A PREDICTION AND VERIFICATION METHODOLOGY FOR SMALL MODULAR REACTOR SAFEGUARDS

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Abstract

The paper proposes a prediction and verification methodology that provides an independent verification of declared small modular reactor (SMR) operating conditions, and therefore fissile material content, for safeguards purposes. This methodology consists of high-fidelity simulations of “indicating parameters” (e.g., neutron flux profiles outside the primary reactor system) coupled with off-line neutron dosimetry of retrievable verification specimens. The methodology is intended to detect erroneous reporting and/or undeclared activities during reactor operation, through discrepancies between simulated and measured indicating parameters, confirmed by offline dosimetry of verification specimens. The indicating parameters are simulated using the Monte Carlo reactor physics code SERPENT. The proposed prediction and verification methodology is demonstrated on a small modular fluoride-salt-based molten salt reactor (sm-FMSR) and a micro-sized high-temperature gas-cooled reactor (m-HTGR). The results show that the flux profiles outside of the reactor primary system, which are unique in value and trend for different SMR designs, are effective indicating parameters for reactor operating conditions. With verified operating conditions, one of the most important nuclear safeguards parameters, the fissile material content in the reactor, can be determined.

KEYWORDS: SMR, safeguards, SERPENT, reactor operating condition, fissile material content

1. Introduction

The small modular reactor (SMR) is considered to be an enabling technology for providing economical and clean energy in remote areas in Canada. One of the strategic priorities of the Canadian Nuclear Laboratories (CNL) is to develop and advance science and technology to enhance nuclear safeguards for the deployment of SMRs at remote sites. SMRs have advantages in nuclear proliferation resistance and safeguards compared to larger power reactors because most of the SMR designs feature sealed module with infrequent or no fueling. However, those advantages also present potential challenges to safeguards because the reactor internals can be accessed only during initial loading. Therefore the fuel status (i.e., fissile material content) in the module may not be easily verified with current technologies for many years of reactor operation.

The accounting for nuclear materials present in an operating reactor core as required by international safeguards is usually achieved by reactor physics simulations (i.e., neutronics and fuel depletion) using the initial core loading and reactor operating power histories as input. Licensees usually have sets of tools (including instrumentation and simulations) to declare core loading and reactor operating conditions and to produce safeguards parameters required by regulators.

The work summarized in this paper aims to provide regulatory agencies with a potential methodology that is independent of licensee reports for safeguards verification purposes. The proposed prediction and verification methodology is demonstrated on two types of SMRs: a small modular fluoride-salt based molten salt reactor (sm-FMSR) with online fueling and without control rods, and a micro-sized high-temperature gas-cooled reactor (m-HTGR) without online fueling and with control rods.

2. Overview of the Proposed Methodology

For a given reactor, the distributions of the fission power, fuel burnup, and composition in the core, as well as the neutron flux and fluence inside and around the reactor, are determined by the history of reactor operating conditions. Many of these parameters vary monotonically with the total energy production in the reactor. For example, the instantaneous neutron flux and fission power are proportionally related to the reactor power. As a result, such parameters at certain locations may serve as "indicating parameters" of the reactor operating conditions that, in turn, is used to determine the nuclear material content in reactor required for nuclear safeguards. Furthermore, consistent discrepancies between simulated and measured values for one or more of such indicating parameters may be signal of unexpected conditions in the reactor.

To be most useful, any indicating and verification parameters considered for reactor operating conditions should

- vary monotonically with reactor operating conditions,
- be reliably simulated,
- be monitored online, and
- have verification samples that are retrievable and measureable during reactor operation.

For accurate simulations, locations for the indicating parameters as close to the core centre as possible are preferred, but the accessibility consideration excludes all locations within the reactor primary coolant boundary unless special measurement channels are included as part of the reactor design.

Suitable choices for the indicating parameters are the neutron flux and fluence outside of the reactor primary system (e.g., outside of the reactor vessel), which can be monitored online using reactor instrumentation (included in the reactor design or provided by the licensee), and verified offline using neutron dosimetry technologies.

Figure 1 illustrates the scheme of the proposed prediction and verification methodology. At regular time intervals, high-fidelity simulations are performed, using the fuel loading data at the end of the previous time interval and the planned operating conditions for the current operating time interval as input, to obtain the simulated indicating parameters, namely the neutron fluxes, flux ratios, fueling intervals etc. The initial core loading is the input to the simulations of the first time interval. A subset of the verification specimens is also regularly retrieved and measured for neutron dose. The simulated fluxes are compared with the online fluxes from instrumentation, and the simulated fluence to the measured neutron dose. If discrepancies between simulated and measured are observed and persist through multiple calculation intervals, additional specimens may be retrieved and measured for verification of the declared operating conditions. Any confirmed discrepancy may indicate either erroneously declared operating conditions or undeclared activities in the reactor, which may warrant further safeguards investigations or actions.

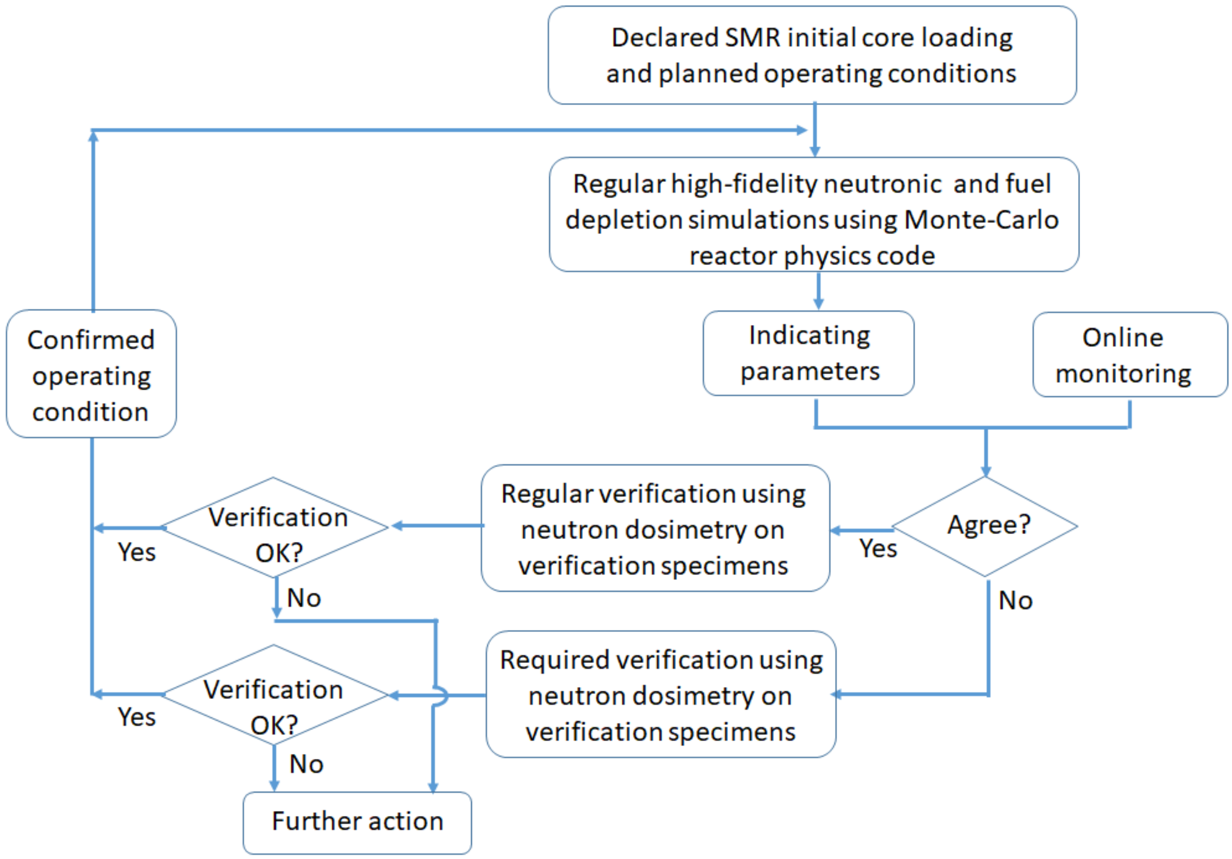

Figure 1: Scheme of the Proposed Methodology

3. Demonstration of the Proposed Methodology

Demonstrations of the proposed methodology are performed on an sm-FMSR and an m-HTGR. The flux profiles outside of reactor primary system corresponding to reactor operating conditions are simulated using SERPENT 2.1.31 [1], a multi-purpose three-dimensional continuous-energy particle transport code developed at VTT Technical Research Centre of Finland, Ltd.

3.1 Safeguards Tools for sm-FMSR

3.1.1 sm-FMSR Model

The SERPENT model of the sm-FMSR, is an integral design in the same concept family as the integral molten salt reactor IMSR®400 [2][3][4] proposed by Terrestrial Energy Inc. While this design is similar in nature with a completely sealed reactor vessel with integrated internal pumps, heat exchangers, and shutdown rods, it should be noted that detailed core design and materials are not representative of the actual IMSR design. The modeled reactor core consists of graphite moderator blocks and molten salt fuel channels arranged in a hexagonal lattice. The core is modeled in four fuel zones (shown in Figure 2): centre, inner, intermediate and outer. The fuel contains low-enrichment uranium fluoride diluted with other carrier salts [3], which constitutes both the fuel and the primary coolant. The fuel–coolant mixture is pumped in the critical core, and then through the integral heat exchangers to transfer its heat to the secondary loop.

Short-term reactivity is controlled through the intrinsic properties of the fuel salt, which has a strongly negative temperature coefficient of reactivity. This is achieved by using the secondary coolant-salt pump to alter the circulation flow rate, which changes the temperature of the fuel salt in the core and thus alters reactivity. Long-term reactivity loss due to fuel burnup is compensated by regular online fuel addition.

The SERPENT model of the sm-FMSR is illustrated in Figure 2, and the major core parameters used in the model are summarized in Table 1. A total of 500 million histories per burnup step were required in SERPENT to achieve statistical errors of less than 5% for calculated parameters in the annulus.

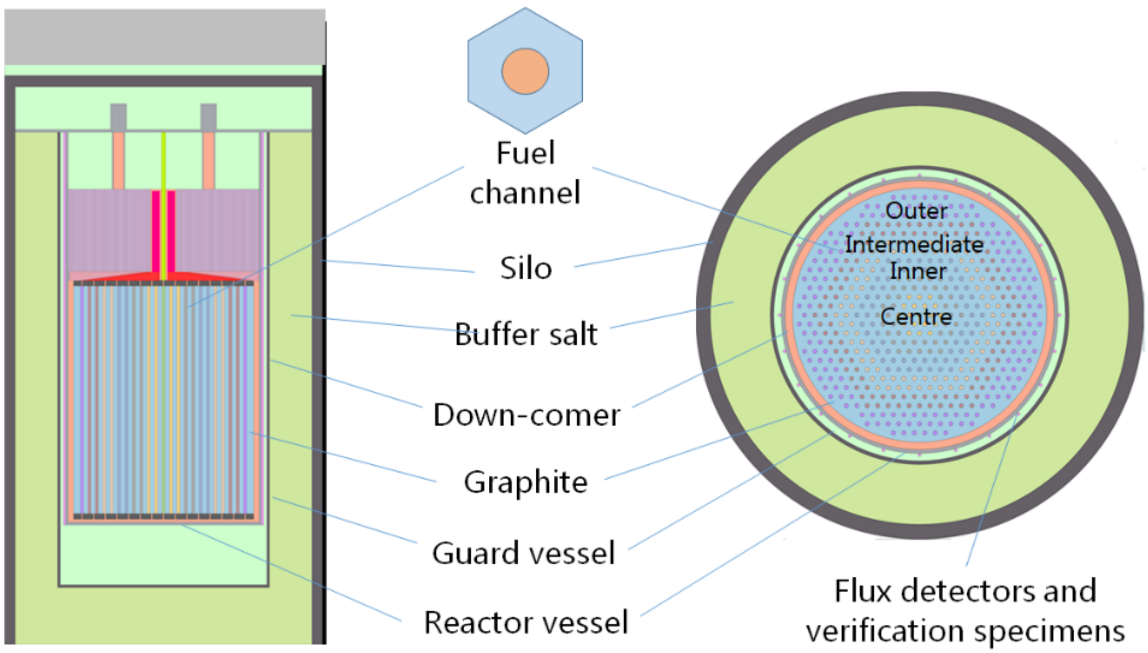

Figure 2: SERPENT Model of sm-FMSR

Table 1: Major Core Parameters for sm-FMSR

Parameter	**Value**
Core Overall Geometry	180 cm diameter 400 cm height
Moderator	Graphite
Fuel	mixture with 60–80 wt% UF4 2.6 wt% ^{235}U in initial fuel 5.5 wt% ^{235}U in fuel addition
Lattice	Hexagonal pitch of 14 cm Fuel/moderator volume ratio of 0.156
Nominal Fuel Temperature	925 K
Reactor Power	400 MW thermal
Initial Fuel Loading (mass of ^{235}U)	471 kg ^{235}U in reactor (292 kg ^{235}U in core)
Service Life of Core Unit	7 years (~2500 FPD)

3.1.2 sm-FMSR Operating Condition and Fuel Burnup

Although the development of physics methods in simulating moving molten salt fuel behaviours has drawn increasing interest and effort [5][6], there is no well-established method to simulate neutron transport and fuel depletion for moving liquid fuel systems [7]. Modeling moving liquid fuel with existing neutron transport and depletion codes is challenging because most of these are designed for analyzing solid-fueled systems. The major challenges presented by liquid-fueled system are the delay-neutron precursor drift as liquid fuel flows through and out of core and the constantly changing fuel material compositions at any core location.

For the purpose of obtaining the flux profile outside of the reactor vessel, approximations are made to simplify the calculation of the depletion of the circulating fuel so that it can be simulated using current fuel depletion codes with some corrections. The assumptions and approximations used in the fuel depletion calculation for the sm-FMSR are:

- The molten salt fuel in the sm-FMSR system is always well mixed (i.e., homogenized with uniform density and temperature). The fuel temperature is 925 K uniformly in the core.
- Fuel irradiation occurs as though the fuel salt were not moving.
- The reactivity regulation range of 25 mk [4][6] is used in this study, i.e., in the SERPENT model, the k-effective limit for fuel addition is 1.10, and fuel addition is required when k-effective decreases to 0.985.

- The reactor is operated continuously at a constant thermal power of 400 MW.

The fission power and fissile material content, as well as the flux and fluence in and around the core, are determined by the reactor operating conditions. With initial core loading and operating conditions, the SERPENT model of sm-FMSR is able to determine fuel compositions at each burnup step. At the end of each burnup step, SERPENT calculated fuel material compositions are homogenized in the whole fuel circulation system, and used as an input to the next burnup step.

Figure 3 illustrates the fissile content, fuel burnup and neutron flux outside of reactor vessel as functions of reactor thermal output (in FPDs, each equivalent to 400 MWd). The step changes in the figure indicate fuel additions. The results show that between fuel additions, the flux increases (right) as fuel burnup increases or as fissile content in the core decreases (left), until the next fuel addition. When fuel addition occurs, the amount of ^{235}U is increased so that the core can achieve criticality, and fuel burnup decreases due to the increase in total mass of heavy elements in the system. At the same time, the thermal neutron flux drops to maintain the reactor power constant. The time interval between fuel additions and the amount of fuel added vary at different reactor thermal outputs (i.e., at different fuel burnups), and they can be predicted.

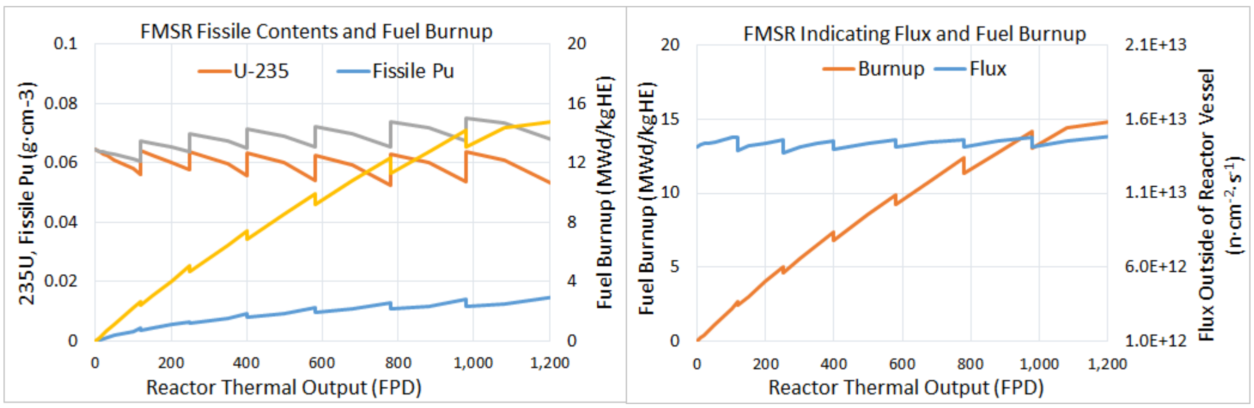

Figure 3: Fissile Contents and Flux as Functions of FPDs

As the fuel material in sm-FMSR is assumed to be homogenized and regular fuel addition is implemented to maintain the fissile material content needed for core criticality, the power and flux profiles in the core do not vary significantly (less than 5% in regional axial flux) during the service life of the reactor core unit (Figure 4).

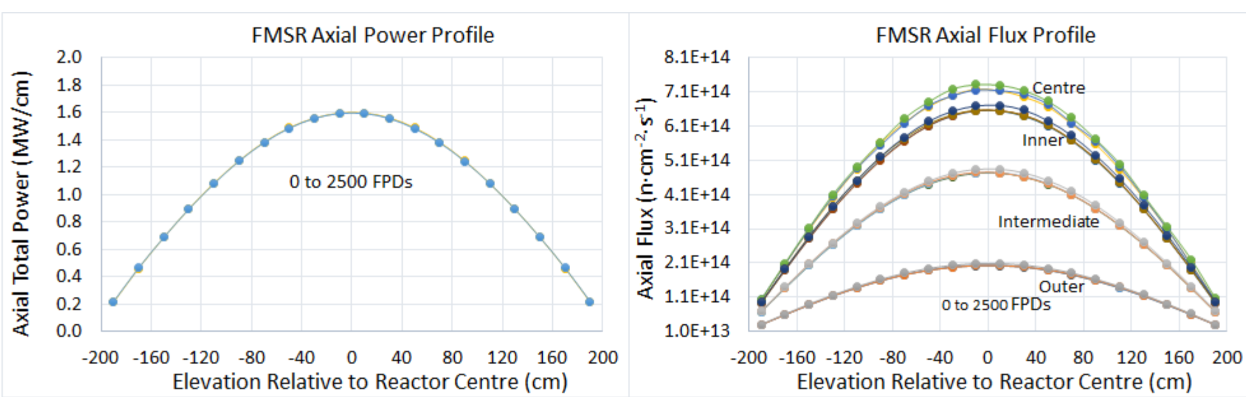

Figure 4: Power and Flux Profiles in Fuel

3.1.3 sm-FMSR Simulations and Safeguards Indicating Parameters

The axial and azimuthal flux distributions in the annulus between the reactor vessel and the guard vessel are illustrated in Figure 5, and they vary for about 5% between two fuel additions. With specified reactivity range, initial fuel loading and planned operating conditions, the flux profile between the reactor vessel and the guard vessel, as well as the fueling amount and frequency can be predicted and used as indicating parameters for operating conditions.

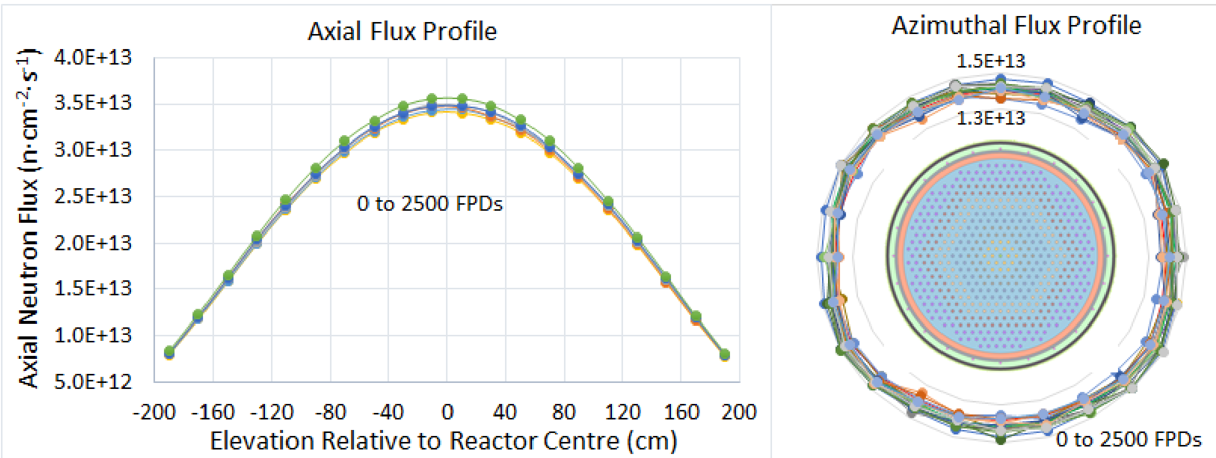

Figure 5: Axial and Azimuthal Flux Distribution in Flux Monitoring Region

3.2 Safeguards Tools for m-HTGR

3.2.1 m-HTGR Model

The SERPENT model of the m-HTGR is shown in Figure 6, and the main design data are given in Table 2. The model of the m-HTGR is based on the published design of the micro modular reactor (MMR®) proposed by Global First Power [8]. The overall design data used in this study were interpreted from referenced open literature publications. They may not be exact copies of the actual designs and may be subject to changes. The model features a reactor core made up of prismatic graphite blocks housing fuel elements and coolant channels. Each fuel element is a stack of pellet-type compacts that contain thousands of macro-fuel TRISO (TRI-structural ISOtropic) particles randomly distributed in a carbon silicide (SiC) matrix. A TRISO particle consists of a fuel kernel coated by multiple layers of ceramic and pyrolytic carbon materials. The core is surrounded by reflectors made of graphite blocks that all fit in a cylindrical core barrel and are cooled by a pressurized inert gas. The reactor pressure vessel (RPV) contains the core and all internal structures, which are inaccessible for the whole lifetime of the reactor, and the core volume is divided into six axial burnup sections (sections 1 to 6 from bottom to top).

As the m-HTGR is designed to operate at a nominal power of 15 MWt for its whole life of 20 years without fueling, fuel is initially loaded in excess, thus requiring the use of burnable poisons (BPs) and control rods (CRs) for compensation. CRs are also used for reactor control and protection (i.e., as shutdown rods).

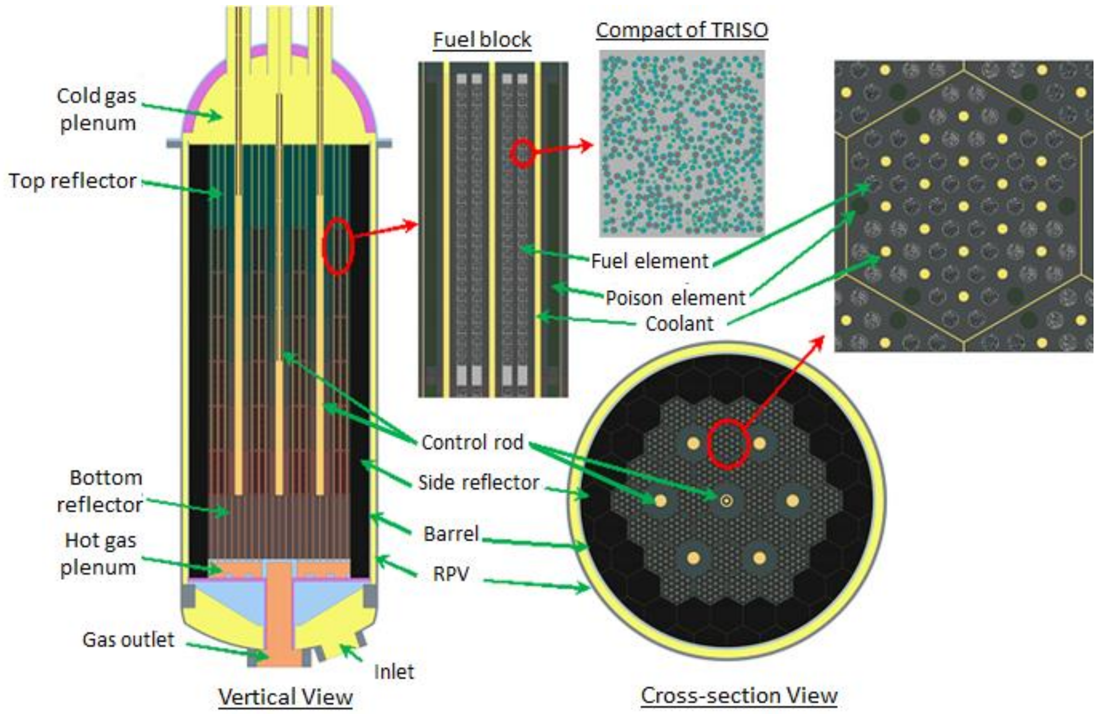

Figure 6: SERPENT Model of m-HTGR

Table 2: Major Core Parameters for m-HTGR

Parameter	**Description**
Reactor power	15 MWt
Service cycle (lifetime)	1 (20 years)
Core design - Number of blocks	Prismatic block (hexagonal, 32 cm d, 70 cm H) 222 (6 sections, each of 30 fuel blocks and 7 control blocks)
Fuel - initial loading - micro-fuel particle ➢ coating (thickness, μm) - packing ratio	UC0.5O1.5 (10.4 g·cm−3, 19.75% ^{235}U /U) 1,148 kg U (226.8 kg ^{235}U) in 180 fuel blocks TRISO (0.0932 cm D) with fuel kernel (0.05 cm D) buffer (100) / iPyC (40) / SiC (36) / oPyC (40) 40.2%
Moderator	SiC matrix and graphite blocks
Reflector	Graphite blocks
Coolant (pressure, inlet/outlet temperature)	Helium (4 MPa, 300°C / 630°C)

Burnable poisons (initial total loading)	B4C in graphite (688 g 10B in 1080 elements)
Number of control rods (absorber)	7 (B4C in SS sheath)

d, hexagon minimum diameter; D, sphere or cylinder diameter; H, height

iPyC/oPyC, inner/outer pyrolytic carbon; SS, stainless steel

3.2.2 m-HTGR operating conditions and fuel burnup

Spatial distributions of flux and power in the core, and therefore the external flux and fluence, are dependent not only on the fuel burnup distribution in the core but also on the CR positions, both evolving with reactor operation. Local fuel burnups and compositions are incremental solutions of the reactor simulation, but CR positions need to be provided as operating conditions. In the m-HTGR model, the central CR is designated for reactor control and compensation for daily fuel consumption. The six off-center CRs (6CRs) are used to supplement the compensation for the BP. For each SERPENT burnup calculation step, the central CR is set to the mid-core position, and the 6CRs are adjusted to keep the calculated k-effective close to 1.0.

Use of CRs will lead to depressions of the core flux shape, leading to overpower in the lower sections but significantly lower burnup of fuel in the upper sections, as shown in Figure 7, which illustrates the section powers (on the right) and section average burnups (on the left) at different reactor outputs (in FPDs, each equivalent to 15 MWd).

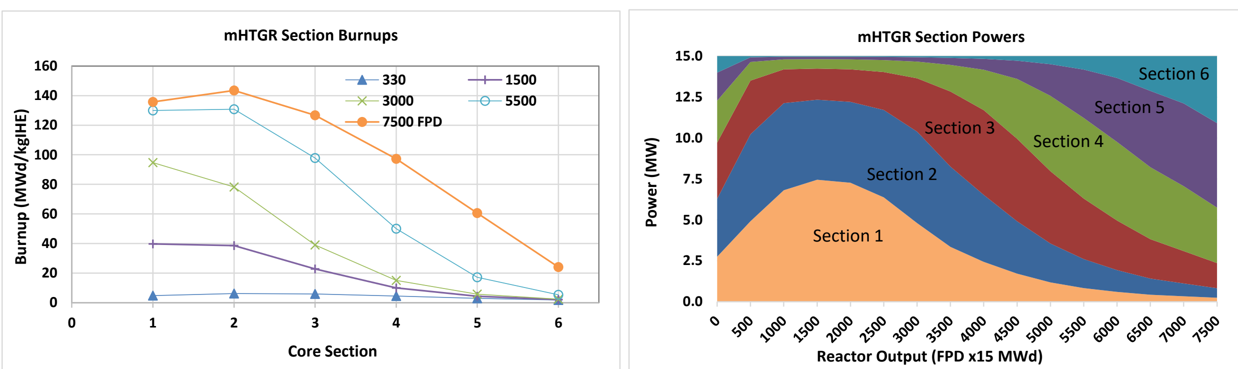

Figure 7: m-HTGR Section Power and Burnup Distributions

Initially the maximum amount of ^{10}B (Table II) is loaded in the BP elements and the 6CRs are fully out of the core. The ^{10}B BP is burning so quickly that the insertion of the 6CRs is required to offset the reactivity that is in excess of what is consumed by reactor burnup during the first 1500 FPDs. After that point, the 6CRs are gradually withdrawn to also provide compensation for burnup, until they are fully out of the core at the end of the service life.

3.2.3 m-HTGR simulation and safeguards indicating parameters

The SERPENT simulation of the m-HTGR, with the initial loading of fuel and BP and under the previously described operating conditions, is used to generate compositions of burnable materials (i.e., fuel and BP) at every incremental burnup calculation step. Figure 8 shows the fissile and BP contents in the reactor extracted from the SERPENT results and presented as a function of reactor output. As shown, the load of the only initial fissile element, ^{235}U, decreases as fuel is burning out but at a decreasing rate (from 1.28 g/MWd at the beginning to 0.97 g/MWd at the end of the reactor life) as a result of production and burnup of fissile plutonium (^{239}Pu and ^{241}Pu), which increasingly contributes to a constant fission rate (i.e., 15 MW power) of the reactor.

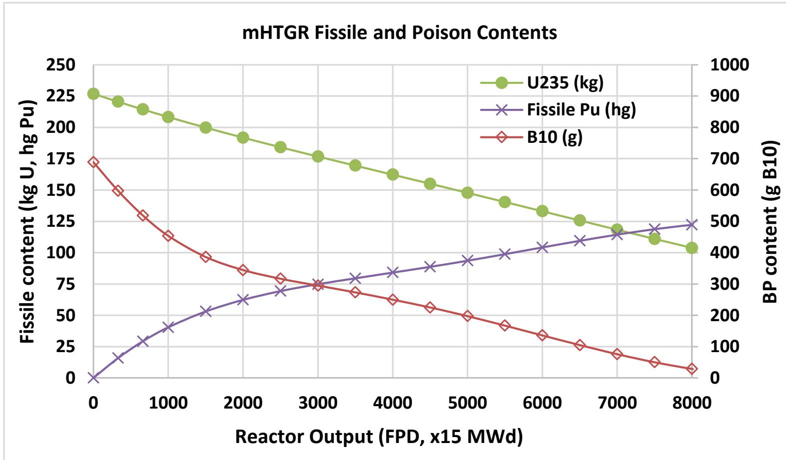

Figure 8: m-HTGR Fissile and Poison Contents Versus Reactor Output

The SERPENT simulation can also provide the indicating flux data (Figure 9, left) in the air space adjacent to the RPV wall at elevations corresponding to mid-points of the six core block sections. The external flux spectra (not shown) appear to be fairly uniform over the whole axial length of the reactor and over the operating time, making it possible to measure part of the spectrum for scaling to a total flux quantity. Absolute flux values (in n/cm2/s) require offline calibrations, but a few ratios of different pairs of measured fluxes can be used to deduce the reactor output at any time. The results shown in Figure 9 (right), indicate that, in this case, two ratios of flux indicators are sufficient to uniquely determine reactor output over the time period up to 8000 FPD. The effect of control rods makes a single ratio non-monotonically related to the reactor output.

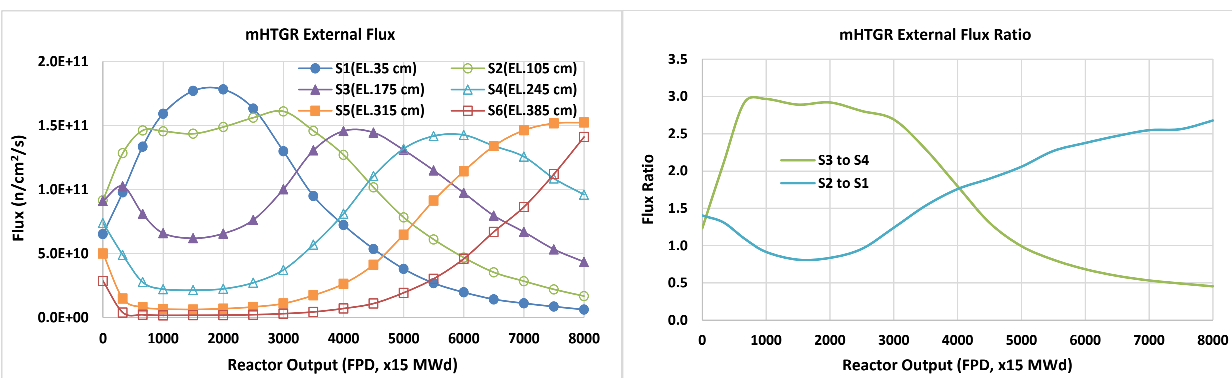

Figure 9: m-HTGR External Flux Versus Reactor Output

In addition, the trends of online measured fluxes can be used for the qualitative assessment of the operation history declared by the reactor operator. As shown, the external flux at a particular elevation tends to increase when the control rods are either approaching from above or moving out of the section upwards. Otherwise, the external flux tends to decrease as the section power is decreasing both with fuel burnup and as the flux peak shifts away. Deviations from such trends in one or more measured fluxes may serve as signals of undeclared activities.

4. Conclusions

Demonstrations of the proposed prediction and verification methodology on the sm-FMSR and m-HTGR show that flux profiles outside of the reactor primary system can be used as indicating parameters for SMR operating conditions and can be, in turn, used to determine nuclear material content of the reactor. The methodology requires comparison of the simulated indicating parameters with online flux instrumentation outside of the reactor primary system and verification of the declared conditions through offline dosimetry of the verification specimens, on a regular basis and/or upon request.

For the sm-FMSR, the flux profile between the reactor vessel and the guard vessel, as well as the fueling amount and frequency can be predicted and used as indicating parameters for reactor operating conditions. The accuracy of the simulation results may be improved by removing some of the assumptions and approximations such as incorporating temperature distributions and more realistic temperature reactivity effects, and using smaller fuel burnup steps, etc.

With varying values and trends, the fluxes and flux ratios in the air space adjacent to the RPV wall of m-HTGR can serve as effective indicating parameters for reactor operating conditions. The trends of indicating flux can also be used for the qualitative assessment of the operation history.

The methodology may be tested in the future in either a research reactor or a power reactor.

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