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Abstract

Water has both advantages and disadvantages as a coolant in conceptual designs of future fusion power plants. In the US, water has not been chosen as a fusion power core coolant for decades. Researchers in other countries continue to adopt water in their designs, in some cases as the leading or sole candidate. In this article we summarize the technical challenges resulting from the choice of water coolant and the differences in approach and assumptions that lead to different design decisions amongst researchers in this field.

1. Introduction

Water is a common coolant used in existing fission reactors throughout the world. A large base of operating experience has been accumulated for heat exchangers, steam generators, chemistry control and other large-scale water systems. There are issues with the use of water in fission reactors, like stress corrosion cracking or steam generator tube wear, but those issues are mostly known and addressed in designs.

Advanced "Gen IV" fission reactor concepts also have been studied for many years. [1] These concepts are pursued because they offer substantial improvements in safety, waste, economics and/or non-proliferation while still considered feasible in the near term (mid-21st century time frame). Most Gen-IV fission reactor concepts rely on alternative coolants, including helium, molten salt and liquid metal to obtain their advantages. One remaining candidate uses supercritical water.

In any case, our experience with fission reactors may have only limited applicability to fusion. For over 30 years, conceptual studies for fusion power plants have described a wide range of design options that include the choice of primary coolant. *Within the US*, water has been avoided in conceptual fusion power plant design studies for over 25 years as a result of factors related to performance and safety. The purpose of this paper is to explain the technical and programmatic reasons for the avoidance of water within the fusion power core.

Design choices involve complex relationships between materials and systems, and also depend strongly on the design *requirements* applied to any particular facility. Unfortunately, we do not have a modern self-consistent US power core design using water to allow an integrated evaluation. While each study must address its own choices in an integrated and self-consistent fashion, here we attempt to generalize the rationale for excluding water based on our experience in several ARIES power plant studies performed over the past 25 years. We restrict our attention to “in-vessel” blanket and divertor components and the vacuum vessel. The choice of heat transport fluid for the power conversion cycle is an important related topic, but is **not** discussed here.

Besides purely technical attributes, it is important to understand the role of programmatic factors in the design of future energy systems. For example, in some parts of the world, government-sponsored research is aggressively trying to compress the timeline for a demonstration of practical fusion energy by mid-century. The technical readiness of the primary coolant system **today** is therefore an important factor in decision-making, and economic competitiveness may play a lesser role. In the US, the Department of Energy supports a basic research program with the goal of resolving the major science and technology challenges for practical and competitive fusion energy. The existence of remaining R&D needs is considered acceptable, and forms the basis to plan the research portfolio. In either case, whether driven by a near-term sense of urgency or a long-term vision, fusion is a speculative future energy source without a clear customer nor an obvious market potential in the US. For that reason, we must maintain focus on the attributes of a fusion energy system that could lead one day to its implementation within the US.

2. Previous design studies

Existing designs of fusion facilities generally fall into one of two categories: (1) conceptual designs of long-term visions for a power plant, and (2) detailed engineering designs for experimental facilities like ITER. Recently, especially in Europe (EFDA) and *via* the EU-Japan Broader Approach activities, increased attention has been given to near-term implementations of a fusion power plant demonstration [2,3]. Although still in an early pre-conceptual phase of study, this machine is intended to proceed through a detailed engineering design phase and construction in the mid-21st century time frame. In the US, an activity was started in 2014 to explore the mission space and requirements for a fusion nuclear test facility called Fusion Nuclear Science Facility (FNSF), leading to its possible construction. FNSF is a plasma confinement facility whose purpose is to bridge the gap between ITER’s plasma and nuclear environment and that of Demo. [4]

In this section, we summarize the long-term concepts developed by the ARIES Team, the design choices made for the near-term ITER burning plasma experiment, and finally documentation from Europe and Asia on their power plant and Demo concept selection processes. These projects all have their own unique goals and ambitions, which affect the design selection process decisively.

2.1 ARIES power plant studies

Design decisions are usually derived from the evaluation of alternative concepts relative to some set of metrics or requirements. Although the requirements for a new source of nuclear energy in the future are uncertain and evolving, it is important to establish a quantitative basis for decision-making; otherwise, decisions can be biased by individual judgment or political pressure. In 1994, an advisory group was formed to provide guidance on the criteria for practical fusion power systems from a US electric utility industry perspective [5]. These relate to economics, public acceptance and regulatory simplicity. Following that, top-level design requirements were derived at a level of detail needed to support continuing design studies [6]. The requirements and attributes of an attractive power plant that impact these requirements are summarized in Table 1. These requirements have formed the basis for design decisions in US conceptual fusion power plant studies ever since they were introduced.

In the years following the establishment of utility-inspired requirements, the ARIES team carried out studies of several different magnetic confinement configurations for electric power plants in the range of 1 GW net electric output. These include a stellarator (ARIES-CS [7]), low aspect-ratio tokamak (ARIES-ST [8]), and several moderate aspect ratio tokamaks (A=4) covering a wide range of design space (ARIES-AT [9] and ARIES-ACT [10]). In those designs, PbLi became the preferred breeder. Both self-cooled and dual cooled (PbLi and He) blanket designs were explored. Divertor designs were developed using PbLi (at lower heat flux levels) or helium. Water has not been adopted for use inside the vacuum vessel in an ARIES study in over 2 decades.

Table 1. Technical requirements and attributes of an attractive fusion power plant

Requirements	**Example Attributes**
Cost advantage over other available options	High thermal conversion efficiency, high component efficiencies, compact (high beta), low recirculating power (*e.g.* high bootstrap fraction), high availability, uncomplicated components with low cost fabrication.
Eased licensing process	Plant standardization, low activation materials, low energy release potential, low tritium inventory.
No need for evacuation plan	Low activation materials, low energy release potential, passive safety, reliable containment, low tritium inventory
Produce no high-level waste	Materials choices
Reliable, available, and stable	Ample design margins, uncomplicated designs, fast and easy maintenance
No local or global atmospheric impact	Low CO_2 emissions, low tritium emissions
Fuel cycle is closed and on-site	Controllable tritium generation; efficient generation, extraction and processing of tritium; tritium control and barriers to losses
Fuel availability is high	
Plant is capable of operation at partial load	
Plant is available in a range of unit sizes	

2.2 The ITER experiment

ITER is an experiment, now under construction, that is expected to demonstrate the creation and control of a burning plasma in the tokamak configuration. Many of the technologies required for a tokamak power plant, such as superconducting magnet systems and tritium fueling systems will be demonstrated at power plant relevant scale. The base blanket does not breed tritium and does not operate at a temperature capable of generating electricity. Small ports allow in-vessel testing of more reactor-relevant technologies for blankets [11]. The total accumulated neutron fluence will be much lower than required in a power plant. The lower fluence and reduced requirements on the base blanket enable the use of more established technology choices.

Water has been selected as the coolant for all in-vessel components of ITER. The blanket and divertor normally operate with inlet water temperature of 70˚C and 4 MPa pressure [12]. The outlet temperature is typically ~50˚C higher than the inlet.

The structural material for all in-vessel components is 316L(N) austenitic steel. This steel is in direct contact with the water coolant within the blanket, whereas a copper alloy is used for the heat sink in the divertor target plates. Both 316SS and copper alloy are compatible with the use of low temperature water coolant in ITER. However, at the higher required operating temperature and higher fluence of a power plant, both of these materials are expected to suffer severe property degradation. In order to use water in a power plant, either alternative materials must be utilized or the performance and safety requirements of the device must be reduced. These issues are described in more detail in Section 3 of this report.

The water coolant in the ITER divertor target plates, which are composed of W as plasma facing material and a Cu-alloy as heat sink material, flows in specially designed small cooling channels, allowing steady state heat fluxes up to ~10 MW/m^2 at the target surface. To improve the heat transfer, either swirl flow in circular channels of ~10 mm diameter or the “hypervapotron concept” with ~3 mm x 17 mm channel dimensions will be employed. The maximum copper temperature is typically less than 500˚C, and the maximum temperature of the tungsten tiles remains below 2000˚C at the location of the peak surface heat flux. There are different options for the attachment of the W-tiles to the Cu-alloy heat sink, including flat tiles brazed to the heat sink, a brush-like concept where small tungsten pins are embedded into a cast Cu plate, or so-called monoblocks where Cu-alloy tubes are embedded into a W-block. Such target plates have been successfully tested with pulsed heat fluxes up to ~20 MW/m^2. However, all these tests were performed without any irradiation impact. The excellent performance relies heavily on the very high ductility of the Cu-alloy in order to compensate for the very different thermal expansion coefficients of Cu and W. This ductility will be reduced severely for neutron doses greater than 1-2 dpa.

2.3 European designs for a power plant and Demo

For many years the EU has considered several alternative concepts for a power plant blanket. The PPCS study, completed in 2005, initially considered four combinations of coolant and breeder (called “Models”) [13]. The original blanket concepts in the PPCS were:

- Model A: water-cooled PbLi
- Model B: helium-cooled ceramic breeder blanket
- Model C: dual coolant PbLi blanket
- Model D: self-cooled PbLi blanket

Later on, a Model AB was added with a helium-cooled PbLi blanket [14]. The EU is continuing to evaluate these blanket options within their Demo project, with the exception of Model D [3].

The Model A blanket uses water coolant at 15.5 MPa with inlet/outlet temperature of 285/325˚C [15]. The saturation temperature of water at this pressure is 343˚C. Eurofer97 is the structural material, with a minimum operating temperature limit between 300 and 350˚C, depending on neutron fluence. Double-walled tubes of Eurofer are used in the breeding zone to reduce the chances of water interaction with PbLi, and the gap between the tubes is filled with copper. Because these water conditions are "PWR-like", the proponents claim that the R&D needed to field such a blanket would be modest. The main issues with this concept are related to the feasibility of copper interlayers, the feasibility of using Eurofer at such a low temperature, tritium management and the need for and effectiveness of tritium and corrosion barriers at the steel/PbLi interface [15].

Different divertor target concepts have been proposed for the five models of PPCS. For Model A with a water cooled blanket, a water cooled divertor concept has been suggested. Models AB, B, and C, based on He-cooled blankets, and model C with a dual-cooled blanket employ a He-cooled divertor with W-alloy as structural and plasma-facing material. Model D with a self-cooled PbLi blanket used a PbLi cooled divertor.

Although much work has been done in Europe to demonstrate helium cooling of divertor target plates with surface heat flux of 10 MW/m^2 or higher [16], only water is being considered at present in the EU divertor within the framework of their near-term DEMO study. According to a recent publication, "It is generally agreed that water should be considered as the divertor coolant for a near-term Demo design as the divertor surface heat flux conditions prove to be beyond present helium power handling capabilities" [3]. The divertor design is likely to be derived from ITER, using copper as a heat sink and Eurofer97 for the supporting structures. The heat flux limit of a water-cooled Demo divertor using Eurofer has been estimated in the range of 8-10 MW/m^2 [17].

Ref [3] also notes that "detailed operational requirements are not yet available" for the EU Demo. Unlike the US approach to adopt requirements established by power-generating utilities and to define Demo as the step that demonstrates commercial viability, the determination of requirements for the EU Demo will depend upon design choices and project timescales [3]. The EU Demo "lies somewhere between ITER and a power plant".

2.4 Asian designs for a power plant and Demo

The oldest and most established program of power plant conceptual studies in Asia resides in Japan. Since the early 1990's, water with PWR-like conditions (285-325°C, 15 MPa) was adopted for the blanket and divertor coolant [18]. In some alternative designs, helium was considered as the blanket coolant. However, the JAEA program remains committed to water coolant, using it in its most recent SlimCS conceptual design [19]. While JAEA retains a focus on water coolant, Japanese universities continue to explore alternative concepts. The most notable example is the FFHR design led by NIFS, which uses Flibe ($LiF\text{-}BeF_2$) or Flinabe ($LiF\text{-}NaF\text{-}BeF_2$) as blanket coolant [20].

More recently, both China and South Korea have begun to explore options and develop more detailed design concepts for their fusion Demo and power plants. For example, South Korea is actively engaged in the conceptual design of K-Demo [21]. Both helium and water are under investigation. In China, extensive research is underway on a "flexible" PbLi blanket concept where either the entire heat (HCLL) or a part of the heat (DCLL) is extracted by helium.

3. Concerns with water as a coolant

The main concerns using water as a coolant relate to safety and performance. Much work has been done within the US fusion program to better understand the safety risks. Section 3.1 addresses chemical reactivity concerns, and Section 3.2 addresses tritium safety. Section 3.3 summarizes performance limitations, which mainly result from the need to avoid phase change. Temperature limits also arise due to the need to operate water together with compatible structural materials; Section 3.4 describes these materials' compatibility concerns with steel and copper alloys. Maintaining ductility under irradiation at low temperature is the primary concern. Although not a primary discriminator, we briefly discuss neutronic characteristics of water in Section 3.5. Water is a superior shielding material, which can lead to more compact systems, but also can degrade tritium breeding.

3.1 Chemical reactivity

It is anticipated that breeding blankets will contain materials operating at temperatures up to ~500˚C (or higher under some LOCA scenarios). At these temperatures, water or steam ingress will oxidize functional materials in both liquid and solid breeder blankets. Concerns specific to each are discussed in the following subsections.

3.1.1 Solid Breeders

Solid breeder concepts invariably employ small pebbles composed of lithium ceramic for breeding, and beryllium or beryllium alloy for necessary neutron multiplication. Beryllium reacts exothermically in air and steam [22]:

$$Be + \tfrac{1}{2}O_2 \rightarrow BeO \quad -609\ \text{kJ/mol}$$

$$Be + H_2O \rightarrow BeO + H_2 \quad -367\ \text{kJ/mol}$$

The production of hydrogen is an obvious safety concern, as coupled with air ingress into the vacuum vessel, an explosive gas mixture results that may be capable of destroying it. Based on the second equation above, steam oxidation of 18 kg of beryllium will generate the 4 kg of hydrogen necessary to produce an explosion in an ITER-sized vacuum vessel that exceeds safety limits [23]. The beryllium multiplier inventory in a single TBM is over 10 times that amount (~240 kg [24]); extrapolating to a full ceramic breeder blanket, it is apparent that such limits will be difficult to meet. As a result, and motivated in particular by the use of beryllium on the ITER first wall, a number of studies have been performed in order to measure the reaction rate of steam with various types of beryllium, including fully dense [25,26], porous [27,26], irradiated [28], and dust [29]. At relatively low temperatures, an oxide layer forms that is essentially protective against further oxidation; the reaction rate is limited by the rate of Be^{++} diffusion through the layer. At and above 600˚C, a large difference in the specific volumes of the oxide and metal lead to a breakdown of the oxide film, permitting diffusion of steam along grain boundaries [30,31]. This regime is characterized by higher reaction rates and a higher dependence on temperature. At still higher temperatures (> 900˚C), the reaction rates are very high (and may become auto-catalytic), and substantial degradation of the metal may occur.

Beryllium-steam reactions are of particular concern when the surface area involved is high, as in the case of beryllium dust, *e.g.* produced *via* plasma-surface interactions on the first wall [32,33]. This will clearly be an issue for ceramic breeder designs, where the surface area of the multiplier (~1 mm beryllium pebbles) is indeed high, and where stresses and friction on the pebbles can also be expected to produce dust. Adequate heat removal from these blankets is a concern under normal operating conditions; proving that they can withstand accident conditions involving steam reactions as described here is a significant obstacle.

Chemical reactivity concerns are greatly reduced by the prospective use of titanium beryllide ($Be_{12}Ti$), rather than pure beryllium, as a multiplier. Oxidation experiments with 1% H_2O in Ar at 1000˚C [34] found that the hydrogen generation rate increased and peaked after ~1.5 hours, before decreasing again to nearly zero over the next couple hours. The reason is supposed to be formation of a protective BeO layer that prevents further oxidation, which remains stable either because of a better match in mechanical properties of the BeO layer and $Be_{12}Ti$ substrate, or perhaps formation of a Ti-rich layer [35]. Note that others have observed continuous, not limited, hydrogen generation [36]. The peak reaction rate reported in [34] (based on the geometric surface area of the sample) was $3x10^{-5}$ mol/m^2-s, more than 1000 times less than was measured for pure beryllium in [28]. To put this into perspective, for a single ITER TBM as described in the preceding paragraphs [24], it would take about a day to generate 4 kg of hydrogen, though only seconds to generate an explosive quantity in the TBM multiplier volume. So, it would appear $Be_{12}Ti$ dramatically reduces, but does not completely eliminate, chemical reactivity concerns.

3.1.2 Liquid Metal Breeders

Liquid breeder concepts favor metals such as lithium or lead-lithium eutectic, though molten salt concepts have been considered in the past. Lithium, though less reactive than other liquid metals such as sodium [37], does react with oxygen, nitrogen, or water, and this reaction can be violent at high temperature. Lithium reactions with steam also produce hydrogen, via the reactions [38]:

$$Li + H_2O \rightarrow LiOH + \frac{1}{2}H_2 \quad -205 \text{ kJ/gmol Li}$$

$$Li + \frac{1}{2}H_2O \rightarrow > \frac{1}{2}Li_2O + \frac{1}{2}H_2 \quad -157 \text{ kJ/gmol Li}$$

$$\frac{1}{2}Li_2O + \frac{1}{2}H_2O \rightarrow LiOH \quad -69 \text{ kJ/gmol Li}$$

Concerns about the reactivity of liquid lithium and resultant hydrogen production were a primary motivator in the development of lead-lithium eutectic as a breeder material [39] (and sometimes coolant), now the favored liquid breeder in US and other fusion power plant designs. The relative hazard of Li or PbLi reactions with water depends on the contact mode; five such modes are identified by [38]. These essentially fall into two categories: "pouring" which results in little mixing or stratified layers, and sprays, which can be either water/steam jets into Li/PbLi, or Li/PbLi sprays into water or air. For PbLi poured into water, the reaction rates may be rather low since a protective oxide layer can form around the PbLi and prevent further oxidation. Sprays, however, result in a much larger contact area and thus are a greater concern, especially for a blanket concept that employs both Li or PbLi and high-pressure water. These concerns prompted a series of large-scale experiments on both lithium and lead-lithium at Hanford in the 1980's. Among these were experiments in which steam was injected into a pool of liquid Li (summarized in [40]) or PbLi [41]. Both were conducted in a similar setup, and it was found that Li pool experienced a higher temperature increase (to 980˚C *vs.* 870˚C for PbLi) despite a steam injection rate of 1/3 and total steam injection mass of ½ the PbLi experiment. However, all the lithium in the PbLi was consumed, which was what ultimately limited the pool temperature increase; the pure Li pool temperature was still rising when the experiment was terminated (to avoid over-pressurizing the chamber *via* hydrogen generation above 1000˚C). Three times as much energy was released in the Li experiment compared to the PbLi experiment.

So while PbLi is only mildly reactive compared to pure lithium, PbLi reactions with water can still be non-trivial. Such a blanket would have inside each blanket module a rather large number of cooling tubes or cooling plates. If there would be any water leak into the liquid metal breeder, the resulting chemical reaction would still result in a pressure and temperature increase that may be capable of rupturing a blanket box [42]. PbLi blanket designs that use water coolant, then, need to ensure that they are capable of surviving such occurances.

The favorite in the EU blanket comparison study of 1995 was a water cooled PbLi blanket [43]. The philosophy there was that the blanket box could be made sufficiently strong to take the full water pressure without rupture, and would stop in this way any water leak into the liquid metal in the box very early. The price to be paid for this safety feature was a rather high steel content in the breeding zone with a negative impact on TBR. Maintaining TBR>1 in the presence of added structure must necessarily require some increase in the size and cost of the reactor. Later on, it had been suggested to use double wall water tubes inside the blanket box to minimize the potential for water leaks. However, it remains to be seen how effectively the growth of a crack in one of the tubes would be stopped by the second tube.

Even if the blanket components can be designed to withstand the initial pressure and temperature increases resulting from steam ingress and subsequent chemical reactions, there remains the issue of hydrogen generation. This will exacerbate the overpressurization inside the blanket box and possibly result in a hydrogen ingress into the vacuum vessel in case of a box failure. Since 0.5 moles of hydrogen can be generated per mole of lithium, it can be generated in sufficient quantities to put the vacuum vessel at risk in the event of air ingress; 3530 kg of PbLi eutectic (~0.68 % Li by weight [39]) would be capable of generating the 4 kg ITER VV hydrogen limit. For comparison, this is less than the lithium inventory of a single ITER TBM (about 2760 kg of PbLi based on the breeding zone volume given by Kleefeldt [24]), but clearly this too becomes an issue when scaling up to a full size blanket. In addition to the risk to the VV, it has been estimated that ~100 m^3 of PbLi can even generate sufficient quantities to reach the 4% lower explosion limit in a representative building (*e.g.* the 250,000 m^3 STARFIRE building [39]). The risk of such a catastrophic hydrogen explosion can only be completely eliminated if water coolant is avoided in lead lithium systems.

3.2 Tritium safety: inventory, control, and extraction

In all high-temperature blanket systems, tritium can permeate through structures and accumulate in coolant streams where it is not desired. Efficient extraction of tritium from the PbLi breeder is one strategy for keeping circulating inventories low, but some permeation into the coolant (whether helium or water) is probably unavoidable. Thus, tritium must also be recovered from these coolants and processed for subsequent reuse as fuel. In systems that use water as a coolant, tritium will readily accumulate there, and continuous detritiation systems will be required. In PbLi blankets, the rather low solubility of tritium in this breeder leads to considerably large tritium partial pressure and consequently to high T-permeation fluxes into the cooling water. This is particularly problematic for HCLL/WCLL designs, which have higher tritium inventories than a DCLL blanket due to a much lower PbLi flow rate (~100 Pa vs <1 Pa for the DCLL). There may additionally be a large contribution from tritium implanted in the first wall, which can permeate through it and ultimately into the cooling water.

Both effects together result in a considerably high tritium flux into the water coolant. Since for safety reasons the allowable tritium concentration in water should be as low as reasonably achievable, continuous on-line extraction is mandatory. This is similar to the case of He-cooling, but tritium extraction is there much easier since either getters or cryopanels can be used. For tritium extraction from water, the water first has to be split into oxygen and hydrogen (*e.g.,* via electrolysis), and then the tritium separated from H and D in an isotopic separation system. Both steps require large facilities with high energy consumption.

A reference concept for the water-cooled lead lithium (WCLL) DEMO design initially assumed that tritium permeation into the water must be limited to 1 g/day in order to keep the tritium inventory below 1 Ci/kg of water [44]. Achieving this figure relied upon the use of tritium permeation barriers to prevent permeation from the PbLi into the water, a strategy not unique to water-cooled systems. Since the performance of permeation barriers in reactor environments has not yet lived up to small scale laboratory experiments [45], a feasibility study was conducted to assess scale-up of the water tritium extraction systems for a WCLL concept without strong permeation barriers (a permeation reduction factor of 10 was still assumed) [44]. The analysis considered three candidate water detritiation concepts (including electrolysis, which was the least expensive) and also the increased demands on the air detritiation system and isotope separation system. It was concluded that, in order to process water at the same rate as in the reference design, the allowable tritium permeation rate must increase to 10 g/day and the circulating inventory be allowed to reach 8.5 Ci/kg. The cost of the systems increased from \$27.2M in the reference design to \$47M, and 5 MW was required for electrolysis. Based on the figures in that reference, adhering to the original limits for the tritium inventory (1 Ci/kg) under the increased permeation rate (10 g/day) would result in a cost increase of ~\$70M, from \$27.2M to ~\$97M; it is not clear what additional energy would be required in this case.

While this configuration is said not to have "prohibitive effects on safety" on the basis of a LOCA analysis [46], it is clear that both occupational and accidental doses under such an increase in allowable tritium inventory can only be larger. We note that 8.5 Ci/kg is substantially larger than the 2 Ci/kg adhered to by CANDU reactors in order to keep occupational exposures and environmental releases acceptably low [47], and far in excess of the values assumed for ITER (0.05 Ci/kg [48]) and observed in light water reactors (~2.2 mCi/kg and ~1.5 μCi/kg for PWRs and BWRs respectively [49]). We do not believe the advantages conferred by the technological maturity of water-cooled systems justify such increases, nor does the perception of technological maturity seem particularly well founded in light of the fact that water-cooled designs ultimately rely on new technologies such as permeation barriers, and extraction systems of unprecedented scale, for their safety.

3.3 Performance limitations

The main drawback of water arises from the need to avoid phase change (*i.e.*, critical heat flux). At a reasonable pressure, the coolant temperature is typically limited to ~325˚C. At this temperature, thermal conversion efficiency is limited to values of 33% or less. For comparison, existing combined-cycle or supercritical water-cooled power plants achieve efficiencies of 45% or higher.

The impact of low conversion efficiency on a fusion power plant is twofold. Low efficiency directly impacts the cost of electricity [10]. For any given net electric output, lower efficiency requires a larger power core with higher thermal output. Both the power core and the power handling equipment become more costly.

Another impact of low efficiency, which is often overlooked, is the impact on other system parameters. System studies have shown that high conversion efficiency helps to reduce the requirements on other system parameters, such as plasma beta. The overall design space opens up, allowing operating margins on some of the key plasma parameters.

3.3.1 <u>Water cooled blankets</u>

Proponents of water-cooled blankets claim that such a concept could be based on the very well-known fission PWR technology. Typical coolant temperatures in such power plants are 285˚C at the inlet and 325˚C at the exit, and the water pressure is 15 MPa. The exit temperature is limited by the requirement to avoid water boiling in the reactor core for neutronic reasons. From the core pressure vessel, the water flows to external steam generators where, at the secondary side of the heat exchanger tubes, saturated steam at a pressure of ~7 MPa is generated. With such steam conditions, the efficiency of the power conversion system based on a Rankine power cycle is limited to values less than 33%.

In the frame of the development of Gen. IV fission reactors there are plans to use supercritical water with a pressure > 25 MPa and exit temperatures > 550˚C. The resulting efficiency in the power conversion system will be ~45%. However, for such steam conditions corrosion of steel is a very critical issue, making the selection of a suitable structural material a difficult task.

Can the PWR experience and the planning for more advanced fission power plants really be extrapolated to the requirements for breeding blankets in fusion power plants?

The main candidate structural material in breeding blankets come from the ferritic martensitic steels of the EUROFER/F82H class. If such a steel is irradiated in the neutron energy spectrum of a fusion plant at temperatures below 350˚C, its ductility would decrease rapidly. A considerably higher water temperature than used in a PWR would be required to maintain the structure at a temperature > 350˚C. This would require an increase of the water pressure to values >19 MPa or, for a higher efficiency in the power conversion system, the transition to supercritical water with a pressure > 25 MPa.

However, already the coolant pressure of a PWR is about twice as high as in He-cooled breeding blankets, and requires for strength and safety reasons a rather large volume fraction of steel, making the achievement of a TBR>1 a difficult task.

Altogether, the use of water cooling in breeding blankets of a fusion power plant would require a different structural material than the main candidate reduced-activation ferritic steel (RAFS), and the pressure and temperature levels typical for a fission PWR would limit the achievable efficiency in the power conversion system to values < 33 %.

3.3.2 Water cooled divertors

For the lay-out of water cooled divertor target plates there are two different design concepts.

a) Low temperature water as for example suggested for ITER.

In this concept, water with a pressure of 4 MPa and inlet temperature of 70˚C is used. Such a low coolant temperature is required to allow the use of a copper alloy in the heat sink. With these coolant conditions and facilitated by the high thermal conductivity of such alloys, the critical heat flux to the divertor target surface is > 20 MW/m^2 for the selected design of the coolant channels.

However, for use in a fusion power plant, such a divertor design would have two very critical issues:

1. The low water outlet temperature does not allow the use of the divertor heat in the power conversion system. This means that in a power plant about 10-20% of the thermal power would be wasted.
2. The Cu-Cr-Zr alloy used in the ITER divertor heat sink would be susceptible to severe radiation-induced changes in mechanical properties under the higher neutron flux and fluence typical for a fusion power plant. For neutron doses >1-2 dpa, radiation hardening accompanied by severe reductions in ductility occurs at irradiation temperatures below ~275˚C. A transition to radiation softening occurs for irradiation temperature above ~300˚C and ~80% of the initial yield strength is lost after a few dpa at ~400˚C (with correspondingly rapid increases in ductility.) As neutron doses increase, a regime of grain boundary helium embrittlement is encountered at ~350˚C as helium concentrations exceed ~60 appm.

b) Cooling of the divertor target plates with water conditions as in a fission PWR.

Typical coolant conditions here are a pressure of 15 MPa and an exit temperature of ~325˚C. These conditions rule out the use of Cu-alloys in the heat sink, and challenge the viability of the primary candidate RAFS due to ductility loss at these low temperatures [50,51,52]. Furthermore, evaluations in the frame of the EU power plant studies have indicated that the maximum surface heat flux to such a divertor target plate must be < 7 MW/m^2.

Limiting issues here are the low thermal conductivity of the steel leading to high thermal stresses, and the critical heat flux at the steel/water interface. Without the use of a Cu-alloy with its high thermal conductivity in the heat sink, the peak heat flux to the cooling water would be considerably higher than the heat flux at the plasma facing surface

3.4 Materials compatibility

3.4.1 Water compatibility with steel alloys

Matching of the temperature windows between the coolant, breeder and structural material is essential in any blanket design concept. The water temperature in the blanket and integrated first wall will likely be in the range of 285 to 325˚C.

The use of ITER's solution-annealed 316L(N) stainless steel has been avoided in the US for power plant applications due to performance limitations (related to poorer thermomechanical properties) and more importantly its high levels of long-lived radioactivity. It has been shown [52] that the ITER divertor will generate high level waste, mainly because of the presence of 316SS. This violates one of the top level utility requirements for fusion. The EU also considers 316SS impractical for a power plant blanket [53]. Resistance to radiation damage is also limited. Ductility and fracture toughness are severely degraded during irradiation around 300˚C and damage of ~10 dpa. At mid-range temperatures of 400–500˚C it is susceptible to unacceptable volumetric void swelling for doses >20 dpa, and has been shown to suffer from severe helium embrittlement at higher temperatures at very low He content (10–100 appm). For these reasons, a reduced activation ferritic steel is probably required.

However, many experiments have shown that the class of reduced activation ferritic steels like F82H and Eurofer requires a minimum irradiation temperature greater than 350˚C in order to avoid excessive embrittlement. The highest impact engineering design risk for the EU water-cooled blanket is the irradiation hardening-induced shift in ductile to brittle fracture behavior for temperatures below ~350˚C. The level of risk is difficult to fully quantify because of heat-to heat variations in fracture behavior of RAFMs related to impurities, distribution of brittle fracture initiation particles, and above all the uncertain influence of helium generation on hardening and fracture behavior in the 14 MeV environment. In addition, the effect of extrinsic variables such as section thickness, crack geometries, loading rate etc. have a major role in determining the effective fracture toughness and structural integrity. Because of these uncertainties, the EU fusion programme's Materials Assessment Group (MAG) assessment [53] has taken the position that they will not pursue a design based on the current EUROFER alloy in a WCLL blanket with 285-325˚C water cooling. They conclude that a water-cooled blanket option would only be viable if it were possible to develop a RAFM variant that exhibits reduced susceptibility to radiation hardening and exhibits ductile fracture behavior during irradiation with fusion neutrons in the 250-350˚C regime. The report recommends that a Risk Mitigation Program be implemented to pursue this goal.

Another issue with water cooling in the 285-325˚C regime is the potential threat from irradiation assisted stress corrosion cracking (IASCC) in the 8-9 Cr ferritic-martensitic steels. Although very few studies have been conducted, there is evidence for the susceptibility of the FM steels to IASCC for certain combinations of material compositions, irradiation parameters and environmental conditions, and this is another potential degradation mechanism that requires careful evaluation in parallel with efforts to mitigate the effects of radiation hardening [54].

As with all 8–9% Cr FM steels, corrosion under irradiation would be an issue if water were to be used in the blanket, and coating and coolant chemistry mitigation will be required.

3.4.2 Water compatibility with copper alloys

For a water-cooled divertor, copper is generally required for performance reasons. Copper must be operated above 200˚C to avoid rapid (1-2 dpa) loss of ductility under irradiation, but not higher than 350˚C where strength is lost [53]. Even then, fluence lifetime is very limited, requiring frequent replacement of the divertor and a large waste stream.

The current material selected for the ITER divertor heat sink is a Cu-Cr-Zr alloy which requires a 3-step thermo-mechanical treatment to develop the optimum combination of strength, conductivity and fracture toughness. Achieving these properties is compromised by joining processes such as HIP or brazing which inevitably result in lowering the solute concentrations during cooling from the joining temperature and thus reducing the volume fraction of precipitation, and hence the strength properties, developed in the final post-joining heat treatment.

The operating temperature window for this alloy is very limited. Below ~285˚C radiation hardening is rapid (saturating at <1 dpa) and is accompanied by severe loss of uniform strain due to dislocation flow localization. For irradiation temperatures >300˚C, coarsening of the precipitate dispersion results in loss of strength.

The back-up materials is GlidCop Al-25, a Cu-Al_2O_3 dispersion strengthened alloy produced by mechanical alloying. While this material is much more tolerant of fabrication heating and cooling cycles because of the nano-scale Al_2O_3 dispersoid, it suffers from severe loss of fracture toughness at temperature >250˚C and enters a regime of rapid creep deformation above 300˚C.

For these reasons, neither of these materials is perceived as being adequate for service as a heat sink material for the water-cooled divertor for the EU DEMO.

The EU MAG assessment [53] concludes that the development of advanced Cu-based materials is urgently needed to ensure an expanded temperature and neutron dose operating regime for the water-cooled heat sink and propose a set of risk mitigation options that could be pursued including fiber and foil reinforced materials, W-Cu laminates and functionally graded tungsten-copper composites.

3.5 Neutronic and activation characteristics of water

Due to its high hydrogen content, water is effective at moderating neutrons. This makes it a superior shielding material in conjunction with absorbers like boron. In some of the earlier ARIES power plant studies, water was used inside of a double-walled vacuum vessel to aid in shielding and to reduce activation of ex-vessel components (most notably the magnets). The latest set of designs, called ARIES-ACT, moved this shielding function outside of the vessel in order to avoid the use of water in the vessel.

The same physical phenomenon that improves shielding can lead to a negative impact on tritium breeding. For example, in a PbLi blanket, moderation and absorption reduces the number of energetic neutrons needed for Pb to perform its primary function of multiplying the number of neutrons. To enhance breeding in PbLi blankets, Pb multiplies energetic neutrons only if their energies exceed ~7 MeV. While the large neutron moderation in water helps enhance tritium breeding from Li-6, the large absorption tends to decrease the total TBR. In typical liquid and solid breeder designs, using 20% water coolant in the FW/blanket system reduces the TBR by up to 7% [55]. Increasing the enrichment of the lithium, or some other measures (like increasing the radial build) become necessary to compensate.

The fraction of steel in a first wall and blanket is also an important determinant of breeding capability. Designs with higher coolant pressure tend to require more steel in order to support the loads. This can be especially troublesome with solid breeder designs that require more internal structure and suffer from lower breeding capability as compared with liquid metal breeders. The extensive use of beryllium is usually employed to counteract this deficiency, as well as enrichment of Li-6.

The ability to breed sufficient tritium in order to close the fuel cycle is an absolute requirement for fusion power plants, and is not assured for any set of blanket materials choices. PbLi was chosen as a breeder/coolant in ARIES studies in large part due to its superior breeding capabilities and ability to adjust breeding on-line via control of Li-6 enrichment. This is a good example for which design decisions result from an integrated assessment of **all** of the requirements for a commercial power plant involving the complete set of materials choices.

On balance, neutronics is important and affects design, but is not a primary discriminator against water. Design solutions are possible to mitigate any undesirable effects on both shielding and breeding.

Coolant activation is another operational concern with any fusion plant using water in the primary loop. N^{16} is produced by 14-MeV neutrons in the $O^{16}(n, p)N^{16}$ reaction. As a result, all parts of the loop will become a strong gamma source due to N^{16} decay with principal lines at 6.14 and 7.12 MeV [56]. Water leaks would provide an occupational exposure risk, and additional shielding may be required around the cooling lines. Besides the unavoidable presence of N^{16}, corrosion products may become activated. A careful evaluation of corrosion product concentration and activation characteristics will be needed for any water-cooled design.

4. Summary

The choice of water as a fusion reactor coolant is based to a large extent on the commercial availability of large components and a vast industrial experience base, although the relevance of this experience base to the unique conditions in a fusion reactor is questionable.

The choice of water leads to several negative consequences, which have been described in detail above. These include:

- Low thermal conversion efficiency, of the order of 33% or less, leading to higher cost of electricity
- Performance limits in the divertor, restricting the heat flux to 8-10 MW/m^2
- Low ductility of structural materials under irradiation, which would restrict fluence lifetime (to perhaps one year of operation for the divertor) and/or require the development of new materials
- For the water-cooled PbLi blanket, the use of more complex double-walled tubes to avoid interaction of water with PbLi
- More difficult tritium management and higher inventories, leading to higher occupational and accidental doses

- The risk of hydrogen explosion during accident scenarios
- Additional challenges on tritium breeding as a result of the higher required structure fraction of steel (due to high coolant pressure) and the moderating effect of water on neutrons.
- Coolant activation from the $O^{16}(n, p)N^{16}$ reaction and corrosion products.

Due to the large number of negative consequences, continued effort to identify and develop more attractive coolants appears to be well warranted.

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