ADVANCED NUCLEAR ENERGY SYSTEMS: HEAT TRANSFER ISSUES and TRENDS

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ABSTRACT
Almost 450 nuclear power plants are currently operating throughout the world and supplying about 17% of the world's electricity. These plants perform safely, reliably, and have no free-release of by-products to the environment. Given the current rate of growth in electricity demand and the ever growing concerns for the environment, the US consumer will favor energy sources that can satisfy the need for electricity and other energy-intensive products (1) on a sustainable basis with minimal environmental impact, (2) with enhanced reliability and safety and (3) competitive economics. Given that advances are made to fully apply the potential benefits of nuclear energy systems, the next generation of nuclear systems can provide a vital part of a long-term, diversified energy supply. The Department of Energy has begun research on such a new generation of nuclear energy systems that can be made available to the market by 2030 or earlier, and that can offer significant advances toward these challenging goals [1]. These future nuclear power systems will require advances in materials, reactor physics as well as heat transfer to realize their full potential. In this paper, a summary of these advanced nuclear power systems is presented along with a short synopsis of the important heat transfer issues. Given the nature of research and the dynamics of these conceptual designs, key aspects of the physics will be provided, with details left for the presentation.

1. FUTURE NUCLEAR POWER: THE CHALLENGE

Nuclear power plant technology in the U.S. can be characterized as three distinct design generations: (1) prototypes, (2) current operating plants, and (3) advanced light water reactors. While the third generation plants have been very successful where they have been built in Europe, Asia and the Pacific Rim, further evolution is needed to make new nuclear energy systems a more attractive option for deployment around the world. In particular, the next generation of nuclear energy systems must be able to be licensed, constructed, and operated in a manner that will provide a competitively priced supply of energy while satisfactorily addressing plant reliability, nuclear safety, waste disposal, proliferation resistance, and public perception concerns of the countries in which they are deployed.

Advanced water-cooled-reactor nuclear energy system concepts have been identified as part of the Generation IV International Roadmap evaluation [1] and R&D planning activity; i.e., involving international laboratories, academia, and industry groups from countries including Argentina, Brazil, Canada, France, Italy, Japan, Korea, Russia, Switzerland, the UK and the U.S. This activity resulted in the proposal of over thirty-eight specific reactor designs. For our purposes, the leading reactor designs can be categorized into two general groups:

- Near-Term Advanced Boiling Water and Pressurized Water Reactors with Passive-Safety
- Longer-Term Advanced Water Reactors – e.g., Supercritical Water Reactor

The first grouping of advanced BWR and PWR systems can be represented by the ESBWR (experimental simplified boiling water reactor) and the AP1000 (advanced pressurized water reactor), while the Supercritical Water Reactor (SCWR) is a unique example of the second grouping.

Advanced reactors have also been proposed that utilize different coolants than water and potentially may allow for more flexibility in operation, improved sustainability and minimizing by-product flows as well as providing the potential for higher outlet temperatures to allow for a wider range of process heat applications; e.g., high-temperature chemical reduction of water to produce hydrogen. Over fifty different concepts have been proposed and the most promising designs can be grouped into:

- Advanced Gas-Cooled Reactors for High Temperatures (PBMR, MHGR, VHTR, GFR)
- Advanced Liquid-Metal Fast Reactors (Sodium-cooled and Lead-alloy-cooled)

The first grouping of advanced gas-cooled reactors can be represented by the VHTR (very-high temperature gas reactor) either with graphite pebbles or with prismatic graphite blocks as moderators. The second grouping can be represented by the integral sodium-cooled fast reactor or the lead-cooled fast reactor, both providing high-temperature process heat with a low pressure cooling circuit.

In order to understand these various advanced nuclear reactor systems and their features, consider the figure below with the average neutron energy and process heat temperature as the key variables.

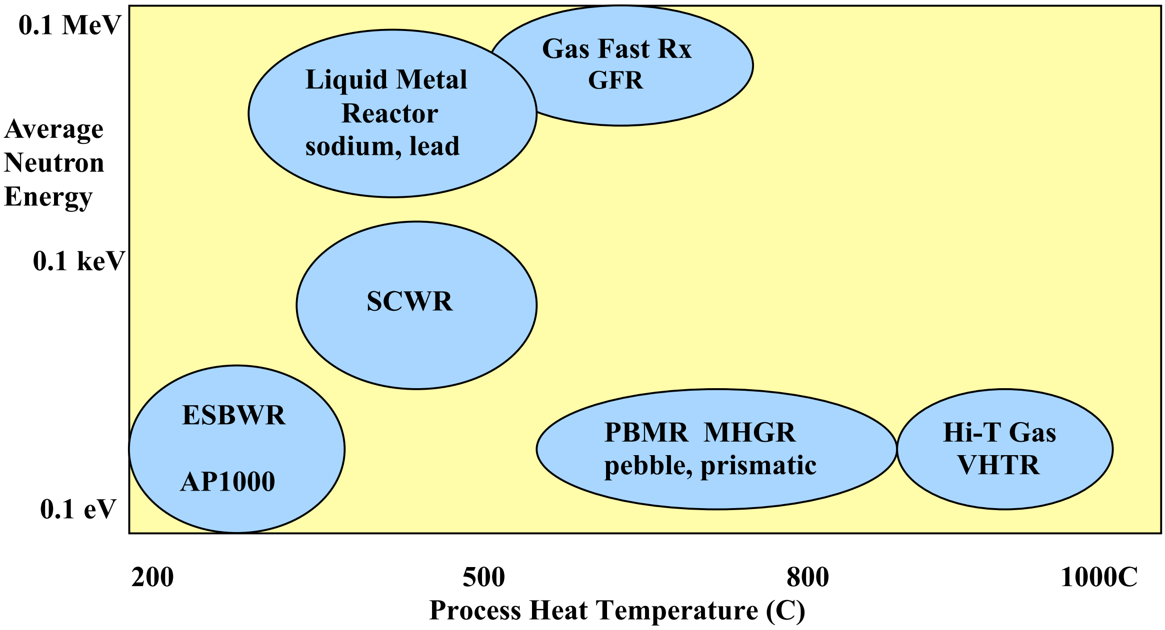


Figure 1: Conceptual Mapping of Generation IV Nuclear Reactor Concepts

If one considers the nuclear reactor as the heat source for a heat engine then the process heat temperature exiting the reactor is indicative of its ability to efficiently produce electricity (via a rankine cycle or brayton cycle), or to be used for producing a variety of energy products; e.g., hydrogen. In addition, the nuclear reactor average neutron energy is an important parameter since it controls the ability of the device to "burn" long-lived radioactive by-products (actinides) as well as provide more flexibility to manage its complete by-product radioactive waste stream. Thus, these various advanced reactor concepts provide a spectrum of possibilities to more safely and efficiently produce high-temperature process heat and increase the neutron energy to manage its by-products.

2. ADVANCED WATER REACTOR SYSTEMS

The major innovation for the ESBWR and the AP1000 over currently operating light-water reactors is the simplification of the safety systems, while retaining the familiar technologies for the boiling water reactor and pressurized water reactor operational concepts. These simplifications are achieved by using passive safety systems that do not need AC power for activation and gravity-driven flows when possible to reduce operator/mechanical actuation.

2.1 ESBWR – An Advanced Boiling Water Reactor

The ESBWR is a 4000 MWth (approximately 1400 MWe), boiling water reactor that uses the same basic passive technology and simplified design as its predecessor (the 2000 MWth SBWR). The system makes use of existing technology whenever possible. The ESBWR plant design relies on the use of natural circulation and passive safety features to enhance the plant performance and simplify the design (such as reductions in the required numbers of control blades and control rod drives). The use of natural circulation has allowed the elimination of several systems—such as the re-circulation pumps. Adequate natural circulation behavior has been achieved using shorter fuel and an improved steam separator (to reduce the pressure drop), and a seven-meter chimney to enhance the driving head.

The overall ESBWR reactor-containment layout is provided in Figures 2 and 3. One important difference between the PWR and the BWR system is that water suppression pools suppress pressure buildup in the containment (drywell) by an increase in the pool sensible energy, since the free volume is much smaller (~ 10000 m^3). Hot gases would be cooled and steam would condense. For long term containment cooling, once the RCS is depressurized, the decay heat released from the fuel must be transported out of the containment drywell to the ultimate heat sink in the environment. To accomplish this task a second isolation condenser system is connected to the drywell and containment atmosphere. Any steam produced by evaporation and containment gases pass through this isolation condenser by natural convection at containment drywell pressures and transfer the heat to an outside water pool, which evaporates and dissipates the heat to the environment. The condensate flows to the GDCS pool while the gases vent to the suppression chamber through the pool.

If a transient event occurs at BWR operating pressures and compromises the normal heat removal path, then the RCS isolation condenser is designed to remove the nuclear fuel decay heat. In the SBWR there are three redundant isolation condensers; any two out of three are required to remove the decay heat from the SBWR at high pressures. They are activated by opening an isolation valve to the condensate return line and natural circulation flow drives the steam through the condenser tubes, transferring the energy to an outside containment water pool. The automatic depressurization system (ADS) is designed to provide a means of depressurizing the RCS given any event that threaten to decrease the RPV water level. For a large break LOCA (such as a main steamline pipe break) the RCS would depressurize to the drywell pressure also without the intervention of the ADS. The ADS system is similar in concept to the current BWR designs in which a number of independent piping runs extend from the reactor vessel upper plenum to the suppression pool.

The major system to insure core cooling at lower pressures is the Gravity-Driven Cooling System (GDCS). The GDCS consists of three elevated pools of water in the containment above the reactor core connected to the vessel by piping and isolated by a check valve (Figure 3) and a squib valve that opens on a low RPV level signal. During a LOCA the RPV would be depressurized by the ADS or by the LOCA, itself, if it is a large break. Under such a circumstance the gravity head of the GDCS would then provide enough pressure to cause the check valve to open and water to flow to the core. Because of the relative elevation of the core, suppression pool provides water injection in the long term when water from GDCS pools is depleted. Three different injection lines, equipped with check valves and squib valves, connect the suppression pool to the RPV. Once this cooling circuit is established, energy is removed from the core over long times by the passive containment cooling system (PCCS).The PCCS is the name given to the containment isolation condenser and associated piping that connects the drywell and wetwell volume to this condenser for eventual heat transport to the environment. It is designed in such a way that no matter where the fuel may reside (in-vessel or ex-vessel) an energy removal path is provided to the ultimate heat sink. Any steam generated by blowdown or water evaporation due to heating would be transported by natural convection upward to the isolation condenser with the containment atmosphere. It then passes through the condenser tube banks where the steam is condensed and the water condensate drains back to the GDCS pools and the noncondensible gases are returned to the wetwell suppression chamber.

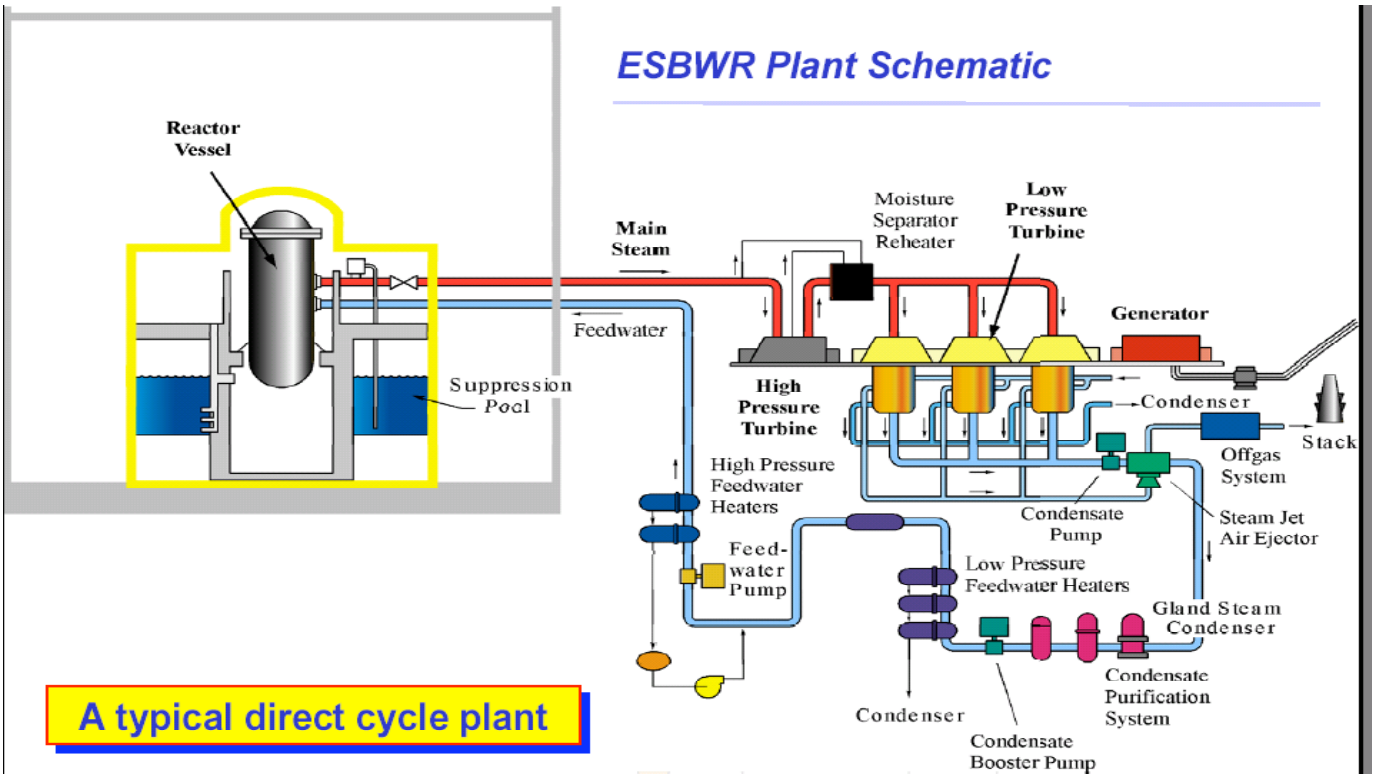
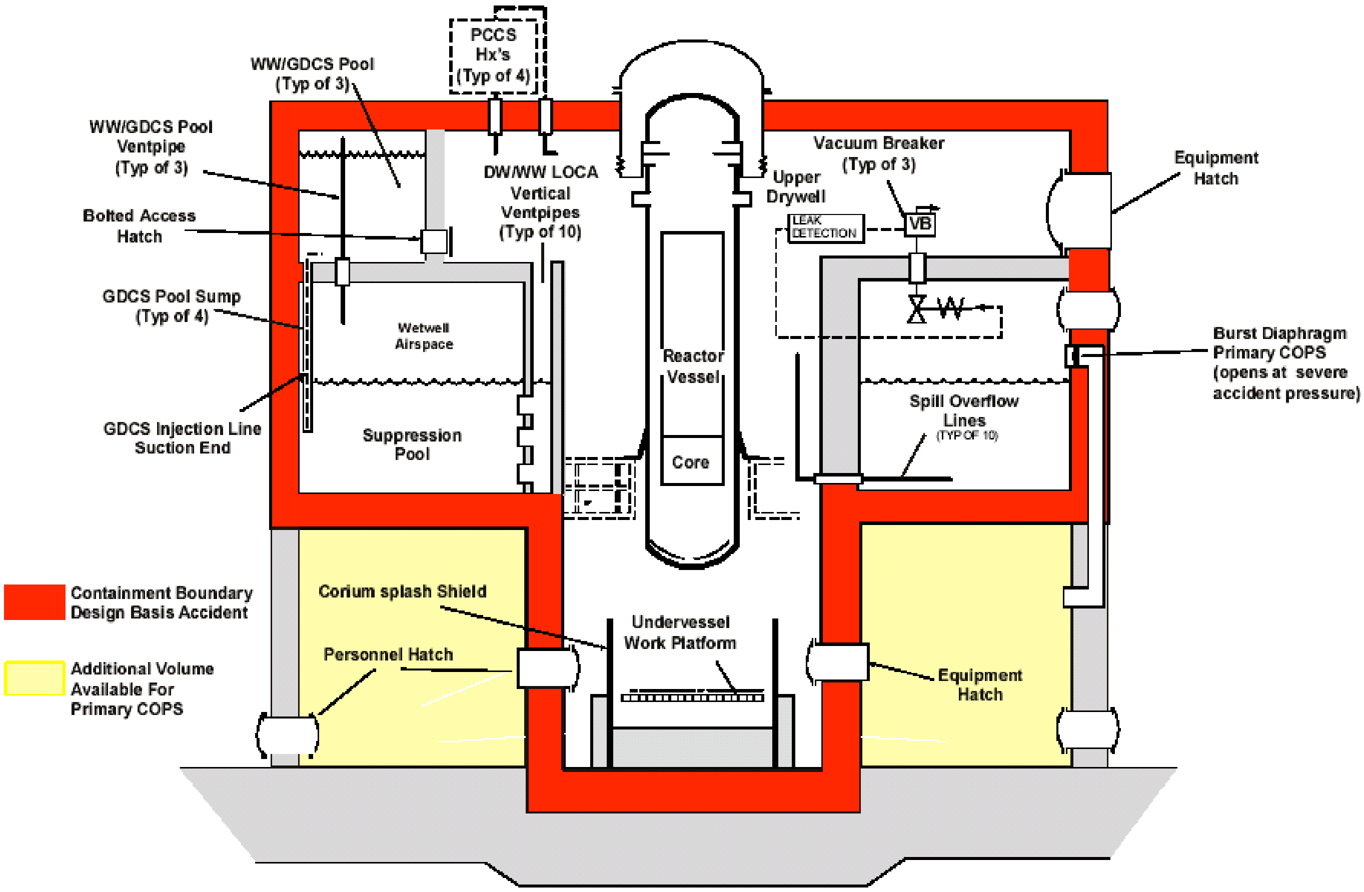


Figure 2 and 3: SBWR Plant and Containment Systems Schematic Diagrams



A severe accident would only occur in the SBWR at elevated system pressures if all of the redundant reactor isolation condensers were to fail to operate or if the ADS system could not depressurize to a level in which the GDCS would discharge water to the core. Similar to the AP1000 the core would boil dry and the first physical process encountered as it overheats would be hydrogen generation due to clad oxidation. Without eventual water addition in the core this would lead to fuel melting and slumping into the RPV lower plenum. Once again fuel-coolant interactions may result. Without fuel quenching in the lower plenum of the vessel, fuel heating would eventually breach the RPV and the core melt pours into the drywell cavity below the vessel. Under the current SBWR design this region of the drywell would be a large area (about 40 m^2) over which the melt could spread. By design it would be partially filled with water (1-3 meters). Thus one must consider the unique multiphase phenomena associated with this design.

(a) System interactions considering condensation with noncondensibles--The SBWR containment isolation condensers require that the steam-gas mixture enter the condenser tube and efficiencly remove steam and pass the gases to the wet well.

(b) Ex-Vessel Fuel-Coolant Interactions--If the fuel melts and penetrates the RPV it will pour into the water-filled lower drywell and may challenge this pressure boundary from FCI dynamic pressures.

(c) Fuel Debris Coolability--The molten fuel which pours into the lower drywell would spread and accumulate on the drywell floor, and must be cooled by the available water to ultimately halt the accident.

2.2 AP1000 – An Advanced Pressurized Water Reactor

The pressurized water reactor (PWR) has historically utilized the safety principle of providing core cooling under all design basis accident circumstances by the ECCS pump system and also allowing for pressure and temperature reduction by mass and energy expansion into the large free volume of the containment following any loss of coolant (LOCA) with mechanical sprays and coolers aiding in pressure suppression. Water was supplied to these active injection systems from a water storage tank with the ultimate heat sink being reached by the component cooling system to a river or lake.

In the AP1000 (formerly AP600) for the core cooling function the high pressure pumps are replaced by the core-makeup tanks (CMT), the low pressure pumps by an elevated in-containment refueling water storage tank (IRWST) and the transition from high to low pressure by a redundant automatic depressurization system (ADS). The logic of this design is to allow for a modest amount of water inventory makeup at high pressure for a small LOCA from the CMT and if water leakage becomes substantial enough then to depressurize through dedicated valves connected to the pressurizer and the hot legs to allow further cooling by water from the IRWST. One notes that because of these system changes the APWR can accomplish core cooling with few active systems.

The overall AP1000 reactor-containment layout is provided in Figures 4 and 5, compared to AP600. The containment design specifically accounts for pressure and temperature suppression by allowing for gas/steam expansion into a large free volume (about 50000 m^3). Any water released from the RCS during an accident is designed to drain into the lower cavity regions of the containment and provide for ex-vessel cooling if needed later in an accident. Any steam or hot gases released to the containment or generated by the fuel decay heat can transport their energy without the need of active systems to the ultimate heat sink. Mass transport of the gas mixture occurs by buoyant natural convection to the upper dome with convective heat transfer to the colder containment walls as well as steam condensation on the cold upper dome and walls. The energy is then conducted through the steel wall evaporating a water film, which can flow down the outer surface of the containment shell. Finally, this heat is transferred to the gas/vapor mixture at ambient conditions by water evaporation and natural convection flow around the steel shell within the external missile shield and discharged to the environment.

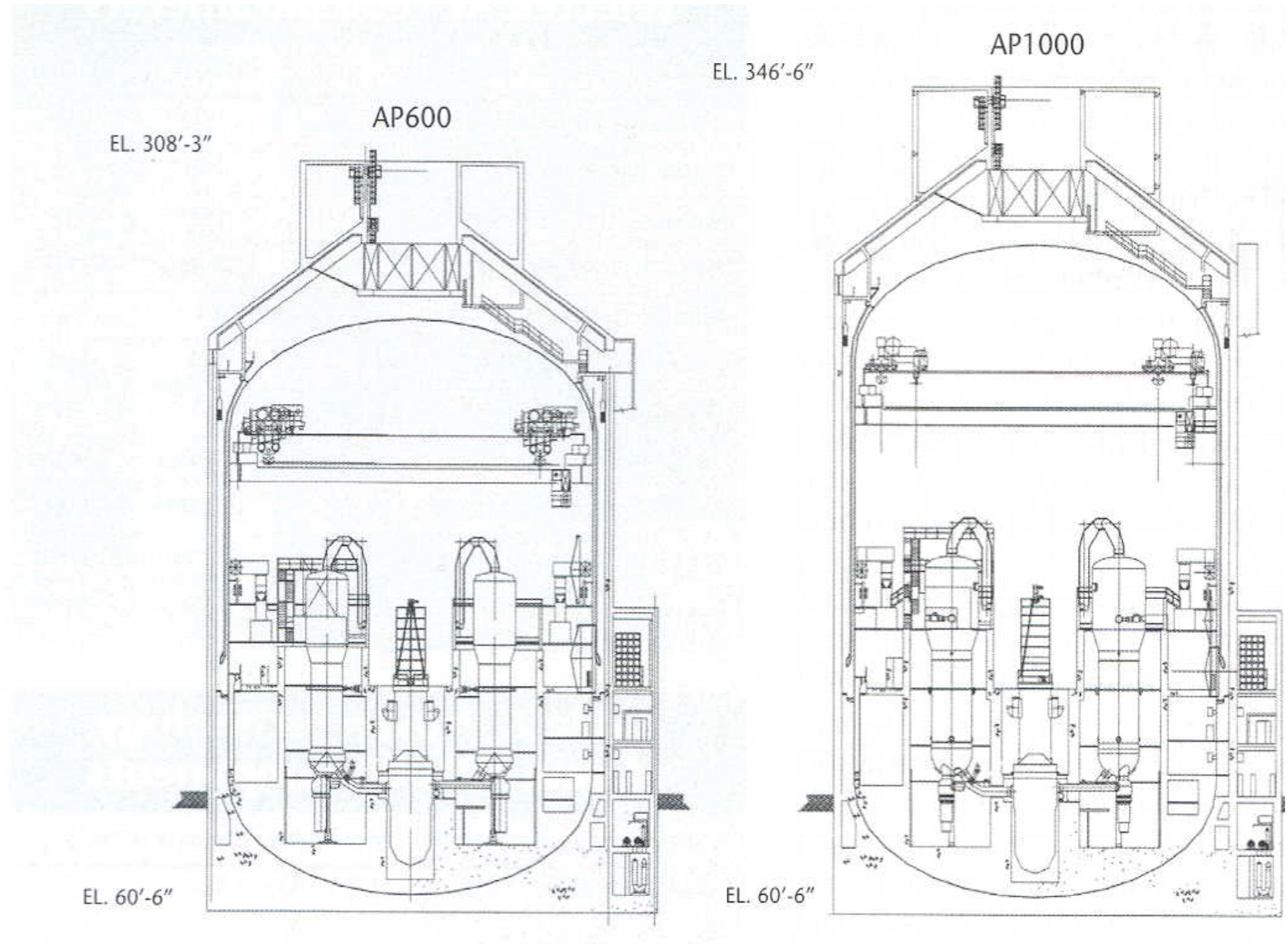
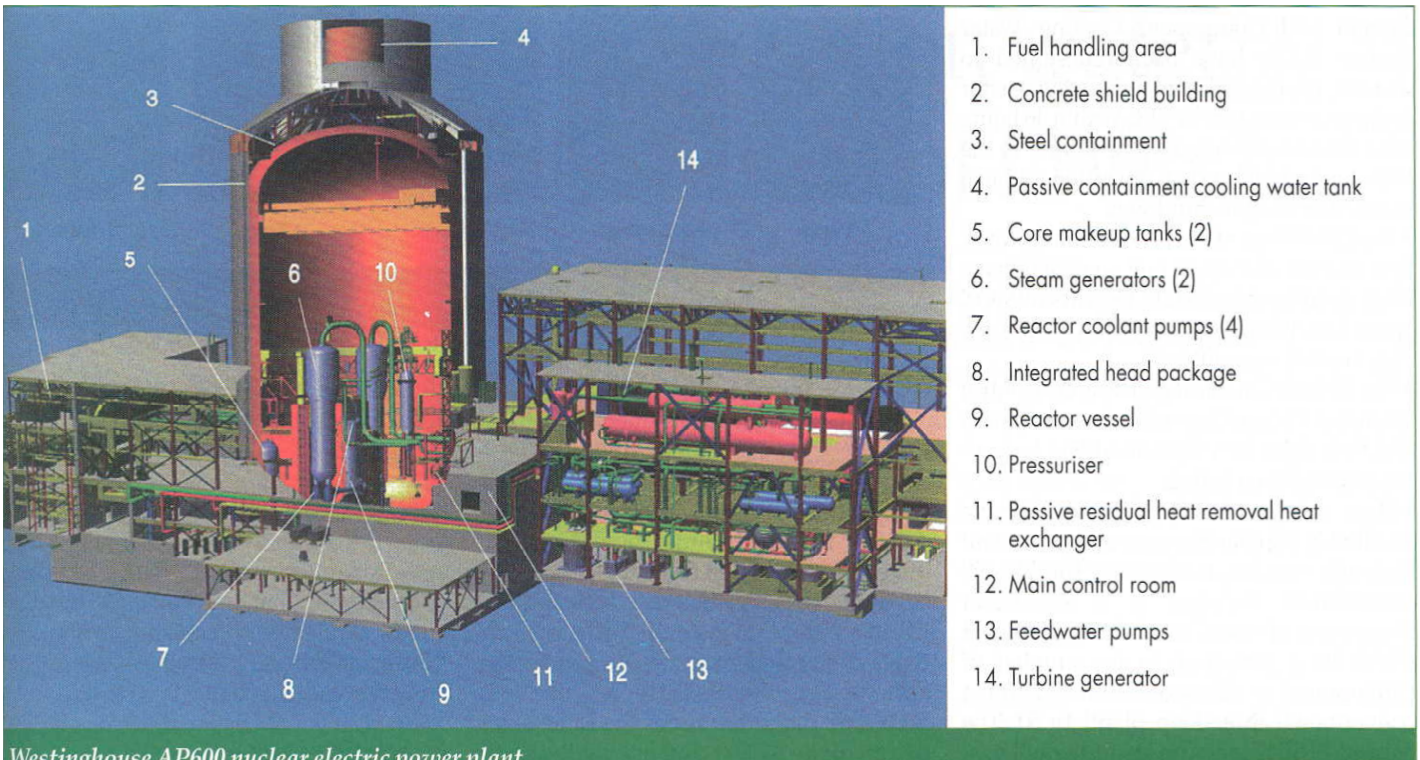


Figure 4: AP600 & AP1000 Comparison and Overall Plant Conceptual Figure



Westinghouse AP600 nuclear electric power plant

Core Makeup Tanks (CMTs) are large high-pressure inventory makeup tanks that supplement small reactor coolant system (RCS) inventory losses at or near the operating pressure. The AP600 design incorporates two such tanks, which supplement the inventory in the RCS. The CMTs are located above the core elevation and allow for passive inventory makeup by a pressure difference and gravity head, through a check valve, between the tank and the RCS. The pressure in the CMTs is maintained with a pressure balance line between the tanks and the top of the pressurizer. If the RCS inventory leakage becomes large the CMT inventory will be depleted, and the depletion of their inventory would signal the activation of the automatic depressurization system (ADS). The ADS is designed to provide a means of depressurizing the RCS for small and inter-mediate size breaks of the reactor coolant piping which cause a LOCA. For a double-ended guillotine (large) break, the RCS would depressurize to the containment pressure before the ADS can respond to the LOCA event and the accumulators would supply the initially needed water inventory.

Such a highly reliable system eliminates the concern of high pressure accident sequences. The In-Containment Refueling Water Storage Tank (IRWST) is a large volume of borated water, located inside containment, that is of sufficient inventory to flood the lower containment above the reactor vessel lower head as well as for refueling operations. The IRWST also serves as a quenching pool for the exhaust of the first three stages of the ADS. In addition, when the RCS is completely vented, the IRWST provides post-accident long-term coolant inventory to the core via a line containing isolation and check valves. The AP600 containment is designed to ensure that the cavity will contain water to some depth regardless of the accident scenarios, from the RCS or the IRWST.

2.3 Advanced Heat Transfer Issues for the ESBWR and AP1000

This advanced reactor-containment design has unique features, which require us to consider multiphase heat transfer and fluid-flow phenomena that had not been considered as crucial in current reactor systems.

(a) Multiphase flows under buoyancy driven and natural circulation conditions – Current light water reactor designs utilize active cooling with pumps and high-pressure tanks, whereas these advanced designs rely on flows generated by modest pressure changes and this can be complicated by boiling phenomena.

(b) Condensation with noncondensibles - The AP600 containment primarily relies on steam condensation in the presence of noncondensibles (air and possibly hydrogen) on the inner containment walls as well as water evaporation off the outer surface as a means to remove the nuclear decay heat from the fuel.

(c) Lower Head Cooling - The RCS and IRWST water can be discharged into containment and flood the reactor cavity above the lower head of the reactor vessel; this downward facing heat transfer phenomena is key to preserving the vessel integrity.

(d) Fuel Debris Coolability - The molten fuel that may accumulate on the inside of the reactor vessel head or be discharged into the cavity must be cooled by available water to halt the accident progression.

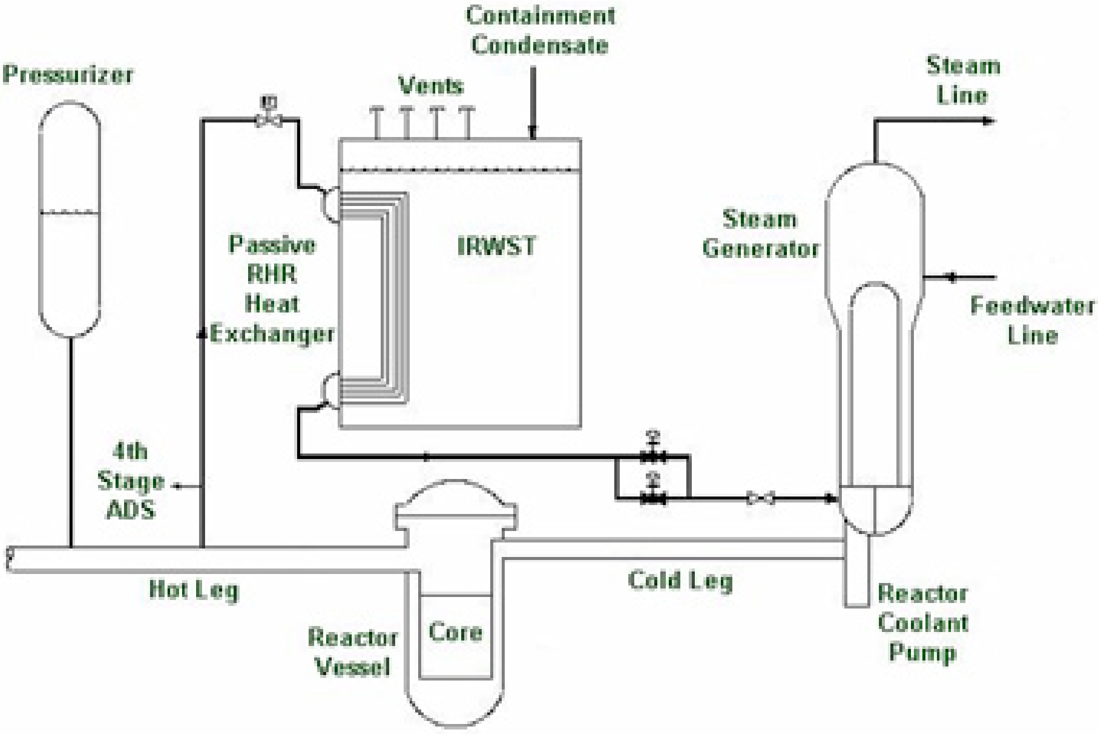
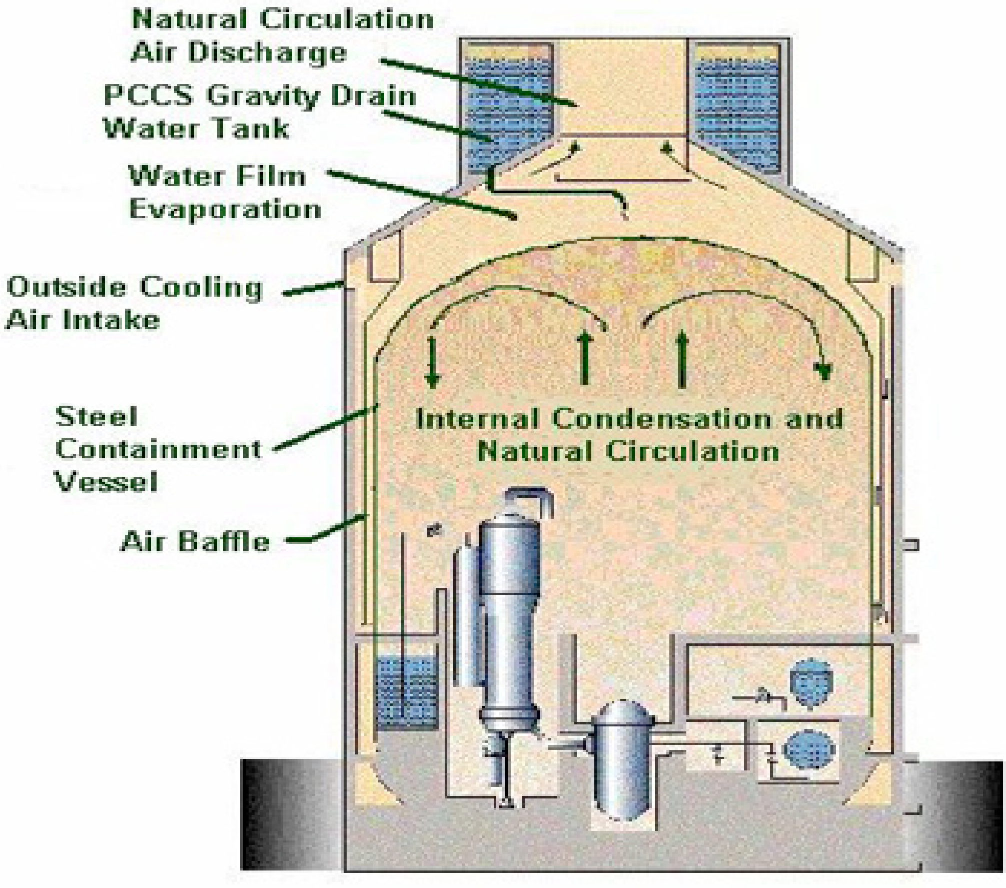


Figure 5: Passive Safety Features of the AP600 and AP1000 Reactor Designs



2.4 Supercritical Water Reactors – SCWR

A supercritical light water reactor would operate with a water coolant above the critical temperature and pressure for water (374°C, 221 bar - 705°F, 3208 psia). Conceptually this is analogous to the supercritical fossil fuel plants now being constructed in the United States – typical operating parameters for coal-burning supercritical plants is 500-600C ant 25-30Mpa. The key advantages to the concept that are derived from the use of higher temperatures during heat addition include:

- Significant increases in thermal efficiency can be achieved relative to current generation LWRs. Estimated efficiencies for supercritical water-cooled reactors are in the range of 40-45% compared to 32-34% for current LWRs.
- A higher heat transfer rate per unit mass flow would result from the large specific enthalpy above the critical point. This leads to a reduction in the reactor coolant pumping power and higher fuel cladding-to-coolant heat transfer coefficients.
- A lower coolant mass inventory results from the reduced coolant density as well as a lower reactor coolant system required heat content. This results in lower containment loadings during a design basis LOCAs and a much smaller containment structure.
- No critical heat flux exists due to lack of a second phase, thereby eliminating heat transfer regime discontinuities within the reactor core. However, an excessive increase in heat flux and/or decrease in coolant flow will cause a predictable heat transfer deterioration in supercritical water-cooled reactors.
- Because the coolant does not undergo a change of phase, the need for steam dryers, steam separators, and re-circulation pumps, as well as steam generators, is eliminated.

The supercritical water reactor concepts are summarized in Table 1 and grouped into three categories: the supercritical, light-water cooled, thermal spectrum reactor design, the supercritical light-water cooled, heavy water-moderated reactor designs, supercritical, light-water-cooled fast reactor designs.

Supercritical Light Water Cooled Thermal Reactors
The Japanese supercritical light water thermal spectrum reactor (SCLWR) has been the subject of considerable development work over about the last ten years. The SCLWR reactor vessel is similar in design to the advanced boiling water reactor (ABWR) or the ESBWR. High-pressure (250 bar) coolant enters the vessel at 280°C. The inlet flow splits, partly to a downcomer and partly to a plenum at the top of the core to flow downward through the core in special moderator water rods to the inlet plenum. This strategy is employed to provide uniform moderation throughout the core.

Table 1: Summary of Generation IV Supercritical Water Cooled Reactor Concepts

Organization	Concept Name	Moderator	Rating MWe	Outlet Temp	Net Efficiency	Comments
University of Tokyo	Thermal spectrum super-critical water cooled	H_2O	1700	508 C	44 %	Once-through, direct cycle
Idaho Nat'l Engr Lab	Fast spectrum super-critical water cooled	H_2O	1500	Varied	38-45 %	Can burn actinides
Atomic Energy of Canada	CANDU- X1	D_2O	950	450 C	40.6 %	Dual-cycle- SCW reactor feeds VHP turbine. VHP turbine exhaust feeds SG with traditional indirect cycle

The coolant is heated to 508°C and delivered to a secondary cycle which looks like a blend of LWR and supercritical fossil technology: high- intermediate- and low-pressure turbines are employed with two re-heaters as in advanced BWRs. The fuel rods are arranged in a tight pitch (square or triangular) with a water rod in the center (Figure 6). Positive reactivity insertion, during core flooding, is managed by control rods, as in a BWR. This is a passively safe, high leakage core that can use either fertile or fertile-free fuel, depending on whether the objective is to maximize the actinide burning or maximize plant capacity factors and minimize fuel cycle costs. The SCWR designs all are designed to use the passive safety features of the ESBWR and the overall cycle design of advanced BWRs.

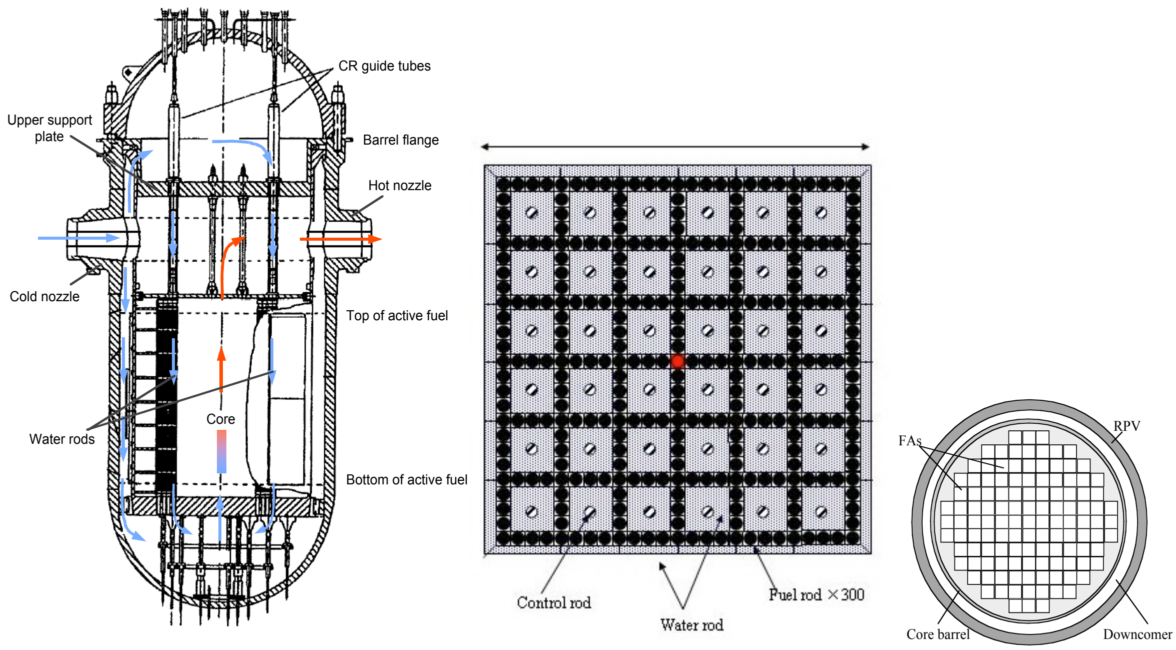
<u>Supercritical Light Water Cooled, Heavy Water Moderated Reactors</u>

The AECL CANDU reactor concept is at a similar level of conceptual maturity as the SCLWR. The AECL has investigated both indirect (steam generator) and combined direct cycles using very high-pressure turbines. They have also examined a lower power system with natural circulation on the primary side. These designs are based on many of the standard CANDU features, including horizontal pressure tubes fueled with short fuel bundles and surrounded by a low-temperature heavy water (D_2O) moderator (on-line refueling is possible but not required in these designs). The major innovations in these supercritical CANDU energy systems relevant to current CANDUs are: (a) a more compact core design (pressure tube spacing and fuel lattice spacing are adjusted to improve overall cost and safety issues), (b) slightly enriched urania fuel in pressure tube bundles, (c) higher thermal efficiency caused by higher outlet temperatures as well as higher pressures in tubes, and (d) enhanced passive safety systems.

<u>Supercritical Light Water Cooled Fast Reactors</u>

Supercritical water reactors can also be designed to operate as fast reactors. The difference between a thermal and a fast supercritical water-cooled reactor is in the lattice pitch and the use of additional moderator material. The fast spectrum reactors use a tight lattice and no additional moderator material. This is a very promising concept that may allow for actinide burning in a water reactor. Researchers have proposed the use of a simple, blanket-free pancake shaped core with streaming assemblies to make a fuel self-sufficient reactor that retains a hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel, while efficiently generating electricity.

Figure 6: SCWR Reactor Vessel, Fuel Rod Assembly and Overall Core Geometry



2.5 Advanced Heat Transfer Issues for the SCWR Concept

The SCWR reactor concept is unique in that it will require designers to demonstrate that the reactor and fuel design limits can be met under supercritical water conditions. These design limits are for the fuel rod cladding and fuel structural integrity as well as stability during startup/shutdown and load variations. This leads one to investigate the following issues in heat transfer and associated fluid flow:

(a) Supercritical heat transfer – Current understanding of heat transfer from heated surface to a supercritical fluid is quite limited. Empirical correlations have been developed by a few investigators over the last two decades (e.g., Bishop, Jackson) but agreement is disappointing. This will require fundamental heat transfer studies under supercritical conditions. Simulant fluids may be useful and reduces the need for prototypic testing under many conditions.

(a) Supercritical flow stability – The ability to vary the reactor power and flow for startup and shutdown as well as for load variations is required for power plant operation. Even though the supercritical fluid does not undergo phase change, the large change in density can cause unstable power-flow behavior. The conditions under which this occurs needs to be studied.

3. ADVANCED GAS-COOLED and LIQUID-METAL REACTORS

As noted previously, advanced reactors have been proposed that utilize various coolants, that potentially may allow for more flexibility in operation, improved sustainability and minimizing by-product flows as well as providing the potential for higher outlet temperatures to allow for a wider range of process heat applications; e.g., high-temperature chemical reduction of water to produce hydrogen. The SCWR is the only water-cooled concept that is flexible enough to achieve these goals, but with gas-cooling and liquid-metal-cooling, the spectrum of possibilities are broadly expanded.

3.1 Advanced Gas-Cooled Reactors

The gas-cooled reactor systems can be grouped into four concept sets representing similarities and common capabilities and attributes among the concepts:

- Pebble Bed Reactor Systems (thermal neutron energy with process heat ~ 600C)
- Prismatic Fuel Reactor Systems (thermal neutron energy with process heat ~ 800C)
- Very High Temperature Reactor Systems (thermal neutron energy at hi-temp. ~1000C)
- Gas Fast Reactor Systems (fast reactor at high neutron energy with process heat ~700C)

The key point to emphasize is that these systems are designed to increase the process heat temperature for the production of energy products and for higher neutron energies for potential actinide “burning”.

<u>Pebble Bed Modular Reactor Systems (PBMR)</u>

The PBMR concepts use helium gas as the coolant, a graphite moderator, and refractory coated particle fuel. The TRISO-coated microspheres are contained in a 6 cm ball configuration as the fuel form (the “pebble”). In contrast to the prismatic fuel, the coated particles are homogeneously distributed within the spherical fuel matrix, which acts as the moderator. There are two generic concepts for pebble bed reactors in terms of refueling. The most common is the online multiple-recirculating feeding system (MEDUL) in which pebbles are continuously removed, controlled with regard to their burn-up and mechanical integrity and then transported back on top of the reactor core if they did not yet reach the burn-up target. Fresh fuel is only added as needed to maintain criticality. Another type is the once through then out (OTTO) concept where the pebbles only perform one passage through the core. For the on-line refueling designs, the pebbles are circulated by gravity in the core, which is surrounded by a graphite reflector and pneumatically transported in the fuel handling system. Normally the graphite reflector has to resist high neutron flux that accumulates in its lifetime.

The PBMR concepts use a thermal neutron spectra and are 'naturally safe', i.e., designed to maintain fuel integrity under all design basis accidents with no active safety system requirements. The all-ceramic core, low power density, and low excess reactivity enable this natural safety. The PBMR design also exhibits high efficiency, generally based upon direct cycle gas turbine power conversion systems, with or without a bottoming cycle using the relatively high exit temperature (about 600 C) helium from the turbine. Reference PBMRs are 250 MW thermal and 115 MW electrical; variations use intermediate heat exchangers to facilitate process heat applications or steam cycle power conversion, while maintaining moisture isolation from the primary coolant circuit.

Prismatic Modular Reactor Systems (MHGR)

Key design characteristics of the Modular Helium Reactor (MHGR) are the use of helium coolant, graphite moderator, and refractory coated particle fuel in a geometric form different than the PBMRs. The refractory coated particle fuel is again the TRISO coated particle fuel, and consists of a spherical kernel of fissile or fertile material, as appropriate for the application, encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an impermeable barrier to the release of gaseous and metallic fission products. The TRISO coated particles are mixed with a matrix and formed into cylindrical fuel compacts, approximately 13 mm in diameter and 51 mm long. The fuel compacts are loaded into fuel channels in hexagonal graphite fuel elements, 793 mm long by 360 mm across flats. Over one hundred columns of the hexagonal fuel elements are stacked 10 elements high to form an annular core. Reflector graphite blocks are provided inside and outside of the active core. The MHGR uses a thermal neutron spectrum and is designed to maintain fuel integrity under design basis accidents with no active safety system requirements. Batch refueling requires periodic refueling shutdowns, but the fuel cycle flexibility is appreciable. High burnup low enriched uranium (LEU) once-through fuel cycle is the reference approach. The high thermal efficiency of the systems leads to better than current generation fuel utilization. The reference power level is 600 MW thermal and 286 MW electrical. Combinations of low enriched uranium, highly-enriched uranium, plutonium recycle, thorium-uranium, and excess weapons material burning are fuel cycle flexibilities for the MHGR. The reference concept has exit temperatures of about 800 C.

Very High Temperature Reactor Systems (VHTR)

Very High Temperature Reactors Systems (VHTRs) are those reactor concepts that have average coolant outlet temperatures almost 1000°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high temperature process heat applications such as coal gasification or thermo-chemical hydrogen production. While all the gas-cooled reactor concepts considered have sufficiently high temperature to support process heat applications such as desalination or cogeneration, as well as some thermo-chemical processes of interest to alternative fuel production, the VHTRs higher temperatures, however, open a broader and more efficient range. These concepts require substantive improvements in the fuel design, especially in material properties. High temperature alloys, fiber reinforced ceramics or compound materials as well as ZrC coatings of the fuel are promising candidates. Not only are the thermal efficiencies improved or the application potentials for nuclear process heat extended, but higher safety standards and safety margins would be required. The reference concept has a prismatic block-type core and is based on the MHGR design discussed above connected to a steam reformer / steam generator unit in the primary circuit. It is an advanced, high efficiency reactor system, which can supply process heat to a broad spectrum of high temperature and can be used in energy-intensive, non-electric processes.

Gas-Cooled Fast Reactor Systems (GFRs)

These concepts offer a closed fuel cycle through high conversion of fissile materials. A breeding capability around unity may be of interest if the fast reactor is used in a synergistic fuel cycle with light water reactors. Fast reactors using a Brayton direct-cycle have the potential to combine the advantages of high sustainability and economic competitiveness, while making nuclear energy benefit from the most efficient conversion technology today. A major challenge is to develop adequate fuel technologies, and associated core design and treatment processes, for preserving most of the attractive safety features of thermal gas cooled reactors as well as including the benefits of actinide “burning”.

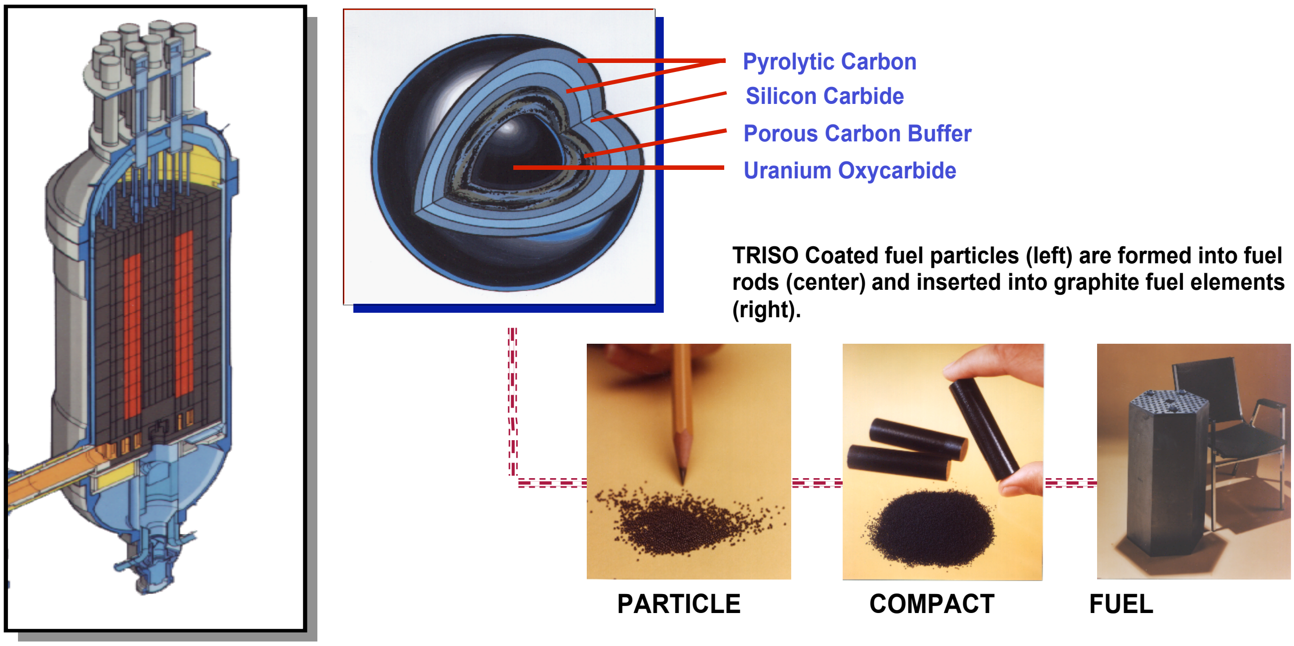


Figure 7: Conceptual Picture of MHGR and TRISO Fuel for Prismatic Core and PBMR Fuel

3.2 Advanced Liquid-Metal Cooled Reactors

A number of diverse liquid-metal cooled reactor concepts have been proposed as part of the Generation IV roadmap process. These reactor concepts are grouped to focus on the coolant type, the fuel-cycle type as well as the reactor safety systems. With these considerations in mind, five groups are defined that encompassed the various liquid-metal cooled reactor concepts:

- Group A: medium-to-large sodium-cooled, mixed-oxide fueled reactors (MOX) with advanced aqueous reprocessing and ceramic fuel pellets with associated fuel fabrication (traditional designs).
- Group B: medium-to-large sodium-cooled, metal-fueled (U-TRU-Zr metal) reactors with electrochemical fuel cycle technology (pyro-processing).
- Group C: medium-sized Pb or Pb-Bi cooled; MOX or Th-U-TRU-Zr metal alloy fueled reactors with a pyroprocess fuel cycle for the metal-fueled concepts, advanced aqueous or unspecified "dry" process for the ceramic fueled concepts.
- Group D: Small, liquid-lead-cooled (or lead-bismuth alloy); metal fuel or nitride fueled reactors with long-life "cartridge" or cassette cores. Fuel cycles vary in specific details.
- Group E: Sodium-cooled concepts that eliminate the traditional secondary sodium loops by development of novel new steam generators that may use direct-contact heat exchange.

Two observations can be made from the grouped concepts. First, technology feasibility and maturity decreases in passing from group A to group E. Group A is nearer-term than group B, which is nearer-term than group C, etc. Second, there is more similarity in the technical features and in the R&D requirements within groups A and B than in groups C, D, and E, particularly because of the use of molten lead or lead alloys in place of or addition to sodium cooling and its associated technologies. Because of this observation, the two liquid-metal-cooled designs that are being seriously considered in the future are a sodium-cooled fast reactor with metal-fuel and pryo-processing for U-PU recycle and actinide “burning” as well as a modular lead-alloy-cooled fast reactor with an extended core-life. Both designs have inherent safety, critical for fast reactor systems and proliferation resistant designs.

3.3 Advanced Heat Transfer Issues for Gas-Cooled and Liquid-Metal-Cooled Reactors

There are a few interesting aspects to these advanced reactors. First, these basic concepts were first examined in the 1950’s and the current designs are improvements upon the prototype reactors and demonstration reactors that have been operated since that time. Thus, the unresolved heat transfer issues affiliated with these core designs are relatively minor. Second, these advanced designs, as modifications of past demonstration projects, seek to innovatively solve the key issues that kept them from going into wide-scale commercial production and those issues are not heat transfer related, but rather deal with fuel-cycle issues and related materials and operational concerns. Finally, the actual heat transfer issues that give rise to interesting future research questions are more related to the power cycle and how the process heat is used to produce electricity or used as a heat source to produce other energy products. Consider two very interesting examples that are now being investigated in the US.

Direct-Contact Heat Exchange: Current sodium-cooled fast reactors use an intermediate loop of liquid-metal to isolate the reactor core and its primary coolant circuit from the rankine power cycle. In these advanced reactor designs, direct-contact heat exchange from a liquid metal to water is currently being considered for steam generation in power production, and could economically remove the need for this intermediate loop in a liquid metal reactor design [2]. The development and evaluation of these direct contact heat transport systems are hampered by the lack of relevant fundamental knowledge of the behavior of multiphase natural circulation under prototypic conditions as well as aerosol generation, to make such devices viable for use in advanced reactor systems. The vigorous agitation of the two fluids, the direct liquid-liquid contact and the resultant interfacial heat transfer area gives rise to large volumetric heat transfer coefficients and rapid steam generation. To optimize heat transfer and flow stability and minimize aerosols for direct-contact heat exchange systems, it is necessary to look at interfacial transport process under reactor-relevant multiphase flow conditions.

Process Heat for Hydrogen Production: The use of hydrogen as a zero-emission portable fuel for combustion as well as for use in fuel-cell technology has recently emerged as a national priority. Fuel cells offer the promise of meeting future global energy needs through the use of hydrogen as a fuel in environmentally clean, quiet and highly efficient devices. While hydrogen fuel cells have a low impact on the environment, current methods to produce hydrogen require high-temperature steam reforming of non-renewable hydrocarbon feedstocks. This approach although technically feasible is economically unattractive and results in overall fuel-processing emissions that exceed what might occur from direct combustion of the hydrocarbon feedstocks. Thus, the challenge to enable hydrogen to be a competitive fuel with gaseous as well as liquid alkanes is to produce it economically and with less by-product emissions (including greenhouse gases) than direct combustion. Nuclear power affords a chance at doing this, since if technologies can be developed [3, 4] that will effectively utilize its process heat for hydrogen production. There are currently three generic ways that seems attractive to pursue:

(a) Low-temperature conversion of carbohydrates from biomass to hydrogen, alkanes and CO_2;
(b) High-temperature condensed phase conversion of hydrocarbon feedstock to hydrogen;
(c) Very-high-temperature thermo-chemical reduction of water to hydrogen and oxygen.

Each method has very unique heat transfer issues associated with it, and could use process heat from nuclear power to minimize or eliminate the generation of by-product emissions from fossil fuel usage. The challenge is to be able to demonstrate this with economics that are competitive with fossil fuels. It is the authors opinion that nuclear power can play an important role in this emerging field.

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