Annals of Nuclear Energy 103 (2017) 49–59

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Preconceptual design of a fluoride high temperature salt-cooled engineering demonstration reactor: Core design and safety analysis ☆

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ARTICLE INFO

Article history:
Received 13 July 2016
Received in revised form 2 January 2017
Accepted 3 January 2017

Keywords:
Fluoride salt
FHRs
Molten salt reactors
Fuel cycle
Safety analysis
Advanced reactors

ABSTRACT

Engineering demonstration reactors are nuclear reactors built to establish proof of concept for technology options that have never been built. Examples of engineering demonstration reactors include Peach Bottom 1 for high temperature gas-cooled reactors (HTGRs) and Experimental Breeder Reactor II (EBR-II) for sodium-cooled fast reactors. Engineering demonstrations have historically played a vital role in advancing the technology readiness level of reactor technologies. This paper details a preconceptual design for a fluoride salt-cooled engineering demonstration reactor. The fluoride salt-cooled high-temperature reactor (FHR) demonstration reactor (DR) is a concept for a salt-cooled reactor with 100 megawatts of thermal output (MWt). It would use tristructural-isotropic (TRISO) particle fuel within prismatic graphite blocks. FLiBe (2 $^{7}LiF\text{-}BeF_2$) is the reference primary coolant. The FHR DR is designed to be small, simple, and affordable.

Core design characteristics, fuel cycle performance, and safety analysis of the FHR DR preconcept have been evaluated. The FHR DR core design features a negative or negligible void coefficient throughout a reactor operating cycle. Both single-batch (cartridge) and multiple-batch fuel cycles can be demonstrated in the FHR DR. The single-batch cycle length of the FHR DR core is estimated at between 12 and 18 months, assuming the successful qualification of composite carbon (C/C) or silicon carbide (SiC/SiC) structural fuel block tie rod material. Fuel cycle performance of the FHR DR is similar to a high temperature gas-cooled reactor. Preliminary safety analysis of the FHR DR indicates that the reactor could be used to demonstrate the inherent safety characteristics of FHR designs.

1. Introduction

This paper is a companion paper to Qualls et al. (2017), which presents a high-level overview of a preconceptual design philosophy for a fluoride salt-cooled engineering demonstration reactor. This paper describes details of the reactor core design and preliminary safety analysis for that engineering demonstration reactor.

In 2015, the U.S. Congress authorized the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) to initiate the Advanced Demonstration and Test Reactor (ADTR) study. The ADTR study evaluated advanced reactor technology options, capabilities, and requirements within the context of national needs and public policy to support innovation in nuclear energy. National laboratories, industry, and other relevant stakeholders of an advanced nuclear reactor conducted the ADTR study.

The fluoride high temperature salt-cooled reactor (FHR) engineering demonstration reactor (DR) is a preconceptual design for a small-scale demonstration reactor with the goal of increasing the technology readiness level (TRL) of the overall system (Qualls et al., 2016). The FHR DR was developed as a part of the broader ADTR study. The notional FHR DR concept, also known as a point design, described in this paper was developed for evaluation as part of the ADTR study. The potential missions of the

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http://dx.doi.org/10.1016/j.anucene.2017.01.003
0306-4549/

demonstration reactor concepts in the ADTR study include: (1) process heat applications to reduce the carbon footprint of U.S. industry, (2) closing the nuclear fuel cycle and extending natural resource utilization, and (3) deploying a small scale engineering demonstration reactor with the goal of increasing the TRL of the overall system. Point designs were developed for several advanced reactor technology options (e.g. FHR, HTGR, SFR, and others) and considered against each potential mission. The FHR DR targets the third potential mission for an advanced engineering demonstration reactor. The methodology, approach, and findings of the ADTR study are outside the scope of this paper, which is focused on the technical details of the point design for the reactor core and safety analysis.

FHRs comprise a class of reactor concepts that use fluoride salts as low-pressure coolants to produce high-temperature heat with a high degree of inherent safety. In recent years, Oak Ridge National Laboratory (ORNL), the University of California, Berkeley (UCB), and the Massachusetts Institute of Technology (MIT) (among others) have developed salt-cooled FHR concepts.

The FHR DR will demonstrate many key features of FHRs including:

- Tritium management, salt handling, and passive safety;
- Active heat exchangers proposed for heat removal during maintenance outages for integral FHRs; and
- Safety system operation to enable licensing and deployment of inherently safe commercial FHRs.

The FHR DR will demonstrate the inherent safety features of FHR designs, as Experimental Breeder Reactor II (EBR-II) demonstrated the potential for inherently safe sodium fast reactor designs with the shutdown heat removal tests (Planchon et al., 1986).

The FHR DR is designed to be an affordable, near-term system for demonstrating technology solutions that bridge remaining gaps to established technical viability. To meet that goal, lower risk technologies identified from previous experimental and design efforts are incorporated into the design. However, the flexibility of the facility also allows inclusion, demonstration, and qualification of other technologies being pursued for commercial development efforts.

Technology gaps for FHRs include demonstration of a fuel form for a commercial plant, including the ability to efficiently install and remove a core, in-core structural material performance, and inherent safety performance. Other technical uncertainties applicable to salt reactors pertain to salt procurement, chemistry control, and tritium management of lithium- and beryllium-containing salts. Operational uncertainties include demonstration of reliable pump and heat exchanger performance at temperatures of interest using relevant salt coolants. In addition, demonstration of control rod drive mechanisms will be an important aspect of the FHR DR.

The operating scheme for the FHR DR will resemble Shippingport, the historic light water reactor test bed that demonstrated three different core designs within the same vessel. Another relevant example of an engineering demonstration reactor is the EBR II. Similar flexibility in the FHR DR allows for testing of different fuel forms and cores as they become available. As part of the present work, the neutronics and thermal hydraulics of multiple fuel forms in an FHR DR core were evaluated and deemed acceptable. Based on an assessment of technology readiness, the baseline fuel form selected for the FHR DR was TRISO fuel compacts within a prismatic graphite block. This fuel form was previously demonstrated in high-temperature gas-cooled reactors.

2. FHR DR core neutronic design

The FHR DR uses prismatic block-type fuel with integral fuel compacts and coolant channels as its base fuel form. The fuel is a typical high temperature gas-cooled reactor (HTGR) particle geometry with 15.5% $UC_{0.5}O_{1.5}$ TRISO kernels with packing fractions of 0.35 within compacts. Selection of prismatic block-type fuel was motivated by the manufacturability of the prismatic fuel for HTGRs (Petti et al., 2009). The recent experience in the advanced gas reactor fuel irradiation program is another reason for this selection (Petti et al., 2010; Grover and Petti, 2014a,b).

"FLiBe" (2 ^7LiF-BeF_2) is the primary loop coolant selected for the FHR DR. FLiBe features relatively attractive coolant properties, such as heat capacity and neutron moderation. This combination results in coolant temperature and void coefficients that are either negative or negligible (Williams et al., 2006; Žáková and Talamo, 2008).

The prismatic block fuel type provides significant flexibility in enrichment zoning, fuel-coolant-moderator ratio, coolant distribution, and other core design parameters. The prismatic block fuel type was selected by MIT for their FHR test reactor concept (Forsberg et al., 2014). The cycle length of the FHR DR core is calculated to be 12–18 months with carbon composite (C/C) or silicon carbide composite (SiC/SiC) structural tie rod components in the fuel blocks, with the following characteristics:

- Prismatic block construction with an assembly pitch of 46 cm,
- 180 fuel compacts and 109 coolant channels per fueled position,
- Core height of 350 cm with a fueled height of 261 cm,
- Core diameter of 324 cm with a fueled diameter of 231 cm, and
- 100 MW thermal power.

The completed core consists of 18 fueled hexagonal positions surrounded by 18 full-sized reflector and test positions, as shown in Fig. 1. Partial hexagonal positions near at core structure periphery are filled with graphite reflector to limit bypass flow between the core and the structure that restrains the core and forms the downcomer region within the vessel.

The baseline prismatic block fuel has the same hexagonal flat-to-flat distance as the reference plank fuel assembly for the Small Modular Advanced High Temperature Reactor (SmAHTR) (Greene et al., 2010) and the Advanced High Temperature Reactor (AHTR) (Holcomb et al., 2011) concepts. Therefore, the FHR DR core could later be exchanged for a similar plank-fueled core. Other proposed FHR fuel forms include uranium oxide, carbide, or nitride fuel

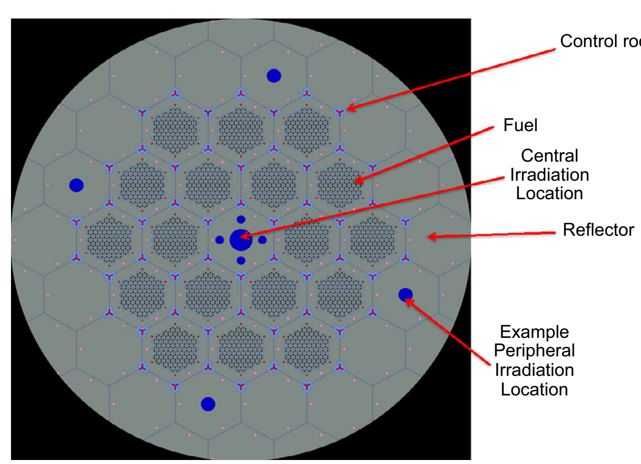

Fig. 1. Planar layout of the FHR DR neutronic model showing key features.