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Reducing the fuel temperature for pressure-tube supercritical-water-cooled reactors and the effect of fuel burnup

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HIGHLIGHTS

- Typical PT-SCWR fuel uses single-region pins consisting of a homogeneous mixture of ThO_2 and PuO_2.
- Using two regions (central for the ThO_2 and peripheral for the PuO_2) reduces the fuel temperature.
- Single-region-pin melting-to-average power ratio is 2.5 at 0.0 MW d/kg and 2.3 at 40 MW d/kg.
- Two-region-pin melting-to-average power ratio is 36 at 0.0 MW d/kg and 10.5 at 40 MW d/kg.
- Two-region-pin performance drops with burnup due to fissile-element buildup in the ThO_2 region.

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Q. New reactor concepts

ABSTRACT

The Pressure-Tube Supercritical-Water-Cooled Reactor (PT-SCWR) is one of the concepts under investigation by the Generation IV International Forum for its promise to deliver higher thermal efficiency than nuclear reactors currently in operation. The high coolant temperature (>625 K) and high linear power density employed by the PT-SCWR cause the fuel temperature to be fairly high, leading to a reduced margin to fuel melting, thus increasing the risk of actual melting during accident scenarios. It is therefore desirable to come up with a fuel design that lowers the fuel temperature while preserving the high linear power ratio and high coolant temperature. One possible solution is to separate the fertile (ThO_2) and fissile (PuO_2) fuel materials into different radial regions in each fuel pin. Previously-reported work found that by locating the fertile material at the centre and the fissile material at the periphery of the fuel pin, the fuel centreline temperature can be reduced by ~650 K for fresh fuel compared to the case of a homogeneous $(Th–Pu)O_2$ mixture for the same coolant temperature and linear power density. This work provides a justification for the observed reduction in fuel centreline temperature and suggests a systematic approach to lower the fuel temperature. It also extends the analysis to the dependence of the radial temperature profile on fuel burnup. The radial temperature profile is determined from the analytical solution of the steady-state heat conduction equation in the fuel pin, ignoring azimuthal dependence and axial heat flow. The temperature dependence of the thermal conductivity is accounted for, as is the radial dependence of the volumetric power density, which is determined from detailed lattice-level transport calculations performed using the lattice code DRAGON. The reduction in the centreline fuel temperature is shown to be caused partly by a reduction in the heat flux in the case of the two-region fuel and partly by the higher thermal conductivity of both pure ThO_2 and pure PuO_2 compared to that of a $ThO_2–PuO_2$ mixture. The centreline temperature of the two-region fuel is shown to be higher for irradiated fuel than for fresh fuel, a fact explained by the depletion of the fissile material in the peripheral region and the buildup of fissile material in the central region.

1. Introduction

The Pressure-Tube Super-Critical Water-Cooled Reactor (PT-SCWR) is a Generation IV Power reactor being developed in Canada. It is anticipated to have a total efficiency in excess of 50% with 2.5 GW anticipated fission power. The PT-SCWR reactor is one of the six Generation IV reactor concepts being developed in the world. It

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Table 1
PT-SCWR fuel characteristics (reproduced from Pencer et al., 2014).

Component	Dimension	Material	Composition (wt%)	Density (g/cm^3)
Central Coolant (inside fl. tube)	4.60 cm radius	Light water	H:11.19; O:88.81	Variable
Flow tube	4.60 cm inner radius (IR) 0.1 cm thick	Zr-modified 310 stainless steel (Zr-mod SS)	C:0.034; Si:0.51; Mn:0.74; P:0.016; S:0.0020; Ni:20.82; Cr:25.04; Fe:51.738; Mo:0.51; Zr:0.59	7.90
Inner ring (32 pins)	0.415 cm radius 5.4 cm pitch circle radius	15 wt% PuO_2/ThO_2	Pu:13.23; Th:74.70; O:12.07	9.91
Outer ring (32 pins)	0.440 cm radius 6.575 cm pitch circle radius	12 wt% PuO_2/ThO_2	Pu:10.59; Th:77.34; O:12.08	9.87
Cladding	0.06 cm thick	Zr-mod SS	As above	7.90
Coolant	n/a	Light water	H:11.19; O:88.81	Variable
Liner tube	7.20 cm IR 0.05 cm thick	Zr-mod SS	As above	7.90
Insulator	7.25 cm IR 0.55 cm thick	Zirconia (ZrO_2)	Zr:66.63; Y:7.87; O:25.5	5.83
Outer liner	7.80 cm IR 0.05 cm thick	Excel (zirconium alloy)	Sn:3.5; Mo:0.8; Nb:0.8; Zr:94.9	6.52
Pressure tube	7.85 cm IR 1.2 cm thick	Excel (zirconium alloy)	Sn:3.5; Mo:0.8; Nb:0.8; Zr:94.9	6.52
Moderator	25 cm square lattice pitch	D_2O	99.833 D_2O; 0.167 H_2O	Variable (1.0851, nominal)

uses heavy water as moderator and light water as coolant. The inlet coolant temperature and pressure exceed those for the critical point of light water, namely 647.096 K and 22.06 MPa, respectively. The operating coolant pressure is 25 MPa, with the inlet and outlet temperatures being 625 K and 900 K, respectively (Yetsir et al., 2013). The PT-SCWR uses a batch-refuelling scheme, with an anticipated discharge-burnup of 40 MW d/kg (Pencer et al., 2014).

A diagram of the PT-SCWR reactor (Yetsir et al., 2013) is shown in Fig. 1. The reactor vessel is filled with heavy-water moderator at low temperature and pressure and contains a rectangular array of 336 parallel re-entrant fuel channels oriented vertically. Each re-entrant fuel channel consists of an external pressure tube, a smaller-diameter liner tube and a central inner flow-tube. A thermal insulator assembly lines the interior of the pressure tube. It consists of a 5.5 mm-thick ceramic insulator sandwiched between two thin (0.5 mm) liner tubes. The 5 m-long fuel assembly is located in the space between the thermal insulator assembly and the central flow tube and consists of two rings, each comprising 32 fuel pins enclosed in collapsible cladding.

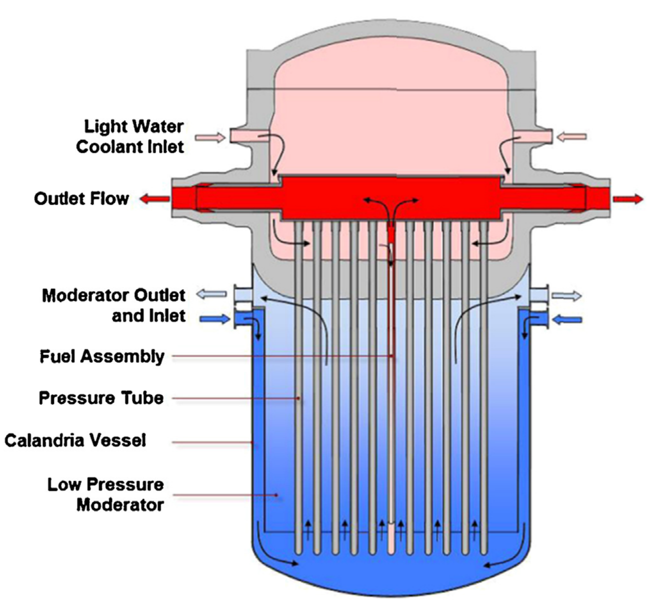

Fig. 1. Preliminary concept of re-entrant channel PT-SCWR (Yetsir et al., 2013).

The coolant inside the flow-tube flows downwards, whereas the coolant between the flow tube and the thermal insulator assembly flows upwards and is heated through contact with the fuel rods. The distance between channels (lattice pitch) is 25 cm. The fuel is a homogeneous (Th–Pu)O_2 mixture, with PuO_2 representing 15% by weight in the inner fuel ring and 12% in the outer fuel ring. The main characteristics of the re-entrant PT-SCWR fuel (Pencer et al., 2014) are presented in Table 1. The fuel temperature profile, as well as axial supercritical-coolant properties have been investigated by several teams (Hughes et al., 2013; Peiman et al., 2012; Kovaltchouk et al., 2015; Lokuliyana, 2014). In particular, Kovaltchouk et al. (2015) found that, by locating the fertile material at the centre and the fissile material at the periphery of the fuel pin, the fuel centreline temperature can be reduced by ~650 K for fresh fuel compared to the case of a homogeneous (Th–Pu)O_2 mixture for the same coolant temperature and linear power density. This work provides a justification for the observed reduction in the centreline temperature and suggests a systematic method of lowering the fuel temperature. It also determines the burnup-dependent radial fuel temperature profile for the Th–Pu fuel rods used by the PT-SCWR.

2. Computational method

The radial temperature distribution inside fuel pins is calculated in two steps. First, the radial distribution of the fission power density inside the fuel rod is obtained from neutronics calculations performed using the lattice code DRAGON (Marleau et al., 2000). Second, the radial fuel temperature profile is calculated by solving the one-dimensional (radial) heat conduction equation analytically inside the fuel pin, accounting for the temperature dependence of the thermal conductivity as well as for the radial distribution of the fission power density.

2.1. Radial power density distribution

The radial distribution of the fission power density inside a fuel pin, as well as the fraction of assembly power generated in each fuel pin, is found from a two-dimensional fuel-assembly transport calculation using the detailed lattice cell geometry and reflective boundary conditions. The code DRAGON (Marleau et al., 2000) is used to solve the 69-group transport equation using the collision probabilities method. The dimensions and material characteristics used in the DRAGON model are listed in Table 1. The DRAGON geometrical model is shown in Fig. 2. To obtain the