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Core design and performance of the FASt TEst reactor (FASTER)

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1. Introduction

The FASt TEst Reactor (FASTER) is a Sodium-cooled Fast Reactor (SFR) point design. It originated from the Advanced Demonstration and Test Reactor (ADTR) study (Petti et al, 2017) chartered by the Advanced Reactor Technology campaign of the U.S. Department of Energy, Office of Nuclear Energy in 2015/2016. The FASTER test reactor plant is a pool-type sodium-cooled fast spectrum test reactor that provides high levels of fast and thermal neutron flux for scientific research and development. It is mostly relying on previously demonstrated Sodium-cooled Fast Reactor (SFR) technologies, and with 300 MWth it would be able to offer the highest levels of both fast and thermal neutron fluxes available in the world, in a single reactor. The reactor plant has a superheated steam power conversion system, which can provide electrical power to a local grid allowing for recovery of operating costs for the reactor plant. In addition, the FASTER reactor plant could be used for isotope production or as a heat source, if desired. A companion paper titled "Fast Test Reactor (FASTER) design overview" (Heidet et al, 2018) provides a more detailed overview of the FASTER concept.

The FASTER core concept and its performance characteristics are presented in this paper. An overview of the core design and layout is provided in Section 2. The performance characteristics are summarized in Section 3. The thermalization of leaking neutrons, enabling thermal spectrum irradiation capabilities, is briefly described in Section 4. Testing capabilities enabled by FASTER are detailed in Section 5. The reactivity coefficients and the quasi-static reactivity balance are discussed in Section 6, leading into the safety behavior of the core. The safety analysis for FASTER is discussed in details in a companion paper titled "Safety Analysis of the FASTER Test Reactor Preconceptual Design" (Sumner et al., 2018). The reactivity control systems, reactivity coefficients and basic thermal hydraulic analysis are discussed in Sections 6, 7 and 8, respectively. A summary of this paper is provided in Section 9.

2. Design characteristics

The FASTER core design is not based on any previously existing fast reactor, but uses materials and dimensions consistent with the U.S. base technology program. The main objective of the FASTER reactor design efforts was to achieve a fast flux of at least 5×10^{15} n/cm^2-s as well as a thermal flux of about 1×10^{15} n/cm^2 while offering a large number of test locations.

Ternary metallic fuel, U-Pu-Zr, is used with HT-9 stainless steel for cladding and structural material. Plutonium-bearing fuel was preferred over LEU fuel due to the higher flux levels achievable (Heidet and Hill, 2017). Although there is no mandated limit on the weight fraction of plutonium that can be used in the fuel, it was decided to limit it to 20%, by weight (wt%), based on the availability of irradiation data for this type of fuel. Another incentive for not resorting to higher plutonium weight fraction is the degradation of the fuel thermal conductivity as plutonium content is increased. This is of particular importance for the FASTER reactor due to the high power density.

In order to optimize the reactor performance and obtain a compact core, the zirconium weight fraction in the fuel is assumed to be 6 wt% and the fuel smear density is assumed to be 85%. Using 6 wt%, instead of the 10 wt% (Hofman et al., 1997) often used for metallic fuel, does not affect the characteristics of the ternary fuel (Hofman et al, 2009);

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additionally, irradiation tests have previously been performed for such a fuel type. The decision to use an 85% smear density, instead of the 75% more often used for metallic fuel, is based on the relatively low peak burnup, compared to other SFR concepts, that will be achieved. Because of the low fuel burnup, the internal stress applied by the fuel on the cladding as a result of irradiation swelling will be lower than observed in metallic fuel that reaches a high burnup. Furthermore, the fission gas plenum length relative to the active fuel length does not need to be as long as what is typically used in SFR core designs, because of the lower fuel burnup achieved. For the FASTER core design, the fission gas plenum length is set to be 65% of the active fuel length.

2.1. Core design approach

The FASTER core layout and design characteristics are the result of a multi-step design process. Initially the smallest core volume resulting in a critical core was determined, using 20 wt% plutonium. Then the power of the core was increased until the thermal hydraulic limits, discussed in Section 7, were reached. Thermal hydraulic and safety analyses were performed for that core configuration, and the core layout and dimensions were progressively modified in order to increase the peak fast flux to above 5.0×10^{15} n/cm^2-s ($E_n > 0.1$ MeV).

This required slightly flattening the core, to reduce the coolant pressure drop across the core, therefore allowing increasing the coolant flow rate and increasing the peak power density. The peak power density correlates almost directly with the achievable peak fast flux. With a flattened core geometry, the volume of the proposed core is larger than the minimum volume needed for criticality. This resulted in a total power level of 300 MW_{th} in order to achieve the targeted peak fast flux of 5×10^{15} n/cm^2, with the selected constraints (fuel type, fissile content, thermal hydraulic constraint, passive safety).

Furthermore, FASTER was designed to be inherently safe for a base configuration, not containing performance-affecting test materials in the test assemblies.

2.2. Analysis tools

The Argonne code suite for fast reactor analysis (DIF3D/REBUS) (Smith et al, 2014; Toppel, 1983) and MCNP6 (Los Alamos Scientific Laboratory Group X-6, 2014) are used to perform the neutronics and depletion calculations. A different set of cross sections is used in the different regions of the core in order to properly capture all the spectra changes, and have been generated using MC2-3 (Lee and Yang, 2012) associated with TWODANT (Alcouffe et al, 1984), based on the ENDF/B.VII library. All assemblies are modeled as having a homogenous composition, which might introduce some slight bias in the region where neutrons are moderated, and might require more detailed assessment.

The PERSENT code (PERturbation and SENsitivity for Transport) (Smith et al., 2013) was used to determine reactivity coefficients. It is based upon the variational nodal method employed in DIF3D-VARIANT. PERSENT provides perturbation theory calculation options for generating the spatial breakdown of reactivity coefficients such as Doppler and computing the point kinetics parameters.

SuperEnergy2-ANL (Basehore and Todreas, 1980) was used to performed thermal hydraulics calculations for FASTER and to determine the orificing strategy. It is a steady state thermal hydraulics code used to optimize flow orificing in SFRs. It assumes a hexagonal lattice of ducted fuel assemblies with wired wrapped fuel pins. It is loosely based upon a porous body medium model and uses correlations based upon experimental measurements of wire-wrapped pin bundles to predict the pressure drop and mixing with each assembly. It has radial conduction models to handle the by-pass flow and heat transfer between adjacent assemblies with different temperatures.

2.3. Core layout and assemblies

The 300 MW_{th} FASTER core layout, shown in Fig. 1, is composed of 55 fuel assemblies, each with the same plutonium weight fraction. The fuel, coolant and structural material volume fractions are 30.93%, 37.36%, and 23.65%, respectively. The active fuel height is 80 cm. Six primary control rod assemblies and three secondary control rod assemblies composed of B_4C rods ensure the safe shutdown of the core. There are 33 fast neutron flux test locations, in addition to the two closed loops also being exposed to a fast neutron flux. It also has three thermal neutron flux test locations and one closed loop being exposed to a thermal neutron flux.

The fuel assembly positions have been chosen to enhance neutron leakage probability toward the moderated zone (brown in Fig. 1). The purpose of the moderator is to take advantage of the neutrons leaking out of the active core region and thermalize them in order to provide thermal spectrum testing capabilities. With the current design, fast neutrons are thermalized by the moderator and do not return into the active core region because of the reflector layer between the two regions. This design approach, discussed in Section 4, prevents a number of potential issues. Canned beryllium is used as the moderator and Zircaloy is used as the structural material in that region to avoid parasitic absorption of thermal neutrons in iron. A detailed discussion and analysis of the thermal zone is available in reference (Heidet et al., 2018). The moderated region does not contain any fuel and is cooled

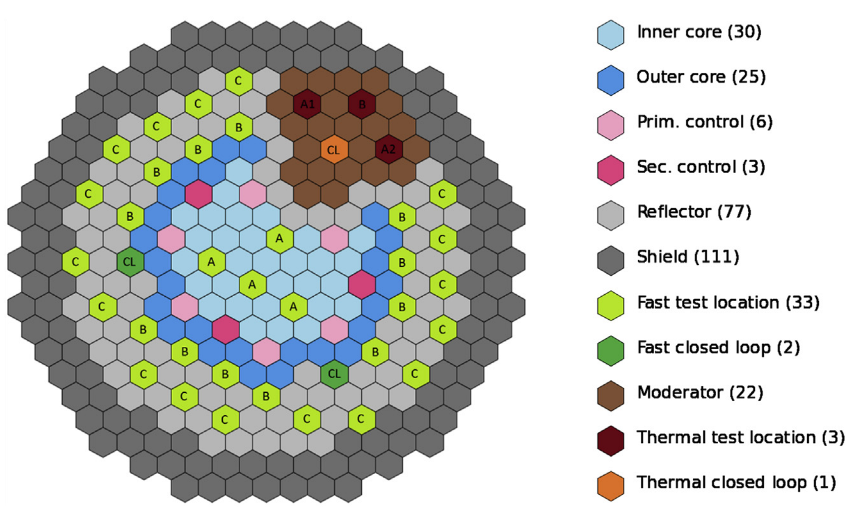

Fig. 1. FASTER core layout.