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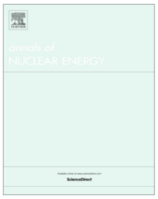

Neutronics and thermal hydraulics analysis of a small modular reactor

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ABSTRACT

The small modular reactor (SMR) offers many feasible pathways for the construction of more nuclear power plants. A physics model of a near term deployable SMR of the integral pressurized water reactor (IPWR) design is developed. Fuel depletion simulations are performed to optimize the active fuel length, fuel enrichment and core loading pattern in order to achieve a uniform core power distribution. The optimized core can produce 500 MW of thermal power with a four year core life-time at a capacity factor of 87%. The core consists of 69 uranium dioxide (UO_2) fuel assemblies; 5 assemblies at 4.4 at% ^{235}U enrichment and 64 assemblies at 4.95 at% ^{235}U enrichment. The active fuel length is 200 cm and the core diameter is 194.55 cm for an active core height-to-diameter ratio of 1.03. As part of the study the active fuel length is increased to 240 cm resulting in an increased capacity factor of 95% at 530 MW of thermal power output for an active core height-to-diameter ratio of 1.23. Rod cluster control assemblies (RCCAs) are placed strategically to reduce the overall core power peaking factor to 1.3. Estimated reactor kinetics parameters such as the delayed neutron fraction and mean neutron generation time are typical of existing larger pressurized water reactors (PWRs) from which much of the IPWR based SMR design is derived. This study showed that Doppler, moderator temperature, void and power reactivity coefficients are all negative over the core life-time of four years indicating the possibility of safe reactor operation. A semi-analytical thermal hydraulics analysis reveals acceptable radial and axial fuel element temperature profiles with significant safety margin from industry standards on peak fuel and clad surface temperature limits. The critical heat flux (CHF) is calculated and is not exceeded even in 10% overpower conditions. In addition the nucleate boiling ratio (DNBR) is calculated and found to be above 4.8 for the entirety of the active core region. These parameters further engender confidence in the safety of the SMR design.

1. Introduction

The small modular reactor (SMR) has enjoyed increased popularity within the nuclear energy community as it represents a plausible reactor type to see construction as part of a renaissance in nuclear energy (Ingersoll, 2009). SMRs offer numerous benefits including inherent safety features such as passive heat removal capabilities for decay heat (in some designs for full power operation), increased security and proliferation resistance with integrated safeguards, and underground construction to address the threats of sabotage, airplane impact and some natural hazard scenarios. SMRs also offer significant economic benefits such as lower capital investment, shorter construction times and greater ability to match plant capacity with demand by adding additional modules as required (Carelli et al., 2010). Both the end user of the electricity and the developer of the power plant stand to profit from scale gains which are not present in a conventional large Pressurized Water Reactor (PWR) as SMRs can be situated in areas with smaller electrical grids, limited supplies of water and/or land allowing greater proximity to the end user and to industry for process heat applications.

All over the world, many SMR designs are under development spanning the entire range of nuclear technologies currently available from thermal, epithermal and fast neutron spectrum reactors to light water, heavy water, gas and liquid metal coolants and electrical power output ranging from ~5 MWe to the 300 MWe limit by SMR definition (IAEA, 2014). In the United States, the integral pressurized water reactor (IPWR) design is the closest to deployment due to the vast operating experience with the larger PWR; its predecessor. In March 2013, the first of two United States Department of Energy (USDOE) funding opportunities worth approximately $226 million was awarded to the mPower SMR, followed by a second award of the same sum to NuScale in December 2013, both belonging to the IPWR type. At the time of writing, the United States Nuclear Regulatory Commission (NRC) had four different ongoing pre-application licensing activities with regards to SMRs (USNRC, 2014); the NuScale Power module from NuScale

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Power, LLC, the BWXT mPower from BWTX mPower Inc. (a Babcock and Wilcox company), the Westinghouse SMR from Westinghouse Electric Company and the SMR-160 from SMR LLC (a Holtec International Company). All four are of the IPWR SMR design concept.

The remainder of the paper is structured as follows. Section 2 of the paper states the research objectives of this work and provides a brief overview of the IPWR SMR design. This is followed by a detailed discussion of the physics modeling employed for the reactor core depletion simulations and thermal hydraulics analysis in Section 3. The results are presented and discussed in Section 4 and the paper concludes with a brief summary of the work and avenues available for future work.

2. Research objectives

The large break loss of coolant accident (LOCA) is amongst the worst accident scenarios for existing PWRs. In this scenario, it is assumed that one of the large coolant pipes connecting the reactor core to the steam generators suffers a large double ended break. Such an event would lead to rapid uncovery of the core (due to sudden loss of a large fraction of the coolant inventory), the possibility of fuel failure and potentially an eventual large scale radioactive fission product and actinide source term release. In order to avoid a release of radioactivity to the environment, a plethora of auxiliary safety systems have been added to the designs of PWRs to ensure that in the case of the large-break LOCA, the core remains covered with coolant and the heat removal systems remain capable of removing the remaining decay heat thereby preventing fuel failure.

The IPWR takes a dramatic approach to mitigating this accident scenario. The IPWR design places the pressurizer, steam generator and coolant pumps along with the core inside the same pressure vessel; thus drastically reducing the large coolant pipes and the associated possibility of a large-break LOCA altogether. This can be done more easily in the case of SMRs as their smaller size facilitates the forging of large enough pressure vessel to house the required components. As mentioned, the proposed SMRs that are near deployment are all of this IPWR type. Not only does this eliminate an entire category of accident scenarios but also increases the coolant inventory in the core allowing heat removal by natural circulation to be applicable over a wider range of operation. Such innovative and inherently safe design features capitalizing on the small core size are characteristic of SMRs.

The objectives of this work are twofold. The first objective is to develop an IPWR SMR core model to analyze core neutronics, reactor safety and fuel depletion performance. The second objective is to perform a thermal hydraulics analysis on the developed model to confirm the heat removal capability of the design. The results of these analyses allow conclusions regarding the feasibility of the design, its performance characteristics and initial dynamics considerations to be made. Fig. 1 is a schematic of the layout of an IPWR SMR.

2.1. Design objectives and model parameters

The main application of the new fleet of SMRs will be for electricity generation. Although they can be used to replace coal fired plants producing <300 MWe for base load purposes, their niche market will be in areas where the existing grid is not capable of accommodating a single large 1000 MWe plant online. Rather a series of six individual SMRs producing 150–200 MWe per module can be employed to meet the same demand. The first module can be built immediately with additional units brought online in direct response to the demand for electricity in the area (Ingersoll, 2009).

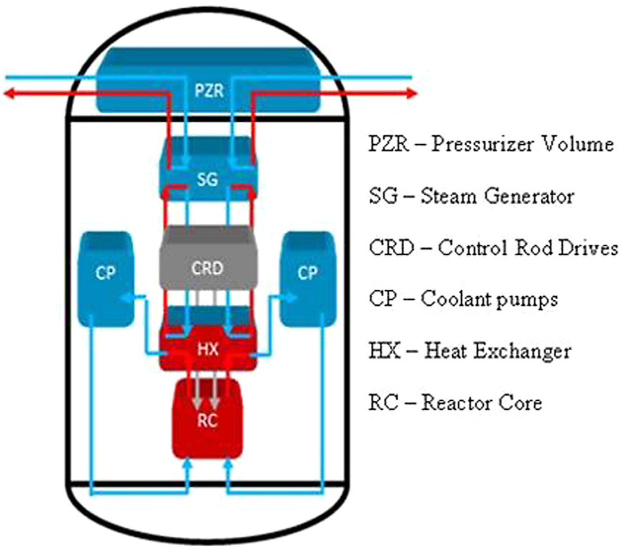

Fig. 1. Schematic showing layout of an integral pressurized water reactor.

The IPWR based SMR design borrows heavily from the existing fleet of large PWRs in operation in the U.S. By leveraging upon technologies used in the currently operating PWRs, the licensing and certification of the resulting IPWR SMR design is anticipated to be faster; with expedited deployment of these reactors driving design choices, as with the mPower SMR from B&W (Halfinger and Haggerty, 2011). The model development in this work is based on the publically available characteristics proposed for the mPower SMR released by B&W in 2010. Specifically; 500 MW thermal power output, 4 year core life-time, fuel enrichment less than 5 at% of ^{235}U and no soluble boron in the coolant. The absence of soluble boron as a means of reactivity control results in an increased importance of Rod Cluster Control Assembly (RCCA) manipulations and the initial burnable poison loading. As such the results presented are uncontrolled reactivity results.

An update to the design was released in 2012 featuring an increased thermal power output (Babcock and Wilcox, 2012). These model parameters are chosen because they represent one of the most advanced SMR designs in the United States. The SMR design is intended for a “battery type” deployment meaning there is no fuel assembly shuffling or refueling over the entire life-time of the core’s operation for four years. Once the core can no longer maintain criticality (sustained controlled nuclear fission chain reaction with neutron multiplication factor equal to one), the entire core is removed and replaced much like replacing a depleted battery (IAEA, 2007). Other parameters needed for developing the core model, such as active fuel length, number of assemblies needed, radius of the core, uranium fuel mass, etc were derived using engineering and physics calculations as this information was not available from open literature or other sources.

3. Physics modeling

A three dimensional full core model of the IPWR SMR design is developed using MCNP5 (X-5 Monte Carlo Team, 2008) and MCNPX2.6 (Pelowitz, 2008) to assess the safety and performance of the reactor core by calculating; 1) Doppler, coolant temperature, void and power reactivity coefficients; 2) the delayed neutron fraction and mean neutron generation time; 3) the core life-time and end of life (EOL) isotopic composition and 4) axial and radial (spatial) core neutron flux and power profiles. These parameters were also followed up at discrete temporal intervals within the