02729	0.00011	0.00021	0.02761	0.07352
Ag-110m	0.00039	0.0	0.0	0.00039	0.00105
Ag-110	0.00005	0.0	0.0	0.00005	0.00014
Te-129m	0.00004	0.0	0.0	0.00005	0.00012
Te-129	0.00006	0.0	0.0	0.00006	0.00015
Te-131m	0.00003	0.0	0.0	0.00003	0.00009
Te-131	0.00001	0.0	0.0	0.00001	0.00003
I-131	0.00512	0.00004	0.00015	0.00531	0.01413
Te-132	0.00009	0.0	0.0	0.00009	0.00024
I-132	0.00054	0.00001	0.00007	0.00062	0.00164
I-133	0.00211	0.00003	0.00038	0.00252	0.00670
I-134	0.00030	0.0	0.0	0.00031	0.00081
Cs-134	0.00370	0.00001	0.00002	0.00373	0.00993
I-135	0.00144	0.00002	0.00041	0.00187	0.00497

Table 5
RELEASES TO DISCHARGE CANAL (CI/YR) CALCULATED BY GALE CODE

Nuclide	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Total Releases(1)
Cs-136	0.00023	0.0	0.0	0.00024	0.00063
Cs-137	0.00496	0.00001	0.00003	0.00500	0.01332
Ba-137m	0.00464	0.00001	0.00002	0.00468	0.01245
Ba-140	0.00203	0.00001	0.00003	0.00207	0.00552
La-140	0.00272	0.00002	0.00005	0.00279	0.00743
Ce-141	0.00003	0.0	0.0	0.00004	0.00009
Ce-143	0.00006	0.0	0.00001	0.00007	0.00019
Pr-143	0.00005	0.0	0.0	0.00005	0.00013
Ce-144	0.00117	0.0	0.00001	0.00119	0.00316
Pr-144	0.00117	0.0	0.00001	0.00119	0.00316
All others	0.00001	0.0	0.0	0.00001	0.00002
Total (except tritium)	0.09398	0.00043	0.00182	0.09623	0.25623
Tritium release		1010 curies per year			

Notes:
1. The release totals include an adjustment of 0.16 Ci/yr added by PWR-GALE code to account for anticipated operational occurrences such as operator errors that result in unplanned releases.
2. An entry of 0.0 indicates that the value is less than 10-5 Ci/yr.

5

SITE RELATED COMBINED LICENSE INFORMATION ITEMS

This section provides a listing of the Combined License (COL) information items identified in the AP1000 Design Control Document (DCD) that are site related. The AP1000 DCD (APP-GW-GL-700) includes identification of information items which must be provided to NRC during a COL application process. In addition to the site related items listed below there are items are related to additional detail in the plant design and to the COL applicant's organization information. It is important for a COL applicant to plan for the submittal of required site related COL information items and include planning for data acquisition in the Early Site Permit process. The following information items and their referenced DCD sections are site related and should be acknowledged during Early Site permit planning.

Item Number	Subject	DCD Subsection
2.1-1	Geography and Demography Combined License applicants referencing the AP1000 certified design will provide site-specific information related to site location and description, exclusion area authority and control, and population distribution. Site Information – Site-specific information on the site and its location will include political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area. Exclusion Area – Site-specific information on the exclusion area will include the size of the area and the exclusion area authority and control. Activity that may be permitted within the exclusion area will be included in the discussion. Population Distribution – Site-specific information will be included on population distribution.	2.1.1
2.2-1	Identification of Site-specific Potential Hazards Combined License applicants referencing the AP1000 certified design will provide site-specific information related to the identification of potential hazards within the site vicinity, including an evaluation of potential accidents and verify that the frequency of site-specific potential hazards is consistent with the criteria outlined in Section 2.2. The site-specific information will provide a review of aircraft hazards, information on nearby transportation routes, and information on potential industrial and military hazards.	2.2.1

2.3-1	**Regional Climatology**	2.3.6.1

Combined License applicants referencing the AP1000 certified designwill address site-specific information related to regionalclimatology.

2.3-2	**Local Meteorology**	2.3.6.2

Combined License applicants referencing the AP1000 certified design will address site-specific local meteorology information.

2.3-3	**Onsite Meteorological Measurements Program**	2.3.6.3

Combined License applicants referencing the AP1000 certified design will address the site-specific onsite meteorological measurements program.

2.3-4	**Short-Term Diffusion Estimates**	2.3.6.4

Combined License applicants referencing the AP1000 certified design will address the site-specific X/Q values specified in subsection 2.3.4. For a site selected that exceeds the boundingX/Q values, the Combined License applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10CFR Part 50.34 and control room operator dose limits given in General Design Criteria 19 using site-specific X/Q values. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plumespread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.

2.3-5	**Long-Term Diffusion Estimates**	2.3.6.5

Combined License applicants referencing the AP1000 certified design will address long-term diffusion estimates andX/Q values specified in subsection 2.3.5. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameter for atmospheric dispersion.

2.4-1	**Hydrological Description**	2.4.1.1

Combined License applicants referencing the AP1000 certified designwill describe major hydrologic features on or in the vicinity of the site including critical elevations of the nuclear island and access routes to the plant.

2.4-2	**Floods**	2.4.1.2

Combined License applicants referencing the AP1000 certified designwill address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.

- Probable Maximum Flood on Stream and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the probable maximum flood on streams and rivers.

- Dam Failures – Site-specific information on potential dam failures.
- Probable Maximum Surge and Seiche Flooding – Site-specific information on probable maximum surge and seiche flooding.
- Probable Maximum Tsunami Loading – Site-specific information on probable maximum tsunami loading.
- Flood Protection Requirements – Site-specific information on flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

No further action is required for sites within the bounds of the site parameter for flood level.

2.4-3 Cooling Water Supply 2.4.1.3

Combined License applicants will address the water supply sources to provide makeup water to the service water system cooling tower.

2.4-4 Groundwater 2.4.1.4

Combined License applicants referencing the AP1000 certified design will address site-specific information on groundwater. No further action is required for sites within the bounds of the site parameter for ground water.

2.4-5 Site Effects of Accidental Release of Liquid Effluents in Ground and Surface Water 2.4.1.5

Combined License applicants referencing the AP1000 certified design will address site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Effects of these releases on existing and known future use of surface water resources will also be addressed.

2.4-6 Flood Protection Emergency Operation Procedures 2.4.1.6

Combined License applicants referencing the AP1000 certified design will address any flood protection emergency procedures required to meet the site parameter for flood level.

2.5-1 Basic Geologic and Seismic Information 2.5.1

Combined License applicants referencing the AP1000 certified design will address the following site-specific geologic and seismic information:

- Regional and site physiography
- Geomorphology
- Stratigraphy
- Lithology

- Structural geology
- Tectonics

Seismicity

2.5-2 Site Seismic and Tectonic Characteristic Information 2.5.2.1

Combined License applicants referencing the AP1000 certified design will address the following site-specific information related to seismic and tectonic characteristics of the site and region:

Correlation of earthquake activity with geologic structure or tectonic provinces

Maximum earthquake potential

Seismic wave transmission characteristics of the site

Safe shutdown earthquake (SSE) ground response spectra

The Combined License applicant must demonstrate that the proposed site meets the following requirements:

The free field peak ground acceleration at the foundation level is less than or equal to a 0.30g safe shutdown earthquake.

The site design response spectra at the foundation level in the free-field are less than or equal to those given in Figures 3.7.1-1 and 3.7.1-2.

2.5-3 Surface Faulting 2.5.3

Combined License applicants referencing the AP1000 certified designwill address surface and subsurface geological and geophysical information including the potential for surface or near-surface faulting affecting the site.

2.5-4 Site and Structures 2.5.4.6.1

Site and Structures – Site-specific information regarding the underlying site conditions and geologic features will be addressed. This information will include site topographical features, as well as the locations of seismic Category I structures.

2.5-5 Properties of Underlying Materials 2.5.4.6.2

The Combined License applicant will establish the properties of the foundation soils to be within the range considered for design of the nuclear island basemat.

Properties of Underlying Materials – A determination of the static and dynamic engineering properties of foundation soils and rocks in the site area will be addressed. This information will include a discussion of the type, quantity, extent, and purpose of field explorations, as well as logs of borings and test pits. Results of field plate load tests, field permeability tests, and other special field tests (e.g., bore-hole extensometer or pressuremeter tests) will also be provided. Results of geophysical surveys will be presented in tables and profiles. Data will be provided pertaining to site-specific soil layers (including theirthicknesses, densities, moduli, and Poisson's ratios) between the basemat and the underlying rock stratum. Plot plans and profiles of site explorations will be provided.

Laboratory Investigations of Underlying Materials – Information about the number and type of laboratory tests and the location of samples used to investigate underlying materials will be provided. Discussion of the results of laboratory tests on disturbed and undisturbed soil and rock samples obtained from field investigations will be provided.

2.5-6 Excavation and Backfill 2.5.4.6.3

Excavation and Backfill – Information concerning the extent (horizontal and vertical) of seismic Category I excavations, fills, and slopes, if any will be addressed. The sources, quantities, and static and dynamic engineering properties of borrow materials will be described in the site-specific application. The compaction requirements, results of field compaction tests, and fill material properties (such as moisture content, density, permeability, compressibility, and gradation) will also be provided. Information will be provided concerning the specific soil retention system, for example, the soil nailing system, including the length and size of the soil nails, which is based on actual soil conditions and applied construction surcharge loads. Information will also be provided on the waterproofing system along the vertical face and the mudmat.

2.5-7 Ground Water Conditions 2.5.4.6.4

Ground Water Conditions – Groundwater conditions will be described relative to the foundation stability of the safety-related structures at the site. The soil properties of the various layers under possible groundwater conditions during the life of the plant will be compared to the range of values assumed in the standard design in Table 2-1 of the DCD.

2.5-8 Response of Soil and Rock to Dynamic Loading 2.5.4.6.5

Response of Soil and Rock to Dynamic Loading – The Combined License applicant will establish the dynamic characteristics of the soil and rock to be used in the soil structure interaction analyses and the foundation design for soil sites. For rock sites the dynamic characteristics will be compared to the assumptions made in the standard design regarding the variation of shear wave velocity and material damping.

2.5-9 Liquifaction Potential 2.5.4.6.6

Liquefaction Potential – Soils under and around seismic Category I structures will be evaluated for liquefaction potential for the site specific SSE ground motion. This should include justification of the selection of the soil properties, as well as the magnitude, duration, and number of excitation cycles of the earthquake used in the liquefaction potential evaluation (e.g., laboratory tests, field tests, and published data). Liquefaction potential will also be evaluated to address seismic margin.

2.5-10 Bearing Capacity 2.5.4.6.7

Bearing Capacity – The Combined License applicant will verify that the site-specific soil static bearing capacity is equal to or greater than the value documented in Table 2-1 of the DCD. The Combined License applicant will verify that the dynamic site-specific bearing capacity is equal or greater than the seismic bearing demand.

2.5-11 Earth Pressures 2.5.4.6.8

Earth Pressures – The Combined License applicant will describe the design for static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities using soil parameters as evaluated in previous subsections.

2.5-12	Static and Dynamic Stability of Facilities	2.5.4.6.10

Static and Dynamic Stability of Facilities – Soil characteristics affecting the stability of the nuclear island will be addressed including foundation rebound, settlement, and differential settlement.

2.5-14	Stability of Slopes	2.5.5

Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of soil and rock slopes, the failure of which could adversely affect the nuclear island.

2.5-15	Embankments and Dams	2.5.6

Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of embankments and dams, the failure of which could adversely affect the nuclear island.

3.3-1	Wind and Tornado Site Interface Criteria	3.3.3

Combined License applicants referencing the AP1000 certified design will address site interface criteria for wind and tornado.

3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3

The Combined License applicant will demonstrate that the site satisfies the interface requirements as described in Section 2.4 of the DCD. If these criteria cannot be satisfied because of site-specific flooding hazards, the Combined License applicant may propose protective measures as discussed in Section 2.4 of the DCD.

3.5-1	External Missile Protection Requirements	3.5.4

The Combined License applicant will demonstrate that the site satisfies the interface requirements provided in Section 2.2 of the DCD. This requires an evaluation for those external events that produce missiles that are more energetic than the tornado missiles postulated for design of the AP1000, or additional analyses of the AP1000 capability to handle the specific hazard.

3.7-1	Seismic Analysis of Dams	3.7.5.1

Combined License applicants referencing the AP1000 certified design will evaluate dams whose failure could affect the site interface flood level specified in subsection 2.4.1.2 of the DCD. The evaluation of the safety of existing and new dams will use the site-specific safe shutdown earthquake.

6.4-1	Local Toxic Gas Service and Monitoring	6.4.7

Combined License applicants referencing the AP1000 certified design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required. Regulatory Guides 1.78 and 1.95 address control room protection for toxic chemicals, and for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with the guidelines of Regulatory Guides 1.78 and 1.95 in order to meet the requirements of TMI

Action Plan Item III.D.3.4 and GDC 19.

Combined License applicants referencing the AP1000 certified design are responsible for verifying that procedures and training for control room habitability are consistent with the intent of Generic Issue 83 (see Section 1.9 of the DCD).

8.2-1 Offsite Electrical Power 8.2.5

Combined License applicants referencing the AP1000 certified design will address the design of the ac power transmission system and its testing and inspection plan.

8.2-2 Plant/Site Technical Interfaces 8.2.5

The Combined License applicant will address the technical interfaces for this nonsafety-related system listed in Table 1.8-1 and subsection 8.2.2. These technical interfaces include those for ac power requirements from offsite and the analysis of the offsite transmission system and the setting of protective devices.

8.3-1 Onsite (Grounding and Lightning) Electrical Power 8.3.3

Combined License applicants referencing the AP1000 certified design will address the design of grounding and lightning protection.
The Combined License applicant will establish plant procedures as required for:

- Clearing ground fault on the Class1E dc system
- Checking sulfated battery plates or other anomalous conditions through periodic inspections
- Battery maintenance and surveillance (for battery surveillance requirements, refer to DCD Chapter 16, Section 3.8)
- Periodic testing of penetration protective devices

Diesel generator operation, inspection, and maintenance in accordance with manufacturers' recommendations.

9.5-2 Fire Protection Analysis Information on Adjacent Structures 9.5.1.8

The Combined License applicant will address qualification requirements for individuals responsible for development of the fire protection program, training of firefighting personnel, administrative procedures and controls governing the fire protection program during plant operation, and fire protection system maintenance.

The Combined License applicant will provide site-specific fire protection analysis information for the yard area, the administration building, and for other outlying buildings consistent with Appendix 9A of the DCD.

The Combined License applicant will address BTP CMEB 9.5-1 issues identified in Table 9.5.1-1 of the DCD by the acronym "WA."

The Combined License applicant will address updating the list of NFPA exceptions after design certification, if necessary.

The Combined License applicant will provide an analysis that demonstrates that operator actions which minimize the probability of the potential for spurious ADS actuation as a result of a fire can be accomplished within 30 minutes following detection of the fire.

9.5-9 Cathodic Protection of External Tanks 9.5.4.7

Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

Combined License applicants referencing the AP1000 certified design will *address site-specific factors in the fuel oil storage tank installation specification* to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

10.4-1 Circulating Water Supply 10.4.12.1

The Combined License applicant will address the final configuration of the plant circulating water system including piping design pressure, the cooling tower or other site-specific heat sink.

As applicable, the Combined License applicant will address the acceptable Langelier or Stability Index range, the specific chemical selected for use in the CWS water chemistry control, pH adjuster, corrosion inhibitor, scale inhibitor, dispersant, algicide and biocide applications reflecting potential variations in site water chemistry and in micro macro biological lifeforms. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room compatibility is addressed in Section 6.4 of the DCD.

10.4-3 Potable Water Biocide 10.4.12.3

The Combined License applicant will address the specific biocide. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room compatibility is addressed in Section 6.4 of the DCD.

11.2-1 Liquid Radwaste Processing by Mobile Equipment 11.2.5.1

The Combined License applicant will discuss how any mobile or temporary equipment used for storing or processing liquid radwaste conforms to Regulatory Guide 1.143. For example, this includes discussion of equipment containing radioactive liquid radwaste in the nonseismic Radwaste Building.

11.2-2 Cost Benefit Analysis of Population Doses (Liquid) 11.2.5.2

The analysis performed to determine offsite dose due to liquid effluents is based upon the AP1000 generic site parameters included in Chapter 1 and Tables 11.2-5 and 11.2-6 of the DCD. The Combined License applicant will provide a site specific cost-benefit analysis to address the requirements of 10 CFR 50, Appendix I, regarding population doses due to liquid effluents.

11.2-4 Dilution and Control of Boric Acid Discharge 11.2.5.4

The Combined License applicant will determine the rate of discharge and the required dilution to maintain acceptable concentrations. Refer to Section 11.5 of the DCD for a discussion of the program to control releases.

The Combined License applicant will discuss the planned discharge flow rate for borated wastes and controls for limiting the boric acid concentration in the circulating water system blowdown.

11.3-1 Cost Benefit Analysis of Population Doses (Gas) 11.3.5.1

The analysis performed to determine offsite dose due to gaseous effluents is based upon the AP1000 generic site parameters included in Chapter 1 and Tables 11.3-1, 11.3-2 and 11.3-4 of the DCD. The Combined License applicant will provide a site specific cost-benefit analysis to demonstrate compliance with 10 CFR 50, Appendix I, regarding population doses due to gaseous effluents.

11.5-2 Effluent Monitoring and Sampling 11.5.7

The Combined License applicant will develop an offsite dose calculation manual that contains the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents. The Combined License applicant will address operational setpoints for the radiation monitors and address programs for monitoring and controlling the release of radioactive material to the environment, which eliminates the potential for unmonitored and uncontrolled release. The offsite dose calculation manual will include planned discharge flow rates.

The Combined License applicant is responsible for the site-specific and program aspects of the process and effluent monitoring and sampling per ANSI N13.1 and Regulatory Guides 1.21 and 4.15.

11.5-3 10 CFR 50, Appendix I 11.5.7

The Combined License applicant is responsible for addressing the 10 CFR 50, Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents.

13.3-2 Activation of Emergency Operations Facility 13.3.1

Combined License applicants referencing the AP1000 certified design will address emergency planning including post-72 hour actions and its communication interface.
Combined License applicants referencing the AP1000 certified design will address the activation of the emergency operations facility consistent with current operating practice and NUREG-0654/FEMA-REP-1 except for a loss of offsite power and loss of all onsite AC power. For this initiating condition, the Combined License applicant shall immediately activate the emergency *operations facility rather than bringing it to a standby status.*

To initially and continuously assess the course of an accident for emergency response purposes, Combined License applicants referencing the AP1000 certified design will address the capability for promptly obtaining and analyzing grab samples of reactor coolant and containment atmosphere and sump in accordance with the guidance of Item II.B.3 of NUREG-0737.

13.6-1 Security Plans, Organization and Testing 13.6.13.1

Combined License applicants referencing the AP1000 certified design will address site-specific information related to the security, contingency, and guard training plans. Those plans will include descriptions of the tests planned to show operational status, maintenance of the plant security system, the security organization, communication, and response requirements.

The Combined License applicant will develop the comprehensive physical security program which includes the security plan, contingency plan, and guard training plan. Each COL applicant will describe in its physical security plan how the requirements of 10 CFR Part 26 will be met. At least 60 days before loading fuel, the Combined License applicant will confirm that the security systems and programs described in its physical security plan, safeguards contingency plan, and training and qualification plan have achieved operational status and are available for the staff's inspection. Operational status means that the security systems and programs are functioning. The determination that operational status has been achieved will be based on tests conducted under realistic operating conditions of sufficient duration to demonstrate that:

the equipment is properly operating;

procedures have been developed, approved, and implemented; and

personnel responsibility for security operations and maintenance have been appropriately trained and have demonstrated their capability to perform their assigned duties and responsibilities.

13.6-3 Site-Specific Security System 13.6.13.3

Combined License applicants referencing the AP1000 certified design will address site-specific information related to the maintenance and testing of the plant security system including the intrusion detection and assessment system, the access control features specified in subsections 13.6.6, 13.6.7.2, and 13.6.7.3 of the DCD, and the vehicle barrier system. The Combined License applicant will address in its safeguards plans how the physical protection system will provide the protection stated in subsection 13.6.3.2 of the DCD.

14.4-5 Testing Interface Requirements 14.4.5

The combined license applicant is responsible for testing that may be required of structures and systems which are outside the scope of this design certification. Test Specifications and acceptance criteria are provided by the responsible design organizations as identified in subsection 14.2.3. The interfacing systems to be considered for testing are taken from Table1.8-1 and include as a minimum, the following:

- storm drains
- site specific seismic sensors
- offsite ac power systems
- circulating water heat sink
- raw and sanitary water systems
- individual equipment associated with the fire brigade
- portable personnel monitors and radiation survey instruments
- equipment associated with the physical security plan

Appendix R: The Global Nuclear Energy Partnership

Information about the Global Nuclear Energy Partnership (GNEP) is provided in the following pages (extracted from ref. [36]).

The Global Nuclear Energy Partnership

GNEP is a federal research and development program headed by DOE that is designed to "effectively address two of the great concerns that have historically been associated with nuclear power" and which have limited the growth of nuclear power: disposal of spent fuel and nuclear weapons proliferation (DOE 2006a). The vision for GNEP is that both of these challenges would be addressed by the development of "proliferation-resistant" nuclear fuel reprocessing technologies that will minimize nuclear waste streams (DOE 2006j, p.61). In addition, the U.S. and other members of the global partnership would launch a fuel leasing program to allow countries to access nuclear power without developing their own uranium enrichment and reprocessing facilities. As described by DOE, the key objectives of GNEP are as follows (GNEP 2007):

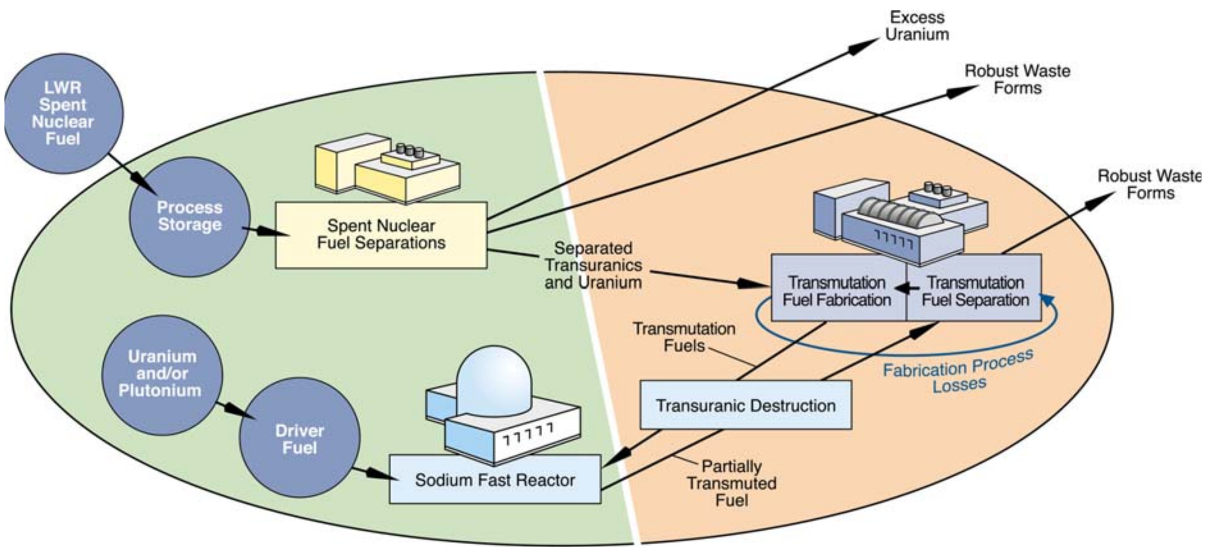
- Recycle nuclear fuel using new proliferation-resistant technologies to recover more energy and reduce waste
- Apply advanced technologies to the nuclear fuel cycle in order to reduce the risk of nuclear proliferation worldwide
- Encourage global economic prosperity and sustainable development by developing and promoting reliable, environmentally friendly energy supplies
- Reduce the use of fossil fuels

Achieving these goals will require a significant effort both domestically and internationally. The domestic components of GNEP will be initiated first, with the international components introduced only after the success of GNEP's domestic reprocessing vision has been proven.

Domestic Components of GNEP

The domestic goal of GNEP is to move from the once-through fuel cycle currently used throughout the U.S. to a closed fuel cycle that incorporates repeated reprocessing of spent fuel. According to the GNEP plan, spent fuel from current reactors would be sent to a reprocessing and recycling facility, where the uranium and plutonium would be separated out. These components would then be sent to a fuel fabrication facility, where they would be recycled into fuel for a new type of reactor, called an advanced burner reactor or a fast reactor. The fast reactor would be used to generate electricity and to convert (transmute) long-lived transuranic elements in the spent fuel into less radioactive elements, thereby reducing the need for disposal at an underground geological repository.[42] Spent fuel from the fast reactor would be reprocessed and recycled into additional fast reactor fuel, which would then be reprocessed and recycled into additional fast reactor fuel. Unlike the reprocessing currently being done in Europe, under the GNEP plan spent fuel would be repeatedly recycled until nearly all the transuranic elements are destroyed (DOE 2006l, p.8). (See Figure 11.)

Figure 11: Domestic Components of GNEP



Source: (DOE 2007h, p.23)

The GNEP program plans to develop new reprocessing technologies instead of relying on the PUREX technology already available and in use in Europe. The primary reason for not using the existing PUREX technology is that it is seen as a potential proliferation threat. New technologies that DOE is exploring may provide some measure of proliferation resistance. They may also provide other benefits, such as the easing of fuel repository requirements and the facilitation of advanced reactor fuel reprocessing. DOE's preferred technologies are shown in Table 11. The reprocessing technologies are further described in Figure 12 and Figure 13 and in Appendix A.

[42] DOE has expressed preference for the sodium-cooled fast reactor, and a pre-conceptual design has been completed for a 250 MW test reactor.

Table 11: New Technologies Required for GNEP

Technology Needed	Preferred Candidate
Proliferation-resistant technology to reprocess spent fuel from LWR reactors	UREX+; COEX also being considered (See Figure 12.)
Advanced burner reactor	Sodium cooled fast reactor
Fuel for the advanced burner reactor (transmutation fuel)	Initially, metal or oxide fuels
Technology to reprocess spent fuel from the advanced burner reactor	Pyrochemical processing ("pyroprocessing"—see Figure 13)

Source: (DOE 2006d, p.10; DOE 2007h. p.28)

DOE has moved forward with planning for these new technologies on two parallel fronts: 1) identifying potential locations to host a fuel reprocessing center and/or an advanced reactor facility, and 2) soliciting early input from industry, government laboratories, and research centers on how best to develop the needed technologies to make GNEP possible. Table 12 identifies 13 locations that have expressed an interest in hosting one or more of the facilities planned under GNEP.

Table 12: Possible Locations for GNEP Facilities

DOE Sites	Non-DOE Sites
Argonne National Laboratory (IL)	Atomic City, ID
Hanford (WA)	Barnwell, SC
Idaho National Laboratory (ID)	Hobbs, NM
Oak Ridge National Laboratory (TN)	Roswell, NM
Paducah Gaseous Diffusion Plant (KY)	Morris, IL
Portsmouth Gaseous Diffusion Plant (OH)	
Savannah River National Laboratory (SC)	
Los Alamos National Lab (N.M.)	

Source: (DOE 2007h, p.39)

DOE is currently in the process of developing a programmatic environmental impact statement for the domestic component of GNEP; a final environmental impact statement may be released in late spring 2008.

Figure 12: PUREX, UREX+, and COEX

The PUREX process is currently the only commercially viable method for reprocessing. The process separates spent fuel into uranium, plutonium, and a nitric acid waste solution containing highly radioactive fission products and other isotopes. A variety of low-level and intermediate-level wastes also result from the process.

The UREX+ (Uranium Extraction plus) reprocessing method is similar to the PUREX process in that it extracts explicit elements from the spent fuel rods via chemical reactions in an aqueous solution. UREX+ differs from PUREX in that more radiotoxic materials are extracted and plutonium is kept mixed with transuranic elements and is not extracted in a pure form. Also, UREX+ reprocessing can be used in conjunction with a fast reactor to allow for repeated reprocessing cycles.

One benefit of the UREX+ process relative to the PUREX process is the extraction of cesium and strontium from the waste stream. Cesium and strontium are initially highly radioactive, and their presence in the waste stream increases the volume requirements for a waste repository. Separating these elements from the waste stream would thus allow for the storage of a much larger volume of spent fuel in a repository. As cesium and strontium lose their radioactivity relatively quickly (after about 300 years), they could theoretically be stored aboveground in a monitored facility until they no longer presented a health concern.

Another benefit of the UREX+ method is that it is more proliferation-resistant than the PUREX method, since plutonium is never isolated. However, as discussed below in the section *GNEP and Nuclear Weapons Proliferation*, there is debate over the proliferation-resistance of UREX+. Some fear that the combination of plutonium and transuranic elements that would be extracted using UREX+ would not be sufficiently radioactive to prevent handling and transport, while it would remain sufficiently radioactive to fuel a nuclear bomb. (UCS 2007a)

The UREX+ process has been demonstrated only in a laboratory environment at Argonne National Laboratory. Preparations for a “scale-up demonstration” are reported to be underway. (ANL 2007b) DOE estimates that the technology could be fully developed as early as 2012 and commercialized in the 2012-2025 timeframe. (DOE 2005a, p.24) NEI is less optimistic, estimating that commercialization could require at least 50 years. (NEI 2006a)

The COEX process is currently under development by AREVA, and it is an intermediate step between PUREX and UREX+. The COEX process co-extracts equal amounts of uranium and plutonium. This adds a measure of proliferation resistance, since pure plutonium is not extracted. However, it does not provide as much proliferation resistance as UREX+. (DOE 2006d, p.8; DOE 2005a) (See further discussion below in the section *GNEP and Nuclear Weapons Proliferation*.)

Figure 13: Pyroprocessing

Pyrochemical processing, also known as pyroprocessing, is an alternative to aqueous processing such as PUREX and UREX+. (The prefix "pyro" indicates that the process happens at relatively high temperatures of around 500°C; there is no flame and no combustion occurs.) The process is primarily being developed to reprocess spent fuel from Generation IV reactors. These reactors, as discussed in Chapter 12, are advanced reactors that are in early stages of research and development. It is currently expected that they will not be LWRs and that their fuel will not be compatible with conventional aqueous processing (DOE 2005a).

A simplified version of pyroprocessing has been demonstrated at Argonne National Laboratory to treat wastes from its experimental breeder reactor (UIC 2005). However, critics question the success of the demonstration. According to Edwin Lyman of the Nuclear Control Institute, "DOE was only able to claim that the demonstration program met or exceeded all key performance criteria by changing the original criteria, in other words, it was only by moving the goal posts that [DOE] was able claim success" (NCI 2000).

Pyroprocessing technology has also been demonstrated in laboratories in Europe and Japan (Venneri 1999). However, the IAEA states that pyroprocessing is "still very much at the R&D stage" and that it would require on the order of 10 to 15 years of additional development before it would be ready for a full pilot-scale demonstration (IAEA 2004, p.109). Other experts estimate that advanced reprocessing technologies, such as pyroprocessing, will not be available for 50 to 60 years (DOE 2006a; Washington Post 2006; DOS 2006).

The IAEA notes that a key non-proliferation feature of pyroprocessing is that it results in impure plutonium, containing a highly radioactive mix of uranium, transuranic elements, and some fission product contamination (IAEA 2004, p.32). However, critics respond that the high radioactivity of the separated product is relatively short lived (on the order of years), after which it loses its nonproliferation benefit (SGS 2005). Another drawback to pyroprocessing is that it does not extract cesium and strontium from the waste stream, which UREX+ does (DOE 2003a).

Global Components of GNEP

A key goal of GNEP is to create an international framework that will allow developing countries and other countries without nuclear infrastructure to harness nuclear power while minimizing proliferation concerns. There are two parts to this framework: an international partnership whereby supplier nations would lease nuclear fuel to countries that agree not to pursue enrichment or reprocessing capabilities, and the deployment of nuclear reactors appropriately sized for the electricity grids and industrial needs of smaller, more rural, and less industrialized regions.

Under the fuel-leasing program, fuel-supplier states would provide fuel enrichment and reprocessing services to fuel recipient countries. Supplier countries would have three primary responsibilities:

1. To offer fuel services at competitive rates in order to provide incentives for fuel recipient countries to lease fuel rather than invest in nuclear infrastructure.
2. To accept spent fuel from fuel recipient countries and reprocess or otherwise dispose of it. This may require facing domestic concerns that land is being used as a nuclear waste dump for other countries' energy production.[43]
3. To continue diplomacy with countries that have been excluded from the partnership and that wish to develop enrichment and reprocessing technologies.

The U.S., the United Kingdom, France, Russia, China, and Japan comprise the initial set of global fuel supplier partners (DOE 2006a).

The goal of the GNEP small-scale reactor research program is to deploy nuclear reactors of 50-350 MW capacities with simple operations, fully passive safety systems, capabilities for remote monitoring by the International Atomic Energy Agency (IAEA), and long-life fuel loads, possibly not requiring any refueling over the reactor's lifetime. The U.S. has done only minimal research on reactors that would have these features, but other countries have been actively researching and developing such technologies. The IAEA leads an International Project on Innovative Nuclear Reactors and Fuel Cycles, which supports development of small-scale reactors for developing countries.[44] (IAEA 2003, p.2) The U.S. role, as currently envisioned under GNEP, is to help form international partnerships to accelerate the commercialization of these technologies (DOE 2007j).

GNEP Timeline

In the near term, DOE is focusing on compiling information and gathering public and industry input to support a decision by the Energy Secretary as to whether to move forward with GNEP. This decision, which may also determine where to locate these facilities, and which technologies to use, is expected to be made in June 2008 (DOE 2007h, p.40). If the Energy Secretary supports moving forward with GNEP, DOE would "build and operate [the] nuclear fuel recycling center and advanced recycling reactor facilities using the latest commercial technology available" as soon as

[43] Current U.S. policy is not to repatriate foreign spent fuel that originated in the U.S. This foreign spent fuel is termed U.S.-obligated, meaning that the countries in possession of the fuel are obligated to follow regulations that the U.S. has imposed with regard to fuel handling. For instance, countries must seek U.S. approval before reprocessing this fuel or transferring it to another country, and the U.S. does retain the right to repatriate it.

[44] Members of the IAEA project include the European Commission, Argentina, Pakistan, Russia, and a dozen other entities. The U.S. has not joined this project.

possible (DOE 2007i, pp.9-10). At the same time, DOE would move forward with an R&D program into advanced reprocessing and transmutation technologies.

If DOE follows this phased approach, using the latest commercial technologies as they become available, limited reprocessing in a LWR could begin before transmutation fuels are available. In addition, reprocessing could begin with the COEX process, rather than the preferred UREX+ process. Indeed, members of academia and industry estimate that achieving the complete domestic GNEP goal could take 50 to 60 years, whereas DOE's goal is to commercialize an advanced reprocessing system and a fast reactor in the U.S. by 2025.[45] The implications of using transitional reprocessing technologies in the near term are discussed below in the sections *GNEP and Spent Fuel Disposal* and *GNEP and Nuclear Weapons Proliferation*.

The global components of GNEP are considered late-stage components. That is, they will only be feasible once a reprocessing technology has been proven that is both proliferation-resistant and effective at minimizing the spent fuel waste problem. Moreover, according to John Deutch, Institute Professor at MIT, the key to GNEP is large-scale global deployment of nuclear power, which he does not anticipate in the near-term. Deutch expects that GNEP will not be fully deployed until about 2150, "a very, very, very, very, very long time in the future" (Greenwire 2007a).

Marvin Fertel, NEI senior vice president and chief nuclear officer, also sees a linkage between GNEP and new reactor deployment. Fertel recommended that key decisions on GNEP wait until 2020 or 2030, at which point industry will have a better idea of the extent of new reactor construction in the U.S. and abroad. By 2020, he said, "we'll have a reasonable idea of deployment" of new reactors, which will indicate whether there will be a market for GNEP's international fuel services portion and whether a tight uranium supply will require the use of reprocessed fuel (Greenwire 2007b).

GNEP and Spent Fuel Disposal

As discussed above, a primary objective of GNEP is to address some of the problems of disposing of nuclear waste in a geologic repository by introducing reprocessing into the fuel cycle. In fact, DOE has predicted that "[technological] advancements through GNEP could reduce the volume, thermal output, and radiotoxicity of waste requiring permanent disposal at the Yucca Mountain geologic repository" (DOE 2007k). These goals and the advancements that will be required to meet them are discussed in this section.

[45] For example, according to a DOE advisory group, it will likely be necessary to fuel a fast reactor initially with a uranium-plutonium fuel (such as MOX fuel or COEX fuel), rather than with fuel that contains transuranic elements, such UREX+ fuel (DOE 2006d, p.2).

Volume

The technologies proposed for the GNEP program are not intended to replace the planned geologic repository for Yucca Mountain. However, GNEP is attempting to address the looming conflict between the statutory limits on the volume of spent fuel that can stored at Yucca Mountain and the actual and projected volumes of spent fuel accumulating around the country at nuclear power plants.

The Nuclear Waste Policy Act (NWPA) of 1982 limits the amount of spent fuel that can be stored at Yucca Mountain to 70,000 MTHM.[46] Of that amount, 63,000 MTHM is reserved for spent fuel from or commercial reactors. As of the end of 2005, the United States had accumulated about 53,000 metric tons (MT) of waste from civilian reactors, with an additional 2,100 MT accruing each year (DOE 2006l, p.7). At this rate of accumulation, the statutory limits of Yucca Mountain will be met by 2010. With the licenses of many of the country's nuclear reactors being renewed for up to another 20 years, spent fuel stockpiles could reach a total of 120,000-130,000 MTHM by around 2040 (APS 2005c, p.17). (License renewal is discussed in Chapter 12.)

Reprocessing spent fuel can reduce the volume of high-level wastes, but it also produces a greater amount of intermediate-level waste and low-level waste.[47] The operators of the British and French reprocessing facilities have reported that, using current technology, reprocessing spent fuel results in four times less volume of high level wastes than the volume of the original spent fuel (Harvard 2003, p.61).[48] But intermediate-level wastes may require storage in a geologic repository just like high-level waste does. If high- and intermediate-level wastes are combined, current reprocessing does not yield a smaller volume of waste when compared to a once-through fuel cycle (Harvard 2003, p.62).

DOE studied the role of different fuel cycle strategies for several different nuclear growth scenarios and considered the implications of these different strategies and growth scenarios on the need for *additional* geological repositories. DOE found that if all existing nuclear power plants are retired at the end of their original 40-year licenses and the fuel cycle does not include reprocessing, then an additional repository will be required simply to store the fuel from current nuclear power plants. Under DOE's highest growth scenario, where nuclear power accounts for a greater share of the electricity supply and reprocessing is not used, the U.S. could need as many as 20 repositories by 2100. However, under the three highest nuclear growth

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[46] Federal legislation has been introduced that would reexamine the capacity limit on the repository planned for Yucca Mountain. (See Chapter 3.) The theoretical maximum capacity is estimated by DOE to be about 120,000 MTHM (DOE 2003c, pp.1-3).

[47] Intermediate-level waste from reprocessing typically needs to be disposed of in geologic repositories along with high-level waste. In the U.S., this waste is referred to as transuranic waste (Harvard 2003, p.61). Low-level and high-level wastes are defined in Chapter 3.

[48] Note that this figure does not include the waste container that would encapsulate the high-level waste.

scenarios, the number of repositories could be cut in half by reprocessing and recycling fuel in current reactors. Additionally, using the new transmutation technologies envisioned under the full GNEP plan, a single repository would be sufficient even in DOE's highest growth scenario (DOE 2007e, p.13). Under all scenarios there would remain a need for long-term geological disposal of radioactive isotopes, and in the reprocessing scenarios there would be significant additional need for low- and intermediate-level radioactive waste disposal (AIADA).

Heat Output

Many of the technical standards established for the proposed repository at Yucca Mountain take the form of temperature limits applied to the overall repository as well as to individual waste packages. By reducing the heat output of nuclear waste, the capacity of a geological repository such as Yucca Mountain could be increased.

In theory, a fast reactor-based fuel cycle would reduce the long-term heat load of a repository by 20 percent 10 years after discharge and by 99 percent 300 years after discharge when compared to storage of spent fuel from a once-through cycle (National Academies 1996, pp.31-34, 100). However, reprocessing spent fuel and using the recycled plutonium in a LWR rather than a fast reactor, as might be done during early phases of GNEP, would actually yield a greater total heat output from the waste than if the same amount of electricity was generated using a once-through fuel cycle. In other words, the GNEP goal of limiting the needed capacity in a geologic repository can only be achieved if "the [reprocessing] soon switches [from limited recycling] to fast-neutron reactors or more complete separation and transmutation of the wastes" (Harvard 2003, p.39).

Radiotoxicity

Another important goal of GNEP is to reduce the duration of radiotoxicity of spent fuel from about 300,000 years to several hundred years, greatly easing the licensing requirements for a geologic repository.[49] DOE investigated the impact of four different fuel cycles on the radiotoxicity of spent fuel: the current once through cycle; a limited recycle scenario, in which enriched uranium and recycled plutonium are used as fuel for existing LWRs and, after a few cycles, the spent fuel is disposed; a transitional recycle scenario, in which spent fuel is recycled continuously using fast reactors until transuranic components are essentially eliminated; and a sustained recycle scenario, in which depleted and recycled uranium are converted into fuel and spent fuel is recycled through fast reactors (DOE 2005a, pp.8-11).

DOE found that limited recycling has no impact on the duration of spent fuel's radiotoxicity, because the long-term radiotoxicity of spent fuel is derived almost

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[49] Radiotoxicity is a measure of the hazard inherent in the waste. Different indices can be used to measure radiotoxicity, for instance: activity per volume, total activity, number of annual limits of intake contained in the material, etc. The duration of radiotoxicity is defined as the amount of time during which the spent fuel radiotoxicity exceeds the radiotoxicity of the source material (uranium ore) (IAEA 1994, p.25 ; DOE 2005a, p.13).

exclusively from the transuranic elements in the waste, and limited recycling leaves these elements intact. However, transitional and sustained recycling in fast reactors would transmute the transuranic elements into shorter-lived or less radiotoxic elements.

GNEP and Nuclear Weapons Proliferation

The U.S. ended efforts to develop commercial reprocessing capabilities in the 1970s when it became evident that reprocessing, if developed by countries or organizations with non-peaceful intentions, could lead to the proliferation of nuclear weapons. GNEP is a reversal of that long-standing U.S. policy against reprocessing. However, GNEP seeks to build in safeguards against weapons proliferation by developing proliferation-resistant fuel cycles and creating a fuel-leasing program that keeps reprocessing facilities in a limited number of countries.

Plutonium extracted from spent fuel via reprocessing can currently be used in one of two ways: as MOX fuel for a nuclear reactor or as fuel for a nuclear weapon. Globally, little of the plutonium that has already been extracted through reprocessing has been made into MOX fuel, and most of the plutonium remains stockpiled. As of the end of 2003, there was approximately 265 MT of plutonium in global military stockpiles and 240 MT of separated plutonium in civil stockpiles. There was an additional 1,300 MT of plutonium within civil stocks of (non-reprocessed) spent fuel (See Table 13.) (ISIS 2005, Tables 1, 3; ISIS 2007). Just 2 to 4 kg of weapons-grade plutonium or about 5 kg of reactor-grade plutonium can produce a 10 to 20 kiloton explosion, similar to the scale of the Hiroshima and Nagasaki bombs (CFR 1998; Greenpeace 2007).

GNEP would eliminate over time these stockpiles of separated plutonium by converting the plutonium into reactor fuel. In addition, the reprocessing technology envisioned under GNEP will be “proliferation-resistant,” meaning that it “would make more difficult, time-consuming, and transparent the diversion by states or sub-national groups of civilian nuclear fuel cycles to weapons purposes” (FAS 2001).

The initial idea under GNEP for achieving a proliferation-resistant fuel cycle was to mix plutonium with other transuranic elements, as is done with the UREX+ process that is under development. According to DOE, “as long as the fissile materials [i.e., plutonium and uranium] remain combined with sufficient quantities of non-fissile materials the product is not directly useable as a nuclear weapon.” However, the UREX+ technology is not expected to be commercially available until after 2020, and it is now expected that DOE would use an alternate process, called the COEX process, at least until UREX+ is available (DOE 2006d, p.8; DOE 2005a). The COEX process keeps plutonium mixed with an equal amount of uranium, but not with other transuranic elements. (See Figure 12.)

Table 13: Worldwide Stockpiles of Plutonium in 2003

Country of Origin	Military Stocks metric tons	Civil Stocks in spent fuel metric tons	Civil Stocks separated metric tons
Belgium		23.1	.4-1.4
China	4.8	5.1	
France	5	183	48.1
Germany		67-70	26
India	.38	12.5-13	1-1.5
Israel	.58		
Italy		4.0	2.5
Japan		111-113	40.6
Netherlands		1-1.4	2-2.5
North Korea	.015-.04		
Pakistan	.04		
Russia	145	88	38.2
Spain		26.6	0.3
Sweden		41	.83
Switzerland		16-17	1.5-3
United Kingdom	7.6	18.5-24.6	74.6
United States	99.5	403	
Other		324-327	2-6
Total	263	1,327-1,337	242

Source: (ISIS 2005, Tables 1, 3; ISIS 2007)

Many experts are concerned that the UREX+ process would not be proliferation resistant. For example, Jungmin Kang and Frank von Hippel investigated whether mixing plutonium with transuranic elements (as done in UREX+) would yield greater proliferation resistance than pure plutonium. They found insufficient improvements in four key areas (SGS 2005):

- A plutonium-transuranic mix would have a higher neutron emission rate than reactor-grade plutonium alone, leading some observers to "conclude that these materials are unusable in nuclear weapons." Kang and von Hippel countered that although a high-neutron emission rate reduces the expected "yield" from a Nagasaki-type weapon from about 20 kilotons to as low as 1 kiloton, the plutonium-transuranic mix could still be used in a weapon since even a 1 kiloton explosion would be devastating.[50]
- Most explosives become unstable at temperatures above 200° C. For this reason, nuclear warheads, which use heat-emitting plutonium, may require a

[50] A plutonium-transuranic mix has a neutron emission rate about twice as fast as the emission rate from reactor-grade plutonium, which is about 10 times as fast as the emission rate from weapons-grade plutonium. Thus, the plutonium-transuranic mix would be less desirable than pure plutonium as a weapons material.

cooling system of some kind. Although reactor-grade plutonium has a rate of heat release significantly higher than weapon-grade plutonium, the IAEA and weapons experts believe that it is possible to use reactor-grade plutonium in combination with a cooling system to make a nuclear warhead. Kang and von Hippel estimated that a plutonium-transuranic mix would have a rate of heat emission only about twice that of reactor-grade plutonium. Thus, if the appropriate cooling system were employed, a weapon could be made using a plutonium-transuranic mix.

- The amount of material required to initiate a chain reaction is greater for the plutonium-transuranic mix (17.9 kg) than for reactor-grade (14.4 kg) or weapons-grade (10.7 kg) plutonium. However, these differences are not significant to prohibit weapons construction.
- The radiation dose for a pure transuranic mix is more than three orders of magnitude lower than the threshold for self-protection.[51] Advanced reprocessing as envisioned under GNEP would increase the radiation dose above the threshold for self-protection by mixing cerium together with the transuranic elements. However, this cerium protection is short-lived. Since the half-life of cerium is less than a year, the radiation dose would remain above the threshold for just over two years.

There are similar (and even stronger) concerns over the proliferation-resistance of the plutonium-uranium mixture from the COEX process. In testimony to Congress, Matthew Bunn of Harvard noted that it would not be difficult to separate out the plutonium from the plutonium-uranium mixture.[52] Moreover, it would not be necessary to do so, since nuclear explosives could be made directly from this mixture. Furthermore, the NRC reviewed this approach 30 years ago and found it to be not significantly more proliferation resistant than pure plutonium.

A Massachusetts Institute of Technology (MIT) study determined that the once-through fuel cycle “defines the baseline for adequate proliferation-resistance,” while advanced closed fuel cycles that mix plutonium with other transuranic elements “need strong process safeguards against misuse or diversion” (MIT 2003, p.67). Moreover, “the development and eventual deployment of closed fuel cycles in non-nuclear weapons states is a particular risk both from the viewpoint of detecting misuse of fuel cycle facilities, and spreading practical know-how in actinide science and engineering” (MIT 2003, p.67). Indeed, a Harvard study questioned the need for reprocessing when there is minimal legitimate demand for plutonium and concluded that “the burden of proof clearly rests on those in favor of investing in reprocessing in

[51] The threshold for self-protection is the radiation dose (100 rads per hour at one meter) above which even short exposures to the material would be very hazardous to human health.

[52] However, the quantity of material that would be required to make a bomb out of the uranium-plutonium mixture is significantly greater than what would be required to make a bomb out of pure plutonium (Bunn 2006).

the near term," due in part to proliferation concerns with respect to separated plutonium (Harvard 2003).

It is debatable whether a plutonium-transuranic mix would be attractive to terrorists seeking to make a nuclear weapon. According to many weapons-design experts, "there is no proliferation-proof nuclear power cycle" because most of the transuranic elements and their oxides are explosive fissionable material (LLNL 1999, p.14). Moreover, as "nuclear weapons design and engineering expertise combined with sufficient technical capability become more common in the world, it becomes possible to make nuclear weapons out of an increasing number of technically challenging explosive fissionable materials" (LLNL 1999, p.14).

Concerns over these reprocessing technologies were echoed by representatives of arms control, consumer, environmental, and public health organizations who wrote in a letter to Congress in January 2006 that the "'proliferation-resistant' reprocessing technologies currently being researched by DOE are not sufficient to prevent theft by terrorists, while the plutonium mix that results from these technologies could be used to make a nuclear weapon" (ANA 2006). However, Dr. Per Peterson of the University of California, Berkeley believes this concern is misplaced. He argues that a plutonium-transuranic mix would not be attractive to terrorists since it is more difficult to develop weapons materials out of reprocessed fuel than out of virgin uranium (NY Times 2006).

The National Commission on Energy Policy (NCEP) reviewed U.S. policy on reprocessing in 2004 and found that reprocessing continues to pose a proliferation risk. It recommended that "the United States do everything it can to minimize access to uranium-enrichment and fuel-reprocessing technologies by countries other than the five de jure nuclear-weapon states" and "that it defer—at least for the next few decades—plutonium separation in its own commercial nuclear-energy operations" (NCEP 2004, p.59). NCEP made this recommendation based on its finding that weapons proliferation concerns were a substantial barrier to the expansion of nuclear energy in the U.S. (NCEP 2004, p.61).

GNEP and Reprocessing: Issues to Consider

If GNEP is pursued, it will substantially change the way that nuclear power is produced and consumed. It will also have a number of other local and national impacts. This section discusses the economic, environmental, and safety implications of the domestic reprocessing component of GNEP, as well as the implications that a large federal reprocessing program could have on competing federal energy programs. The implications of the global component of GNEP are not considered, as this is considered to be a late-stage component and too speculative at this time.

Economics of the Reprocessing Fuel Cycle

There are three major cost categories to the reprocessing fuel cycle: transportation of spent fuel from the reactor to the reprocessing facility, reprocessing, and final disposal of reprocessing waste by-products. A number of studies have compared the cost of the reprocessing fuel cycle using commercially available reprocessing technologies with the cost of the once-through fuel cycle currently in use in the U.S.

- The OECD compared the costs of nuclear power generated with a once-through fuel cycle to the costs of a fuel cycle that includes reprocessing and a one-time recycling of recovered plutonium into MOX fuel for a pressurized water reactor. The study found the reprocessing fuel cycle to be 14 percent more expensive than the once-through fuel cycle (OECD 1994, pp.40, 53, 115).
- A 2003 study by Harvard University found that the cost of reprocessing using the PUREX technology would be between $1,350 and $3,100 per kgHM.[53] They also found that even if the cost of reprocessing was reduced to $1,000 per kgHM, nuclear power-generated electricity costs would increase by at least 0.13 cents per kWh (Harvard 2003, p.28).
- Researchers at MIT concluded that reprocessing would increase the cost of electricity by 0.28 cents per kWh compared with electricity costs in a once-through fuel cycle scenario (MIT 2003, p.148).
- A study by the National Academies concluded that the cost of reprocessing the 63,000 MTHM of civilian spent fuel intended for Yucca Mountain using existing technologies would be about $2,100 per kilogram of heavy metal (kgHM) in 1992 dollars, which is equivalent to a total cost of $180 billion in 2006 dollars (National Academies 1996, p.7).
- In a study for AREVA, the Boston Consulting Group concluded that “the overall cost of recycling used fuel is in the order of $520 per kg, comparable to the cost of a once-through strategy,” which is estimated to be around $500 per kg of spent fuel (BCG 2006, p.12).

The cost of the reprocessing fuel cycle using advanced reprocessing technologies remains highly uncertain at this time. DOE expects that UREX+ will be less costly to implement than PUREX because the amount of liquid waste requiring solidification is less and the scale of processing equipment that must be included in the plant design is smaller (DOE 2005a). DOE estimates that a plant capable of reprocessing 2,000 MT of spent fuel per year using UREX+ technology could cost $6 billion to construct with an annual operating cost of $280 per kilogram of material treated (DOE 2003a). However, the National Academies found that the cost to reprocess and transmute

[53] The variation in estimated cost is due to financing costs for a reprocessing facility. A government-owned reprocessing facility would be able to access very low-cost financing whereas a private entity would face higher financing costs. (The reprocessing facilities built in France, Great Britain, and Japan all relied on some level of government funding.)

the spent fuel sufficiently to affect the need for a second repository would cost about $500 billion (in 1992 dollars) over 150 years (National Academies 1996, p.82).

Opportunity Costs of GNEP

President Bush's 2008 budget proposal requested $405 million in funding for GNEP, an increase of $155 million above the 2007 budget request[54] (DOE 2007a).[55] DOE anticipates that $2 billion will be spent on the program through FY 2009, at which point a determination will be made on whether or not to proceed with the program (E&ETV 2006). If the program is pursued, its lifetime federal funding is projected to total $20-$100 billion. This level of funding raises three concerns:

1. Other DOE programs that support renewable energy, energy efficiency, and demand side management may receive less funding if the "pie" remains the same size overall.
2. DOE may be underestimating the true cost of the complete GNEP program over its expected lifetime.
3. If funding is focused on GNEP, the efforts to license and operate a geologic repository at Yucca Mountain may suffer.

The first concern raises the issue of whether the concentration of energy funds on advanced fuel cycle technologies may result in fewer funds for energy efficiency, renewable technology, demand side management, and other competing programs that may more directly benefit California and the nation as a whole. This type of fund shifting may be seen in DOE's FY 2008 budget request for energy supply and conservation R&D: DOE counterbalances requested funding increases of 10 percent or more for hydrogen and nuclear technologies with requested funding decreases for all other renewable energy and energy efficiency technologies (AAAS 2007).

The second concern reflects criticisms of GNEP cost estimates. For example, Thomas Cochran and Christopher Paine of the Natural Resources Defense Council (NRDC) have pointed out that GNEP cost estimates do not include the cost to build the new fast reactors that are a critical component of the GNEP closed fuel cycle vision. They have estimated that building enough new fast reactors to transmute the fuel discharged from existing U.S. power reactors could cost between $80 and $100 billion (NRDC 2006, p.6). In testimony before Congress, Matthew Bunn of Harvard University urged legislators to consider whether DOE projects of comparable scale and complexity have remained within initial cost estimates (Bunn 2006). Finally,

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[54] The House Appropriations Committee's fiscal year 2008 Appropriations Bill, released June 6, 2007, allocates just $120 million to GNEP. The committee explained: "It is unnecessary to rush into a plan that continues to raise concerns among scientists and has only weak support from industry given that there are reasonable options available for short term storage of nuclear waste and that this project will cost tens of billions of dollars and last for decades." This bill had not been voted on by the full House of Representatives as of the release of this draft report (Congress 2007c).

[55] It should be noted that legislators failed to complete an appropriations bill for DOE's 2007 budget. GNEP funding for 2007 was $167.5 million under a continuing resolution.

John Deutch of MIT said that while he believes it is essential to make nuclear power as affordable as possible, "all these fancy closed-cycle systems will add to the cost of nuclear power. It's not a cost-saver" (Greenwire 2007a). Japan's experience with developing reprocessing capacity may add to these concerns. (See Figure 14.)

Figure 14: Japan's Experience Developing Reprocessing Infrastructure

In the 1980s, Japan embarked on a project to develop domestic reprocessing capabilities. Japan planned to construct its first large-scale reprocessing plant by the mid-1990s, with an additional reprocessing plant to be completed in 2010. It also planned on developing breeder reactors that would be able to burn plutonium recovered from spent nuclear fuel. However, lengthy delays and massive cost overruns ensued. The first plant, called Rokkashomura, is now expected to become commercially available in November 2007 at a cost of $17-$25 billion, and a decision on whether or not to construct the second plant will not be made until 2010. The plans to build breeder reactors have been all but abandoned in favor of a program to develop MOX fuel that will fuel LWRs.

The delays in developing a large-scale reprocessing plant and breeder reactors have led to large and growing stockpiles of spent nuclear fuel in Japan. Stockpiles of recovered plutonium (from Japanese spent fuel that was reprocessed in Europe) are also growing. The accumulation of spent nuclear fuel and recovered plutonium has led to concerns over domestic nuclear safety as well as concerns that Japan may use stockpiled plutonium in a nuclear weapons program. China in particular has expressed concerns about Japan's accumulation of plutonium stockpiles. In 1987 the government addressed the spent nuclear fuel stockpiles with a "partial reprocessing" policy that recognized that interim storage facilities would be needed due to delays in constructing a reprocessing facility. Interim storage of spent nuclear fuel will add to the lifecycle cost of nuclear power.

Meanwhile, public confidence in nuclear power has eroded over the past two decades due to a series of accidents and cover-ups at other Japanese nuclear facilities. One notable accident occurred at a site with a reprocessing plant but did not directly involve the reprocessing plant. The erosion of public confidence has created difficulties for the government in licensing storage and waste facilities and even shipping routes, and it may influence the government's future decisions on nuclear infrastructure research and development.

Japan's vision of a closed fuel cycle was similar to, but much less ambitious than, the vision put forth in GNEP. Twenty years into the process, they have scaled back their near-term plans to one reprocessing facility, which will cost as much as the lower estimates for the entire GNEP plan. While the Japanese government remains committed to reprocessing, given its difficulties with the Rokkashomura plant and growing public dissent, it is unlikely to endeavor on large nuclear infrastructure projects in the near future. The U.S. cannot rely on Japan to be an early adopter of advanced reactor designs or reprocessing technologies.

Source: (Harvard 2001; FEPC Japan 2003; FEPC Japan 2006; Global Security 2005; Japan METI 2007, p.11; Japan NCDI 2001; AIADA 2006; UIC 2006)

The third concern reflects the fear that the GNEP program will divert resources from the continuing effort to develop and license Yucca Mountain to an effort to develop reprocessing technologies that are unlikely to be available for several decades (Washington Post 2006). For example, Representative Boucher said in a September 2006 hearing that he is "somewhat skeptical about the ability of DOE simultaneously to fund and staff [GNEP and centralized interim storage projects] while continuing to meet the new schedule for opening Yucca Mountain" (Congress 2006c, p.4).Initial reactions to the GNEP proposal from some members of Congress support this concern. Senator Burr of North Carolina called for a "pause" on spending on Yucca Mountain in order to explore whether reprocessing may be a better route. Senator Pete Domenici of New Mexico suggested that the $20 billion Yucca Mountain fund be partially redirected for research on reprocessing (LVRJ 2006c).

Reliability and Safety Issues

Because reprocessing spent fuel involves handling highly radioactive wastes, the safety of any reprocessing facility is of critical importance.[56] Unfortunately, the safety record of reprocessing facilities is not stellar. A recent MIT study noted that "the historical accident frequency [i.e., accidents per year] of reprocessing plants is much larger than reactors… Furthermore, the number of reprocessing plant-years of operation is many fewer than in the case of reactors. Therefore the accident frequency [i.e., accidents per plant] of reprocessing plants is much higher" (MIT 2003, p.51).

The higher accident rate at reprocessing facilities than at reactors may in part be due to the difference in safety measures at these facilities. At a reprocessing facility, "fissile materials and waste are handled, processed, treated and stored in easily dispersible forms…using chemicals which can be toxic, corrosive or combustible" (IAEA 2005, p.9).As a result, human intervention and administrative policies, which are prone to human error, play a significant role in safety. At a nuclear power plant, on the other hand, active and passive engineered controls provide most of the safety support.

A recent safety violation at a modern reprocessing facility occurred in January 2005, when about 20 MTHM of uranium and plutonium dissolved in concentrated nitric acid internally leaked at the Sellafield facility in Great Britain. The leak occurred in a contained area, and no radiation was released into the atmosphere. However, the leak continued for three months before being discovered. Repairing the pipes and recovering the spilled liquids is expected to take months and may need special

[56] Although a country's government has ultimate jurisdiction and control of safety regulations for a reprocessing facility located within its borders, international safety standards are under development. In 1997 a number of countries agreed to a Joint Convention related to safety standards at reprocessing facilities. The Joint Convention, which went into force in 2001 and which currently has 42 signatories, is legally binding under international law. The U.S. ratified the Joint Convention in 2003.

robots, which will have to be built. Other significant safety events at commercial reprocessing facilities are described in Table 14.[57]

Table 14: Significant Safety Events at Commercial Reprocessing Facilities

Location and Year	Description of Event
Chelyabinsk, Former Soviet Union, 1957	Chemical explosion in concrete waste storage tank; 20 million curies[58] of radioactivity were released[59]
Tokai, Japan, 1999*	Uncontrolled chain reaction during fuel fabrication causing the deaths of two workers

* The criticality event that occurred in 1999 at the Tokai complex in Japan, in which worker error caused an uncontrolled chain reaction in a solution containing enriched uranium, was not associated with the reprocessing facility. Rather, it was associated with the experimental fast reactor also located on the site (UIC 2000).
Source: (NWMO 2003, p.35)

In addition to process-based safety concerns, a reprocessing program would necessitate a significant high-level waste transportation program, which could have a variety of security and environmental impacts. The GNEP program would require an international high-level waste transportation program as well. In a letter to DOE, the Western Interstate Energy Board (WIEB) raised concerns about the impacts on these shipments of potential malevolent acts or transportation accidents involving long-duration high temperature fires. WIEB also outlined a series of transportation-related impacts warranting investigation (WIEB 2007). For example, WIEB called for an assessment of the number and type of shipments that would be expected both domestically and internationally and an examination of origin and destination points and estimated shipment routes.

Another safety issue raised by GNEP is the potential need for longer interim storage of spent fuel. The GNEP facility would have a planned capacity of 2,500 to 3,000 MT per year and handle all the spent fuel from commercial nuclear power plants. With such a facility, it would require 30 to 40 years to reprocess the 63,000-105,000 MT of spent fuel from current reactors. Since this reprocessing is not expected to begin until at least the 2020s, some of the spent fuel would not be reprocessed for another half century or more. This spent fuel would likely remain in interim storage, which could be located at reactor sites, at several regional locations, or at the reprocessing

[57] Additional safety events have occurred at defense reprocessing plants in the U.S.

[58] The original unit for measuring the amount of radioactivity was the curie (Ci), first defined to correspond to one gram of radium-226 and more recently defined as: 1 curie = 3.7×10^{10} radioactive decays per second (LBL 2000).

[59] By comparison, the Chernobyl reactor accident released about 50 million curies of radioactive matter.

site. Alternatively, the spent fuel could be buried in a repository in a manner that allows it to be retrieved for reprocessing.

Environmental Impacts

The environmental impacts of reprocessing are much greater than the impacts of spent fuel storage. Reprocessing creates multiple waste streams and releases radioactive isotopes, such as carbon-14, krypton-85, iodine-129, tritium, and technetium-99, from spent fuel into the atmosphere (Schneider 2001, p.23). In a conventional PUREX reprocessing plant, these elements are released to the atmosphere.[60] The proposed UREX+ process would capture some of the radioactive off-gases for disposal (IPS 2007).

Historically, these radioactive releases have been significant. DOE found that the radiation dose within 50 miles of the Savannah River military reprocessing site in South Carolina is "four to five million times greater from reprocessing than from interim storage" (IEER 1996; DOE 1995b).The Institute for Policy Studies found that radionuclides stored at the Hanford reprocessing facility "pose potentially significant risks to health and natural resources for 300 to more than 200,000 years" (IPS 2007, p.10).

Significant releases of radioactivity have also been identified from European reprocessing facilities. In a report to the European Parliament, Mycle Schneider of World Information Service on Energy -Paris noted that "reprocessing operations release considerably larger volumes of radioactivity than other nuclear activities, typically by factors of several 1,000 compared with nuclear reactors," with radioactive discharges from the Sellafield and LaHague reprocessing facilities ranking "among the largest anthropogenic sources of radioactivity to the world" (Schneider 2001, pp.2-3).Impacts of the Sellafield discharges include "significant concentrations of radionuclides in foodstuffs, sediments and biota" in the Irish Sea, "very large" volumes of contaminated lands, significant contamination of groundwater, tritium levels in drinking waters exceeding World Health Organization limits, and contaminated sediments for hundreds of kilometers along the Irish Sea coast (NDA 2007; Schneider 2001, pp.5-6). Local residents and opponents of Sellafield suspect that these discharges are responsible for the increased incidence of cancer along the eastern coast of Ireland and the western coast of England (TED 2007).

Reprocessing waste also contaminated the waters in the vicinity of some U.S. reprocessing facilities. Waste disposal practices at the Savannah River military reprocessing site led to severe contamination of portions of the surface and groundwater. Operation of the West Valley commercial reprocessing facility led to a plume of groundwater contamination beneath the reprocessing building, as well as extensive infrastructure contamination (GAO 2001, p.7). Many of the tanks storing

[60] Scrubbers capture about 90 percent of the iodine-129 that is produced, but none of the other gases.

high-level radioactive waste at the Hanford military reprocessing facility have been found to leak (IEER 2004, p.8; DOE 1995a).

Cleanup efforts at these sites have been difficult. Cleanups of the Savannah River and Hanford sites have been bogged down for decades by technical and management issues and have not yet been completed. Cleanup has been similarly difficult at West Valley, which generated over 600,000 gallons of liquid high-level waste during just six years of operation. Cleanup was originally expected to be completed by 1990; however, there have been numerous delays, and significant cleanup efforts remain to be completed (GAO 2001, p.1; NRC 2007ai).

Conclusions

The advanced reprocessing fuel cycle envisioned under GNEP would prevent the need for a second repository for the foreseeable future, even if the use of nuclear power significantly increases. However, many are skeptical about whether this goal is achievable over the coming decades and are concerned that a limited reprocessing fuel cycle using readily available technologies could be instituted instead. Depending on the technologies used, such a fuel cycle could result in an increase in combined high- and intermediate-level nuclear waste, an increase in the risk of nuclear weapons proliferation, and an increase in the cost of nuclear power.

Even with the advanced GNEP technologies, environmental and safety impacts of a reprocessing fuel cycle could be significant. Reprocessing releases radioactive emissions during routine operations, has a higher accident rate than spent fuel storage does, and in some cases has generated significant contamination. A reprocessing fuel cycle also could require the long-term interim storage of large amounts of spent fuel at reprocessing facilities. These concentrated interim storage sites could present security hazards.

Accordingly, there is substantial opposition to the GNEP program. However, the program remains undefined in key respects, and it is far from certain that the proposal will be sustained over the next several years or, if it were, that it would ultimately be successful.

APPENDIX A: COMPARISON OF REPROCESSING TECHNOLOGIES

	PUREX	COEX	UREX+	Pyroprocessing
Product streams	Uranium; Plutonium; Waste stream of minor transuranic elements and fission products	Uranium and Plutonium; Waste stream of minor transuranic elements and fission products	Uranium; Technetium; Strontium and Cesium; Plutonium and neptunium; Americium and curium (together) Waste stream of remaining fission products	Uranium, Plutonium, and other transuranic elements; Waste stream of strontium, cesium, and remaining fission products
High-level waste, kg, per kg spent fuel input	0.25 kg per kg glass logs; 0.95 kg per kg U	N/A	0.12 kg per kg glass logs	0.25 kg per kg ceramic form waste
Weapons-grade plutonium created?	Yes	Uranium-plutonium mix could be used directly in a nuclear weapon	No[149]	No
Short-lived fission products separated from long-lived transuranic elements?	No	No	Yes	No
Useful in LWR	Yes, to create MOX fuel	Yes, to create MOX fuel	Yes, to create MOX	No
Technology maturity	Commercially available	Under development; could be commercially available in the near term	Demonstrated on a Laboratory scale; Potentially commercially available in the 2020-2030 timeframe	Demonstrated on a engineering scale; Potentially commercially available between 2025 and 2055
Can be used for repeated reprocessing?	No	No	Yes	Yes
Estimated construction cost[150]	$8 billion	N/A	$6 billion	$7 billion (highly uncertain)
Estimated operating cost	$400 per kg material	N/A	$280 per kg material	$280 per kg material (highly uncertain)

Source: (Bunn 2006; DOE 2006d, p.8; DOE 2003a; DOE 2005a)

[149] Some experts argue that it is technically feasible to create bomb material from the plutonium-neptunium mixture coming from a UREX+ reprocessor.

[150] Plant capable of processing 2,000 metric tons per year.

References for Appendix R

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Appendix S: MIT Economic Analysis

An economic analysis for the integration of nuclear energy with Oil Sands projects for reduced greenhouse gas emissions and natural gas consumption (Massachusetts Institute of Technology, June 2000) is provided in the following pages (ref. [10]).

8 Economic Analysis

Economic analysis is performed for two scenarios in detail in this section: electricity and steam production. Hydrogen was not included since it was deemed that the best option was to continue to use steam methane reforming in the short term with the future possibility of using nuclear heat in that process but it was not evaluated for cost effectiveness.

8.1 Electricity Production

A comparison is made among the three nuclear reactors considered in this report and a combined cycle natural gas plant (100 MWe) for the purpose of supplying electricity to the oil sands industry. The levelized cost of each option was calculated, and sensitivity analysis was performed on the natural gas price and the capital costs of the nuclear plants. The assumptions made in this analysis are detailed in Tables 14 through 19. All dollars are in Canadian dollars unless stated otherwise, and where an exchange rate was used to convert from US dollars, the rate of $0.90 USD per CAD was used. For simplicity, construction for any project was assumed to start in 2010 in the Edmonton area where it is most likely such a plant might be built. Regional labor adjustments were made to the base costs for overnight capital and for operations and maintenance. Overnight capital was assumed to be 40% labor-related, and for the location of an electric plant in Edmonton, the labor rates were assumed to be 50% above the base rate provided for a site in Ontario for CANDUs and at a coastal location for the PBMR. Thus, the overnight capital costs were increased by 20%. Similarly, O&M was assumed to be 50% labor, and so was increased 25% over the base cost.

Table 14: Assumptions Made in Calculating the Capital Charge Rate for the Nuclear Plants

General Inflation	2.00%
Term, years	40
Federal Tax Rate	22.1%
Provincial Tax Rate	8.00%
Debt Ratio	50%
Loan Term, yrs	40
Interest Rate	8.00%
Equity Return	14.75%
Prop Tax & Insurance	1.50%
Tax Credit Rate	0.00%
Tax Life, Years	20
Declining Balance Rate	100%
Real Return	12.50%
Resulting Capital Charge Rate	0.144 in current dollars (Canadian)

Table 15: Assumptions Made in Calculating the Capital Charge Rate for the Natural Gas Electric Plant

General Inflation	2.00%
Term, years	20
Federal Tax Rate	22.1%
Provincial Tax Rate	8.00%
Debt Ratio	50%
Loan Term, yrs	20
Interest Rate	8.00%
Equity Return	12.71
Prop Tax & Insurance	1.50%
Tax Credit Rate	0.00%
Tax Life, Years	20
Real Return	10.50%
Resulting Capital Charge Rate	0.152 in current CAD

Table 16: Assumptions Specified for the Combined Cycle Natural Gas Plant

Generation (MWe)	100
Overnight $/kWe in Ontario	900
Overnight $/kWe in Edmonton, Alberta[2]	1080
Construction Period	2 years
Construction Interest	12.71% on ½ of construction period escalation of overnight costs
O&M in Ontario	$8 million per year[1]
O&M in Edmonton[3]	$10 million per year
Heat Rate (btu/kWh)	6800
Natural Gas Price	Varies
Natural Gas Price Nominal Escalation	2% above inflation

[1] Source: "Electricity Generation Technologies: Performance and Cost Characteristics" Prepared for the Ontario Power Authority by the Canadian Energy Research Institute, August 2005.

[2] A 20% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 40% of overnight costs, and labor rates are 50% higher in Edmonton than Ontario

[3] A 25% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 50% of O&M costs, and labor rates are 50% higher in Edmonton than Ontario.

Table 17: Assumptions Specified for the Enhanced CANDU 6 Nuclear Electric Plant

Generation (MWe)	728
Overnight $/kWe in Ontario	3375[1]
Overnight $/kWe in Edmonton, Alberta[2]	4050
Construction Period	6 years[1]
Construction Interest	14.75% on construction capital outlay sequence - yr1: 8%, yr2: 21% yr3: 27.1%, yr4: 19.6%, yr5: 12%, yr6: 7.2%, yr7: 5.1%[1]
O&M in Ontario	$90 million per year [1]
O&M in Edmonton[3]	$112.5 million per year
Nuclear Fuel Cost	3.75 $/MWh [1]
Nuclear Fuel Price Nominal Escalation	0.5% above inflation

[1]Source: “Electricity Generation Technologies: Performance and Cost Characteristics” Prepared for the Ontario Power Authority by the Canadian Energy Research Institute, August 2005.
[2]A 20% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 40% of overnight costs, and labor rates are 50% higher in Edmonton than Ontario.
[3]A 25% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 50% of O&M costs, and labor rates are 50% higher in Edmonton than Ontario.

Table 18: Assumptions Specified for the ACR-700 Nuclear Electric Plant

Generation (MWe)	703
Overnight $/kWe	2740 (CERI) [1]
Overnight $/kWe in Edmonton, Alberta[2]	3288
Construction Period	6 years[1]
Construction Interest	14.75% on construction capital outlay sequence - yr1: 8%, yr2: 21% yr3: 27.1%, yr4: 19.6%, yr5: 12%, yr6: 7.2%, yr7: 5.1%[1]
O&M in Ontario	$100 million per year[1]
O&M in Edmonton[3]	$125 million per year
Nuclear Fuel Cost	5.45 $/MWh[1]
Nuclear Fuel Price Nominal Escalation	0.5% above inflation

[1] Source: “Electricity Generation Technologies: Performance and Cost Characteristics” Prepared for the Ontario Power Authority by the Canadian Energy Research Institute, August 2005.
[2]A 20% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 40% of overnight costs, and labor rates are 50% higher in Edmonton than Ontario.

[3]A 25% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 50% of O&M costs, and labor rates are 50% higher in Edmonton than Ontario.

Table 19: Assumptions Specified for the PBMR Nuclear Electric Plant

Generation (MWe) [1]	172
Overnight $/kWe for a 4-module plant	3333
Overnight $/kWe for a single module plant[2]	4000
Overnight $/kWe in Edmonton, Alberta[3] (single module)	4800
Construction Period	3 years
Construction Interest	12.71% on ½ of construction period escalation of overnight costs
O&M at the Base Labor Rate	$10.5 million per year[1]
O&M in Edmonton[4]	$13.13 million per year
Nuclear Fuel Cost	$21.25 million year[1]
Nuclear Fuel Price Nominal Escalation	0.5% above inflation

[1] Source: Pebble Bed Modular Reactor (Pty) Ltd.
[2]A 20% penalty is applied to account for the increase in costs for a single-module plant over a 4-module plant. This penalty is due to the loss of economies of shared systems.
[3]A 20% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 40% of overnight costs, and labor rates are 50% higher in Edmonton than in the base case.
[4]A 25% penalty is applied to account for the increase in labor rates for Edmonton. This is based on the assumption that labor costs account for 50% of O&M costs, and labor rates are 50% higher in Edmonton than Ontario.

The reader may note that the operating and maintenance costs for the PBMR are unusually low for a nuclear power plant. Low O&M cost is a design objective for the PBMR and for Generation IV systems, and is based on the reduction in the number of systems needed to run the reactor safely.

Given the assumptions detailed above, the analysis showed that the breakeven natural gas prices where each of the nuclear plants are competitive with the combined cycle natural gas plant are at approximately $10.15, $12.10, and $12.65 for the ACR-700, CANDU 6, and PBMR, respectively. This analysis assumes that natural gas prices are assumed to escalate at 2.0% above inflation over the life of these projects. These results are illustrated graphically in Figure 17.

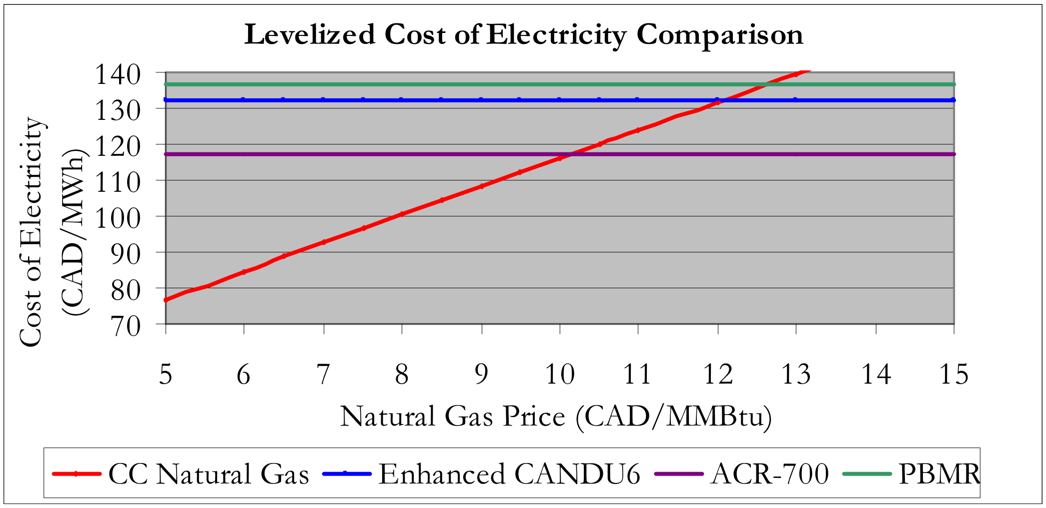


Figure 17: Levelized Cost of Electricity Comparison

A sensitivity analysis was also performed on the overnight capital costs of the nuclear power plants since there is much speculation as to what the capital costs might actually be. While the cost of the natural gas plant and all other factors were kept constant, the overnight costs of the nuclear plants were all raised by 20%, 30%, 40%, and 60% in turn. This was done to show the impact of a cost overrun on the ultimate cost of the electricity produced. The analysis was performed first at $5/MMBtu natural gas, and then at $11/MMBtu natural gas, and the results are shown below in Figure 18 and Figure 19.

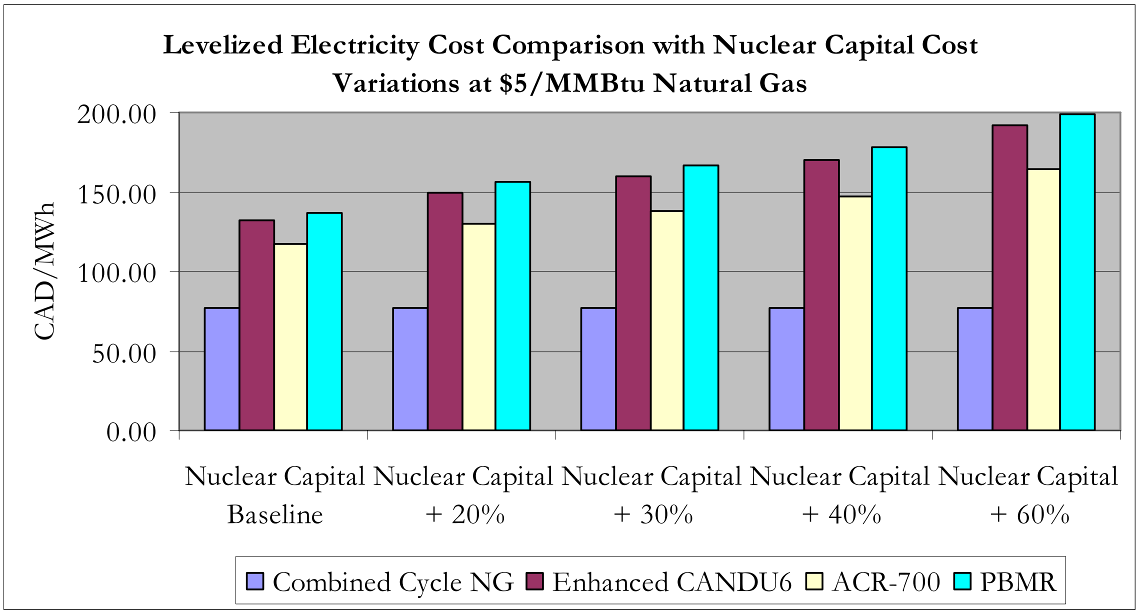


Figure 18: Levelized Cost of Electricity with Varying Nuclear Capital Costs at $5/MMBtu Natural Gas

In the $5 gas case, none of the nuclear plants were found to be competitive at the baseline capital cost.

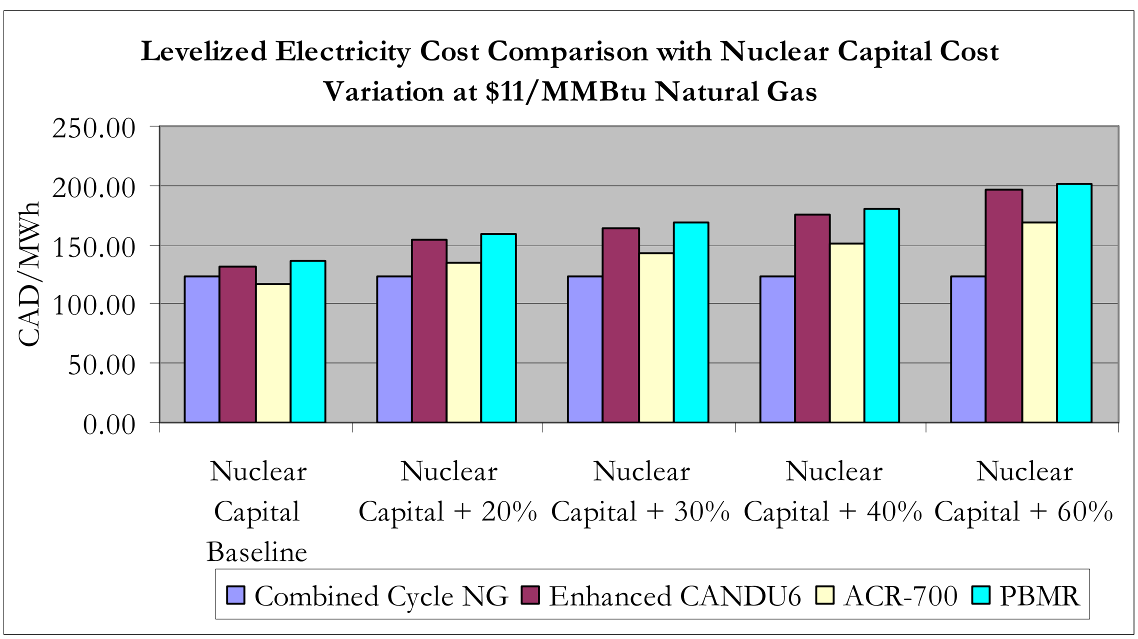


Figure 19: Levelized Cost of Electricity with Varying Nuclear Capital Costs at $11/MMBtu Natural Gas

In the $11 gas case, the ACR-700 was found to be competitive at the baseline capital costs, but at a 20% overrun it was slightly more expensive than natural gas.

It should be noted that other sensitivities should be considered in the economic evaluation. The cost of capital is a significant parameter affecting the cost of nuclear and other capital intensive projects. Alternative financing mechanisms that reduce the cost of capital will have a dramatic impact on the levelized cost. Should public or government support in the form of loan guarantees, low interest loans, or low interest environmental bonds be made available, the cost of the nuclear option would be greatly reduced. In addition, the future rate of natural gas price growth is also a very important parameter for which sensitivity studies need to be made to fully appreciate the economics of alternatives.

8.2 Steam Production

Estimating the costs of the steam production plants was difficult because the data available publicly is generally applicable to electric plants. Adjustments were made to account for two effects. First, the movement from Edmonton (for an electric plant) to Fort McMurray (for a steam plant) was predicted to increase labor rates from 50% over base rates to 100% over base rates. Additionally, the conversion from an electric power plant to a steam plant eliminates a number of expensive systems, reducing the overall cost of the plant. For the sake of consistency, in each nuclear plant case it was assumed that the costs associated with the electricity generation accounted for 1/3 of the overnight capital costs of the nuclear plants. The cost of that equipment is dominated by the turbine-generator, moisture separators and reheaters, oil lubrication systems, and the electrical switchyard. The basis for that assumption is that the typical light water reator has approximately a 60/40 division between the steam plant and the electricity generating plant, as illustrated in Table 20. Thus, the assumption that the nuclear heat plant has a cost two-thirds that of the nuclear electric plant is conservative, since it is less favorable to the economics of the steam plant than a 60/40 split. The cost adjustments made to the nuclear plants are shown in Table 21.

Table 20: Typical Allocation of Costs for an LWR

Project Cost Component	Percentage of Overnight Project Costs	Overall Percentage Allocated to the Steam Plant
Reactor Equipment	30	30
Balance of Plant Equipment	24	4
Structures and Construction	20	13
Owner’s and other Indirects	26	13
Total	100	60

Table 21: Cost Adjustments for the Nuclear Steam Plant

	Enhanced CANDU 6	ACR-700	PBMR
Overnight $/kWe (equivalent) [1]	3150	2557	3733
O&M	$135 million/yr	$150 million/yr	$15.75 million/yr

[1] Equivalent represents the 'would-be' electric power of the plant using the actual MWth and the efficiency of that plant's conversion cycle in the electric case. This notation is chosen so that the relative cost can be compared with that of the nuclear electric plant.

The steam production assumed for each plant is given in Table 22 below. The plants are rated in this case based on their thermal capacity, but the thermal capacity used was the net capacity after providing the heat needed for the house load. The cost of the steam generated from a natural gas boiler was approximated from a reference and is shown in Figure 20 [59].

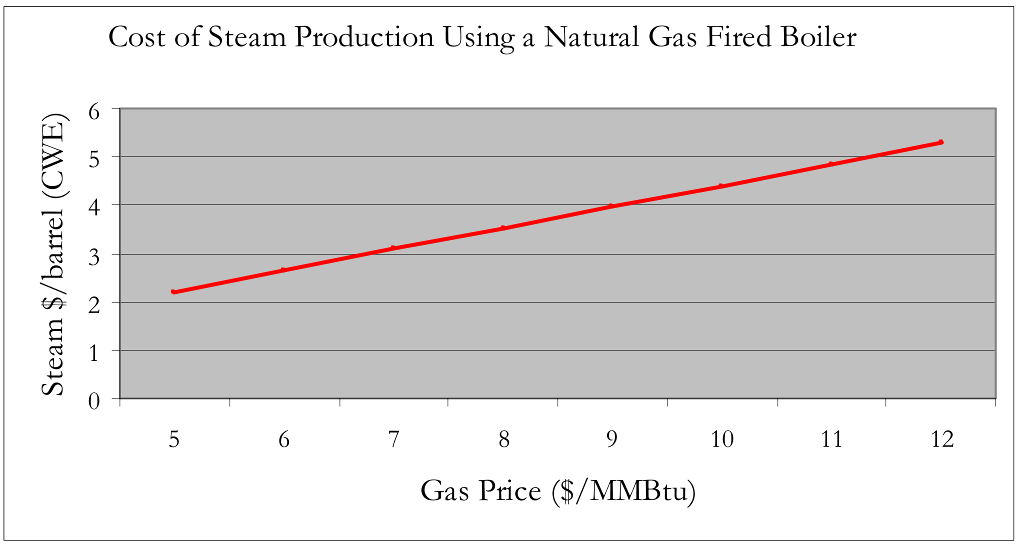


Figure 20: Cost of Steam Production from a Natural Gas Fired Boiler

Table 22: Levels of Steam Production for each Generation Option

Plant Type	Steam Production (bpd)
2030 MWth Enhanced CANDU 6	653,000
1895 MWth ACR-700	697,000
500 MWth PBMR	130,000

The baseline cost to produce one barrel of steam (Cold Water Equivalent, or CWE) from the nuclear reactors was $3.02 for the Enhanced CANDU 6, $2.49 for the ACR-700, and $2.97 for the PBMR. For the natural gas plant, at $5/MMBtu gas, the cost found was $2.20. The breakeven natural gas prices were $6.85/MMBtu for the Enhanced CANDU

6, $5.65/MMBtu for the ACR-700, and $6.75/MMBtu for the PBMR. These results are shown in Figure 21 below. For reference, the June 2007 average NYMEX natural gas price was approximately $ 7/MMBtu.

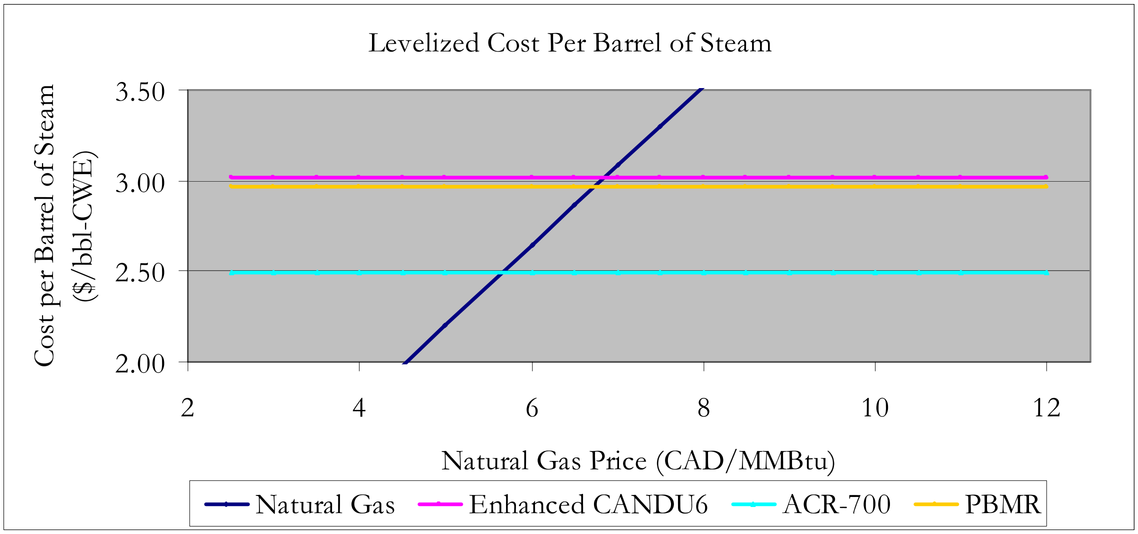


Figure 21: Levelized Cost per Barrel of Steam

A sensitivity analysis was again performed on the overnight capital costs of the nuclear power plants. While the cost of the natural gas plant and all other factors were kept constant, the overnight costs of the nuclear plants were all raised by 20%, 30%, 40%, and 60% in turn. This was done to show the impact of a cost overrun on the ultimate cost of the steam produced. The analysis was performed for $5/MMBtu natural gas and for $11/MMBtu natural gas, and the results are shown below in Figure 22 and Figure 23.

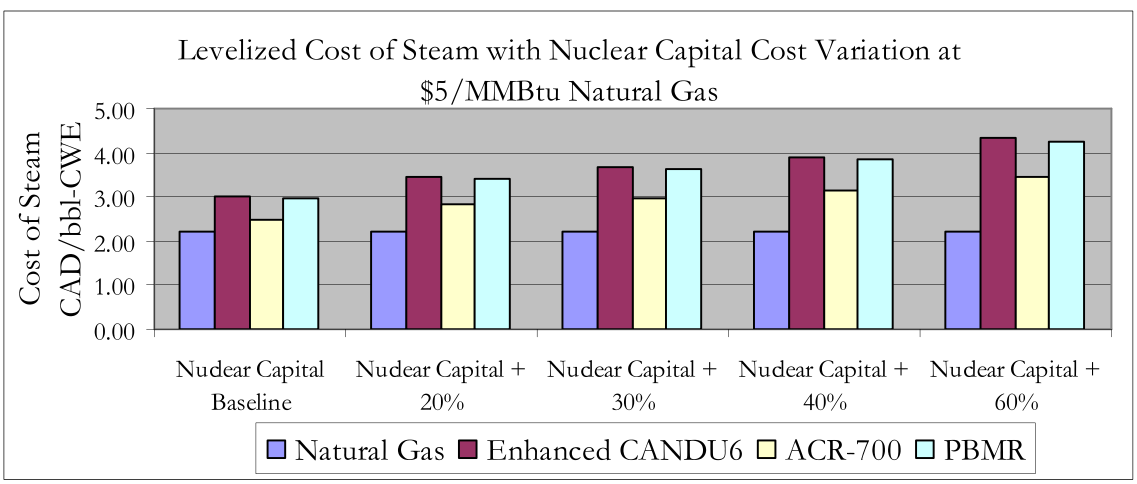


Figure 22: Levelized Cost of Steam Production with Varying Nuclear Capital Costs ($5 NG)

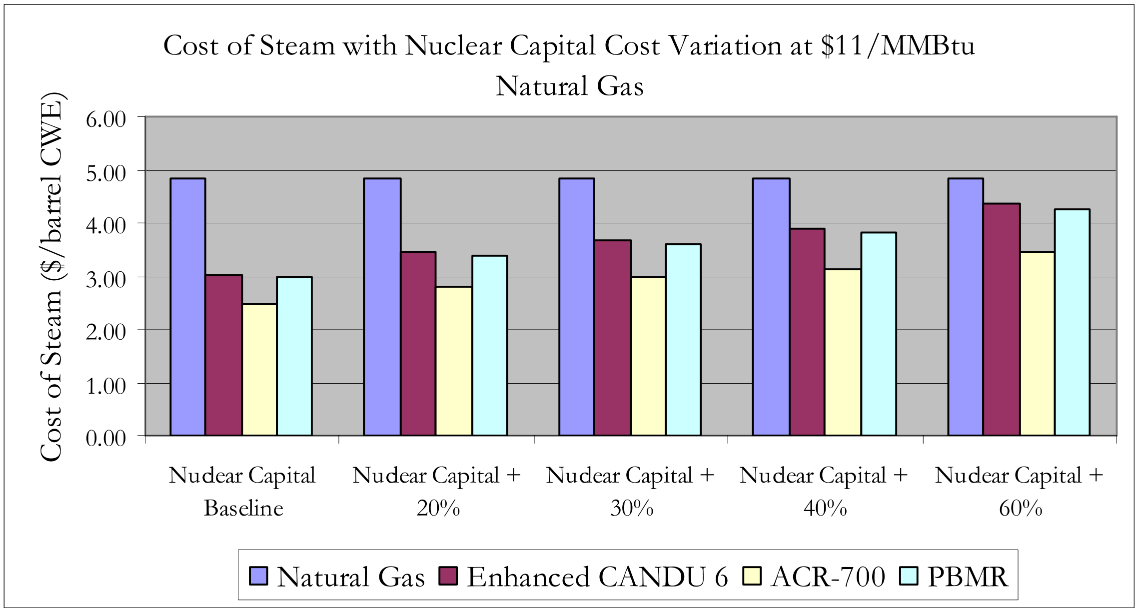


Figure 23: Levelized Cost of Steam Production with Varying Nuclear Capital Costs ($11 NG)

In the $5 gas case, none of the nuclear plants proved to be more economic than a natural gas plant. In the $11 gas case, the results showed that the costs for producing steam with a nuclear plant were much less expensive than natural gas fired production, even when the capital costs were overrun by 60%. It is clear that nuclear steam can be competitive with natural gas at foreseeable gas prices, even when great risks are assumed in the capital costs. Nuclear generation at the assumed costs is not shown to be competitive with natural gas for production of electricity until gas prices are as high as $ 10 /MMBtu. The likely reasons for this distinction lie in the very high efficiencies of the natural gas combined cycle electric plant versus the lower efficiencies and wasted heat associated with a nuclear electric power plant. In the steam case, however, it is much simpler to utilize the full heat output of the nuclear plant, and the comparison with a one-through natural gas boiler is favorable.

This economic analysis has been based on firm foundations with capital costs that are believed to be accurate given the commodity prices at the time of their estimation. However, the recent surge in materials costs affects all large construction projects, and will likely raise the costs of any project, including coal and natural gas plants. When Duke Energy began planning for the construction of two 800 MW coal plants in North Carolina (2004), the cost estimate was for $2 billion. In 2006 it was $3 billion, and in 2007 one unit was canceled and the price for a single unit was projected to be $1.83 billion. This is indicative of the general trend of escalating prices on materials costs throughout North America. When combined with the elevated labor costs of the Fort McMurray area, the resulting project will tend to be much more expensive now than may have been expected ten years ago.

Appendix T: ManTurbo Steam Compressor Information

Overview

Initial evaluation by ManTurbo has been focused on the 1,000,000 lbs/hr turbine steam compressor, which is an appropriate size for meeting 30,000 barrels per day SAGD operational requirements. This evaluation assumes that steam is being compressed from 5.8 MPa to 10.0 MPa.

From an application standpoint, this service is more demanding than most units that are built by ManTurbo. However, from a component, rating and size standpoint, the unit is well within our range of experience with existing machines.

In our opinion, the most logical unit would be an integrally geared compressor consisting of two (2) stages mounted on one (1) pinion shaft each, and entrained by the central bull gear (see Figures T-1 to T-3). The reason for this selection is to limit the concentration of power on each pinion shaft. The unit being described is essentially the ManTurbo model RG 63-2. For performance curves, see Figures T-4 and T-5.

Our engineering group is confident that a machine with the above configuration is feasible and consistent with our current technology base. ManTurbo is prepared to work with SLN in order to refine and optimize the selection, and to undertake the design and demonstration unit construction and performance testing.

As a budgetary cost estimate, the machine described above is valued at US$ 5.5M, including the support structure and lube-oil system, but excluding the steam turbine driver. Note that standard designs for steam turbine drivers are also available.

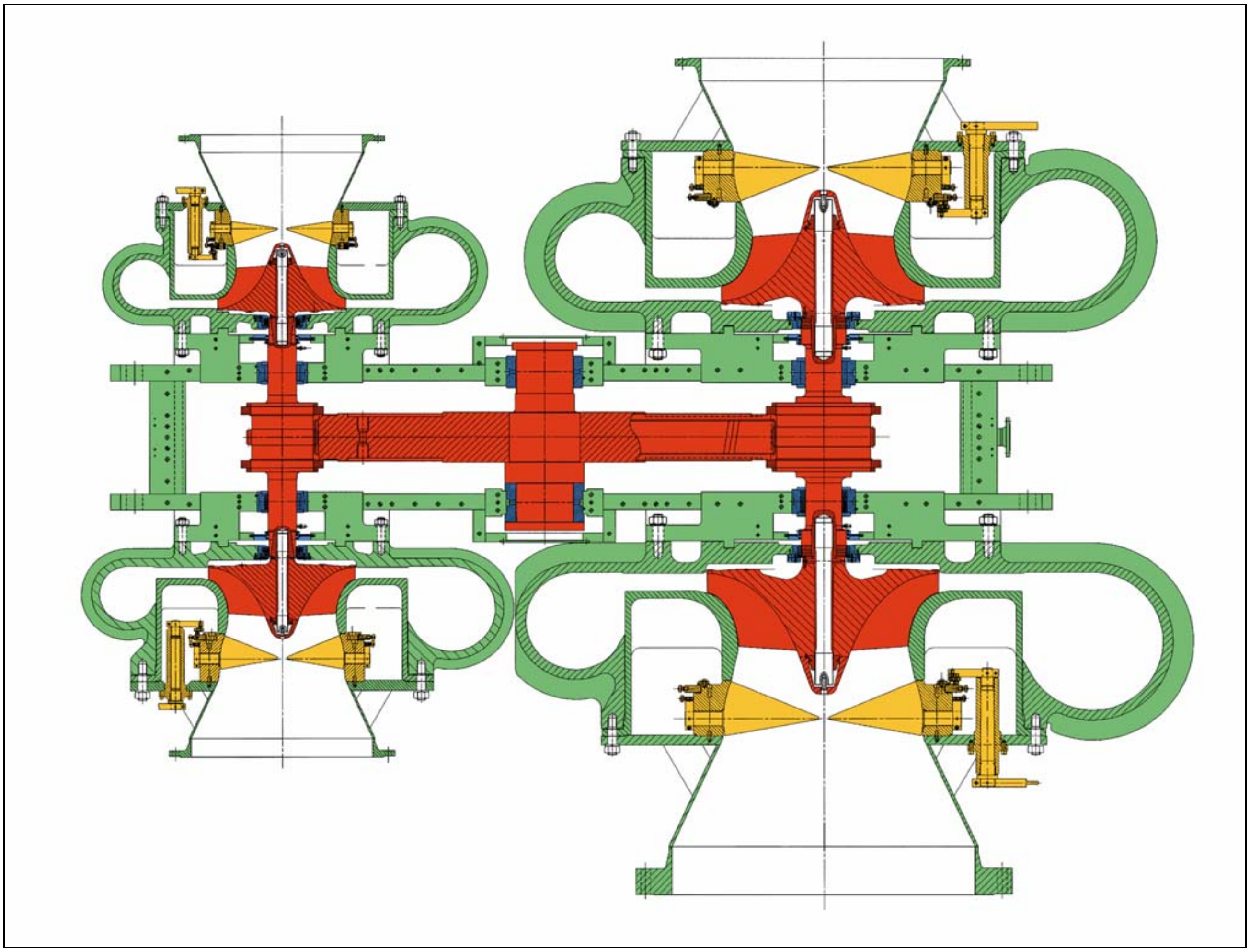


Figure T-1, ManTurbo Integrally Geared Compressor Design: General Design

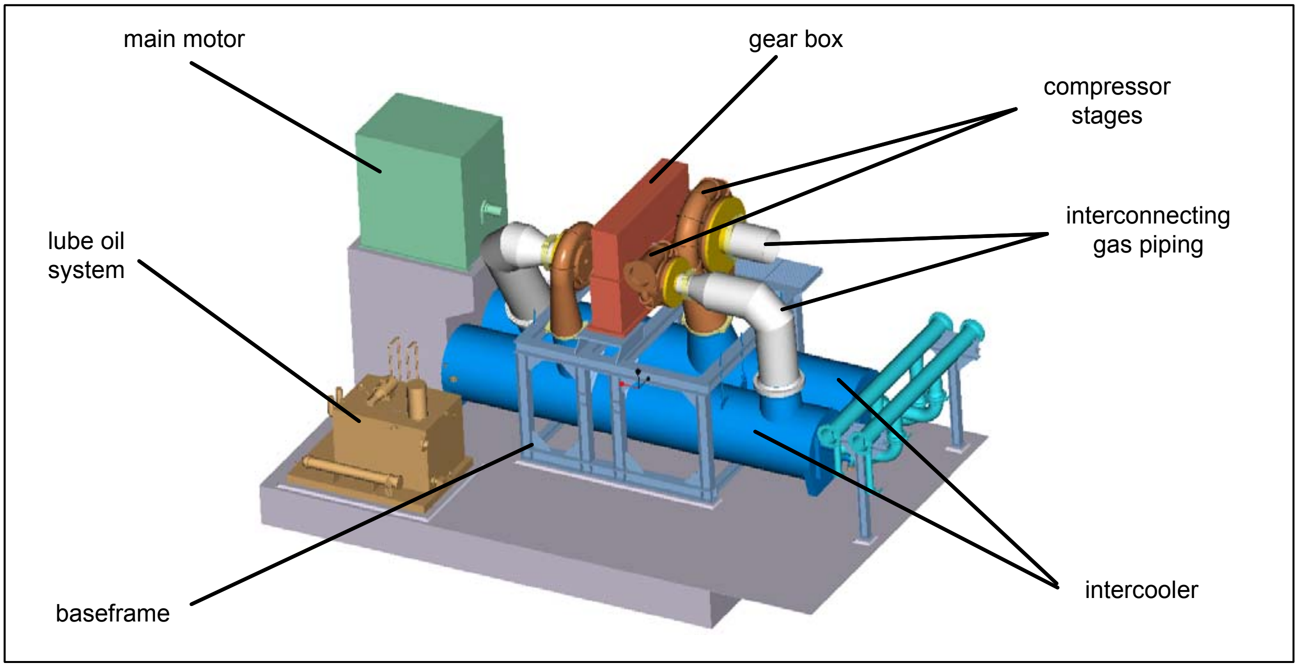


Figure T-2, ManTurbo Integrally Geared Compressor Design: General Arrangement

5 Stage Integrally Geared Compressor (Dry Air) for Air Separation

Steam Turbine Driver: Type RG 53-5

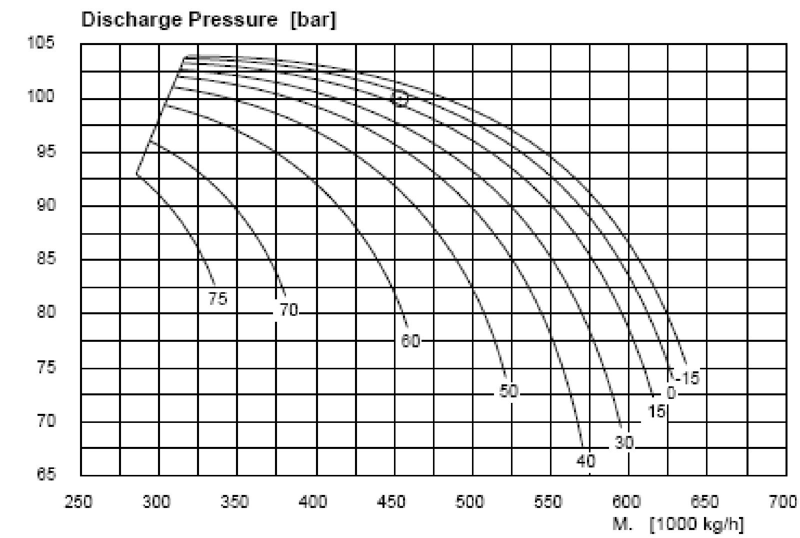
Flow: 29,500 m³/h
Pressure: 6.4 bar - 76 bar
Power: 16,000 kW
ST-Power: 52,000 kW

Figure T-3, ManTurbo 5 Stage Integrally Geared Compressor

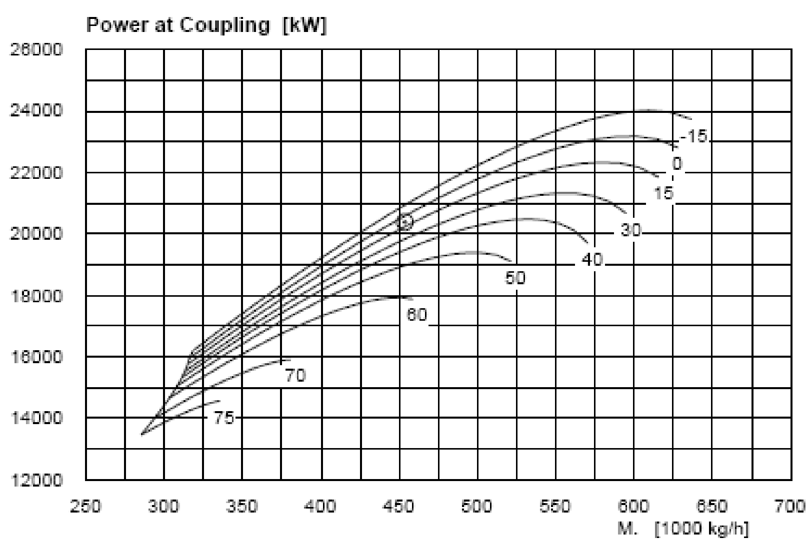
MAN	Centrifugal compressor Type RG 63-2 Predicted Performance Curves	Client: SNC Lavalin Item: Oil Sands Job: 2x 50 % Quote: 5107229

Adjustable inlet guide vanes, stage(s) 1, Guide vane angles are for information only!

					Case
Gas	H2O Vapour	Suction Pressure	[bara]	57.955	1
Molar Mass [kg/kmol]	18.016	Suction Temp.	[°C]	300	
Is. Vol-exponent	1.2963				
Compress. Factor	.83707				



Case 1
M = 453600
Pd = 99.984



Case 1
M = 453600
Pk = 20386

Prepared: M.Heine Checked:	Dept.: SC51	Time: 08:35	Date: 28.01.08	Remarks: Design Estimate	Page 1 / 1

Figure T-4, ManTurbo Centrifugal Compressor Type RG 63-2: Predicted Performance Curves

GHH BORSIG Turbomaschinen GmbH — Integral Gear Compressor, Type RG 63-2

client:	SNC Lavalin	name:	M.Heine
item:	Oil Sands	department:	SC51
quote:	5107229	date:	28.01.08
remarks:	Design Estimate		

conditions at inlet

pressure	[bara]	57,955
temperature	[°C]	300
volumeflow	[m3/s]	4,8139

conditions at exit

pressure	[bara]	100,6
temperature	[°C]	393,13
volumeflow	[m3/s]	3,2305

total for all stages

polytr. efficiency	[%]	84,102
isoth. efficiency	[%]	
inner power	[kW]	
power at coupling	[kW]	20582

gas analysis, vol%

WATER	1

stage data

stage	IGV [-]	LSD [-]	RPM [1/min]	D2 [mm]	U2 [m/s]	MASS [kg/s]	VN0 [Nm3/h]
1	x	-	11763	586	360,92	126	0
2	-	-	12552	537	352,93	126	0

Figure T-5, ManTurbo Integral Gear Compressor Type RG 63-2: Design Estimate Data)

Appendix U: Levelized Unit Energy Cost Details

Detailed LUEC cost information for the NPPs considered in this evaluation is contained in the Excel worksheets listed below. These worksheets are provided together with the report package as separate attachments.

OilSands_LUECfile1.xls

1) ABWR (Single Unit);
2) ACR-1000 (Twin Unit);
3) ACR-1000 (Single Unit);
4) AP1000 (Twin Unit);
5) AP1000 (Single Unit);
6) CANDU 6E (Twin Unit);
7) CANDU 6E (Single Unit).

OilSands_LUECfile2.xls

1) EPR (Single Unit);
2) ESBWR (Single Unit);
3) GA-HTR (Four Unit);
4) GA-HTR (Twin Unit);
5) PBMR (Four Unit);
6) PBMR (Twin Unit).