University of California, San Diego

UCSD-CER-14-03

Technology readiness of helium as a fusion power core coolant

M. S. Tillack, P. W. Humrickhouse,
S. Malang and R. E. Nygren

December 2014

Center for Energy Research
University of California, San Diego
9500 Gilman Drive
La Jolla, CA 92093-0417

Center for Energy Research

Technology readiness of helium as a fusion power core coolant

M. S. Tillack[1], P. W. Humrickhouse[2], S. Malang[3], and R. E. Nygren[4]
December 2014

[1] University of California San Diego, 9500 Gilman Drive, La Jolla, CA 92093
[2] Idaho National Laboratory, PO Box 1625 MS 3840, Idaho Falls, ID 83415
[3] Fusion Nuclear Technology Consulting, Fliederweg 3, 76351 Linkenheim, Germany
[4] Sandia National Laboratories, P.O. Box 5800, Albuquerque, NM 87185

1. Introduction

Helium is an attractive coolant for fusion power plant applications due to its chemical inertness (resulting in safety and performance advantages), compatibility with other reactor materials, low neutron cross section, and high temperature capability to enable high thermodynamic efficiencies. Worldwide, a large number of fusion power plant studies have proposed using helium as a coolant in the blanket [1-4], divertor [5-8], and recently even in the vacuum vessel [9,10].

The technology needed for large-scale high-temperature helium-cooled systems already has been developed and implemented in the fission industry [11,12]. Prismatic and pebble bed reactors have operated in the US, Europe and Asia, with electric power generation demonstrated up to levels of 300 MW or more.

Notwithstanding the advantages and past experience base, concerns have been expressed over the use of helium as a coolant in fusion power plants, including limitations in cooling capability [13], impact on power core size (due to the transparency of He to neutrons) and coolant manifold size, a limited industrial supply chain [14], and even limitations in helium resources [15].

The purpose of this document is to summarize existing design concepts and operating parameters for fusion power plant components that use helium as a coolant, describe issues with the use of helium, and explain our rationale for concluding that the issues either can be avoided by design or solved through further R&D. The emphasis is on in-vessel components, including the blanket and divertor, and their required ancillary systems (manifolds, piping and heat exchangers). Power cycles using helium are excluded from our consideration.

Section 2 summarizes recent conceptual studies of fusion power plant blankets and divertors, together with the main design parameters and justification of design choices. Section 3 provides a more detailed examination of the performance issues and existing database. Section 4 reviews the experience base and lessons learned from both the fission industry, which has operated helium and CO_2 gas-cooled reactors since the 1960's, and the international fusion program that has been exploring helium-cooled concepts for several decades. Finally, Section 5 summarizes the main conclusions of this assessment.

2. Review of previous design studies

2.1 Blanket designs using helium as a coolant

Early explorations of the various blanket options for a fusion power plant, such as the BCSS "Blanket Comparison and Selection Study" in the US [16] and the European blanket study [17], considered the possible use of helium as a coolant. Subsequently, several conceptual power plant studies in the US [1-3] and Europe [4] continued to rank helium-cooled options highly. These designs include He-cooled solid breeder (HCSB), He-cooled PbLi (HCLL), and a dual-cooled concept in which both PbLi and He are used as coolants (DCLL). For all of these concepts, reduced-activation ferritic/martensitic steel (RAFS) is used for the structural material, and the first wall is integrated with the blanket in the form of a strong double-walled helium-cooled box with strengthening ribs. R&D programs have been focused on the development of ITER test blanket modules [18] and the subsequent implementation of a Demo blanket [13].

As examples of recent helium-cooled blanket designs, Figures 1 and 2 show the general layout and cross section of the ARIES-ACT2 DCLL blanket [19] and Figures 3 and 4 show the general layout of the EU PPCS power core and a cross section of the HCPB [20] blanket. In the ACT2 dual-cooled blanket, the coolant flows in series through the first wall and then the grid plates. This is done to provide the full flow to the first wall at the lowest temperature possible in order to maintain the structure within its temperature limits. The first wall inlet and outlet manifolds carry the coolant vertically into the individual channels where the coolant flows horizontally. Flow in the grid plates is vertical. A significant fraction of the volumetric heating in the PbLi, perhaps 10%, is conducted into the He grid plates through the flow channel inserts. Approximately 40% of the blanket thermal power is removed by helium.

In most He-cooled blanket concepts, a maximum coolant pressure between 8 MPa and 10 MPa is used. Usually the He inlet temperature is maintained above ~350˚C in order to avoid steel embrittlement under irradiation, and the He outlet temperature is limited to ~480˚C in order to avoid thermal creep that becomes an issue above ~550˚C in conventional RAFS. For example, helium cooling parameters from ACT2 are summarized in Table 1 [19]. Due to the modest surface heat flux and neutron wall loading, the flow speed of He is not very high. The velocity in the first wall channels is constrained in part because larger channels allow for lower stresses in the first wall. (Other studies have shown that He can remove surface heat fluxes up to 1 MW/m^2 or more if needed [21].) The modest velocity leads to He pumping power of ~3% of the thermal power removed by He.

Table 1. Helium parameters for the ARIES-ACT2 blanket

Peak/average neutron wall load	2.2/1.46	MW/m^2
Peak/average surface heat flux	0.28 / 0.22	MW/m^2
Helium pressure	8	MPa
Helium inlet/outlet temperature	385/470	˚C
FW heat transfer coefficient (w/roughening)	6500	W/m^2-K
He velocity in first wall	50	m/s

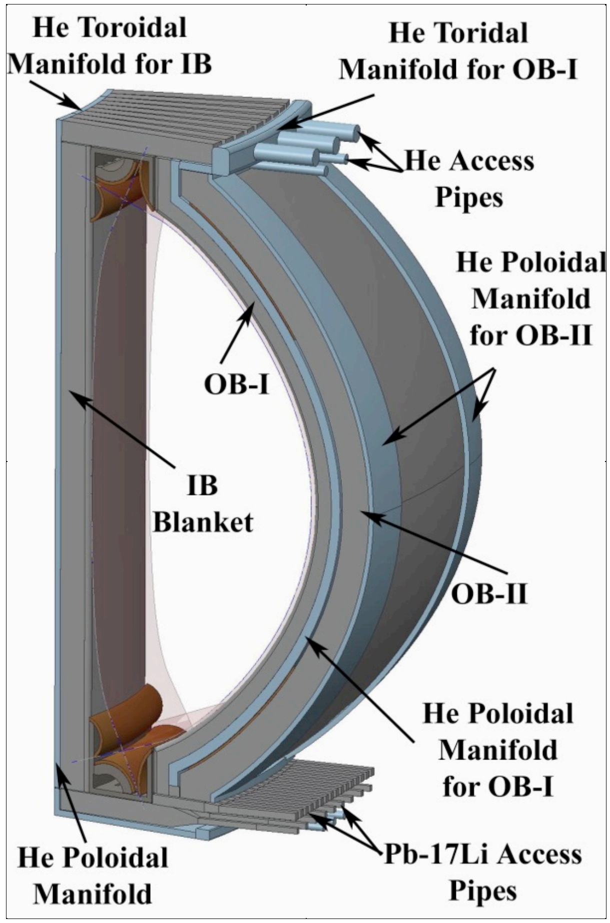

Figure 1. Layout and manifolding for the ARIES-ACT2 blanket [19]

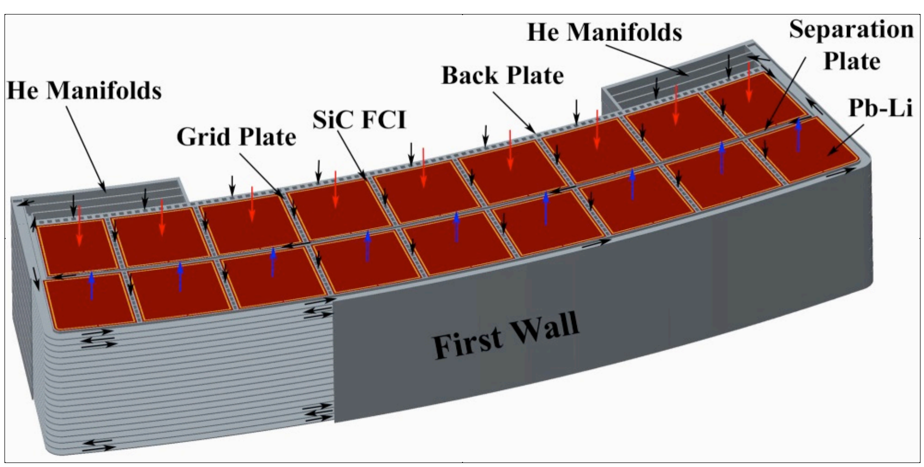

Figure 2. Helium and Pb-17Li flow paths for the ARIES-ACT2 inboard blanket [19]

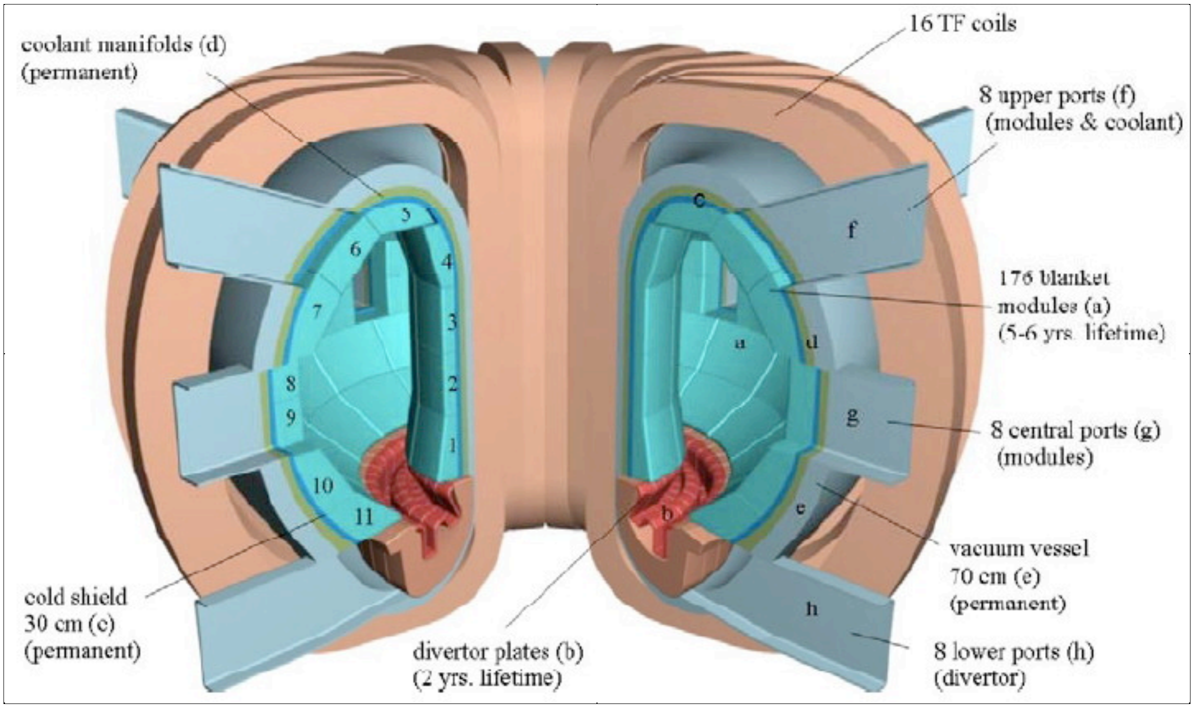

Figure 3. Cut-away view of the fusion power core of the PPCS model C; the other models are broadly similar. [4]

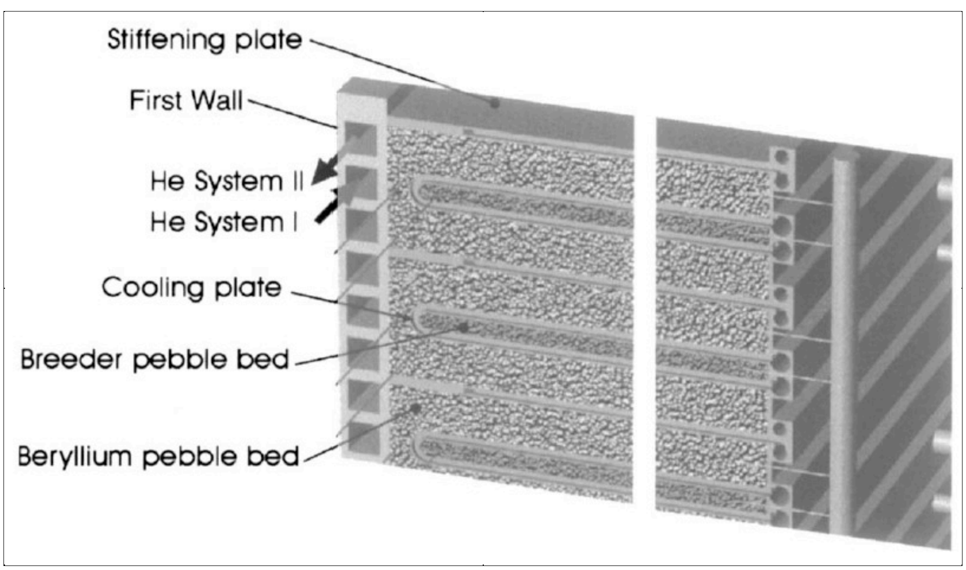

Figure 4. Cross section of the first wall and breeding zone of the HCPB blanket [20]

2.2 Divertor designs using helium as a coolant

Probably the most investigated concept for a steady-state high-power divertor is the water-cooled plate under development for ITER. This concept is based on the use of a heat sink made of a copper alloy (*e.g.,* CuCrZr) and cooled with water at 4 MPa pressure and inlet/outlet temperature in the range of 70-120˚C [22]. Due to the low operating temperature and limited fluence capability of the materials, this concept is considered unsuitable for power plants. The EU PPCS suggested for the Model A (water cooled lead lithium blanket) a divertor concept with water cooling parameters similar their water-cooled blanket (as used in PWR reactors) [23]. However it is very challenging to find suitable structural and heat sink materials for the divertor plates that avoid irradiation induced embrittlement at the relatively low operating temperature of water.

Helium has been explored for over a decade as an alternative divertor coolant. [6,24,25] Several internal cooling configurations have been developed, including impinging slot jets or circular jet arrays [5,8,26,27], porous metal heat sinks [28], normal flow heat exchangers [29] and others. As a general rule, there is a tradeoff between higher performance and complexity.

In most of these designs a tungsten alloy is used for the divertor target plate structural and heat sink material. This reduces thermal stresses arising from differential thermal expansion and provides the best possible thermal contact with the coolant. The plasma-facing tiles are either an integral part of the structure or brazed to it. In order to minimize the thermal stresses in these tiles, they are usually castellated to a size of about 10 mm × 10 mm.

One of the most promising solutions is a modular design in which cup-shaped thimbles with an outer diameter of ~20 mm are brazed into a plate with ~8 mm thickness. To each of these thimbles a tungsten tile is brazed at the front surface, and the inner surface is cooled by high velocity helium jets. This concept (see Figures 5 and 6) is described for example in Wang *et al.* [27] and offers an allowable surface heat flux of about 12 MW/m^2. Typical parameters for these helium-cooled W-alloy divertors are listed in Table 2.

Table 2. Typical parameters for a helium-cooled tungsten alloy divertor

Parameter	**Value**	**Units**
He pressure	8-10	MPa
He inlet temperature	600-700	˚C
He outlet temperature	800	˚C
Pumping power ratio*	<10	%
Maximum allowable temperature of the W-alloy structure	1200	˚C
Maximum allowable temperature of the W armor	2300	˚C
Maximum allowable surface heat flux	10-12	MW/m^2

* ratio of coolant pumping power to thermal power removed

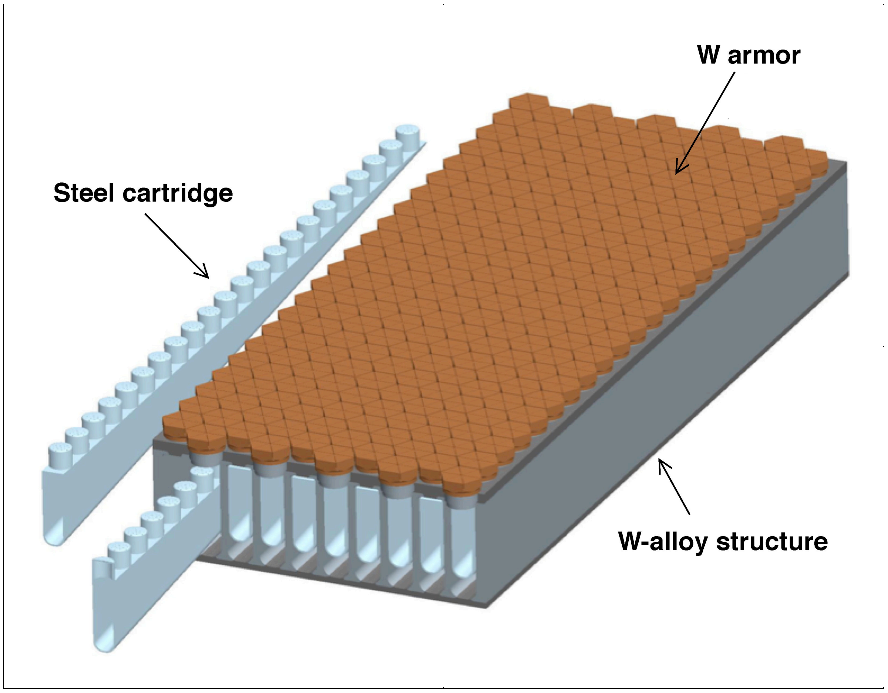

Figure 5. ARIES-ACT1 finger divertor geometry [30]

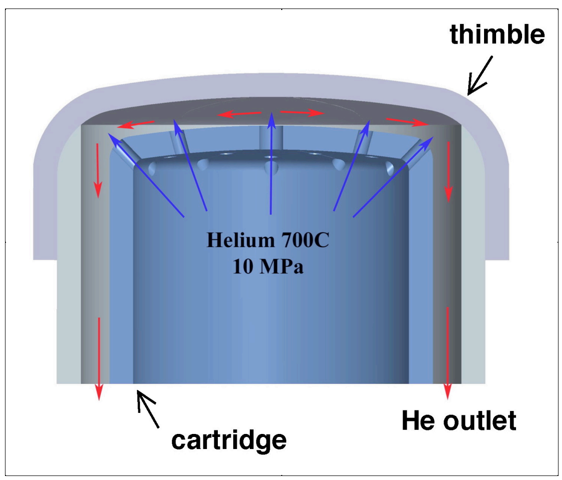

Figure 6. Details of the impinging jets in the finger-type divertor [31]

Optimization of these designs involves the requirement to maintain all of the structural materials within their allowable temperature and stress limits, and to keep the pumping power low enough so as not to unreasonably impact plant efficiency. Typically the maximum allowable pumping power is chosen to be ~10% of the thermal power removed. Considering the high thermal conversion efficiency of helium-cooled systems and the fact that pumping power is mostly recovered as useful heat in the coolant, 10% pumping power ratio corresponds to a net loss of about 20 MW.

Thermomechanical studies have shown that the temperature window of the tungsten structure is usually more constraining than the allowable stresses (although stress limits in irradiated W-alloy are not well understood). This window is determined at the lower end by radiation embrittlement and at the upper end by recrystallization. These temperature limits depend strongly on the tungsten alloy; our best estimates are 700-800˚C on the lower end and 1200-1300˚C on the upper end.

Since tungsten alloy is an improbable choice for the coolant supply lines and heat exchanger, a transition to steel is anticipated. At this location, the temperature limit of steel will influence the design parameters. Current reduced activation steels (such as Eurofer and F82H) are generally limited to temperatures below 550˚C due to loss of creep strength, advanced steels, such as ODS variants, may be required in order to maintain tungsten within its operating temperature window.

All these limits make the selection of a suitable tungsten alloy a very difficult challenge. One promising alloy is VM-tungsten doped with a small amount of potassium (40-70 wppm). This material, which is used extensively for the wires in light bulbs, can be fabricated only as thin wires or sheets. Wire diameter and sheet thickness have to be limited to values < 2 mm, and the mechanical properties depend strongly on the thickness (the thinner the better). Fortunately, for modular finger concept, the W cups can be fabricated from sheets with a thickness of ~1 mm.

3. Performance issues and R&D needs

3.1 Introduction

The claim that helium is limited in performance is often cited as a reason to avoid its use. This concern arises from the relatively low volumetric heat capacity of helium, even at an operating pressure of 8-10 MPa, and the lower heat transfer coefficient in Poiseuille-type transverse flow configurations as compared with liquids. As a result, flow speeds are higher than liquids and the pumping power can be higher in most cases.

However, the performance of helium is very design dependent. Designs that are optimized for water cooling may not perform well with helium. Especially in the divertor, more sophisticated flow geometries are needed to obtain adequate heat removal. For example, impinging jet flows have been studied extensively in the past 10 years and shown to provide adequate cooling for surface heat fluxes beyond 10 MW/m^2 [5-8, 32-34] . The coolant in a fusion power core not only removes heat, but also transfers that heat to the power cycle. The ability to remove heat at very high temperature is a significant advantage for helium.

Any evaluation of performance should be based on several important factors:

1. Is a particular design capable of meeting its design requirements and maintaining all materials within their operating limits.
2. What is the maximum thermal conversion efficiency possible?
3. What is the impact of pumping power on the overal power balance?

Below we address these questions for blankets and divertors in Sections 3.2 and 3.3. In addition, chemistry and tritium control (Section 3.4) as well as safety aspects of He coolant (Section 3.4) are discussed.

3.2 Blanket heat transfer and pumping power

Predictions of the maximum surface heat flux in a power plant first wall are typically in the range of 0.25 MW/m^2. [3] These values assume a separation between the plasma outermost flux surface and the first wall so that plasma does not directly contact the surface. During startup and shutdown operation, specially-design limiters may be required to maintain this separation. For example, ITER specifies as much as 5 MW/m^2 heat flux during startup on some first wall panels that are used as temporary limiters.

Short timescale transients such as ELMs or disuptions do not significantly impact heat transfer into the coolant. The transient typically affects only the near-surface region in the first wall, and is not felt as deeply as the coolant-wall interface.

Based on expected values of heat flux in a power plant, a helium-cooled first wall and blanket can be designed with relatively simple transverse flow paths; more exotic flow schemes are not required. However, simple enhanements have been proposed to provide a robust operating point that maintains all structures within their operating temperature windows. For example, single-sided artificial wall roughening has been shown to enhance the heat transfer coefficient by a factor of 2, allowing heat flux as high as 1 MW/m^2 with a pure ferritic steel wall [35]. The main constraint on the design window comes from the solid wall and not the coolant. With enhancements to the wall conductivity, such as embedded W pins, heat flux as high as 2 MW/m^2 could be removed with simple transverse flow paths. [21] Another possible method to maximize the allowable surface heat flux to the first wall is to replace the ordinary steel with an ODS-steel layer ~3 mm thick on the plasma-facing side with a higher maximum allowable temperature of ~650˚C.

With flow velocity of the order of 20 m/s and Reynolds number of the order of 10^5, a heat transfer coefficient of ~4000 W/m^{2}K is obtained. (For comparison, values for water in the ITER blanket shield block are of the order of 10^4 W/m^{2}K [36].) Including volumetric heating in the wall, which contributes significantly to the surface heat flux, this results in a film temperature rise of about 75˚K. With a steel temperature window of 200˚C, this leaves adequate margin for the temperature rise in the bulk coolant and in the plasma-facing steel wall.

An important concern with helium cooling is the required pumping power. For the design of power plant blankets, usually a limit of 3 to 5% of the thermal power is applied. This would result in a minimal impact on the overall plant efficiency. The pumping power causes heating in the coolant, which can be extracted in the power cycle and recovered at the cycle conversion efficiency. The required pumping power for the He-blowers is compensated by the higher power conversion efficiency of a plant with He-cooled blankets compared to water-cooled systems, enabled by the higher coolant outlet temperature from He-cooled blankets. For example the total efficiency is below ~33% with water cooled blankets, ~37% with He-cooled blankets, and ~45% with DCLL blankets.

The heat flux that would correspond with a ratio of 5% is plotted in Figure 7 as the bottom curve. A heat flux below this curve would result in a ratio above 5%. The top curve in Figure 7 shows the surface heat flux that would result in a peak steel temperature of 550˚C, as a function of the velocity. Any heat flux above this curve would violate the peak streel temperature limit. Therefore, the design window exists between the two curves. As can be seen, for a modest velocity in the range of 40 m/s there is a wide design window extending up to 1 MW/m^2.

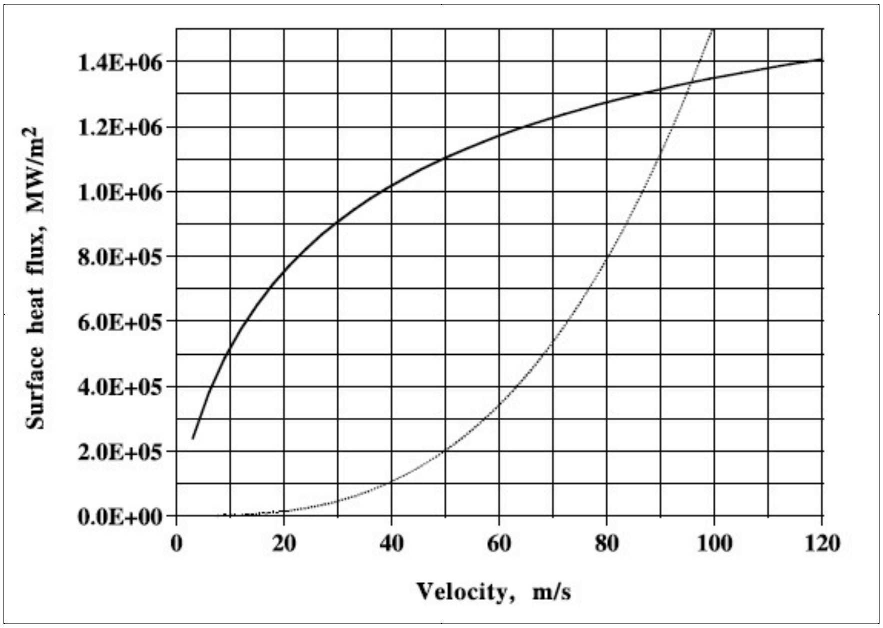

Figure 7. Heat flux window of operation in the ARIES-ST outboard first wall using 5% allowable pumping power [35]

3.3 Divertor heat transfer and pumping power

Values of heat transfer coefficient in the range of 50-100 kW/m^{2}K have been predicted and experimentally verified for helium impinging jet configurations (as opposed to 10 kW/m^{2}K for water in Poiseuille flow or 20-30 kW/m^{2}K in the hypervapotron configuration [37]). With a heat flux of 10 MW/m^2, a film drop of 100-200˚K is predicted. The design window of tungsten is approximately 300-400˚K, leaving margin for conduction in the tungsten wall (provided the structural wall is not too thick) and bulk coolant temperature rise of ~100˚K.

As with the blanket, heat transfer can be improved at the expense of increased pumping power. Due to the greater challenge of heat removal in the divertor, a higher allowable pumping power fraction is applied. For example, Figure 8 shows the pumping power fraction in two leading design concepts: a plate divertor with slot jets and a finger divertor with circular jet arrays. If 10% pumping power is allowed, then the plate divertor can handle up to 9 MW/m^2 while maintaining the structural materials within their limits, whereas the finger divertor may achieve as high as 15 MW/m^2. The dark and light curves assume different inlet He temperature (600 or 700˚C), which is uncertain due to uncertainties in the minimum allowable tungsten alloy temperature.

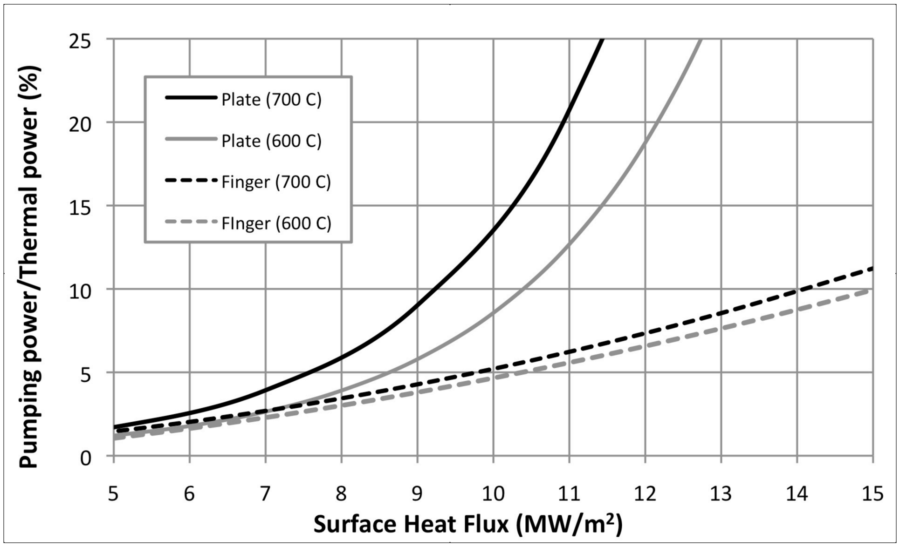

Figure 8. Heat flux temperature window for the ARIES-ACT divertor [31]

3.4 Chemistry and tritium control

While the chemical inertness of helium is one of its principal advantages as a coolant, deleterious effects can arise from impurities in the coolant, and it will be necessary to ensure that any such impurities are kept to acceptably low levels. There is a considerable experience with these issues in helium cooled fission reactors, and we summarize the salient points of some recent reviews [38, 39] in this section, and assess the applicability of this experience to fusion. A more general review of gas-cooled fission reactor operating experience is given in Section 4.

The helium cooled fission reactors all had circulating impurities at ~ppm levels (see [38, 39] for details) of H_2O, H_2, CO, CO_2, CH_4, and N_2. The carbon-based impurities may arise from both the large amount of graphite in the core and from oil-based lubricants if these are used. As a primary objective of this reactor type is to obtain high (up to 950°C) helium outlet temperatures, structural materials suitable for use at these temperatures must be used.

Materials of interest include the nickel alloys Incoloy 800H, and more recently, Inconel 617 and Haynes 230 for their superior creep resistance above 850°C. Incoloy 800 is used in the EU HCPB DEMO design in the steam generator/heat exchanger [40], and so the HTGR experience and knowledge base with these alloys is directly applicable to fusion.

Depending on the relative concentrations of the above mentioned impurities, these alloys may undergo corrosive oxidation, carburization, or de-carburization, which all impact mechanical stability. All of these are prevented by the formation of a protective Cr_2O_3 layer at the surface of the metal, and this layer is successfully maintained provided that the coolant environment is slightly oxidizing. All the helium cooled fission reactors have used purification systems to maintain this condition, and were successful in doing so. These were all similar in nature, and consisted of (1) a filter to trap particulates and plate out fission products, (2) a catalyst be (e.g. copper oxide) to oxidize H and CO, (3) molecular sieves to remove H_2O and CO_2, and (4) cold traps (e.g. charcoal, cooled to -200 to -300°C) to remove remaining noble gases, CO, N_2, CH_4, and H_2. Fort St. Vrain additionally included a titanium getter for tritium removal as a final step, though it did not function properly in the presence of nitrogen, a point worth noting for fusion systems.

Coolant purification systems for the ITER TBMs are largely modeled on these gas-cooled fission reactor systems, tailored to the somewhat different needs of fusion systems (i.e. extraction of much larger amounts of tritium) [41]. It is not certain what impurities and impurity levels can be expected in fusion coolants [41], though absent the massive amounts of graphite present in an HTGR core, and provided that the use of oil in the circulator or other components can be avoided, minimized, or at least appropriately contained (see section 4), it can be expected that carbonaceous impurities will be minimal. The system envisioned for the HCPB and HCLL TBMs consists of [41,42]:

1. Oxidation of Q_2 and CO on a copper oxide bed;
2. Adsorption of Q_2O and CO_2 on a molecular sieve (e.g. via pressure-temperature swing adsorption)
3. Removal of remaining impurities and tritium on a heated getter

It should be capable of reducing any anticipated 1-10 ppm level impurities to ppb levels [41], and so this appears to be a manageable problem for helium as a coolant provided that water ingresses can be avoided (see section 4).

The use of molecular sieves for removal (tritiated) water is a proven technology; however, the heated getter beds may not be feasible at the scale necessary for DEMO [42]. However, this scaling problem is common to all breeder/coolant configurations (this is a significant technical challenge for fusion), and thus does not constitute a strong argument for or against helium.

In both PbLi and ceramic breeders that use helium as coolant, significant tritium permeation from the breeder (either PbLi of helium purge gas) can be expected. Tritium present in the helium coolant may then permeate through pipe walls or through the heat exchanger, and such losses must be strictly minimized (to ~1 g/year). Regarding permeation through the heat exchanger, the aforementioned Cr_2O_3 layer may serve an important (perhaps essential) function as a permeation barrier [43]; it has been shown for the HCPB [43] that the presence or absence of this layer may results in losses that differ by orders of magnitude, and may be a determining factor in the acceptability of the design. In ceramic breeders, the ~0.1% hydrogen added to the purge gas will permeate along with tritium into the coolant, where it may be the most significant impurity; this may actually necessitate the *addition* of some small amount water to the helium coolant [44] in order to maintain an environment that is slightly oxidizing.

The chemistry control needs are dictated by the materials used and their environment, and the discussion above is focused on nickel alloys envisioned for use in ancillary systems such as the heat exchanger. Inside the reactor, it is envisioned that reduced activation steels will be used, and the use of helium for very high temperature components (i.e. the divertor) is reliant upon the further development of advanced steels (e.g. nanostructured) suitable for use at those temperatures. Any potential impact of impurities on these materials will need to be understood as a part of that development.

3.5 Safety assessment

The chemical inertness of helium avoids some of the worst safety issues that exist with water coolant (namely energetic chemical reactions and hydrogen generation), and it can therefore be regarded as superior in this regard. Nevertheless, its non-condensible nature at room temperature, and lower convective heat transfer capacity, result in some unique safety issues. These can be adequately designed for, and previous ARIES studies, which have favored helium coolant for the reasons outlined in this paper, have identified such design solutions. We summarize these safety issues and design mitigations in this section.

The first of these is overpressure resulting from a loss of coolant accident (LOCA). In water-cooled systems, overpressure resulting from a loss of coolant (e.g. steam in the vacuum vessel) can be alleviated by recondensing the steam in a pressure suppression pool. The inability to do so for helium implies a need for a large expansion volume to accommodate a helium LOCA. This can be partially mitigated by dividing the helium coolant system into separate loops, each cooling a fraction (e.g. 1/3) of the reactor; a design-basis pipe break accident, then, implies a loss of only a fraction of the coolant inventory. This approach has been employed in previous ARIES designs [45,46]. Still, since the vacuum vessel is not expected to be able to withstand high pressures, it must be equipped with a rupture disk in order to vent to an adequate expansion volume. This volume serves as a secondary confinement boundary, and as such must be designed to a given size and maximum pressure dictated by the coolant inventory it must accommodate. In past ARIES designs, the cryostat has additionally served this purpose [45,46].

In the event of a beyond-design-basis accident, e.g. one involving multiple coolant loop breaks, some consideration has to be given to transport of radionuclides beyond even this secondary confinement boundary. The primary radioactive source term in in the vacuum vessel for this scenario is activated dust generated by plasma-surface interactions, which can be mobilized and transported during a LOCA. Condensation acts to sequester some of the resuspended particulate, but again this cannot be relied upon for helium systems. However, a final filtered, vented confinement, as considered in lieu of a containment for gas-cooled fission reactors [47], might be sufficient for this purpose.

Finally, we note that during any loss of flow accident (LOFA, e.g. resulting from a loss of offsite power), it is desired that all decay heat can be removed passively by natural convection. Given the aforementioned lower heat capacity and heat transfer coefficient of helium, it might be more difficult to meet this requirement in helium cooled systems. In ARIES-ACT1 [3], the use of water was avoided everywhere inside the vacuum vessel, including the vacuum vessel itself. However, it was ultimately necessary to include a water-filled component outside the vacuum vessel for the sole purpose of shielding the magnets from neutrons. This water was then additionally used to remove decay heat via natural circulation during a LOFA, and the safety analysis verified that it was successful in doing so [46]. Significant decay heat removal may also be achieved by natural convection in the helium or PbLi (in liquid breeders), though the adequacy of the water system in ARIES-ACT1 precluded further investigation of this. Another strategy to assist in removing decay heat is to inject gas into the cryostat, which provides a path for conduction and convection of heat away from the vacuum vessel; this approach is adopted by ITER and the ARIES-CS study [45].

4. Experience base and lessons learned

4.1 Fission industry experience

Inexperience with helium-cooled systems (particularly relative to water-cooled systems) is an oft-cited reason in support of water-cooled blanket designs. In the fission industry, while the majority of reactors have indeed been water-cooled, it is important to realize that many gas-cooled reactors have also operated, and a number of these used helium. The experiences with these reactors are in many ways relevant to fusion as well. Interest in high temperature gas-cooled reactors revived in the last decade with the advent of the Next Generation Nuclear Plant (NGNP) project, which sought to build a demonstration reactor that would produce not only electricity but also high-temperature process heat for a variety of potential applications. It was envisioned that NGNP would use helium as a primary coolant and achieve outlet temperatures of 700-900˚C. The robust research effort in support of the program included comprehensive reviews (*e.g.* [11]) of gas-cooled reactor operating experience, which we draw upon in this summary. Some of these experiences are also recounted in a text on this reactor type [12].

There have been seven helium-cooled, high temperature reactors whose primary purpose was to develop the concept for commercial electricity generation (some others operated previously, primarily for other purposes, see [12]). These all employed a coated-particle fuel (TRISO and similar) in which a kernel is surrounded by layers of pyrolitic carbon and/or silicon carbide, which act as a fission product transport barrier. These fuel spheres (slightly less than 1 mm in diameter) are embedded in either graphitic pebbles of about 6 cm in diameter, which slowly circulate through the core (pebble bed reactor) or cylindrical fuel compacts which are then fixed in place within prismatic graphite blocks (prismatic reactor). Both reactor types are attractive in that they can be made passively safe, and they can achieve high efficiencies by operating at high helium temperatures.

Following operation of the Dragon reactor in the UK, whose primary purpose was to test a variety of fuels and materials for use in gas-cooled reactors, the development of prismatic and pebble-bed type reactors proceeded in the US and Germany (respectively) with the smaller scale prototypes at Peach Bottom (Unit 1) and AVR. These were followed by the full-scale demonstration plants at Fort St. Vrain in the US and THTR in Germany. Both of these shut down around 1990 (FSV for technical and economic reasons, THTR entirely for political reasons). There nevertheless remains a strong interest in the concept for its passive safety, high temperatures, and high efficiency, and since that time two more helium-cooled reactors have come on line and are presently operating: HTTR in Japan (prismatic) and HTR-10 in China (pebble bed). While the demonstration plant originally envisioned under the NGNP project is no longer planned, an industry alliance nevertheless remains interested in the concept and its applications, which has recently selected a frontrunner prismatic concept by AREVA. A summary of the seven helium-cooled reactors and their operating parameters is given in Table 4.

A recent review of helium-cooled reactor operating experience is given in [11]. That review identifies 68 "lessons learned" that can be applied to the NGNP project, in the following categories:

- Ingress or leakage events such as moisture ingress
- Primary coolant flow issues such as bypass flow and flow induced vibrations
- Fuel performance, fission product release, and graphite dust generation.

Table 4. Helium-cooled reactors and their operating parameters (adapted from [11, 12]).

Reactor	Years Operational	Thermal Power (MW)	Electrical Power (MW)	Fuel Element Type	Helium Pressure (MPa)	Helium Outlet Temperature (°C)
Dragon (UK)	1964-1975	21.5	-	Prismatic	2	750
Peach Bottom (US)	1966-1974	115	40	Prismatic	2.3	700-726
AVR (Germany)	1967-1988	46	15	Pebble Bed	1.1	950
Fort St. Vrain (US)	1976-1989	842	330	Prismatic	4.8	777
THTR (Germany)	1985-1991	750	300	Pebble Bed	4	750
HTTR (Japan)	1998-present	30	-	Prismatic	4	850-950
HTR-10 (China)	2000-present	10	-	Pebble Bed	3	700

Many of these are not directly related to the use of helium as a coolant, or are otherwise not relevant to fusion reactors (i.e. those in the last category; note that dust in fusion reactors has an entirely different origin than in gas-cooled reactors). Nevertheless, a few of the lessons learned related to the helium systems are relevant and we review these here.

Most of these are the many problems that occurred in Fort St. Vrain that were related to water ingress into the helium coolant [11,48,49]. The helium circulator at FSV had water-lubricated bearings and a steam turbine drive, which were sources of the water. Problems related to water ingress included:

- Plugging of helium pressurization lines due to corrosion
- Failure of control rods to insert during a SCRAM, a result of corrosion products in the drive motor bearings
- A control rod cable break resulting from chloride-induced stress corrosion cracking; moisture increase precipitated the leaching of volatile chlorides
- Stress-corrosion cracking occurred in the circulator seals and bolts for the same reason
- Failure of the reserve shutdown system to deploy; boronated graphite balls were fused together by boric acid crystals formed in the presence of moisture
- A variety of instrument failures
- Formation of ice in the helium purification system

All of the above were contributors to the very low capacity factor (~15% [50]) of the plant over its 11 year commercial operation and the decision to shut it down at that time. The use of water in the helium circulator, however, was unique to FSV, and not characteristic of the considerable operating experience with oil-lubricated circulators obtained before and since. While there exists the possibility of oil ingress from such devices (Peach Bottom experienced an oil ingress), even this possibility may be precluded by the use of dry lubricants and magnetic bearings, as are being investigated in HTR-10. It is also important to note that water ingress via heat exchanger tube failure (AVR experienced a large water ingress resulting from such a failure) is still possible in any system that employs water as a secondary coolant.

While the above outlined issues do indicate a clear need to understand, monitor, and control impurities in the helium coolant (a point not lost on the NGNP project [38,39]), it must be emphasized that these do not reflect the overall operating experience with such devices. This includes not just the seven aforementioned helium-cooled reactors, but also the large fleet of

CO_2-cooled reactors of both the MAGNOX and Advanced Gas Reactor (AGR) designs [51]. There have been *fifty-two* such reactors; 37 Magnox (and similar) and 15 AGRs. 26 of the Magnox reactors were commercial plants in the UK, many of which operated for over 40 years. All but one of these (Wylfa) have now been shut down, for economic reasons or simply because they reached their end of life. 14 of the 15 AGRs were commercial plants (the first, at Windscale, was a prototype) which came online in the late 1970s or 1980s, and all are still operating today. The UK commercial reactor fleet is made up almost exclusively of AGRs (save one PWR and the aforementioned Magnox reactor at Wylfa), and they provide ~9 GWe to the power grid. Most have received license extensions in recent years.

While the coolant chemistry of these reactors is obviously different those cooled with helium (aside from using CO_2 they operate at lower temperatures, 414°C for Magnox and 640°C for AGRs), they clearly provide an enormous operating experience base for components of gas-cooled systems. Even 30 years ago, such technologies were regarded as mature: the MHTGR helium-cooled design (while acknowledging the problems at FSV) cites the 100,000s of hours of successful circulator operation in the other helium-cooled reactors, and millions of successful operating hours (even at that time) in the Magnox and AGR plants, in support of the concept; circulator efficiencies of 99.4% are noted. Steam generator reliability increased over time in the AGRs as design and manufacture were improved over the earliest reactors [11], reflecting the maturation of the design and components.

The 1984 text by Melese [12] on helium-cooled reactors notably includes a final chapter on gas cooling of fusion reactors. It weighs the pros and cons of helium as a coolant, and cites amongst the advantages the fact that helium heat transfer, power conversion, and purification systems are "Developed Technology". An additional 30 years of AGR operating experience and VHTR research and development add validity to these statements today.

4.2 Fusion program experience with helium cooling

As described in section 2, there have been a rather large number of studies on blanket and divertor designs using helium as a coolant. Most of these were conceptual studies including comparisons with water or liquid metal cooled systems. More detailed investigations and developments have been performed especially in the EU for helium-cooled blanket test modules for ITER, and in the EU and US on concepts of helium-cooled divertor target plates based on W-alloys as structural material as well as plasma facing material. For both components, R&D work including design, thermal-hydraulic analyses, and experimental investigations have been made since the year 2000.

4.2.1 Helium-cooled blanket studies

The most extensive fusion experience with helium facilities exists in Europe, in connection with the ITER test blanket module development program. Facilities at KIT include the large-scale HELOKA helium facility [52,53], HEBLO blanket test loop [54,55], and HETRA experimental section attached to the HEBLO loop [56]. HELOKA consists of both a low-pressure (0.3 MPa) loop for application to IFMIF, and a high pressure (10 MPa) loop for fusion applications. Once fully operational, the HELOKA-HP facility is expected to operate at pressures up to 10 MPa and temperatures as high as 500˚C. With one circulator it will be able to reach a mass flow rate of 1.4 kg/s. The loop could be extended adding a second circulator increasing the maximum flow rate through the test section up to 2 kg/s. [57]. At ENEA Brasimone, a loop called He-Fus3 has been installed for TBM testing [58]. The He circulator with pneumostatic helium shaft supporting system is able to deliver a maximum flow-rate of 0.35 kg/s at 8 MPa.

As a result of blanket thermal-hydraulic investigations to date, it can be concluded that there is excellent agreement between the measured results and the values predicted by sophisticated 3D CFD models as used for example with the computer code STAR-CD. The concepts of He-cooled breeding blankets with integrated FW are feasible from a thermal-hydraulic point of view for the conditions of ITER and a moderate DEMO power plant [59,60]. However, it remains to be seen if in a fusion power plant the impact of irradiation with fusion typical neutron energy spectrum and fluence on the mechanical performance of the selected structural material (EUROFER) is tolerable. The rather high helium generation by the high energy neutrons at the FW conditions for example probably requires an increase of the He inlet temperature to at least 350 C, and a FW surface heatflux > 0.5 MW/m^2 will require some modifications.

4.2.2 Helium-cooled divertor studies

In the frame of the EU power plant study (PPCS), R&D work was initiated around 2000 on helium-cooled divertor concepts based on tungsten alloy as structural material. Since this time, a rather large number of theoretical and experimental tasks have been performed at KIT with extensive collaborations with the D.V. Efremov Institute and the Georgia Institute of Technology. The main subject of this R&D work is a modular concept of a He-cooled divertor target plate where the heat transfer from the W-structure to the He-coolant is maximized by employing an array of high speed He-jets hitting the back-side of the plasma facing surface.

Tests have been performed in small scale loops with high pressure, high temperature helium cooling typical for the conditions in a fusion power plant, with surface heat fluxes up to 10 MW/m^2, at facilities in the US and Russia. Collaborative work between KIT and Efremov utilized a helium loop and e-beam facility at Efremov [61]. The closed loop operates at 10 MPa pressure up to 600˚C with 25 g/s flow rate. Testing was performed with 10 MW/m2 heat flux with 1000 load cycles. A new loop, the Karlsruhe Advanced Technologies Helium Loop (KATHELO), is planned with capability of operating at high pressure (10 MPa) and high temperature (800˚C) with flow rates up to 200 g/s. This will allow testing of HEMJ divertor mockups with up to 5 modules with 9 fingers each. [62].

Within the US, application-oriented research has been performed for three decades at Sandia National Laboratory, including modeling of He flow, heat transfer and stresses that have been used as the basis for design studies (described elsewhere in this paper) and the testing and modeling related to performance tests on small modules, most of which were developed though DOE/s SBIR and CRADA[1] programs. Table 5 lists the companies involved and the designs tested. Ref [5] summarizes much of this research.

Dual-channel tests with flow through an internal structure with open, well connected porosity revealed a flow instability at high gas density (pressure) and temperature under high heat loads. [63] Tests on a tubular tungsten target with porous tungsten inside reached a maximum absorbed heat load was 22.4 MW/m^2 with helium at 4 MPa, flowing at 27 g/s and with inlet and outlet

[1] The Small Business Innovative Research (SBIR) and Creative Research and Development Agreement (CRADA) programs are processes through which small businesses are funded to develop ideas. In the cases here these grants involved the use of testing facilities at Sandia

temperatures of 40 and 91 °C and a pressure drop of 0.07 MPa. [64] Tests on single channel Mo targets, each with a differing value of open volume in the porous tungsten media inside the channels provided results of pressure drop versus porosity. [64] Extensive modeling has accompanied the testing. A very recent paper by Youchison compares the susceptibilities of jet flow and flow through porous media to flow instabilities and concludes that the jet flow is more resistant. [65] His work shows that jet flow on a smaller scale, as is used in the electronics industry, would reduce the thermal stresses from those in the current leading design for jet flow for fusion PFCs.

Table 5. He-cooled targets tested at Sandia National Laboratory

Company Name	**Targets Tested**
Thermacore	Porous metal heat exchanger (HX) He-cooled RF Faraday shield He-cooled dual-channel divertor module
General Atomics	He-cooled vanadium HX micro-channel He-cooled HX
Ultramet	tungsten foam HX single-channel He-cooled Mo HX multichannel He-cooled tungsten HX He/He refractory regenerator Li/He refractory HX
Plasma Processes, Inc.	Helium-cooled plasma-sprayed refractory target
Creare	normal flow He-cooled HX (NFHX)

The Georgia Institute of Technology (GIT) also has been involved in fundamental and applied research on helium cooling in impinging jet configurations. [32-34] They developed a helium loop for high heat flux testing, and have performed tests on target with a configuration similar to that of the HEMJ thimble developed at Karlsruhe [7] but made with the capability to vary the distance between the flow jets and the cooled surface. Initially they used a torch as the heat source but recently have developed an inductively headed plug that couples to their primary target and has the potential to deliver applied heat fluxes above 6 MW/m^2. Their loop is capable of delivering He at mass flow rate up to 10 g/s, pressure of 10 MPa and temperature up to 400˚C.

Overall, the thermal and thermal-hydraulic investigations show good agreement between measured results and the values predicted by semi-empirical computer models. Cyclic tests up to 1000 cycles using optimized mock-ups based on the selected tungsten alloys and representative cooling conditions and a surface heat flux of 10 MW/m^2 demonstrated the thermal-mechanical feasibility of the concept. However, all these tests have been performed without any neutron irradiation, and it remains to be seen to which degree the mechanical performance of the selected structural material will be degraded by the high fluence irradiation with high energy neutrons, leading to considerable He-generation and increased embrittlement. Obviously, the minimum irradiation temperature of the tungsten structure is an important issue. Probably the helium inlet temperature must be raised from 600˚C, as used in these tests, to values of at least 700˚C in order to maintain the structure temperature above 800˚C.

5. Summary

Helium has been used as a coolant in several fission power plants around the world with coolant temperature, pressure and flow rate similar to those of fusion power plant blanket and divertor designs. An industrial basis exists. Helium offers unique challenges as compared with water, and those challenges have been addressed in conceptual design studies. Experiments have been performed on small divertor mockups to validate heat transfer models and demonstrate performance under high heat flux. Larger validation tests have been performed for the several ITER test blanket modules that use helium as coolant. Heat removal capabilities are not worse than water in a properly designed system. The main penalty is increased pumping power, but this can be managed to levels that are small compared with the recirculating power needed for plasma sustainment (and is compensated by the higher efficiency of He-cooled plants enabled by the higher coolant exit temperature).

The use of helium allows several important advantages to the designer, including chemical compatibility with pressure vessel materials, tritium safety and the ability to operate at high temperature without constraints from the coolant itself. Helium provides a pathway to improvements, allowing the introduction of fusion power core technology at modest temperature levels without precluding advances enabled by future materials development. Due to its advantages in the near term, as well as its long-term prospects, we believe that the choice of helium as a fusion power core coolant is well justified.

References

1. F. Najmabadi and the ARIES Team, "Overview of the ARIES-ST Spherical Torus Power Plant Study," *Fusion Engineering & Design* **65**, 143-164, 2003.
2. F. Najmabadi and the ARIES-CS Team, "The ARIES-CS Compact Stellarator Fusion Power Plant," *Fusion Science and Technology* **54** (3), 655-672, 2008.
3. C. E. Kessel *et al.,* "The ARIES Advanced and Conservative Tokamak (ACT) Power Plant Study," *Fusion Science and Technology* **67**, **to appear** (2015).
4. D. Maisonnier, D. Campbell, I. Cook, L. Di Pace, L. Giancarli, J. Hayward, A. Li Puma, M. Medrano, P. Norajitra,M. Roccella, P. Sardain, M.Q. Tran and D.Ward, "Power plant conceptual studies in Europe," *Nuclear Fusion* **47** (2007) 1524-1532.

   D. Maisonnier, I. Cook, P. Sardain, R. Andreani, L. Di Pace, R. Forrest, L. Giancarli, S. Hermsmeyer, P. Norajitra, N. Taylor and D. Ward, "A Conceptual Study of Commercial Fusion Power Plants: Final Report of the European Fusion Power Plant Conceptual Study (PPCS)," EFDA Report EFDA-RP-RE-5.0, April 13th, 2005.
5. M. S. Tillack, A. R. Raffray, X. R. Wang, S. Malang, S. Abdel-Khalik, M. Yoda and D. Youchison, "Recent US Activities on Advanced He-Cooled W-Alloy Divertor Concepts for Fusion Power Plants," *Fusion Engineering and Design* **86** (2011) 71-98.

6. S. Hermsmeyer and S. Malang, “Gas-cooled high performance divertor for a power plant,” *Fusion Engineering and Design* **61–62** (2002) 197–202.

7. P. Norajitra *et al*., “European development of He-cooled divertors for fusion power plants, *Nuclear Fusion* **45** (2005), 1271-1276.

8. T. Ihli, R. Kruessmann, I. Ovchinnikov, P. Norajitra, V. Kuznetsov, and R. Giniyatulin, “An advanced He-cooled divertor concept: design, cooling technology,and thermo-hydraulic analyses with CFD,” *Fusion Engineering and Design* **75–79** (2005) 371–375.

9. L. El-Guebaly, T. Huhn, A. Rowcliffe, S. Malang and the ARIES-ACT Team, “Design Challenges and Activation Concerns for ARIES Vacuum Vessel,” *Fusion Science and Technology*, **64**(3) (2013) 449-454.

10. H. H. Toudeshki, F. Najmabadi, X. R. Wang and the ARIES Team, “Vacuum Vessel Analysis and Design for the ARIES-ACT1 Fusion Power Plant,” *Fusion Science and Technology*, **64**(3) (2013) 675-679.

11. J. M. Beck and L. F. Pincock, “High Temperature Gas-Cooled Reactors Lessons Learned Applicable to the Next Generation Nuclear Plant,” INL external report INL/EXT-10-19329 (April 2011).

12. G. Melese and R. Katz, “Thermal and Flow Design of Helium-Cooled Reactors,” ISBN: 0-89448-027-8, American Nuclear Society, La Grange Park, IL, 1984.

13. G. Federici, R. Kemp, D. Ward, C. Bachmann, T. Franke, S. Gonzalez, C. Lowry, M. Gadomska, J. Harman, B. Meszaros, C. Morlock, F. Romanelli, R. Wenninger, “Overview of EU DEMO design and R&D activities,” *Fusion Engineering and Design* **89** (2014) 882-889.

14. M. Porton, H. Latham, Z. Vizvary and E. Surrey, “Balance of plant challenges fora near-term EU demonstration power plant,” Proc. 25th Symposium on Fusion Engineering, 10–14 June 2013, San Francisco, USA, 2013.

15. A. M. Bradshaw and T. Hamacher, “Nuclear fusion and the helium supply problem,” *Fusion Engineering and Design* **88**, Issues 9-10 (October 2013) 2694-2697.

16. D. L. Smith, *et al*., “Blanket Comparison and Selection Study Final Report,” Argonne National Laboratory Report ANL/FPP-84-1, 1984.

17. E. Proust, L. Anzidei, G. Casini, M. Dalle Donne, L. Giancarli and S. Malang, “Breeding Blanket for DEMO,” *Fusion Engineering and Design* **22** (1993) 19-33.

18. L. M. Giancarli, et al. “Overview of the ITER TBM Program,” *Fusion Engineering and Design* **87** (2012) 395-402.

19. X. R. Wang, M. S. Tillack, C. Koehly, S. Malang, H. H. Toudeshki, F. Najmabadi and the ARIES Team, “ARIES-ACT2 DCLL power core design and engineering,” *Fusion Science and Technology* **67**, **to appear** (2015).

20. L.V. Boccaccini, L. Giancarli, G. Janeschitz, S. Hermsmeyer, Y. Poitevin, A. Cardella, E. Diegele, “Materials and design of the European DEMO blankets,” *Journal of Nuclear Materials* **329–333** (2004) 148-155.

M. Gasparotto, L.V. Boccaccini, L. Giancarli, S. Malang, Y. Poitevin "Demo blanket technology R&D results in EU," *Fusion Engineering and Design* **61/62** (2002) 263-/271.

21. X. R. Wang, S. Malang, M. S. Tillack and the ARIES Team, "Innovative First Wall Concept Providing Additional Armor at High Heat Flux Regions," *Fusion Science and Technology* **60** (2011) 308-312.
22. M. Merola, F. Escourbiac, R. Raffray, P. Chappuis, T. Hirai and A. Martin, "Overview and status of ITER internal components," *Fusion Engineering & Design* **88** (2014) 1798-1801.
23. M. Richou, A. Li-Puma and E. Visca, "Design of a water cooled monoblock divertor for DEMO using Eurofer as structural material," *Fusion Engineering & Design* **89** (2014) 975-980.
24. S. Hermsmeyer and K. Kleefeldt, "Review and Comparative Assessment of Helium Cooled Divertor Concepts," FZKA report 6597, November 2001.
25. E. Diegele, R. Kruesmann, S. Malang, P. Norajitra and G. Rizzi, "Modular He-cooled divertor for power plant application," *Fusion Engineering and Design* **66–68** (2003) 383–387.
26. A.R. Raffray, L. El-Guebaly, T. Ihli, S. Malang, X. Wang, the ARIES-CS Team, "Engineering design and analysis of the ARIES-CS power plant," *Fusion Science and Technology* **54** (2008) 725–746.
27. X. Wang, S. Malang, A. R. Raffray and the ARIES Team, "Design optimization of high-performance helium-cooled divertor plate concept," *Fusion Science and Technology* **56** (2009) 1023–1027.
28. S. Sharafat, A. Mills, D. Youchison, R. Nygren, B. Williams and N. M. Ghoniem, "Ultra Low Pressure-Drop Helium-Cooled Porous-Tungsten PFC", *Fusion Science and Technology* **52** (2009) 559-565.
29. M. G. Izenson and J. A. Valenzuela, "Normal Flow Heat Exchanger for Divertor Panel Cooling," *Fusion Science and Technology* **21** (1992) 1828-1834.
30. X. R. Wang, M. S. Tillack, C. Koehly, S. Malang, F. Najmabadi, H. H. Toudeshki and the ARIES Team, "ARIES-ACT1 system configuration, assembly and maintenance," *Fusion Science and Technology* **67**, **to appear** (2015).
31. M. S. Tillack, X. R. Wang, D. Navaei, H. H. Toudeshki, A. F. Rowcliffe, F. Najmabadi and the ARIES Team, "Design and analysis of the ARIES-ACT1 fusion power core," *Fusion Science and Technology* **67**, **to appear** (2015).
32. J. B. Weathers, L. Crosatti, R. Kruessmann, D. L. Sadowski, S. I. Abdel-Khalik, "Development of modular helium-cooled divertor for DEMO based on the multi-jet impingement (HEMJ) concept: Experimental validation of thermal performance," *Fusion Engineering and Design* **83** (2008) 1120-1125.
33. L. Crosatti, J. B. Weathers, D. L. Sadowski, S. I. Abdel-Khalik, M. Yoda, R. Kruessmann, P. Norajitra, "Experimental and Numerical Investigation of Prototypical Multi-Jet Impingement (HEMJ) Helium-Cooled Divertor Modules," *Fusion Science and Technology* **56** (2009) 70-74.

34. B. H. Mills, B. Zhao, S. I. Abdel-Khalik, and M. Yoda, “An experimental study of the helium-cooled modular divertor with multiple jets at nearly prototypical conditions,” *Fusion Science and Technology,* **submitted** (2015).

35. M. S. Tillack, X. R. Wang, J. Pulsifer, S. Malang, D. K. Sze, M. Billone, I. Sviatoslavsky and the ARIES Team, “Fusion power core engineering for the ARIES-ST power plant,” *Fusion Engineering and Design* **65** (2003) 215-261.

36. X. Zhang, J. Chen, W. Kang, J. Wu and T. Yuan, “Heat transfer coefficient calculation for analysis of ITER shield block using CFX and ANSYS,” *Fusion Engineering and Design* **86** (2011) 312-317.

37. P. Chen, T. A. Newell and B. G. Jones, “Heat transfer characteristics in subcooled flow boiling with hypervapotron,” *Annals of Nuclear Energy* **35** (2008) 1159-1166.

38. R. N. Wright, “Kinetics of Gas Reactions and Environmental Degradation in NGNP helium,” INL external report INL/EXT-06-11494 (June 2006).

39. B. Castle, “NGNP Reactor Coolant Chemistry Control Study,” INL external report INL/EXT-10-19533 (November 2010).

40. L. A. Sedano, “Tritium Cycle Design for He-cooled Blankets for DEMO,” CIEMAT report 1110, June 2007.

41. A. Ciampichetti et al., “The coolant purification system of the European test blanket modules: Preliminary design,” *Fusion Engineering and Design* **85** (2010) 2033-2039.

42. I. Ricapito, P. Calderoni, Y. Poitevin, and A. Aiello, “Technologies and modelling issues for tritium processing in the European Test Blanket Systems and perspectives for DEMO,” *Fusion Engineering and Design* **89** (2014) 1469-1475.

43. F. Franza, L. V. Boccaccini, A. Ciampichetti, and M. Zucchetti, “Tritium transport analysis in HCPB DEMO blanket with the FUS-TPC code,” *Fusion Engineering and Design* **88** (2013) 2444-2447.

44. I. Ricapito, A. Ciampichetti, P. Agostini, and G. Benamati, “Tritium processing systems for the helium cooled pebble bed test blanket module,” *Fusion Engineering and Design* **83** (2008) 1461-1465.

45. B. J. Merrill, L. A. El-Guebaly, C. Martin, R. L. Moore, A. R. Raffray, D. A. Petti, and the ARIES-CS Team, “Safety assessment of the ARIES Compact Stellarator design,” *Fusion Science and Technology* **54** (2011) 838-863.

46. P. W. Humrickhouse and B. J. Merrill, “ARIES-ACT1 Safety Design and Analysis,” *Fusion Science and Technology* **67** (2015) 9.

47. L. Loflin, “Containment Versus Confinement for High-Temperature Gas Reactors,” EPRI report 1011948, 2005.

48. D. L. Moses and W. D. Lanning, “The analysis and evaluation of recent operational experience from the Fort St. Vrain HTGR,” *Specialists’ Meeting on Safety and Accident Analysis for Gas-Cooled Reactos,* Oak Ridge, TN, May 13-15, 1985.

49. D. A. Copinger and D. L. Moses, “Fort Saint Vrain Gas Cooled Reactor Operational Experience,” NUREG/CR-6839, 2004.

50. IAEA Power Reactor Information System, http://www.iaea.org/pris.

51. D. E. Shropshire, "Lessons learned from Gen I carbon dioxide cooled reactors," *12th International Conference on Nuclear Engineering* **1** (2004) 463-473.

52. B.E. Ghidersa, M. Ionescu-Bujor, G. Janeschitz, "Helium Loop Karlsruhe (HELOKA): A valuable tool for testing and qualifying ITER components and their He cooling circuits," *Fusion Engineering and Design* **81** (2006) 1471–1476.

53. B.E. Ghidersa, V. Marchese, M. Ionescu-Bujor, T.H. Ihli, HELOKA facility: Thermo-hydrodynamic model and control, Fusion Engineering and Design 83 (2008) 1792–1796

54. P. Norajitra, D. Piel, G. Reimann, R. Ruprecht, "HEBLO, a blanket test loop for small test sections of helium cooled solid breeder blankets," 19th International Symposium on Fusion Technology, Lisbon Portugal, September 16-20, 1996.

55. P. Norajitra, D. Piel, G. Reimann and R. Ruprecht, The first HEBLO experiments based on the EU HCPB blanket concept, Fusion Eng. Des. 58–59 (2001) 623-627.

56. M. Ilić, G. Messemer, K. Zinn, B. Kiss, "HETRA experiment for investigation of heat removal from the first wall of helium-cooled-pebble-bed test blanket module*," Fusion Engineering and Design* **86** (2011) 2250-2253.

57. A. Aiello, *et al*., "Mock-up testing facilities and qualification strategy for EU ITER TBMs," *Fusion Engineering and Design* **85** (2010) 2012–2021.

58. M. Utili, A. Tincani, A. Aiello, I. Ricapito, G. Coccoluto, "The European Breeding Blanket Test Facility: An integrated device to test European helium cooled TBMs in view of ITER," *Fusion Engineering and Design* **84** (2009) 1881-1886.

59. F. Hernández, F. Cismondi, B. Kiss, "Fluid dynamic and thermal analyses of a HCPB TBM Breeder Unit mock-up," *Fusion Engineering and Design* **86** (2011) 2278–2281.

60. M. Ilić, G. Messemer, K. Zinn, R. Meyder, S. Kecskes and B. Kiss, "Experimental and numerical investigations of heat transfer in the first wall of Helium-Cooled-Pebble-Bed Test Blanket Module, Part 1: Presentation of test section and 3D CFD model, Part 2: Presentation of results," *Fusion Engineering and Design* (2014) **in press**.

61. P. Norajitra, R. Giniyatulin, V. Kuznetsov, I.V. Mazul, G. Ritz, "He-cooled divertor for DEMO: Status of development and HHF tests," *Fusion Engineering and Design* **85** (2010) 2251.

62. B. E. Ghidersa, X. Jin, M. Rieth, M. Ionescu-Bujor, "KATHELO: A new high heat flux component testing facility," *Fusion Engineering and Design* **88** (2013) 854-857.

63. D. L. Youchison, M. T. North, J. E. Lindemuth, J. M. McDonald and T. J. Lutz, "Thermal Performance and Flow Instabilities in a Multi-channel, Helium-Cooled, Porous Metal Divertor Module," *Fusion Engineering and Design* **49-50** (2000) 407-415.

64. D.L. Youchison, T. J. Lutz, B. Williams, R. E. Nygren. High heat flux testing of a helium-cooled tungsten tube with porous foam. FED 82/15 (2007) p1854-1860.

65. D. L. Youchison, "Flow Instabilities in Non-uniformly Heated Helium Jet Arrays Used for divertor PFCs," ANS 21st Topical Meeting on the *Technology of Fusion Energy, Anaheim Nov. 10-13, 2014*, to be published in *Fusion Technology.*