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Core design optimization and analysis of the Purdue Novel Modular Reactor (NMR-50)

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ABSTRACT

A BWR-based SMR called the Novel Modular Reactor (NMR-50) is being developed at Purdue University. NMR takes the advantages of the two-phase flow driving head, which allows a much smaller and simpler reactor pressure vessel (RPV) compared to the integral PWRs. In this study, through a systematic stepwise optimization approach including a simulated annealing based optimization method, an optimum core design that meets a 10-year cycle length with a minimum fuel cost while satisfying safety related criteria was derived and analyzed. The lattice code CASMO-4, the whole core analysis code PARCS and the thermal–hydraulics code RELAP5 were used to perform calculations from pin cell up to whole core depletion calculations. The NMR-50 optimized core design is able to achieve a 10.2 year cycle length with an average fuel enrichment of 4.61 wt% of ^{235}U in a 10×10 lattice fuel assembly. The minimum critical power ratio (MCPR) and the maximum fuel linear power density (MFLPD) during the cycle are 1.99 and 18.25 kW/m, respectively, providing large margins to thermal design constraints. The NMR-50 control system design is able to provide a sufficient cold shutdown margin of 1.7%. With its small reactor core size, large negative void coefficient, and low operating thermal neutron flux, an enhanced xenon stability characteristic is possible. Peak fast neutron fluence of 8.8×10^{21} n/cm^2 was below the industry standard limit, which from extensive plant data records, should not be a major concern to channel distortions from a radiation damage point of view.

1. Introduction

Since the establishment of nuclear power generation for commercial use, reactor size had been consistently increasing from 60 MWe to around 1600 MWe (IAEA, 2012). In recent years, there has been an increase in interest in small modular reactors (SMRs), below 300 MWe, due to the significant capital cost reduction, simpler design units, reduced construction time, and power accessibility to regions away from large grid systems. SMR has a market that includes countries with moderate sized grid system, many islands, and remote islands as well as countries planning to fine tune the electricity generation capacity. The United States, Argentina, Brazil, France, Japan, Republic of Korea, and Russia are one of the countries with ongoing interest and development of SMRs (IAEA, 2012). In terms of nuclear design principles, SMRs are not a foreigner to current large LWR's. In fact, it is based upon extensive nuclear operational experience of current LWR's (Worrall, 2015).

By taking the advantages of the two-phase flow driving head, which allows a much smaller and simpler reactor pressure vessel compared to the integral PWRs, a BWR-based SMR called the Novel Modular Reactor (NMR) is currently being developed at Purdue University. The NMR system design was derived from the 600 MWe simplified boiling water reactor (SBWR-600) design of General Electric (GE) based on a three-level scaling methodology (Ishii et al., 2013). The NMR has a significantly reduced reactor pressure vessel height. Precisely, its height is reduced by a factor of three compared to conventional BWR RPVs and has a net electrical power output of 50 MWe.

The key objective of the NMR-50 core design is to be able to achieve a 10-year cycle length for the deployment in remote sites while satisfying thermal–hydraulics and materials related design criteria (Wu et al., 2016). Wu et al. (2016) developed a core design that meets these design objective and constraints through parametric studies starting from the AREVA Atrium 10B assembly

Abbreviations: SMR, small modular reactor; NMR, Novel Modular Reactor; BOC, beginning of cycle; EOC, end of cycle; VC, void coefficient; MCPR, minimum critical power ratio; MFLPD, maximum fuel linear power density; SDM, shutdown margin; PMAXS, Purdue Macroscopic Cross-Section; S3C, SIMULATE-3 case matrix; CHF, critical heat flux; BP, burnable poison; Gd, gadolinium; SA, simulated annealing; FA, fuel assembly; AO, axial offset; O-2, Oskarshamn-2.

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design. In this work, a systematic step-by-step optimization approach, including a simulated annealing (SA) based optimization method (Kirkpatrick et al., 1983) was performed to deliver an optimum core design that meets these design goals with a minimum fuel cost.

This paper is structured as follows: in Section 2, the design objectives, constraints, and main variables are presented. Section 3 describes the neutronics and thermal–hydraulics analysis methods and models, and Section 4 presents the systematic step-wise optimization approach, tools used, and the results obtained. In Section 5, the NMR-50 design and performance characteristics are discussed. Section 6 discusses the main conclusions.

2. Design objectives and constraints

The main design objective of the NMR-50 is to obtain a cycle length of 10 years while minimizing the fissile loading (i.e., fuel cost). The current commercial BWR operates at a maximum specific power of around 24 W/gU. However, at this specific power level, it is not possible to achieve a cycle length of 10 years under the current industrial constraint of 5 wt% fuel enrichment. Therefore, it is necessary to reduce the specific power to achieve the targeted 10-year cycle length.

To ensure safe plant operation, two types of design constraints were imposed: reactivity bases and overpower bases. One of the main reactivity constraints on the core design is to ensure negative reactivity feedback coefficients. For a low enriched uranium fuel, the fuel temperature coefficient of reactivity is negative mainly due to the Doppler effects of U-238 resonance cross sections. In BWRs, voiding is the predominant factor in moderator density changes. Since a large coolant void coefficient (VC) can lead to an excessive power tilting, the minimum VC was constrained to be around −50 pcm/%void at the beginning of cycle (BOC) to limit the power tilting while providing a sufficient negative reactivity feedback. In addition, the cold shutdown margin (SDM) was required to be greater than 1%. As the overpower-based constraints, the minimum critical power ratio (MCPR) and the maximum fuel linear power density (MFLPD) were considered. The MCPR characterizes the "flow boiling crisis" phenomenon, in a two-phase fluid flow, which is strictly prohibited in the operation of BWRs. The MFLPD limits the peak cladding temperature during a loss of coolant accident (LOCA). The limiting values of the GE SBWR-600 were adopted in this study, which are 1.3 and 45.3 kW/m for the MCPR and MFLPD, respectively (Ishii et al., 2013).

Another constraint in the design of NMR-50 takes into account the effect of a 10-year cycle length on the performance and integrity of structural material. Since irradiation damage strongly depends on fast neutron fluence, the peak fast fluence was limited to the current BWR standard. Typical operational peak condition of structural components, such as the fuel assembly (FA) channel box, is 2×10^{22} n/cm^2 (Bradley and Sabol, 1996).

These design objectives and constraints are shown in Table 1 along with the design variables considered. The core design variables in Table 1 are divided into two groups, main and other variables. Since NMR-50 was derived from SBWR-600, large fractions of the design variables are fixed such as the core radius, number of assemblies, fuel height, etc. As a result, the primary core design parameters that significantly affect the core performance are those included in the "main variables."

3. Analysis methods and models

In this section, the analysis methods and computational tools for creating a model on the NMR-50 are presented. Important geometrical and thermal–hydraulics design parameters of NMR-50 are shown in Table 2.

Table 1
Design objectives, constraints, and variables of NMR-50 core.

Parameter	Value
Core design objectives	
Cycle length	⩾10 years
Fuel cost	Minimize
Core design constraints	
Maximum fuel enrichment of U-235	5 wt%
Minimum critical power ratio	>1.3
Maximum fuel linear power density	<45 kW/m
Fast neutron fluence ($E > 1$ MeV)	$<2 \times 10^{22}$ n/cm^2
Reactivity coefficients	Negative
Void coefficient at BOC	<−50 pcm/%void
Sufficient shutdown margin of control system	⩾1%$\Delta\rho$ (Oka, 2010)
Core design variable	
Main variables	Other variables
Fuel enrichment	Active fuel height
Fuel rod diameter	Clad thickness
Water-to-fuel volume ratio	Core radius
	Gap thickness
	Number of control blades
	Assembly size

CASMO-4 (Ekberg et al., 1995), a well-established, industry-standard lattice physics code was used to perform pin cell and assembly calculations during the assembly design optimization and to prepare the burnup dependent cross section libraries for the core calculations. The assembly design step mainly involved varying pin size, fuel enrichment, water-to-fuel volume ratio, and void fraction. The optimum assembly enrichment split was determined using simulated annealing optimization approach. In addition, the optimum amount of burnable poison (BP) and number of BP pins were determined.

The computer code named Generation of Purdue Macroscopic Cross-Sections (GenPMAXS) is an interface between lattice physics codes and the whole core simulator PARCS (Downar et al., 2010). It provides the so-called Purdue Macroscopic cross section (PMAXS) file (Xu and Downar, 2012), which includes macroscopic and microscopic cross-sections, discontinuity factors, kinetic data, and yields for the poisons. The cross sections are linearized as a function of state variables except for the moderator density and

Table 2
Geometry and thermal–hydraulics design parameters for the NMR-50 core.

Parameter	Value
Geometry	
Number of fuel assemblies	256
Fuel assembly arrangement	18 × 18
Equivalent core diameter (m)	2.73 m
Total fuel length	1.706 m
Active fuel length (m)	1.372 m
Number of control blades	57
Thermal–hydraulics	
Reactor thermal power	165 MWt
Core coolant flow rate	2.23×10^6 kg/h
Steam flow rate	3.19×10^5 kg/h
Nominal pressure in steam dome	7.19 MPa
Core inlet temperature	278.5 °C
Coolant saturation temperature	287.5 °C
Average power density	20.75 kW/L
Average linear power density[a]	7 kW/m
Total core flow area[a]	3.7 m^2
Core bypass flow area	1.12 m^2
Average core exit quality	0.143

[a] Varies depending on final optimum design developed.