Nuclear Engineering and Technology 54 (2022) 877–887

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original article

Design of a direct-cycle supercritical CO_2 nuclear reactor with heavy water moderation

Robert Petroski*, Ethan Bates, Benoit Dionne, Brian Johnson, Alex Mieloszyk, Cheng Xu, Pavel Hejzlar

TerraPower, LLC, USA

ARTICLE INFO

Article history:
Received 19 July 2021
Received in revised form
17 September 2021
Accepted 26 September 2021
Available online 4 October 2021

Keywords:
Advanced reactor
SMR
Supercritical CO_2
Gas cooled reactor
Heavy water

ABSTRACT

A new reactor concept is described that directly couples a supercritical CO_2 (sCO_2) power cycle with a CO_2-cooled, heavy water moderated pressure tube core. This configuration attains the simplification and economic potential of past direct-cycle sCO_2 concepts, while also providing safety and power density benefits by using the moderator as a heat sink for decay heat removal. A 200 MWe design is described that heavily leverages existing commercial nuclear technologies, including reactor and moderator systems from Canadian CANDU reactors and fuels and materials from UK Advanced Gas-cooled Reactors (AGRs). Descriptions are provided of the power cycle, nuclear island systems, reactor core, and safety systems, and the results of safety analyses are shown illustrating the ability of the design to withstand large-break loss of coolant accidents. The resulting design attains high efficiency while employing considerably fewer systems than current light water reactors and advanced reactor technologies, illustrating its economic promise. Prospects for the design are discussed, including the ability to demonstrate its technologies in a small (~20 MWe) initial system, and avenues for further improvement of the design using advanced technologies.

1. Introduction and background

Supercritical CO_2 (sCO_2) power cycles have been an active area of research for the past two decades, following the realization that modern high-effectiveness compact heat exchangers could greatly improve their performance and compactness [1]. One application area is nuclear energy, where sCO_2 cycles can yield high efficiencies at temperatures produced by advanced Generation-IV reactors. While numerous studies have investigated the use of sCO_2 cycles in existing Generation-IV designs (e.g. SFRs [2] and LFRs [3]), it has also been recognized that significant simplification and cost savings can be achieved by using sCO_2 in a direct-cycle – i.e., having sCO_2 reactor coolant directly drive the power cycle, similar to in a BWR.

Recent examples of direct-cycle sCO_2 reactor concepts are [4,5], and [6], which are all gas-cooled fast reactors (GFRs). A GFR avoids the materials challenges and large volume requirements associated with the graphite moderators typically employed in gas-cooled reactors, such as the UK's Advanced Gas-Cooled Reactors (AGRs), a presently operating fleet of 14 CO_2-cooled reactors. However, GFR design challenges include fuel and material development, the need for a large high-pressure vessel, and difficulty maintaining cooling following a loss-of-coolant accident (LOCA).

This paper describes a new approach to the design of a direct-cycle sCO_2 reactor that avoids these challenges: use of a pressure-tube core configuration with heavy water moderator, like in Canadian Deuterium Uranium (CANDU) reactors. Heavy water moderator requires less volume than graphite, and the separation between the moderator and coolant avoids compatibility issues between the two. Importantly, the high surface-to-volume ratio of a pressure-tube core allows the moderator to be an effective heat sink for decay heat removal, even if all gas cooling is lost. The use of a thermal spectrum permits existing AGR fuel designs to be employed and avoids the need for high assay low-enriched uranium (HALEU).

Past examples of direct-cycle pressure-tube reactors have used light water coolant in a steam cycle, such as Gentilly 1, Winfrith,

* Corresponding author.
E-mail address: rpetroski@terrapower.com (R. Petroski).

https://doi.org/10.1016/j.net.2021.09.030

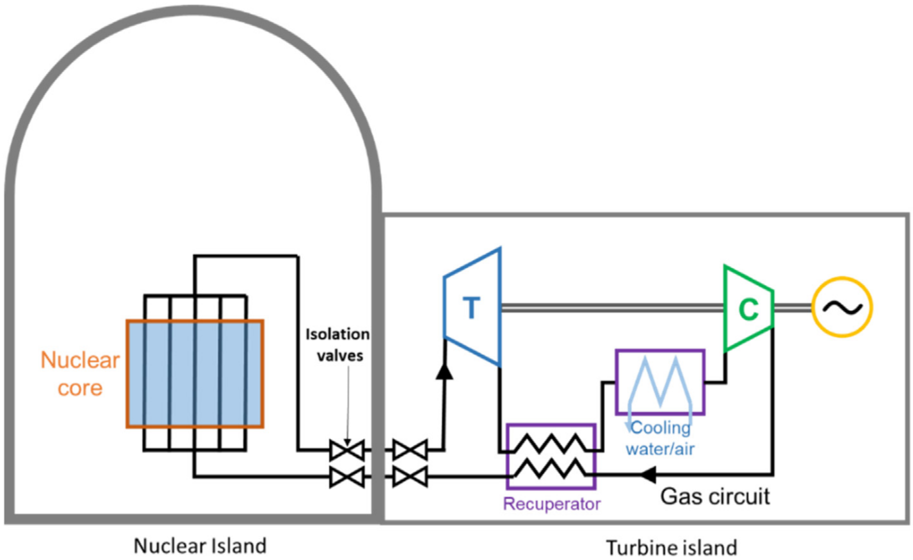

Fig. 1. Schematic of the HWGTR design.

and Fugen. These Steam Generating Heavy-Water Reactors (SGHWRs) share features with Boiling Water Reactors (BWRs), including recirculation pumps, steam driers, and emergency core cooling systems. There have also been examples of heavy-water gas-cooled reactors (HWGCRs), most notably the 250 MWt CO_2-cooled French EL4 reactor at Brenellis [7], which operated from 1967 to 1985. Past HWGCRs used gas-to-water heat exchangers (steam generators) and a steam power cycle, although it was recognized that direct-cycle operation could be a possible advancement [8].

Several historical concepts proposed direct-cycle gas turbine architectures. These include the helium- and CO_2-cooled concepts for marine propulsion investigated in Ref. [9], which considers a range of different moderator options, including heavy and light water. More recently, versions of the Generation-IV High-Temperature Gas-cooled Reactor (HTGR) incorporate a very high temperature (850 °C) helium Brayton cycle [10]. The only example of an operating nuclear reactor using a gas turbine to generate power is the 300 kW ML1 reactor from the U.S. Army Gas Cooled Reactor Program [11]. The ML1, intended for mobile applications, employed nitrogen coolant, light water moderation, and high enriched uranium fuel.

Since the use of a gas turbine in a direct-cycle architecture is a key distinguishing feature, the design described in this paper will be referred to with the acronym HWGTR (heavy water gas-turbine reactor), to differentiate it from HWGCRs that use gas for core heat removal but not for power conversion. This paper provides an overview of a 200 MWe HWGTR design, including the nuclear island, core, and power cycle systems. It also provides analyses that describe how the HWGTR responds to accidents, including LOCAs. The resulting design has the potential to be simpler, more efficient, and more economic compared to present day LWRs and other advanced reactor concepts. Compared to liquid-metal and liquid-salt based advanced reactors, the HWGTR avoids the need for additional coolant loops (with their associated coolant and cover gas processing systems), intermediate heat exchangers, trace heating systems to avoid coolant freezing, and remote operations systems to accommodate activated, toxic, or chemically reactive fluids. Further, the HWGTR employs existing technologies from commercially operating CANDU and AGR reactors, reducing the amount of technology development needed to demonstrate the concept.

2. Overall architecture

A schematic diagram of the HWGTR plant coolant loop is shown in Fig. 1. Heated sCO_2 coolant exits the core via main CO_2 outlet lines and proceeds to the turbine island, where its energy is converted to electrical power. Cold coolant is compressed, pre-heated in a recuperator, and returned to the core through main CO_2 inlet lines. Notably, the reactor primary circuit is considerably simpler in a HWGTR compared to an LWR. It does not include primary pumps, steam generators, steam driers, or a pressurizer.

As a direct-cycle reactor, the HWGTR employs features similar to those in a BWR. Paired isolation valves are used to ensure that the containment can be isolated. Like in a BWR, concrete shielding is used around power conversion equipment due to short-lived activation of oxygen as coolant transits the core. Finally, coolant radioactivity levels are monitored and kept at a low enough level to permit an ex-containment loss of coolant without exceeding personnel or public dose limits.

For simplicity, Fig. 1 shows a simple recuperated power cycle, when in fact a variety of cycle options can be employed, such as a recompression cycle [1] to raise efficiency or a split-expansion cycle [12] to reduce the pressure in the reactor. These options are discussed further in Section 3. Major parameters for this HWGTR are shown in Table 1 and their bases are described further in Sections 3 and 4.

In a heavy water moderated reactor, a fraction of core power is deposited in the low temperature moderator and shielding instead of the primary coolant. Therefore, Table 1 distinguishes between plant “thermal efficiency,” which only considers power supplied to

Table 1
Major plant parameters of a 200 MWe class HWGTR.

Thermal power (MWt)	540
Power supplied to primary coolant (MWt)	500
Neutron and gamma heating to calandria and shield (MWt)	24
Thermal losses to calandria and shield (MWt)	16
Gross power (15 °C/25 °C ambient) (MWe)	217/205
Net power (15 °C/25 °C ambient) (MWe)	206/194
Gross thermal efficiency (15 °C/25 °C ambient) (%)	43.4/40.9
Net thermal efficiency (15 °C/25 °C ambient) (%)	41.3/38.9
Overall net efficiency (15 °C/25 °C ambient) (%)	38.2/36.0

the power cycle, and "overall net efficiency," which includes the thermal power that is lost to the moderator. About 40 MWt are deposited in the moderator and shield by neutron and gamma radiation, as well as thermal leakage from the fuel channels. This small degree of thermal leakage is tuned to permit passive decay heat removal from the fuel in accident scenarios.

Gross power is equal to the shaft power generated by the turbines minus the shaft power consumed by the compressors. Net power is equal to gross power minus house loads (~11 MW), which include generator losses (~2.3 MW); cooling tower fans and cooling water pumps (~5 MW); and moderator pumps and auxiliary loads (~3.5 MW). Two large house loads found in Pressurized Water Reactors (PWRs) and Heavy Water Reactors (HWRs), primary coolant pumps and feedwater pumps, are not present in the HWGTR.

Two sets of efficiency numbers are provided, corresponding to cycles designed for ambient air temperatures of 15 °C and 25 °C respectively, with a relative humidity of 60%. As shown in Table 1, ambient air temperature can have a strong effect on cycle efficiency. For the purposes of this paper, a 25 °C ambient air temperature is used for the design point. However, efficiency values at 15 °C ambient are also shown to allow better comparisons to other power cycle studies, since 15 °C is a more typical reference condition (e.g., as found in ISO 2314:2009).

3. Power cycle description

The power cycle employed in this HWGTR is illustrated in Fig. 2. For the most part, it is a standard recompression cycle, which uses a pair of recuperators ("HT" and "LT" denoting High and Low Temperature) and compressors to better match sCO_2 heat capacities in the recuperators and improve cycle efficiency.

One unique feature of this HWGTR is the use of a split-expansion cycle described in [12]. This cycle option introduces a high-pressure turbine (or turbines) upstream of the reactor. While doing this reduces cycle efficiency, it also lowers reactor operating pressure which helps improve the design of the reactor. Too high of a reactor coolant pressure can worsen tube rupture behavior (described in Section 6), and make the design of valves and seals more challenging. In particular, a split expansion cycle avoids combining peak cycle temperature (550 °C) with peak cycle pressure (22.3 MPa), which for steel piping would require pipe thicknesses in excess of schedule 160 piping.

In this HWGTR, the high-pressure turbines are used to drive the compressors, while the low-pressure turbine is the power turbine that drives the generator. Power turbine speed is chosen to allow direct drive of the generator without needing a gearbox. For simplicity, Fig. 2 shows only one drive turbine driving both compressors on a single shaft; however, the actual design employs two drive turbines on two separate shafts, allowing the speed of the compressors to be varied independently. Placing the power turbine downstream of the reactor helps protect the reactor from events affecting the power turbine, such as a turbine bypass or loss of load. This arrangement places the reactor at a high enough operating pressure (15.0 MPa) such that a trip of the power conversion equipment does not cause the reactor to overpressurize.

Steady state operating points for the cycle are provided in Table 2, for an ambient temperature of 25 °C. The total mass flow rate though the reactor is 2758 kg/s, with 1160 kg/s (42%) of this flow passing through the high temperature compressor. Pressures are shown at both the inlet and outlet of components because pressure losses through piping and valves are important to account for, e.g. 0.33 MPa between the outlet of the reactor and the inlet of the low-pressure turbine. Without these losses, the cycle would generate an additional 12.5 MWe (2.5% higher thermal efficiency).

At the outlet to the cooler, the CO_2 temperature is below the critical temperature of 31.0 °C. However, since the pressure is above the critical pressure of 7.37 MPa, there is no distinct phase transition from gas to liquid within the cooler, and the CO_2 behaves like a high-density supercritical fluid in the main compressor. For analysis of this cycle, the power turbine shaft efficiency used is 92%, while the drive turbine efficiencies average to 89%. The low-temperature and high-temperature compressors have efficiencies of 92% and 85% respectively.

To design this cycle, recuperator and cooler parameters were coarsely optimized to provide the best balance between effectiveness, pressure drop, and equipment cost. This optimization tended to favor a larger number of recuperator and cooler modules with high effectiveness (~98%) to increase overall cycle efficiency. The recuperators and coolers were modeled with sufficient nodalization to check for internal pinch points. The flow split between the

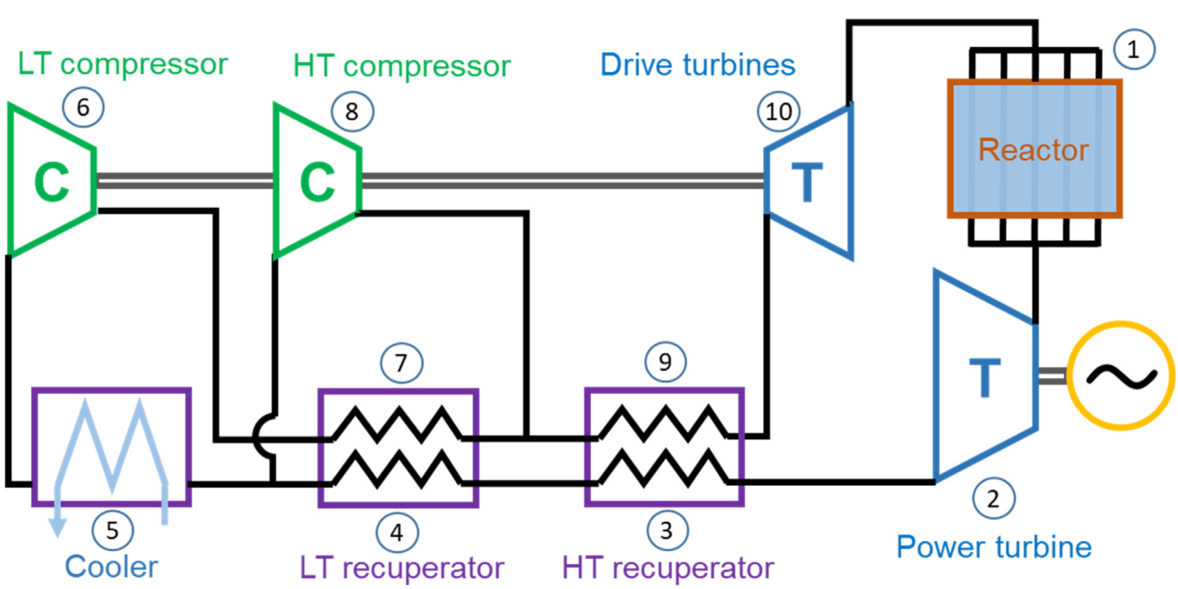

Fig. 2. HWGTR power cycle schematic.

Table 2
Steady state cycle parameters.

Index	Component	Inlet		Outlet		Flow rate (kg/s)
		Pressure (MPa)	Temp. (°C)	Pressure (MPa)	Temp. (°C)	
1	Reactor	15.00	400.9	14.08	550.0	2758.
2	Low-pressure turbine	13.65	549.7	7.97	483.6	2758.
3	HT recuperator (hot stream)	7.91	483.5	7.81	173.3	2758.
4	LT recuperator (hot stream)	7.77	173.1	7.67	65.0	2758.
5	Cooler	7.65	64.9	7.55	29.4	1598.
6	LT (main) compressor	7.50	29.3	22.35	55.3	1598.
7	LT recuperator (cold stream)	22.22	55.2	21.91	169.9	1598.
8	HT (re-) compressor	7.65	64.8	21.99	163.6	1160.
9	HT recuperator (cold stream)	21.81	167.0	21.70	440.6	2758.
10	High-pressure turbines	21.22	440.2	15.07	401.0	2758.

two compressors was selected to match cycle conditions at the merge point upstream of the high temperature recuperator, which tends to maximize cycle efficiency. Reactor inlet pressure was constrained to a maximum of 15 MPa, and the main compressor outlet pressure of 22.35 MPa is set so that the shaft power generated by the drive turbines (~111 MW) equals the power required by the compressors. The reactor outlet temperature of 550 °C is selected to permit compatibility with standard grades of steel such as 316 Stainless and Grade P91. Lower temperature portions of the cycle between the HT recuperator and cooler can be constructed with low alloy steel such as A106 Grade C.

Another viable option for an HWGTR is to use a simple recuperated (SR) cycle in place of a recompression cycle. While cycle efficiency is reduced, the SR cycle benefits by having a lower coolant flow rate (which reduces pressure losses), lower compressor power and less recuperation (which reduce power cycle equipment cost), and more straightforward control and operation. Economic scoping analyses suggest that an HWGTR employing an SR cycle would have a cost of energy only 5–10% higher than one using a recompression cycle. Together with this modest cost penalty, the simplicity and lower technical risk of SR cycles make them a good option for both demonstration and early generation HWGTR systems.

Another important set of CO_2 cycle options are condensing cycles, in which sub-critical CO_2 is cooled below its saturation temperature and liquified [13]. Condensing cycles offer higher efficiency and greater simplicity because they pump liquid CO_2 instead of compressing supercritical CO_2, but require a consistently low ambient temperature for heat rejection. Therefore, condensing cycles may be of particular interest for certain marine propulsion and floating nuclear energy system [14,15] applications, where a cold oceanic heat sink can be perpetually available.

4. Nuclear island description

Fig. 3 shows a general arrangement drawing of the HWGTR reactor building, which is approximately 25 m in diameter and 35 m high. This drawing depicts the reactor core, the main CO_2 lines connecting it to the turbine building, the fuel handling equipment and building crane above the reactor, and the fuel handling space and moderator reflood tanks located beside the reactor. The CO_2 lines pass out the “north” end of the reactor building to the turbine building, while fuel and equipment pass “south” to a fuel building. The moderator equipment and shutdown cooling system (not shown in Fig. 3) are located on the “east” and “west” sides of the reactor building on either side of the reactor core. Similarly, the reflood tanks are located east and west of the reactor to permit passage of the fuel handling machine between them.

The reactor itself uses vertically oriented channels like in past SGHWRs, not the horizontal channels found in CANDU reactors. Because the HWGTR uses enriched fuel and offline refueling, it does not require double-ended channel access like in a CANDU. The vertical orientation reduces footprint, simplifies refueling, and avoids channel sag. Control rods are inserted vertically from the top of the core.

Fig. 4 shows the direction of flow through the reactor core, the inlet and outlet headers, feeder piping, and the reactor itself. The top refueling face of the reactor resembles that of a CANDU, while the bottom face is simpler, with feeder piping attached directly to the ends of the fuel channels. The direction of coolant flow is from the top of the core to the bottom, which helps hold down the fuel assemblies and keeps cooler inlet flow at the top refueling face of the reactor. Downward flow is possible because the HWGTR does not rely on natural circulation of CO_2 for decay heat removal.

Main CO_2 pipes pass straight from the inlet and outlet headers through sliding supports in the containment wall. Thermal expansion of the CO_2 pipes is accommodated by expansion loops outside of the reactor building. This configuration minimizes the CO_2 inventory present in containment and the containment design pressure. A shutdown cooling system, consisting of a small CO_2 blower and cooler, is connected to the main CO_2 pipes. The shutdown cooling system permits CO_2 flow and fuel temperature to be controlled when the power conversion system is unavailable or when the main isolation valves are closed.

HWGTR employs moderator systems like those in a CANDU reactor, which cool the moderator, maintain its chemistry, and remove radiolytically generated deuterium from its helium cover gas. Soluble poisons in the moderator are used to compensate for core reactivity changes over a cycle, like in a PWR. Due to the use of enriched fuel instead of natural uranium, neutron flux in HWGTR is lower relative to CANDU, so tritium concentrations in the moderator build up at approximately 1/6th the rate as in a CANDU, toward an equilibrium value that is also 1/6th as large. The HWGTR employs a passive moderator cooling system that consists of two moderator reflood tanks filled with light water. Operation of the passive moderator cooling system is discussed further in Section 6.

Refueling is accomplished with a rail-mounted vertical fuel handling machine, resembling a smaller version of those used in AGRs. Since refueling occurs when the reactor is offline and depressurized, the cooling, shielding, and pressure requirements of the refueling machine can be greatly reduced relative to an online-refueling system. The refueling machine seals onto the end of a fuel channel, opens the fuel channel, and withdraws a fuel assembly into a CO_2-filled fuel transfer tube. The refueling machine contains a carousel of two transfer tubes, allowing a used assembly to be withdrawn and a new fuel assembly to be inserted in a single visit. Because the fuel assemblies can be cooled via radiation and conduction alone (see Section 6), the fuel transfer tubes do not

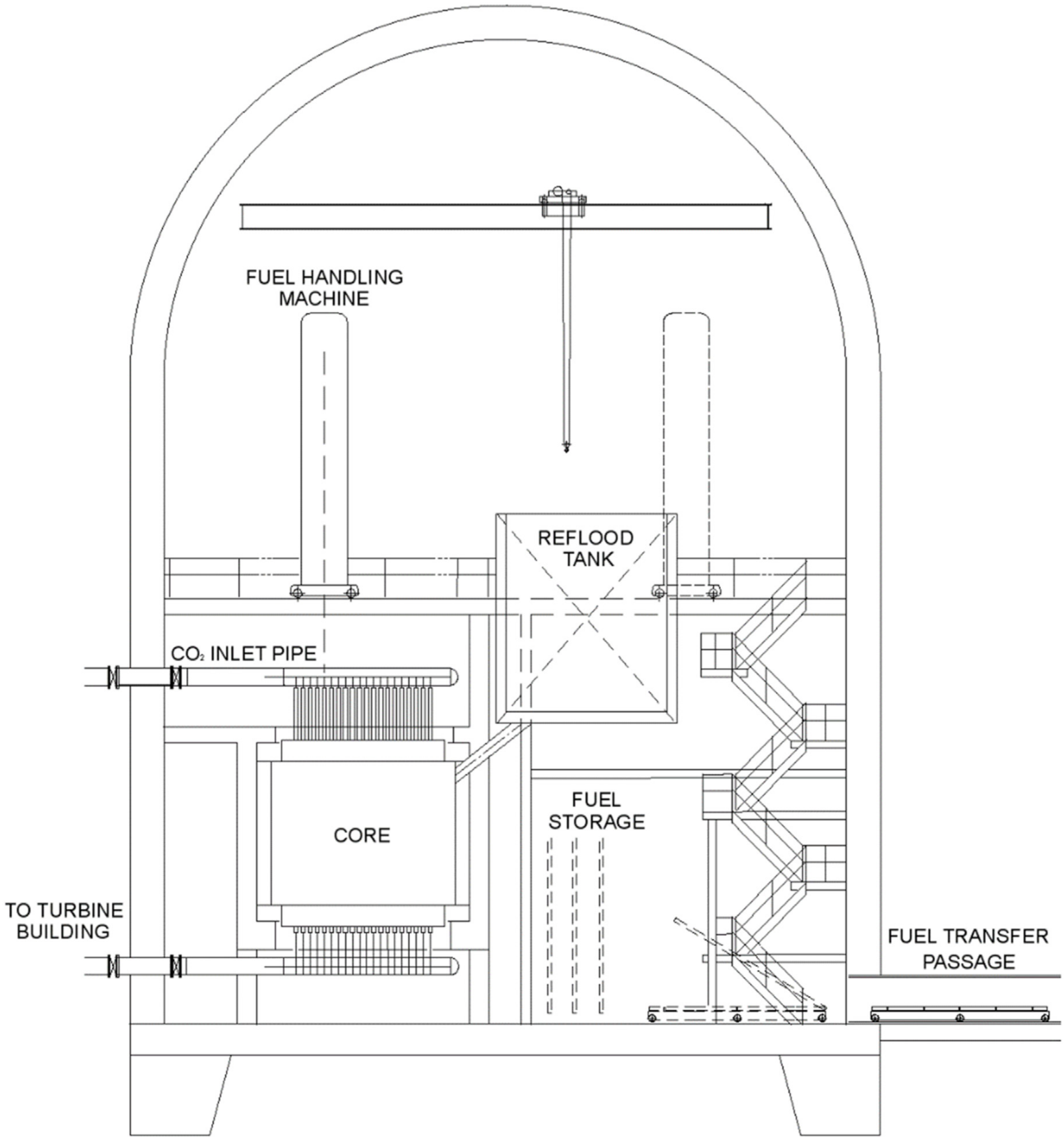

Fig. 3. General arrangement drawing of a HWGTR reactor building.

require internal forced cooling and can rely on external air cooling alone. Fuel assemblies leaving the core are placed into transfer tubes located in the fuel handling pool, which are sealed, upended, and transferred to a spent fuel pool in the adjacent fuel building. The use of a transfer tube protects the irradiated cladding from stress corrosion cracking and thermal shock in water while providing an extra barrier for failed fuel.

5. Core Description

The HWGTR reactor core resembles a vertically oriented, sCO_2-cooled CANDU core that uses fuel developed for the UK CO_2-cooled AGRs. It is made up of 332 vertical fuel channels arranged in a circular cylinder. The height and equivalent diameter of the core are each approximately 5 m. The calandria, a tank of unpressurized heavy water moderator, surrounds the channels. The calandria is in turn surrounded radially by a light water shield tank or vault, and axially by two end shields filled with steel spheres and light water. Fig. 5 shows examples of representative quarter-core core maps used for neutronic modeling, in which different shuffling arrangements were investigated to minimize radial power peaking. The right side of Fig. 6 illustrates the geometry of the fuel channel within each unit cell.

Within the core, fuel channels are made up of Zr-2.5% Nb pressure tubes, the same material used in CANDU pressure tubes. Like in CANDU, the ends of these channels are joined at their ends to thick, hardened steel end fittings via a rolled joint. The top end fitting passes through the top end shield and is attached to it via a bellows, which permits axial growth of the pressure tube. The end fitting terminates in a replaceable closure plug that can be opened by the fueling handling machine for access to the channel. The closure plug contains a valve that permits gas flow rate to the channel to be controlled, similar to in AGR. Near the top of the end

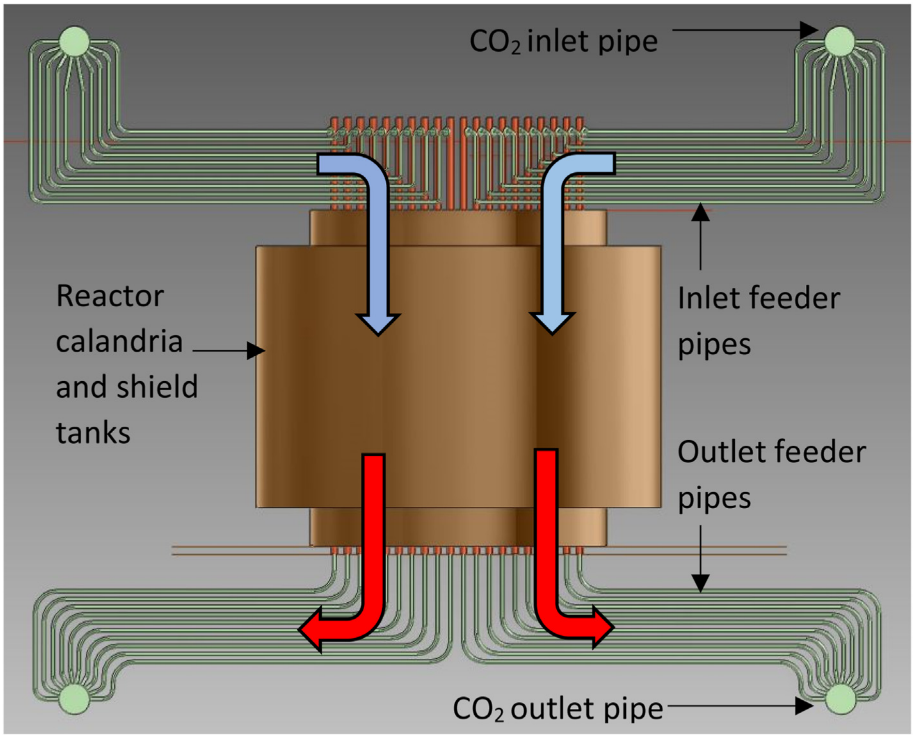

Fig. 4. HWGTR reactor core schematic (elevation view).

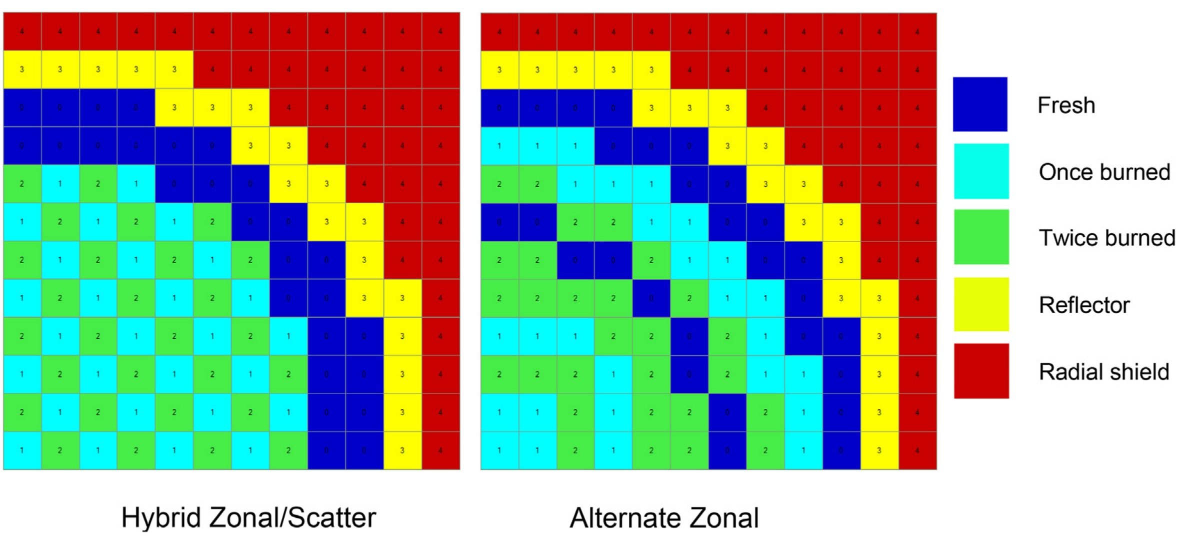

Fig. 5. HWGTR core maps showing example refueling patterns.

fitting, there is a coolant port that is connected via a bolted flange to an inlet feeder pipe, the same type of connection used in CANDU reactors. These feeder pipes are connected to a pair of inlet header pipes that bring in sCO_2 from the turbine building.

The bottom end fittings resemble shorter versions of the top end fitting. Since there is no access into the fuel channel via the bottom end, the outlet feeder piping can be connected directly to the ends of the bottom channel extensions, simplifying the coolant flow path relative to the top of the reactor and allowing the bottom channel extensions to be shorter. The outlet feeders are attached to a pair of outlet header pipes that route CO_2 flow back toward the turbine island.

Each channel contains a replaceable fuel assembly, which consists of a fuel bundle, two axial shields, and a thermal insulation sleeve. The axial shields are helical plugs that permit coolant passage while attenuating neutron and radiation streaming. The fuel bundles resemble smaller versions of those found in UK AGRs, and comprise 21 fuel pins arranged in two rings and a central tie rod. The geometry of the fuel bundle is illustrated on the right side of Fig. 6. Like in an AGR, the fuel pin design consists of annular uranium oxide pellets clad with niobium-stabilized stainless steel containing 20% Cr and 25% Ni, which can tolerate very high temperatures in a CO_2 environment. Fuel temperatures, linear heat rates, and burnups are selected to stay within known AGR fuel

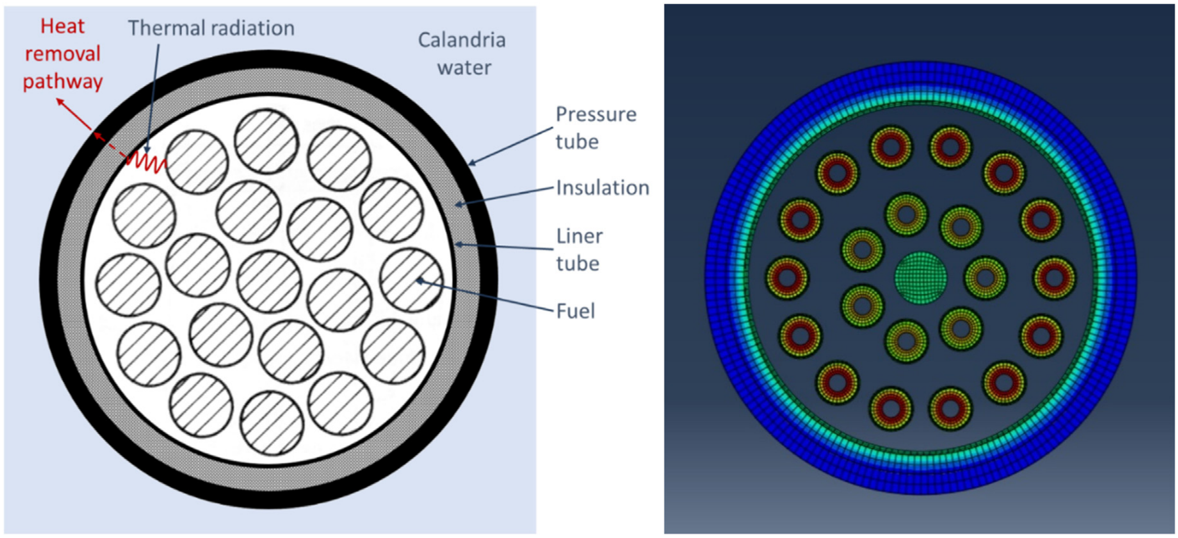

Fig. 6. Calandria heat removal route, with ABAQUS model shown on right.

conditions. Information about the fuel and core are given in Table 3. Reactor physics were analyzed using a combination of Serpent [16] and DIF3D [17] within the ARMI® framework [18], and steady-state thermal hydraulics were analyzed with 1D pressure-drop and heat transfer calculations.

Because the fuel assemblies are surrounded by moderator and shield water in the core, they have an inner insulating sleeve that runs the entire length of the fuel column including both axial shields. The thermal resistance of this insulating sleeve is chosen to minimize the amount of heat lost to the moderator during normal operation while still permitting enough heat transfer for decay heat removal. Similar insulating sleeves were employed in past water-moderated gas-cooled reactors, including EL4 and ML1, which used embossed metal foil [19] and silica fabric insulation respectively.

The presence of the insulation internal to the pressure tube allows the pressure tube to be in direct contact with the moderator and operate at a lower temperature. This is unlike in a CANDU reactor, in which there is a separate calandria tube and gas annulus outside of each pressure tube and the pressure tube operates at the coolant temperature. The lower operating temperature of the HWGTR pressure tube increases its strength and prevents the dissolution of deuterium and subsequent formation of deuterides in the pressure tube, removing the mechanism responsible for past pressure tube ruptures. A leak-before-break assessment was performed to confirm the ability to detect a pressure tube leak before it grew to critical size. The size of the pressure tubes is chosen so that rupture of any one tube doesn't lead to failure of additional tubes or deformation of the calandria. A comparison of pressure tube conditions and dimensions between the HWGTR and other fuel channel reactor designs is given in Table 1 Table 4.

The HWGTR fuel cycle is a once-through, low assay (<4% ^{235}U)

Table 3
Core and fuel channel parameters.

Thermal power (MWt)	540
Power supplied to primary coolant (MWt)	500
Approximate neutron and gamma heating to calandria and shield (MWt)	24
Approximate thermal losses to calandria and shield (MWt)	16
Core cycle length (EFPY)	0.9
Number of fuel channels	332
Average power per channel (MWt)	1.6
Approximate peak channel power (MWt)	2.0
Total fuel mass (kg HM)	25,400
Refueling batches	3
Average discharge burnup (MWd/kg)	21.
Average specific power (MWt/MTHM)	21.
Average channel power density (MW/m^3)	4.9
Channel pitch (m)	0.258
Calandria diameter (main/sub- shell) (m)	6.5/5.9
Calandria height (main/sub- shell) (m)	3.0/5.0
Calandria volume (m^3)	156.
Number of fuel pins per bundle	21
Fuel pin clad OD/ID (m)	0.012/0.011
Fuel pellet OD/ID (m)	0.011/0.0048
Fuel mass per bundle (kg)	15.3
Average fuel linear heat rate (LHR) (W/m)	15,400
Approximate peak LHR (W/m)	29,300
Pressure tube OD (mm)	113.7
Pressure tube ID (mm)	104.3
Average moderator temperature (°C)	70

Table 4
Pressure tube comparison with other reactors.

Reactor type/name	CANDU 6	ACR-700 [20]	Winfrith	EL4	HWGTR
Coolant	D_2O	H_2O	Boiling H_2O	CO_2	sCO_2
Operating pressure (MPa)	11	13	6.5	6	15
Coolant inlet/outlet temperature (°C)	266/310	278/325	280/288	260/500	400/550
Pressure tube temperature (°C)	310	325	288	<100 °C	<100 °C
Pressure tube OD (mm)	111.8	118	140.5	113.4	113.7
Pressure tube thickness (mm)	4.2	6.5	5	3.2	4.7
Approx. Pressure tube stress (MPa)	141	112	88	103	175

enriched uranium fuel cycle, similar to that used in light water reactors (LWRs). With stainless-steel-clad fuel, fuel utilization is similar (within 50%) to that of PWRs, while future ceramic-clad (e.g., SiC–SiC composite clad) fuels can yield fuel utilization superior to that of PWRs. Used fuel would undergo traditional spent fuel pool and dry cask management techniques, with an additional step of sealing the fuel in an CO_2-filled tube prior to immersion in the spent fuel pool to avoid stress corrosion cracking of irradiated cladding in water or air.

6. Safety systems and analyses

The HWGTR safety approach is to use the large thermal inertia of the moderator as an extremely reliable passive heat sink for decay heat removal. This permits HWGTRs to use fewer safety systems and have simpler safety analyses, similar to those found in other Generation IV reactors, such as pool-type sodium-cooled fast reactors. In addition, the HWGTR has significant redundancy and defense in depth for each critical nuclear safety function: core shutdown, cooling, and containment.

6.1. Core shutdown

HWGTR uses a combination of burnable poisons in the fuel and soluble poison shim in the moderator to limit the amount of excess reactivity in the core. Similar to CANDU reactors and BWRs, two diverse mechanisms can be used to shut down the core. The first are physical control rods above the core that drop vertically into the core. The second is a system that injects liquid poison into the moderator. Together these systems provide extremely high assurance of core shutdown.

In addition to these shutdown systems, disruptions to the heavy water moderator will also cause the core to shut down. For example, a pressure tube rupture would cause CO_2 gas to push a portion of the heavy water out through calandria relief ducts, which would render the core deeply subcritical. Subsequent reflooding of the calandria with light water from the passive moderator cooling system would cause the core to remain deeply subcritical, due to the higher moderating power and neutron absorption of light water relative to heavy water.

6.2. Cooling

HWGTRs have four independent systems that can remove decay heat from the core:

1) Power conversion system (CO_2, active, full power)
2) Shutdown cooling system (CO_2, active, decay heat)
3) Moderator cooling system (heavy water, active, decay heat)
4) Passive moderator cooling system (light water tanks, passive, decay heat)

These systems remove heat via two independent routes: via the gas coolant and via the moderator.

The power conversion system removes reactor power during normal operation, and has a motor-driven compressor that can continue to circulate coolant when the reactor is shut down. The shutdown cooling system is similar to the residual heat removal system in a BWR and consists of a smaller circulator and heat exchanger that can remove decay heat from the CO_2 when the power conversion system is unavailable, such as during a maintenance outage or if reactor containment is isolated.

If gas cooling is lost, e.g., in the event of a LOCA, then the moderator heat removal route is available to ensure decay heat removal. In such an event, heat radiates from the surface of the fuel to the liner tube, and then conducts through the liner tube, insulation, and pressure tube into the moderator, as shown schematically in Fig. 6. A thermal analysis of this cooling pathway is provided in Section 6.3.

The moderator cooling system removes 40 MW of power during normal operation, which is sufficient to remove all decay heat, and it employs redundant pairs of pumps and heat exchangers. If the moderator cooling system becomes inoperable (e.g., due to a station blackout) then the passive moderator cooling system ensures decay heat removal. The passive moderator cooling system consists of a pair of elevated light-water-filled reflood tanks in containment that can refill the calandria via gravity. During a prolonged loss of active cooling, the moderator system (which is initially subcooled) will heat up, reach boiling, and pressurize, which automatically opens a set of pressure relief ducts connected to the reflood tanks. The volume and thermal inertia of the reflood tanks suffices to delay further boiling within the calandria for over 12 h. Beyond 12 h, boiling reflood water would introduce steam to the containment, which would be condensed and returned to the calandria, similar to the natural circulation of steam in a Gen-III LWR containment. This containment cooling permits passive decay heat removal for as long a period as desired.

The reflood tanks also come into play if a pressure tube rupture occurs in the core. Such a rupture would pressurize the calandria, causing the pressure relief ducts to automatically open and allow water to flood back into the calandria. In such an event, a coolant depressurization system actuates to reduce coolant pressure and permit more rapid reflooding of the calandria.

The response of these cooling systems to different events is summarized in Table 5. First CO_2 cooling is employed whenever possible to minimize the thermal transient experienced by the fuel. If CO_2 cooling is unavailable, active moderator cooling is used to prevent boiling in the calandria. Finally, moderator flooding via reflood tanks is employed, which together with in-containment recirculation allows for long term passive heat removal. The safety-related passive moderator cooling system serves as a highly reliable backstop that permits the other cooling systems to be non-safety-related.

Table 5
Safety equipment response to different events.

Event	Initial actions	Normal cooling system	Backup cooling system	Safety-related cooling system
Loss of offsite power (LOOP)	Shut down core, start backup power	Shutdown cooling system	Moderator cooling system	Passive moderator cooling system
Loss of flow accident (LOF)	Shut down core	Shutdown cooling system	Moderator cooling system	Passive moderator cooling system
Loss of coolant accident (LOCA)	Shut down core, close containment isolation valves	N/A	Moderator cooling system	Passive moderator cooling system
In-core LOCA (tube rupture)	Close containment isolation valves	N/A	Moderator cooling system	Passive moderator cooling system
Transient overpower (TOP)	Shut down core	Shutdown cooling system	Moderator cooling system	Passive moderator cooling system
Spurious isolation valve closure	Shut down core	Shutdown cooling system	Moderator cooling system	Passive moderator cooling system

6.3. Passive cooling analysis

If gas cooling is lost, due to either loss of flow or loss of coolant, heat from the fuel can still radiate to the inner surface of the pressure tube, as illustrated in Fig. 6. Due to the very high temperature tolerance of AGR fuel cladding [21], this heat removal route is effective at preventing fuel failures. The scenario was modeled in ABAQUS with a radiation heat transfer model, and representative temperature profile results are illustrated on the right panel of Fig. 6. Since CO_2 is a strong absorber of certain infrared wavelengths, participating medium effects are conservatively approximated by reducing the modeled emissivity of all surfaces by 40%, from 0.8 (representative of oxidized metal) to 0.48. This assumption is especially conservative for LOCA scenarios in which CO_2 pressure is lost and therefore its infrared absorption is greatly reduced as well.

Fig. 7 shows peak cladding temperature histories (from the maximum linear heat rate location) resulting from a large-break LOCA, in which all gas cooling is lost. First, the cladding heats up as it is heated by stored thermal energy in the fuel. Within minutes, the fuel cladding approaches a high temperature (~1000 °C) that enables effective radiation heat transfer and decay heat removal via the calandria. As the decay heat falls, the temperatures in the fuel and cladding go down as well.

Based on the temperature histories in Fig. 7, stress on the clad is calculated based on the fission gas pressure within the pin, and clad creep is estimated based on the creep strength relationship given in Ref. [22]. Clad creep was correlated to pin failure probability by fitting a Weibull distribution to the failure probability calculations presented in Ref. [23]. Overall, the cladding's high creep strength and low stress both act to limit the peak cladding strain experienced in the modeled LOCA event to just 0.044%, corresponding to a calculated pin failure probability of approximately 0.02%. In the case of a loss of flow (LOF) event, in which coolant pressure is retained, the inward pressure of the coolant would lower the stress experienced by the clad and reduce both cladding creep and failure probability.

6.4. Containment

The HWGTR containment resembles the large dry containments used in PWR designs. Its design pressure of around 0.4 MPa is typical for PWRs, and is sufficient to contain the entire nuclear island CO_2 inventory. This design pressure is adequate to handle both

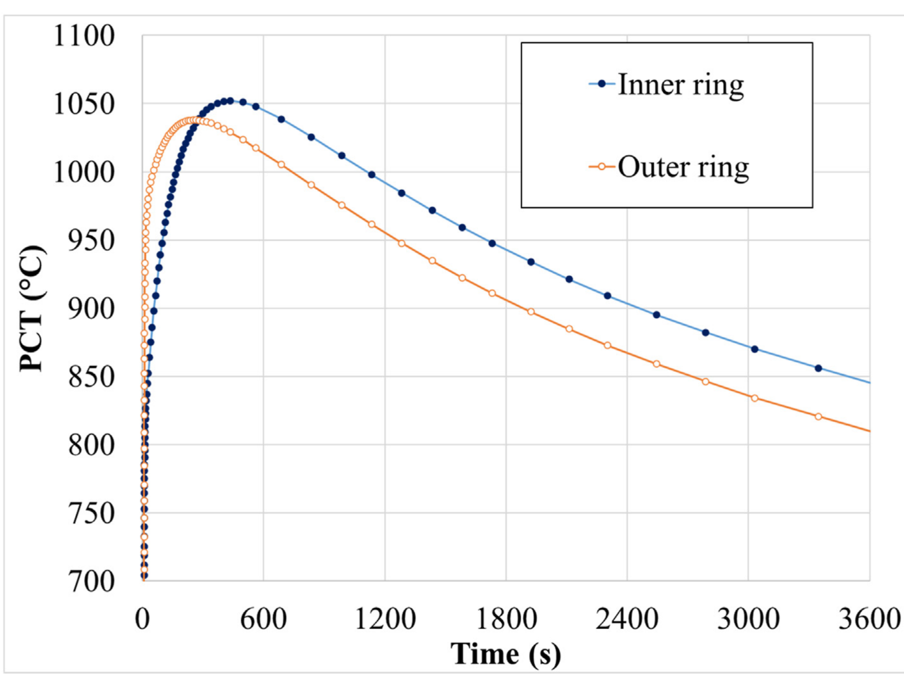

Fig. 7. Peak cladding temperature during a LOCA.

the initial pressurization from CO_2 that would occur due to a LOCA, and any later pressurization that may occur if there is boiling of the moderator or reflood water. Like in a BWR, containment isolation valves are located both inboard and outboard of containment on the main CO_2 lines, allowing containment to be isolated in an accident. Unlike in a PWR, HWGTR does not require containment sprays to reduce containment temperature due to the low mass and thermal energy of the CO_2 coolant. HWGTR also does not require hydrogen recombiners because there are no events in which zirconium comes into contact with superheated steam.

Passive containment cooling, like in a Gen-III LWR, is employed to ensure long-term decay heat removal without the need for safety-related AC power sources. Since the reflood tanks provide a large inventory of cooling water, boiling inside the calandria would not occur for over 12 h. If boiling were to occur, steam would be condensed inside containment and returned to the calandria. Approaches for condensing steam inside an HWGTR containment include cooling via the wall of a steel containment shell (like in the AP1000 design) or via dedicated heat exchangers (like in the Hualong One design). The relatively low height of the containment structure (35 m) allows multiple options for supplying external cooling water, including elevated tanks, ground-level pressured tanks, and fire trucks.

Since the safety approach for HWGTR is to prevent fuel failures by keeping the core submerged in unpressurized water, this lends itself well to emergency approaches if a beyond design basis event were to disable HWGTR safety equipment. For example, fire trucks can be used to add makeup cooling water to the reactor via standpipes, or to the passive containment cooling system if powered cooling cannot be restored. The large thermal inertia of the moderator and its makeup tanks allows ample time for such emergency measures to be implemented.

7. Conclusions and prospects

The preceding discussion provides a high-level introduction to a new nuclear reactor concept: the HWGTR, which couples a pressure-tube reactor core directly to a supercritical CO_2 power conversion cycle. Information is given on the configuration of the power conversion systems, including the power cycle's process conditions and efficiency. The nuclear island systems and their layout are described, and the reactor core and its key parameters are shown in greater detail. Attention is given to the HWGTR safety systems, and how decay heat can be removed via either the CO_2 coolant or through the moderator water. An example thermal analysis is presented that illustrates how fuel failure is avoided even in the case of a total loss of primary cooling, thus allowing a passive, low-pressure water system to serve the safety-related cooling function. The resulting design illustrates how an HWGTR can employ high efficiency direct-cycle power conversion while still employing simple passive safety systems.

Present day LWRs are challenging to deploy due to their numerous safety systems and large, complex nuclear structures. While Gen-IV reactors generally can employ simpler safety systems, they typically require very large pressure vessels and equipment (e.g., HTGRs) or complex process and fuel handling equipment (e.g., liquid-metal, liquid-salt-cooled, and molten salt reactors). Unlike many other Gen-IV reactors, the HWGTR also uses a conventional once-through fuel cycle that avoids the use of HALEU or reprocessing. With its unique combination of high efficiency, simple and compact process systems, and simple safety systems, the HWGTR has strong potential to become a very low-cost form of nuclear generation. The largest open question facing the HWGTR is the economic viability of sCO_2 cycles, since they are a new technology that plays a central role in the HWGTR concept.

In addition to its economic potential, the HWGTR has potential to be developed with less cost and time than other advanced nuclear technologies. It builds strongly on commercial heavy water and gas-cooled reactor technologies, reducing the amount of new development effort required. It employs a familiar industrial fluid (CO_2), greatly expanding the number of available equipment suppliers and existing equipment designs (e.g. valves, instruments, compressors, heat exchangers) that can be employed or adapted. The area that requires the greatest technology development is sCO_2 power cycles, which have been under active development for two decades, and are currently being demonstrated at relevant scale [24] and beginning to see commercial deployment [25]. These projects will result in operating sCO_2 power cycles that will provide valuable information on the economic potential and operational characteristics of sCO_2 cycles and equipment.

To continue the development of HWGTR technology, a smaller demonstration reactor (e.g., ~20 MWe) would be valuable for demonstrating the viability of larger (~200-MWe class) commercial units. This is because the novelty of using direct-cycle gas power conversion, versus currently operating reactors which all use steam cycles to generate power. A pressure tube core would be readily scalable to higher power levels by increasing the number of fuel channels, and the technologies used in a 20 MWe sCO_2 cycle would resemble those used in a larger commercial unit [26]. Therefore, it should be possible to transition from a 20 MWe demonstration reactor directly to 200 MWe commercial units, not unlike the transition from Nuclear Power Demonstration (20 MWe) to Douglas Point (200 MWe) for HWRs, or from Windscale AGR (24 MWe) to commercial AGRs (660 MWe). Due to the HWGTR's thermal spectrum and conventional oxide fuel, a small demonstration reactor would be able to use low-assay (<5% enriched) uranium. Finally, since HWGTR employs technologies that have been developed in the US (sCO_2 cycles), Canada (heavy water reactors) and the UK (CO_2-cooled reactors and fuels), it can represent a unique opportunity for international collaboration toward demonstration and deploying a highly competitive nuclear technology.

Longer term, the HWGTR has strong prospects to become even more competitive. Development of high endurance titanium-nitride strengthened [27] or ceramic-composite claddings can lead to major improvements in fuel temperature tolerance. This can permit higher power density cores with shorter fuel columns, higher temperature more efficient power cycles, and elimination of the shutdown cooling system due to greater tolerance for fuel thermal cycling. With their compact turbomachinery, water-compatible materials, and ability to benefit from a cold ambient temperature, HWGTRs can also be extremely well suited for marine propulsion and offshore power applications. Finally, HWGTRs operate at temperatures compatible with thermal storage salts, allowing them to be coupled to energy storage to increase the flexibility of the plant and its ability to integrate with renewables. This can be done by placing an energy storage system in parallel with the CO_2 power conversion system, so during times of low energy demand, CO_2 would bypass the power conversion system and heat up an energy storage medium instead. Together, the advent of supercritical CO_2 power cycle technology and thermal energy storage technology have the potential to lead to a new generation of highly competitive CO_2-cooled reactors.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgements

The authors are grateful to their colleagues for their numerous technical contributions, including Greg Markham (leak-before-break assessment), Sam Miller (heat transfer modeling), Olu Omotowa (safety event identification), and Michael Pierson (plant layout). We are also grateful to Tim Held, Jason Miller, and David Buckmaster at Echogen Power Systems for their contributions to sCO_2 cycle design and analysis.

References

[1] V. Dostal, A Supercritical Carbon Dioxide Cycle for Next Generation Nuclear Reactors, MIT thesis, 2004.
[2] J.J. Sienicki, A. Moisseytsev, L. Krajtl, A Supercritical CO_2 Brayton Cycle Power Converter for a Sodium-Cooled Fast Reactor Small Modular Reactor, Proceedings of the ASME 2015 nuclear forum collocated with the ASME 2015 power conference, the ASME 2015 9th international conference on energy sustainability, and the ASME 2015 13th international conference on fuel cell science, engineering and technology, ASME, Nuclear Forum. San Diego, California, USA. June 28–July 2, 2015. V001T01A004. ASME, https://doi.org/10.1115/NUCLRF2015-49185, 2015.
[3] Moisseytsev, A, Kulesza, K P, Sienicki, J J, Nuclear Engineering Division, and Oregon State Univ. Thu . "Control system options and strategies for supercritical CO2 cycles.". United States. https://doi.org/10.2172/958037. https://www.osti.gov/servlets/purl/958037.
[4] P. Hejzlar, M.J. Pope, W.C. Williams, M.J. Driscoll, Gas cooled fast reactor for generation IV service, Prog. Nucl. Energy 47 (No.2) (2005) 271–282.
[5] Wright, Steven Alan, Parma, Jr, Edward J, Suo-Anttila, Ahti Jorma, Al Rashdan, Ahmad, Tsvetkov, Pavel Valeryevich, Vernon, Milton E, Fleming, Darryn D, and Rochau, Gary Eugene Sun . "Supercritical CO2 direct cycle Gas Fast Reactor (SC-GFR) concept.". United States. https://doi.org/10.2172/1013226. https://www.osti.gov/servlets/purl/1013226.
[6] Seong Gu Kim, Min Gil Kim, Seong Jun Bae, Jeong Ik Lee, Preliminary design of S-CO2 Brayton cycle for KAIST micro modular reactor. Transactions of the Korean Nuclear Society Autumn Meeting, 2013.
[7] "EL4 – an advanced natural uranium reactor", Nucl. Eng. (September 1963) 312.
[8] Fonteray J, Martin R, Ringot C, Roche R. "Etudes sur les Piles a Eau Lourde Refroidies par Gaz Carbonique (Research on Heavy-Water Reactors with CO_2 Coolant)" Heavy–Water Power Reactors, Proceedings of a Symposium, p 331, 11–15 September 1967, Vienna, Austria.
[9] General Dynamics Corporation, Evaluation of Coolants and Moderators for the Maritime Gas-cooled Reactor, Office of Technical Services, Dept. of Commerce, Washington, D.C., 1958. http://hdl.handle.net/2027/mdp.39015086531806.
[10] H.L. Brey, Development History of the Gas Turbine Modular High Temperature Reactor, 2001.
[11] Aerojet-General Nucleonics, The ML-1 design report. Army gas-cooled reactor systems Program report No, 1960. IDO-28550, 16, 5.
[12] Y. Muto, Y. Kato, Optimal Cycle Scheme of Direct Cycle Supercritical CO_2 Gas Turbine for Nuclear Power Generation Systems, International Conference on Power Engineering, Hangzhou, China, 2007.
[13] Steven A. Wright, R.F. Radel, T.M. Conboy, Gary Eugene Rochau, Modeling and Experimental Results for Condensing Supercritical CO2 Power Cycles, Sandia Report, Jan 2011.
[14] R. Petroski, N. Touran, L. Wood, International Seminars on Nuclear War and Planetary Emergencies 48th Session, in: Near Term Applications and Benefits of Ship-Based Nuclear Energy Systems, Erice, Italy, 2016, pp. 595–613.
[15] J. Buongiorno, et al., The offshore floating nuclear plant concept, Nucl. Technol. 1 (2016) 194.
[16] J. Leppänen, "Serpent – a continuous-energy Monte Carlo reactor physics burnup calculation code: user's manual" VTT technical research centre of Finland, June 18 (, 2015).
[17] K.L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, Applied Physics Division, Argonne National Laboratory, 1984.
[18] N. Touran, J. Gilleland, G.T. Malmgren, C. Whitmer, W.H. Gates, "Computational Tools for the Integrated Design of Advanced Nuclear Reactors", Engineering 3 (Issue 4) (2017), https://doi.org/10.1016/J.ENG.2017.04.016, 518-526, ISSN 2095-8099.
[19] J. Bernard, et al., "Structures du doeur du reacteur 'eau-lourde-gaz EL4', " CEA - R 2665 Report, Geneva, 1964. Conf. A 28/P/69.
[20] Atomic Energy of Canada Limited, "ACR-700 Technical Description", 10810-01371-TED-001 Revision 1, 2004.
[21] M. Bridge, M. Kendal, High-temperature loop experiments - WAGR concluding experiments, in: Gas-cooled Reactors Today, 1982. Bristol (UK).
[22] R.C. Ecob, Primary creep of 20/25/Nb stainless steel AGR fuel cladding under conditions of non-steady stress and stress state, Nucl. Energy 29 (4) (1990) 279–286.
[23] A.P. Thomas, J.B. Hughes, P.K. Holland, Alternative methods for calculating AGR pin failure probabilities in depressurization faults. Gas Cooled Reactors Today, 1982.
[24] Macadam, W. Scott, Follet, Michael kutin, and ganesan subbaraman. "Supercritical CO2 power cycle projects at GTI, in: 3rd European Conference on Supercritical CO2 (sCO2) Power Systems, 2019, pp. 300–309, 2019: 19th-20th September 2019.
[25] Siemens energy utilizing sCO2 technology in new Canadian waste heat-to-power plant, Power Eng. (Feb 15 2021).
[26] C. Turchi, NREL Final Report "10 MW Supercritical CO2 Turbine Test", NREL Nonproprietary Final Report, DE-EE0001589, January, 2014.
[27] H.E. Evans, The development of high-strength fuel cladding for the commercial advanced gas-cooled reactors, Gas Cooled Reactors Today (1982) 55–59.