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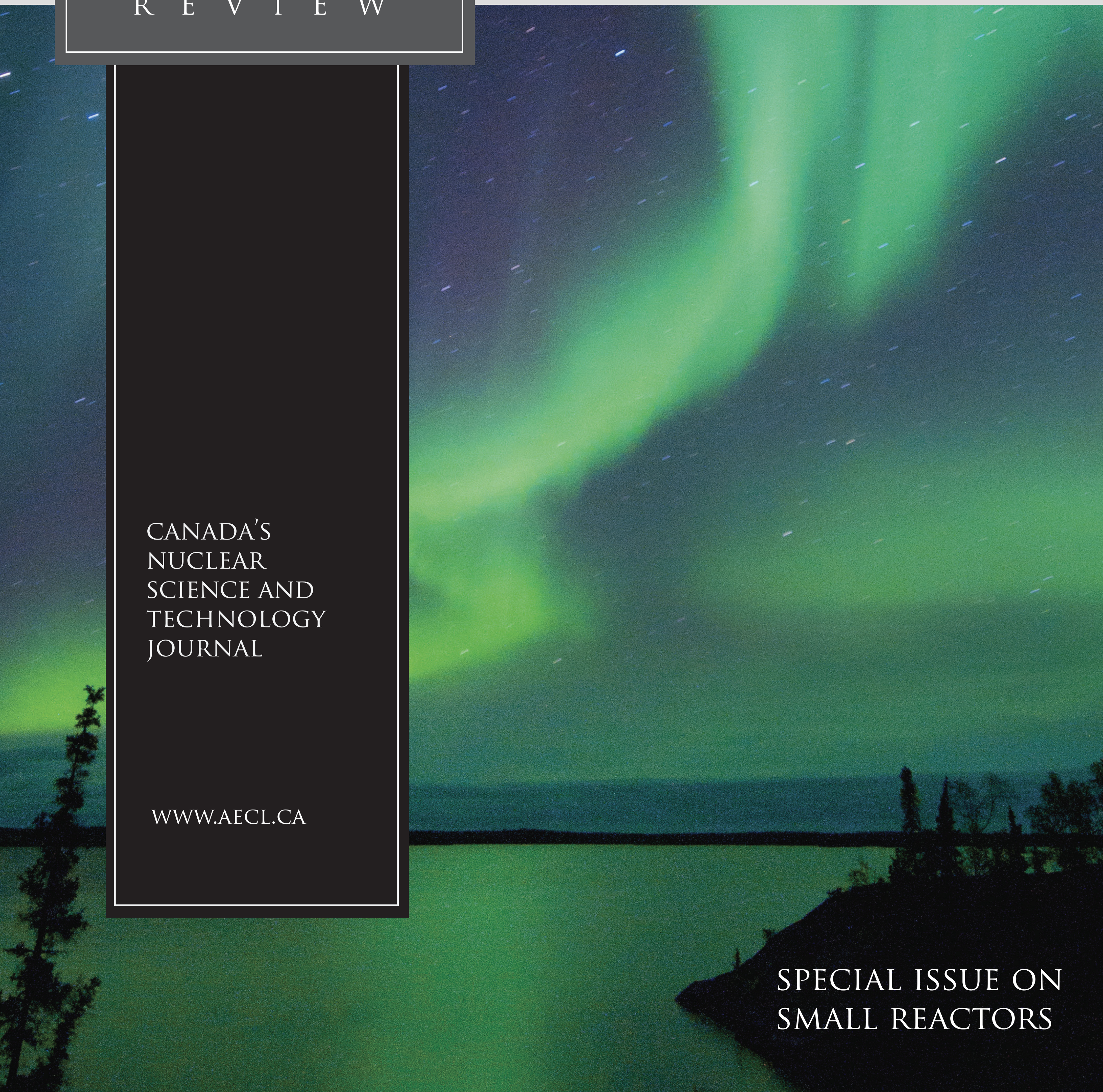

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SPECIAL ISSUE ON SMALL REACTORS

AECL Nuclear Review

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AECL
NUCLEAR
REVIEW

Volume 1, Number 2
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EDITORIAL

INVITED ARTICLES

FULL ARTICLES

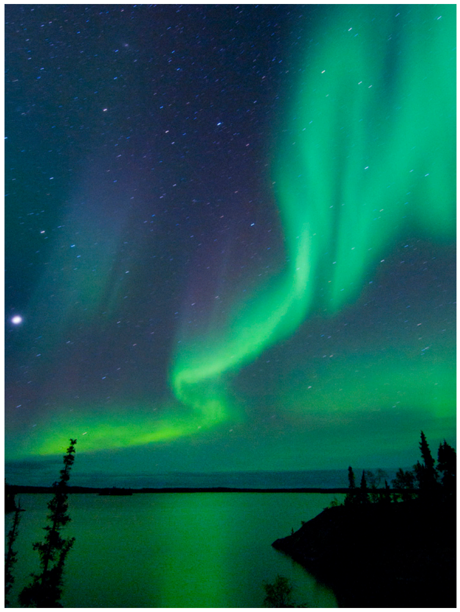
TECHNICAL NOTES

EDITORIAL

THINKING SMALL IS BIG AGAIN

C. Butler, R. Didsbury, G.L. Strati and B. Sur

Editorial Board, AECL Nuclear Review



It is a pleasure to introduce this special issue of the *AECL Nuclear Review*. Renewed interest in small reactors in Canada and worldwide makes this issue timely and relevant. Globally, there is interest in small reactors to produce electricity from nuclear power. Small reactors are once again being seen as a viable, innovative solution for clean energy power. Recently, AECL co-hosted the 2nd International Technical Meeting on Small Reactors. This technical meeting was dedicated to the achievements, capabilities, and future prospects of small reactors. Participation in the meeting exceeded all expectations, underscoring the groundswell of interest and activity, particularly in the area of small modular reactors (SMRs). Reactor designers from across the world presented more than ten small reactor concepts and designs. The small reactor technical meeting also celebrated the 50th anniversary of the Nuclear Power Demonstration (NPD) reactor, which was the first reactor to generate electricity (20 MWe) in Canada. Indeed, in today's terms, NPD would be considered the equivalent of a small power reactor.

Activity in the field of small reactors has been on the upswing in North America. In response to renewed scientific interest in the United States, the US Department of Energy announced plans in January 2012 to support the development of one or two US light-water small reactor designs, allocating $450 million over five years. The major advantages of a small modular reactor are improved safety, lower cost, and a smaller-risk venture. One significant line of development pursued in the US is in very small fast reactors of under 50 MWe; some are conceived for areas away from transmission grids and with small loads while others are designed to operate in clusters in competition with large units.

In Canada, the province of Saskatchewan is expecting its annual electricity demand to nearly double over the next 10 years. To ready itself, Saskatchewan is faced with the need to replace, rebuild and renew its entire power system. Furthermore, Saskatchewan is challenging itself to add more value to its uranium resources, making SMRs a natural fit for its energy renewal initiative.

Canada's northern provinces and territories represent approximately 0.3% of Canada's population and the same in energy usage. With a population of just over 100,000 people dispersed over 3.5 million square kilometres, the costs and logistics of energy distribution are a major issue. Energy costs are the major factor contributing to the high cost of living in the north. Current 'green' power government initiatives to lower northern carbon footprints have considered biomass, solar and wind thus far. However, solar and wind may not be practical options on their own because of the possibility of intermittent supply. Because of the harsh environment, there cannot be any supply interruptions in the northern communities. A lack of heat in winter would lead to immediate emergency evacuation. Thus, the business case for small reactors in the north becomes an option that should be seriously pursued. The many questions surrounding small reactors are discussed in the invited article *Small Modular Reactors – A Solution for Canada's North?*. In addition, the article entitled *Proliferation Resistance Considerations for Remote Small Modular Reactors* discusses the unique challenges faced by small modular reactors at the low end of energy production (on the order of 10 MWe or less), which are becoming known as very small modular reactors or VSMRs.

Earlier this year, the Canadian Nuclear Safety Commission organized a one-day workshop that focussed on technical input to policy issues of particular importance for VSMRs. Although focused on VSMRs, the outcomes of such policy discussions are likely to benefit other SMR vendors and customers in Canada. Representatives from suppliers, academia and the nuclear industry engaged in discussions on a variety of technical issues including security and emergency response for remote sites, design requirements for factories to produce modular units, and inspection and maintenance requirements for small reactors. The paper entitled *Challenges of SMR Licensing Practices* discusses the regulatory practices and challenges currently being faced in a number of European and American jurisdictions including Finland, France, the UK, Canada and the USA.

Canada has a long history of small reactor designs - both research reactors and power reactors - on which to build. The Zero Energy Experimental Pile (ZEEP) reactor was designed in 1944 and first went critical in 1945 at the Chalk River Laboratories site, achieving the first self-sustained nuclear reaction outside the United States. ZEEP produced one thermal watt of power. In 1947, National Research Experimental (NRX), the most powerful reactor in the world at the time, went critical. NRX originally had a design power rating of 10 MWth, which increased to 42 MWth by 1954. NRX was used successfully for producing radioisotopes, undertaking fuel and materials development work for CANDU reactors, and providing neutron beams for basic physics experiments.

The National Research Universal (NRU) was a landmark achievement in Canadian science and technology when it went into service in 1957. This reactor, which was the most powerful research reactor in the world at the time, provided intense beams of neutrons for research as well as irradiation facilities for producing isotopes, testing materials and developing fuels. Built to operate with natural uranium fuel at 200 MWth, the NRU operates today with Low Enriched Uranium (LEU) fuel at 135 MWth.

The Zero Energy Deuterium (ZED-2) reactor, a larger version of ZEEP, went online in 1960 to facilitate measurements on larger, more representative CANDU reactor lattices. The reactor is still operating at the Chalk River Laboratories where it is used for reactor physics research. Several papers discuss the application of state-of-the-art experimental and analysis techniques at these well-established reactors.

The WR-1 organic-cooled research reactor was built at AECL's Whiteshell site in the early 1960's. The 60 MWth WR-1 research reactor was designed and built by Canadian General Electric. The reactor was unique, in that it had vertical fuel channels, and the fuel was cooled by an organic liquid rather than water. The neutrons were moderated by cool heavy water in a large calandria vessel surrounding the fuel channels. The reactor first achieved criticality in 1965, and was used as a test reactor for the proposed organic-cooled CANDU power reactor. When that program ceased in 1972, WR-1 was used for irradiation, experimentation and heating the Whiteshell site. The WR-1 is currently in an interim decommissioning stage and will eventually return to a "green-field" state.

SLOWPOKE (an acronym for Safe Low-Power Kritical Experiment where "K" is the traditional symbol for "criticality" in the field of reactor physics) represents a passively safe, compact-core reactor technology developed by AECL in the late 1960's. This type of reactor was developed for Canadian universities and research institutions, providing a higher neutron flux than available from small commercial accelerators, while avoiding the complexity and high operating costs of existing nuclear reactors. Between 1976 and 1984, seven SLOWPOKE-2 reactors with highly enriched uranium (HEU) fuel were commissioned in six Canadian cities and in Kingston, Jamaica. Since then, three of these reactors have been decommissioned and one has been converted from HEU to LEU. The paper entitled *The Status of HEU to LEU Core Conversion Activities at the Jamaica SLOWPOKE* discusses the status of the SLOWPOKE-2 conversion in Kingston, Jamaica, as part of a US-led program to reduce the amount of HEU in use around the world in civilian reactors.

With a long history of small reactors and an understanding of the relevance of small reactor technology to today's issues, particularly in Canada's North, AECL continues to support the application of new nuclear technologies through R&D activities. The Generation IV International Forum, or GIF, is a collaborative effort by leading nuclear technology nations to develop next-generation nuclear energy systems to meet the world's future energy needs. The Super-Critical Water Reactor (SCWR) is a Generation IV reactor concept that uses supercritical water (referring to the critical point of water, not the critical mass of the nuclear fuel) as the working fluid. The paper entitled *The SUPERSAFE© Reactor: A Small Modular Pressure Tube SCWR* describes the concept of a small modular version of the Canadian SCWR.

The collection of articles in this special issue of the *AECL Nuclear Review*, and the success of the 2nd International Technical Meeting on Small Reactors, showcases the revival of interest in small reactor technology in Canada and the rest of the world. With immediate challenges in economics, licensing, technology readiness, and public acceptance, there is still some work needed to address current needs and to drive SMR technology into the 21st century. We hope that the articles and technical notes in this issue provide some input into these challenges and inspire innovative solutions to enable small reactor technology as a viable, clean, safe and economical energy source for Canadians and the world.

INVITED ARTICLE

SMALL MODULAR REACTORS – A SOLUTION FOR CANADA'S NORTH?

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I) Introduction

In recent years, there has been a renewed and growing interest in nuclear energy solutions for North America's current and future energy needs. That interest has ranged from nuclear technology solutions examining options for replacement or upgrading of current aging nuclear reactors, with more advanced versions of the same reactor design or designs that offer greater output capacity, to the development of a new generation of Small Modular Reactors (SMRs). In March 2012, the United States' White House Administration and Department of Energy reignited interest in nuclear energy and nuclear innovation with their announcement of $450 million of funds targeted at the design and development of SMRs. The announcement states, "The Obama Administration and the Energy Department are committed to an all-of-the-above energy strategy that develops every source of American energy including nuclear power…the Energy department and private industry are working to position America as the leader in advanced nuclear energy technology and manufacturing" [1]. Because of this announcement, there has been resurgence in research interest, both in North America and in the developed world, towards the development of small nuclear reactors.

The desire for SMRs stems from two sources: minimization of capital requirements and the ability to achieve manufacturing scale. Minimization of capital requirements is of particular interest to governments because building a large-scale nuclear facility requires significant capital to finance the project. With tenuous economic sentiments and instability around the globe, the acquisition of capital, particularly large sums traditionally needed for large-scale nuclear facilities, can be difficult to obtain. In addition, and a somewhat compounding factor, there is the potential for escalation of the "all-in cost"[1] [2], which would result in an expanding overall cost for the large-scale nuclear project. SMR builds will be able to obtain funding more readily, and they may be able to minimize cost overruns. To support this belief, scientific proponents must demonstrate that SMR technologies can be both cost competitive with other energy sources and be able to control overall project costs. In addition to potential difficulties in acquiring capital, there is a need for researchers to demonstrate that SMR technologies can achieve manufacturing scalability. Scalability refers to the process of creating a technology that can be manufactured in a manner similar to a highly efficient, or highly scaled, production line. To achieve this scalability, the concept of modularity is required so that SMRs can be "made in factories and shipped to sites – to reduce costs" [1]. The modularity of an SMR offers the benefit that one module requires a smaller capital investment and could be producing electricity while a second module is being built. Currently, the knowledge required to deliver SMRs, a highly complex technology, in a true modular format, is still in its infancy. For SMRs to be viewed as a viable cost-effective solution, innovation in this technology will be required. It is conceivable that this innovation push will generate another nuclear renaissance.

[1] All-in-costs, shorthand for all included costs, include "the spread, commission, interest payments, and any other fees that may result from transactions. For example, some banks may quote an all-in cost of a loan, expressed as a percentage of the loans face value." [13]

The opportunity to explore a future nuclear renaissance via SMR technology is particularly suited to Canada. For example, the Canadian market is unique in that there is a need for electricity and heating independence in remote and sparsely populated Northern communities, a requirement only a handful of nations share. It has been proposed that nuclear technologies, namely SMRs, could be a potential solution to resolve the energy needs in the North. The North is considered the northernmost region of Canada and contains the Northwest Territories, Yukon Territory and Nunavut. The Canadian arctic[2], with its geographical nuances, is also present within this region. The geographical area of Nunavut resides mostly within this arctic region. Population distribution within the North clusters in several medium-sized cities, small mining towns, ice road communities (communities that are only accessible by ice roads in the winter), and fly-in communities.

The North presents unique challenges in providing both electricity and heating to its citizens due to population size and distribution, as well as geographic isolation and the potential for exorbitant infrastructure costs. The need for future energy solutions in this region raises the question, *'Is the SMR a viable solution for powering Canada's North?'* This paper attempts to provide an answer to this question. In addition, several market readiness questions are raised in this paper and are examined through the lens of Pankaj Ghemawat's CAGE framework [3], whereby the Cultural, Administrative, Geographical and Economic distances regarding SMR entry into the North will be examined. The response to these questions for the larger SMR community need to be considered, but will require further research and analysis to be fully explored and resolved, and are therefore not addressed in this article.

II) Reference Community, Northwest Territories

To assist in developing a deeper understanding of the target market for improved, reliable power, and a point of entry for SMRs, a reference community was developed for this article using data provided by the Government of the Northwest Territories. The Northwest Territories' government has collected significant amounts of data regarding energy usage in each of its communities. The data highlights the power needs of an ice road access community, located in the Northwest Territories, that runs on diesel for its electricity and heating needs. Ice-road communities are only accessible by ice roads in the winter, and range in population from approximately 100 to 2000 residents; the majority of the communities in the Northwest Territories are ice road communities [4]. For the purposes of this paper, and to address the needs in arctic regions, it is assumed that the analysis of this reference community can be extrapolated to similar Northern communities.

To develop the needs of the reference community and understand the market, the following has been assumed true for the reference community: (A) energy demand is within a fixed range, (B) heating is sourced electrically, and (C) the current socio-economic conditions within the reference community are at average territory levels. The validity of the assumptions made for the reference community is explored below .

(A) Electricity demands range from 0.5 – 1.5 MW, with most diesel generators sourced at approximately 1.0 MW, as illustrated in Figure 1[3]. While the data does show that there are three source communities that are above this predetermined range, these data points have been treated as outliers because their values are greater than one standard deviation (σ) from the mean (μ) (μ=0.5 MW, σ=0.9 MW).

(B) Heating needs in the NWT, and more generally in the North, are usually electrically-sourced because of the prohibitively expensive costs associated with infrastructure (e.g., piping to transmit waste heat, power lines for transmission). These infrastructure costs result in constraints regarding heating and electricity in communities similar to the reference community. For example, only those closest to the diesel generator will have access to waste heat (for heating a dwelling) because of infrastructure costs associated with piping the heat to dwellings located farther from the generator site. The additional infrastructure costs required to distribute waste heat beyond the region

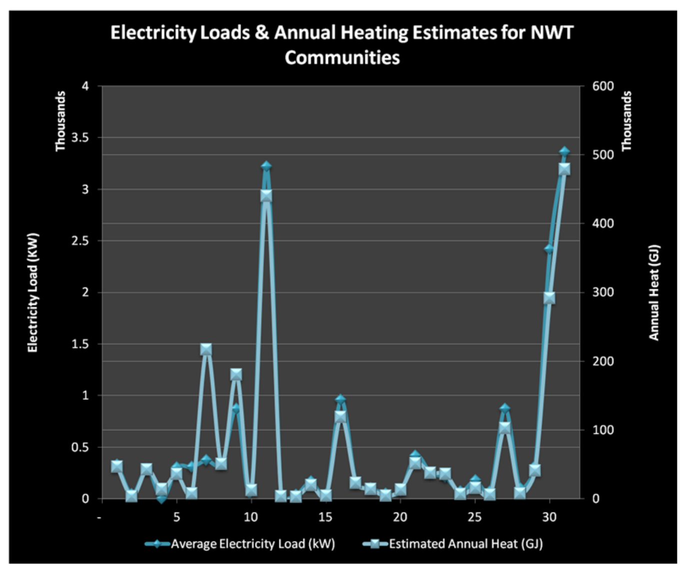

FIGURE 1
Comparison of average electricity and heating need in Northwest Territories used to create the reference community. Note that the ice-road towns used in the data set are numerically listed on the x-axis to provide anonymity to individual locations.

[2] The southernmost limit of arctic regions has commonly been placed at latitude 66 degrees, 32 minutes North. This latitude is considered the Arctic circle where the sun will not set on the date of the summer solstice and will not rise on the date of the winter solstice. In addition, this region is also north of the tree line, and is located where the average daily summer temperature does not exceed 10°C [14].
[3] The Government of the Northwest Territories – Environment and Natural Resources provided the data used to model the reference community.

proximal to the heat source can negate the overall net cost savings of having the waste heat piped in.

(C) To determine the socio-economic environment for the reference community, an Auditor General's report regarding education and employment opportunities within the NWT [5] was sourced. The report states: "Statistics Canada data from 2006–07 shows that while the graduation rate for secondary schools across Canada was 71 percent, in the Northwest Territories it was 55 percent (compared with 65 percent in Alberta, 68 percent in Yukon, and 30 percent in Nunavut)… People in smaller, remote communities may not want to move away to work, and there are fewer opportunities for long-term, full-time employment where they live." [5]. In addition, the socio-economic conditions within the NWT are also affected by the following: "low literacy rates; the shortage of housing, which can result in overcrowded houses with limited space for study and sleep; the continued reliance on traditional economies (especially in smaller communities), which can prevent students from attending the prescribed number of school days in a year; and the effects of substance abuse and foetal alcohol syndrome, which can impact attendance and student performance. Finally, the legacy of the residential school system has had, and continues to have, a negative impact on support for formal education." [5]. Therefore, it has been assumed that the relative socio-economic conditions within the reference community reflect the average education and income levels in the territory.

With the reference community assumptions established, the unique challenges of this community related to accessibility of energy are discussed in the following sections. While it is true that heating by fuel oil is expensive for the reference community, with fuel oil at μ=\$1.53/L, σ=\$0.092/L[4] , it is also true that any SMR, or other energy technology, must be cost competitive. The SMR technology must be able to achieve some degree of manufacturing scale and modularity to minimize costs [1]. In addition, there is the potential for the current socio-economic conditions in the reference community to create potential barriers to entry for complex technologies, as the technical skills to support these technologies can be unique and hard to source. For example, if a biomass plant required a highly skilled tradesperson such as a steam engineer, and this skill set was not available within the community, the cost of this expertise would need to be included in the cost of installing and operating the plant. If the cost was deemed too expensive, the plant may not be built, and less technically proficient energy options, such as a new diesel generator, would be pursued.

III) The 'Great White' Northern Market – Small & Niche

Canada's North can be considered a small market: the region accounts for only 0.3% of Canada's population, estimated at around 100,000 persons, and creates a comparable energy demand [6]. The size of this market has the potential to be quite small. As illustrated in the reference community, the average community size ranges from 100 to 2000 residents and has a power demand between 0.5 MW and 1.5 MW. Current SMR designs may be 'over powered' for the North; proposals range from as small as 50 MW for a CANDU-based design [7] to 25 to 335 MW for well-advanced designs based on non-Canadian reactors [8]. Even at their smallest proposed range of 25 MW, current SMRs exceed the demand of a typical community similar to the ice road community. A much smaller reactor, closer to the demand of approximately 1.0 MW, would need to be developed to meet the current energy requirements for these communities. Development of such a reactor would also need to take into account the specific maintenance and reliability concerns of these communities.

Canada's Northern market can be considered a niche market not only because of the market's small size, but also because of the highly specific customer needs of the customer base. As established in the reference community, many homes in this region are heated with electricity. The winter temperatures routinely drop below -20°C, and the need for uninterrupted energy supply during winter conditions is of paramount importance: if there is an interruption in power, then there is no way to heat the dwellings thereby making an emergency evacuation necessary.[5] This dependence creates unique energy reliability requirements. In addition, it also creates an environment where any power-production maintenance activities that could interrupt the power supply can only occur during warmer conditions. SMR technology is a good candidate for both providing a reliable heating source and energy independence to residents in the North because there would not be any reliance on transportation of generator fuel to the community, and there are no concerns regarding the need for additional fuel in winter months when access to the communities is limited or impossible.

IV) Economic Considerations

Economic conditions in the North indicate that energy costs are the major component of both the cost of operating commercial sites, such as mines, and the cost of residential living, as either heating or electricity, or both [4]. The majority of the electricity used in the North is hydroelectric [6]. However, distribution of power from large-scale energy projects is very limited because of the high cost of infrastructure and the unique design constraints encountered in the region (e.g., permafrost, extreme temperatures). As a result, most isolated communities, such as the reference community, rely on diesel generators as their source of either heat and/or electricity. In these communities, there is a significant amount of diesel

[4] These prices are used as estimates by the Government of the Northwest Territories – Environment and Natural Resources and have been incorporated into this paper as a pricing assumption.
[5] The Government of the Northwest Territories – Environment and Natural Resources stated that the necessary response to this emergency scenario would be an evacuation.

generator usage, which results in an increased carbon footprint for the area.

To reduce the overall carbon footprint of these communities, the NWT government has put forward initiatives that support green power projects [4]. Biomass, a carbon neutral solution, is currently the preferred option, due in part to the slow development of solar and wind technology (both of which are still in the testing phases for use in extreme environments). Nuclear is also considered a carbon neutral solution. An SMR would align with the green power initiatives that Northern governments are considering by offering a solution that reduces the overall carbon footprint of these communities.

V) Market Entry – SMRs and Their Fit in Northern Canada

Entry to the Northern market for SMR technology may have some barriers, or distances, that need to be examined. Using Ghemawat's CAGE framework [3], the Cultural, Administrative/political, Geographic, and Economic challenges relating to the Northern market and reference community will be discussed in this section. Questions stemming from this analysis will be posed; however, these questions will require further discussion and analysis. The resolution of these questions is not addressed in this paper.

When considering cultural distances, one examines the different languages, ethnicities, and values, norms and dispositions existing in the region. Examination of the North illustrates a vast and vibrant diversity. The demographics within the Northwest Territories, on which the reference community is based, illustrate that 45.6% of the population is of Aboriginal descent [9]. In addition, "The territory recognizes 11 official languages, including English and French, with Aboriginal languages spoken more often in smaller, remote communities." [5]. Therefore, the inclusion of Aboriginal cultural requirements and accessibility needs into design considerations for SMRs should be considered prior to entry into this market. In addition to the inclusion of cultural and accessibility requirements, there may be a need to source the highly specialized skills required for the technology locally from communities such as the reference community. The Canadian government has been supportive of Aboriginal development and has introduced measures of improvement, including "measures to improve regulatory systems across the North, to address infrastructure needs including housing, to create the Canadian Northern Economic Development Agency, and to support improvement in indigenous skills and employment." [10]. Therefore, prior to entry into this market, the question that needs to be explored is '*What is the receptiveness of Aboriginal persons and other local persons to the introduction of SMRs into their environments and what can be offered to help realise any opportunities or bridge any concerns?'.*

Examining administrative/political distances relative to the Northern market and reference community requires analysis of the current government's directionality and strategy, and political receptiveness of government policies. The National Energy Board has projected that energy demand will to continue to grow in all regions of Canada, including in Northern regions [11]. The rise in demand of energy could be considered an indicator of a projected growth in the Canadian economy. When energy demand and the economy grow, the relative amount of Green House Gas (GHG) output should be projected to grow. Because the Northern and arctic regions of Canada are more sensitive to the effects of climate change, the Canadian government is supportive of green technologies to help reduce and develop a low carbon economy for the North [10]. To capitalize on the current direction of the NWT government, the SMR community needs to address the following for the Canadian North and the reference community, *'Are SMRs a strategic fit into the plan for Northern development opportunities, specifically as a solution to create a low carbon economy?'.* In addition to needing to demonstrate that SMRs are a low carbon solution for Northern Canada, Canadian sovereignty requirements must also be considered. With the Canadian Arctic rapidly experiencing the effects of climate change, the Northern landscape is becoming the focus of many nations, particularly with respect to increased access by ship because of melting polar ice. "Over time, increased access to the Arctic will bring more traffic and people to the region. While mostly positive, this access may also contribute to an increase in environmental threats, search and rescue incidents, civil emergencies and potential illegal activities." [10] Considering that the introduction of SMR technologies into the North could introduce non-proliferation concerns, the question arises, *'Does having SMRs in the North strengthen or hinder sovereignty and security needs?'.*

The geographic and economic distances are inter-related for the Northern market. Geographically, the North has high physical distance, the region is remote, and there are weak transportation links (reference community is an ice-road access community and others are fly-in only communities). These geographic distances directly influence the economic distances present in the market. For example, the costs of materials and food in Northern Canada are much greater, at a minimum 28% more than the rest of Canada [12]. This price increase is because of both the geographic distance that is covered to deliver the materials and food, and the transportation methods used to deliver the materials and food (e.g., ice-road trucks, airplanes). In addition, the relative market size is limited to the relative income of the market inhabitants. Therefore, unless the geographic distance can be reduced between inhabitants in the North, the economic constraints on the market will remain. A question that needs to be considered when examining the Northern market is: *'Is the North the correct market for SMR development in Canada, or should another market opportunity be considered either as a substitute or as a complementary solution?'.*

VI) Conclusion

There is resurgence in the interest of nuclear technologies as part of a future energy strategy. In particular, SMRs could be an energy solution for remote areas of Canada, such as Northern Canada. Technical and socio-economic challenges, however, will need to be overcome to make SMRs a viable technology for these regions. The analysis of the reference community, Northwest Territories, which represents a typical ice-road access community, identifies some of the technical and socio-economic challenges that exist in this niche market for SMRs, a market where this technology could potentially act as a heating and electricity energy source.

As illustrated by the data, this market is very small and has small power demands. Development of the design of an SMR with a power output closer to the needs of Canada's north will be needed to make this technology truly viable.

In addition to improvements in SMRs to meet the Northern energy needs, the relative socio-economic environment in these communities needs to be developed. This could be achieved in part with specific skill training to enhance independence of the community on the technology as well as implementation of the necessary infrastructure.

At their current design and sizing focus, SMRs are a sub-optimal choice for most communities in Northern Canada. To meet the needs of this small niche market, significant innovation in the technological design of SMRs and their relative power transmission methods will be required, as well as development of robust responses to the posed market entry questions. If these technological and socio-economic challenges can be overcome, the benefit to the Northern ice road communities could be significant, providing a reliable, independent heating and electricity solution that minimizes the carbon footprint and provides opportunities for technical skill enhancement of the population.

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INVITED ARTICLE

PROLIFERATION RESISTANCE CONSIDERATIONS FOR REMOTE SMALL MODULAR REACTORS

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Abstract

Remotely located Small Modular Reactors at the low end of energy production (on the order of 10 MWe, referenced here as Very Small Modular Reactors or VSMRs) present unique proliferation resistance advantages and challenges. Addressing these challenges in the most efficient manner may not only be desirable, but necessary, for development of this technology. Incorporation of safeguards considerations early in the design process (Safeguards by Design) along with safety, security, economics and other key drivers, is of importance. This approach raises the possibility of increased monitoring of operational data for verification purposes (Operational Transparency), which may become a useful aspect of the safeguards approach for such systems.

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1. Introduction

Small Modular Reactors (SMRs) are an enabling technology that supports all four pillars of the Government of Canada's four-tiered "Northern Strategy": sovereignty, environment, economic development, and self-governance [1].

Increased proliferation resistance is a goal of advanced nuclear reactor design; for example, it is one of the technology goals of the Generation IV International Forum (GIF) [2]. The ability of SMRs to meet these broad expectations of advanced reactor design is clearer in some cases than in others, and proliferation resistance is one case where either uncertainty exists due to the early stage of development, or a known gap exists that will require a technical solution.

As an example of this uncertainty, SMRs that incorporate a sealed core will present a high level of resistance to the threat of both technology misuse and material diversion. On the other hand, a core that is sealed for many years (possibly decades) presents a potential challenge to regular physical inventory verification, as well as verification of the absence of misuse.

This paper provides an assessment of proliferation-resistance issues facing SMRs, particularly as they relate to remote deployment in significant numbers throughout the Canadian arctic. The assessment is qualitative in nature and is drawn from expert judgment; as such it is likely not comprehensive.

The scale of such units is expected to be on the order of 10 MWe (30 MWth), in accordance with typical heating and electricity supply needs of remote mining, military, and municipal operations. In this aspect, Canadian deployment of SMRs in the arctic will differ from most other jurisdictions in the world (a notable exception being northern Russia [3]), where unit powers at least an order of magnitude greater make more sense. In this paper, the SMRs in this smaller size range will be referred to as Very Small Modular Reactors (VSMRs).

The range of design possibilities for VSMRs is considerable, ranging from evolutionary LWR technology to more advanced fuel cycles. Cores may be factory-fuelled and shipped to site, or fully assembled on site, and refueling may take place on a regular basis on site, or in a batch replacement process (often referred to as the "nuclear battery" concept). Operational modes may also differ widely. These design and operational parameters will all affect the assessment of proliferation resistance, and ultimately the safeguards approach implemented by the IAEA.

2. Proliferation resistance and IAEA safeguards

"Proliferation resistance" is defined as "that characteristic of a Nuclear Energy System that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices" [4]. Proliferation resistance has both intrinsic components, such as the attractiveness of the nuclear material for diversion or the amenability of its operation to undetected and undeclared uses, and extrinsic components, such as the amenability of its design to inspection and safeguards implementations.

The International Atomic Energy Agency (IAEA) in Vienna provides international verification of nuclear activities in a Host State, through the implementation of nuclear safeguards that include inspections to verify facility design and nuclear material inventory, and also instrumentation and other measures that provide "continuity of knowledge" between inspections. The IAEA safeguards system is viewed as a key instrument of non-proliferation, and cooperation between stakeholders can make implementation of this safeguards system more cost-effective and minimize its impacts on operations.

3. Impact of VSMRs on proliferation resistance and the implementation of safeguards

VSMRs have the following characteristics that potentially impact the implementation of international safeguards:

• *Low thermal signature*. Having a thermal footprint similar to other energy technologies deployed in remote northern locations implies that it will be challenging to use satellite or other forms of remote sensing to verify operation. However, indirect indicators such as "lights on" or the operation of powered equipment in the absence of alternative power sources may be useful. Also useful will be reliable sealing in the case of long-life cores.

• *Remote location with limited access*. Difficult access to the facility sites themselves provides a measure of proliferation resistance by increasing the cost and difficulty of diversion or covert misuse. On the other hand the difficulty of access also applies to safeguards inspectors, increasing the cost and reducing the potential for unannounced inspections. This could be mitigated by reliable year-around off-site monitoring of redundant authenticated sensors (see Section 4).

• *Number of reactor sites*. VSMRs lend themselves to distributed installation, implying that many sites over a large geographic area could require inspection, all with the issues of difficult access and inspection described here.

• *Long-life reactor core, possibly sealed*. Reduced core access and reduced refuelling frequency makes misuse of operation and diversion of spent fuel (respectively) much more difficult. However, this will need to be reconciled with the current IAEA practice of annual physical inventory of each reactor core, performed when access to the core is possible. For sealed cores, reliable monitoring of authenticated sensor data may provide "virtual" access (see Section 4).

• *Advanced fuel cycle*. In some cases the nature of the fuel cycle will be unfamiliar to the IAEA, and require significant analysis to understand the most efficient and effective safeguards approach. This presents an opportunity for safeguards experts to collaborate on the design, the importance of which is discussed in Section 4. A fuel cycle that represents an increase in proliferation resistance may allow a less stringent safeguards approach, providing that IAEA safeguards objectives are still met.

• *Enrichment*. Reactors designed to minimize size (and thus transportation costs) and maximize time between refuelling will require significant levels of enrichment, typically encountered in research reactors (e.g. up to 20% LEU). Widespread popularity of such reactors would therefore increase the amount of enriched uranium on the planet, and might provide incentive for new enrichment facilities, which are "dual-use" technology. When designs require uranium enriched above 20%, the issue becomes even more politically challenging to address.

• *Excess reactivity*. A reactor designed for low refuelling frequency would require significant excess reactivity and burnable absorbers. Such a system might tolerate target irradiation without significantly affecting key operational parameters, and from an observer's viewpoint, neutronic management with burnable absorbers would look similar to neutronic management with target material (while providing a potential diversion route). Verifying that there is no possibility of access for target insertion or removal could be a design requirement. Potentially, these concerns could be mitigated with a pre-operation design verification activity by the IAEA coupled with reliable sealing and surveillance measures – including the potential for in-depth monitoring of operational data, as discussed in Section 4.

• *Fuel element size*. Depending on design, core length could be significantly smaller than conventional designs, leading to shorter fuel elements with two opposing impacts on diversion difficulty: small size tends to render concealment easier, while increasing the number of elements that must be successfully diverted to achieve a Significant Quantity of material. Reduced refueling frequencies and sealed cores, as well as comprehensive containment and surveillance of spent fuel handling, can mitigate some of these issues.

• *Spent fuel storage geometry*. Should spent fuel be stored at the site, smaller elements would most likely need to be stored vertically for cooling purposes, with a strong economic incentive to stack fuel and reduce the storage footprint. This geometry potentially challenges the current safeguards inspection activities due to lack of direct-line visibility from above. Adding IAEA safeguards requirements to the design requirements can potentially lead to an alternate optimal design, such as the use of IAEA-sealed spent fuel baskets.

• *Fissile inventory*. VSMR core loads, whether in sealed-core designs or otherwise, will be small compared to conventional power reactors, providing an additional barrier to diversion or misuse (tempered somewhat by the potentially significant quantity of sites, and therefore total inventory under similar conditions in one state).

• *Environmental considerations*. It is typically difficult to maintain reliability of communications and power supply at remote northern sites. This infrastructure would necessarily need improvement in support of a distributed network of VSMRs, but the situation will continue to present a diversion pathway opportunity that depends on loss of IAEA instrumentation or time needed to affect repairs. Presumably the use of VSMRs would only be considered if reliability issues could be substantially mitigated.

• *Contingency planning*. Contingency planning can take into account the fact that natural hindrances would impact both adversary and normal operations in similar ways. Contingency planning has an opportunity to address the health and safety, security and safeguards issues as an interrelated set. Including safeguards (and other) considerations in the emergency response planning can avoid some of the issues and reduce the impact of others.

4. The Importance of Safeguards by Design

"Safeguards by Design" is a concept that encourages the earliest possible inclusion of safeguards considerations in the design process of Nuclear Energy Systems, in order to achieve greater efficiency and effectiveness of the safeguards implementation [5, 6]. Safeguards by Design was introduced by the IAEA as a way of achieving safeguards goals with limited resources, particularly in light of a possible expansion of the global nuclear industry in terms of both number and types of reactor installations [7].

It is conceivable that Safeguards by Design will not just be desirable, but necessary, in the development of VSMR technology, given the considerations raised in the previous section. In other words, just as it becomes increasingly important to take into account security, environmental, economic, and social acceptance considerations into the design process, so does it become important to factor in proliferation resistance. This doesn't just mean aspects that aid the application of safeguards themselves, but aspects that make the technology less attractive to proliferation interests (i.e., extrinsic as well as intrinsic components, as discussed in Section 2).

For example, the safeguards implementation in remote VSMRs could take greater advantage of the operator's own data, a concept known as Operational Transparency [8]. With an authenticated data stream extracted from the operational systems (e.g. power levels, temperatures, control system configuration, etc.), monitored remotely with trending and analysis software, the IAEA might directly monitor the operation of multiple sites, potentially circumventing many of the challenges listed above that are related to operational misuse. This is an area of current development that builds upon the IAEA's growing experience with remote process monitoring, which has direct applications to fuel processing installations. With appropriately authenticated data and validated simulation software, the IAEA can gain an unprecedented level of "virtual access" to an SMR core. The IAEA has a significant amount of experience with Remote Monitoring, and has developed sufficient confidence in the effectiveness and efficiency gains presented by the technology to move towards a broader implementation of the concept of Remote Safeguards Inspections [9 - 11].

5. Conclusion

Very Small Modular Reactors (VSMRs) offer a number of advantages to remote development such as in Canada's north, and proliferation resistance will need to be considered at an early stage of the design process (Safeguards by Design). Certain features of VSMRs offer potential advantages to safeguards implementation, while others present challenges that may not have been widely appreciated to date in the discussion of SMR implications. Consideration of security and safety needs in an integral approach along with safeguards (the "3S" approach), particularly in contingency planning, will lead to efficiencies. Operational Transparency allows high-level remote monitoring that presents a significant barrier to misuse scenarios, and can improve confidence in the authorized usage. Reactor designs with a significant level of intrinsic proliferation resistance are desired, and sealed, long-life cores present a number of advantages in this respect.

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FULL ARTICLE

THE SUPERSAFE© REACTOR: A SMALL MODULAR PRESSURE TUBE SCWR

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Abstract

The SUPERSAFE© Reactor (SSR) is proposed as a small modular version of the Canadian supercritical water-cooled reactor (SCWR). The SCWR is Canada's primary contribution to the Generation-IV (GEN-IV) International Forum's (GIF) research and development effort toward the study and eventual deployment of advanced nuclear energy systems. All GEN-IV concepts, including the SCWR, have enhanced safety, improved economics, improved sustainability and enhanced security compared to contemporary reactors. The SUPERSAFE© Reactor (SSR) concept incorporates the enhanced features of the SCWR in a smaller core which could be deployed in areas with sparsely distributed population bases where it is impractical to have a full scale SCWR or large centralized energy grid. An overview of the SSR concept is presented in this work.

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1. Introduction

The supercritical water-cooled reactor (SCWR) is one of six concepts selected by the Generation-IV International Forum (GIF) for study within its collaborative research and development program [1]. All GEN-IV concepts are intended to have enhanced safety, improved economics, improved sustainability and enhanced security compared to contemporary reactors. Safety enhancements in the SCWR include passive safety through a negative power coefficient and coefficients of reactivity, as well as passive decay heat removal through the moderator. The SCWR achieves improvements in both economics and sustainability through enhanced thermal efficiency. Using a thermodynamic cycle similar to that used in supercritical fossil plants enables a dramatic increase in thermal efficiency compared to conventional nuclear power plants, from approximately 33% to as much as 48% [2]. Enhanced security in the SCWR is achieved through the use of fuel cycles with increased intrinsic proliferation resistance and appropriate safeguards.

The primary application of the SCWR is electricity production. Potential supplementary products of the SCWR include process heat, hydrogen, industrial isotopes and drinking water. A preliminary concept for a small modular version of the SCWR, called the SUPERSAFE© Reactor (SSR) was proposed recently by Duffey et al. [3]. While the full-scale SCWR is intended to provide power to meet the high energy demands of a large population base (which may also include industrial energy needs), the SSR could be deployed to provide electricity to sparsely populated regions where it is impractical to draw electricity from a large centralized energy grid. The SSR could also be deployed near regions rich in oil sands deposits, where it could provide electricity and co-generated process heat and hydrogen for oil extraction and refining. The work presented in this paper expands upon the initial SSR concept and includes an overview of the core design, safety, fuel cycle options and proliferation resistance.

2. Reactor Core

Cross sectional views of the SSR core concept are shown schematically in Figure 1. The various panels in Figure 1 show the coolant flow and location of various components such as the coolant plenum, heavy water moderator calandria vessel, fuel assemblies, pressure tubes, etc. Light water coolant is contained in a pressurized inlet plenum located at the top of the core, which is attached to a low-pressure calandria vessel (at the bottom) containing heavy water moderator (Figure 1A). The moderator surrounds an array of pressure tubes, each of which contain a full length fuel assembly. The pressure tubes are oriented vertically and are connected to a tube sheet (Figure 1B), which separates the coolant from the moderator. The reactor is batch refuelled based on a three batch cycle.

Simplified views of the coolant and moderator flow streams are also shown schematically in Figure 1. The light water coolant flows from inlet nozzles into the inlet plenum (Figure 1A) and from there flows into the fuel channels (Figure 1C). The fuel assembly is a re-entrant or double flow pass configuration, with a central flow channel through which the coolant is forced vertically downwards (Figure 1C). Near the bottom of the channel, the coolant exits the central flow tube and is redirected upwards and flows up through the fuel elements (Figure 1D). The coolant is initially at sub-critical temperature in the inlet plenum, at a pressure of approximately 26 MPa and a temperature of 350 °C [4] and remains subcritical until it is heated by the fuel during its path upward through the fuel channel. The streams of supercritical water exiting the fuel channels mix in the outlet plenum, which is located inside the inlet plenum (Figures 1A and C). The expected average exit temperature of the coolant is 625°C at a pressure of 25 MPa. To maximize cycle efficiency, while keeping cladding temperature at an acceptable level, channel flows are controlled through channel-specific orifices, such that the exit temperature from each channel will be about 625°C.

The design of the reactivity control and shutdown devices is ongoing. Taking advantage of the low pressure calandria vessel, these devices will be inserted through the calandria vessel walls. The low calandria pressure of 300 kPa virtually eliminates the control rod ejection scenario that is a concern in pressure-vessel reactors. In traditional heavy water moderated pressure-tube reactors, control devices are inserted radially as insertion at the two ends of the calandria vessel is not feasible. In the SSR core, one end (the bottom) of the calandria vessel is available for reactivity control and shutdown devices. The axial insertion of such devices from the bottom of the calandria vessel opens the possibility of using larger devices than would be possible with radial insertion only. For example, water displacement tubes inserted axially from the bottom of the calandria vessel can be used for reactivity control and for the shutdown of the reactor. This arrangement would allow the use of gravity to shut down the reactor in an emergency.

The present SSR concept is intended to generate approximately 670 MW of thermal power, 300 MW of electric power, assuming a thermal efficiency of 45%. The SSR core contains 120 fuel channels arranged in a 25 cm square lattice pitch. The core diameter is 400 cm. This includes a 50 cm thick D_2O radial reflector region. The present concept does not include an axial reflector region. However, the heavy water moderator could be extended 50 cm below and above the lowest and highest elevations of the fuel, respectively, for the purposes of providing additional neutron reflection and achieving flatter axial power profile.

The SCWR and SSR share the same fuel assembly and channel designs. The fuel assembly and fuel channel specifications are based on those provided in Pencer et al. [5]. The fuel channel, shown schematically in Figure 2, is based on the high efficiency channel (HEC) concept [6]. The outermost component of the HEC is an Excel (a zirconium based alloy) pressure tube, which is in direct contact with the moderator. A porous zirconia insulator is located directly inside the pressure tube and insulates it from the high temperatures in the coolant. This insulator is supported on its inner surface by a perforated liner tube. The present concept includes one significant change relative to earlier descriptions [5]. As described above, the solid centre pin used previously has been replaced by a coolant flow tube and the channel now utilizes a bi-directional re-entrant coolant flow.

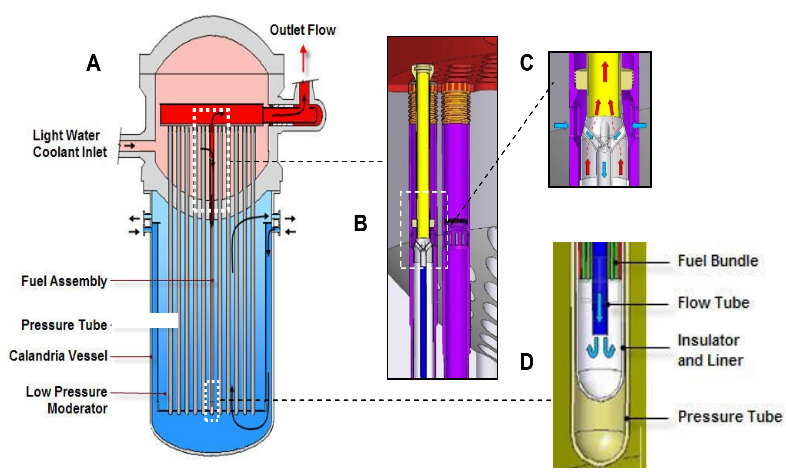

FIGURE 1
Cross-sectional side views of the SSR reactor core and flow streams: A - Core layout, B - Pressure tube connection to tube sheet, C - Coolant flow from inlet plenum and flow to outlet plenum, D - Redirection of coolant flow.

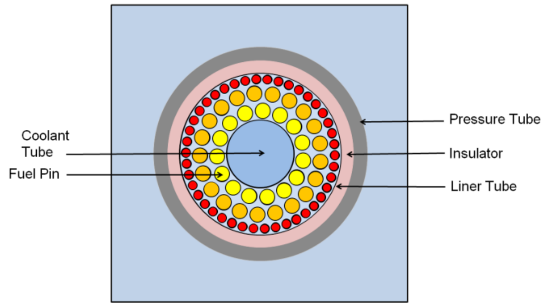

FIGURE 2
Cross-sectional view of the 78-element Canadian SCWR fuel assembly, high efficiency channel and lattice cell.

The 78-element assembly contains three concentric rings of 15, 21 and 42 fuel elements (see Figure 2). In the reference fuel cycle case discussed in this paper, all the fuel elements are composed of 13 wt% reactor grade (RG) PuO_2 and 8 wt% ThO_2 (the Th is assumed to be 100 wt% Th-232).

Optimization studies of the fuel channel and fuel assembly design are ongoing and require an iterative approach because of the coupling of physics, thermal hydraulic and mechanical aspects of the conceptual design. The target parameters for optimization are the core average fuel assembly exit burnup, enrichment, reactivity coefficients, axial and radial power peaking factors, fuel and cladding temperatures, linear power rating, reactivity, and stability.

3. Advanced Safety

The advanced safety of the SSR is based on the key concept of “walk away safety”, that is, during an accident scenario, the reactor will return to a safe state without requiring human intervention. Specific features that meet this goal are discussed below.

3.1 Negative Reactivity Coefficients

A core design that incorporates negative feedback avoids the potential for run-away reactions. The SSR fuel and fuel assembly are specifically designed by choice of enrichment, burnable neutron absorber, and fuel-to-moderator ratio to ensure a negative coolant void reactivity (CVR) coefficient and negative overall power coefficient of reactivity (PCR) throughout reactor operation [4]. Negative CVR and PCR ensure that the onset of a power pulse is self-correcting and, as a result, the peak power never exceeds the maximum threshold value.

3.2 Passive Heat Removal

Providing multiple paths for heat removal from the core minimizes potential heat increase in the event of a loss of coolant accident (LOCA). As in most other pressure-tube type nuclear reactors, the SSR has a low-pressure moderator that surrounds the high-pressure fuel channels. This configuration provides a unique safety enhancement with respect to other reactor types, in which unwanted excess heat (i.e., decay heat of the fuel under accident conditions) can be independently removed by the surrounding moderator as well as by the circulation of the primary coolant. Hence, the SSR is equipped with two independent cooling systems.

The first heat removal system for the SSR is the primary-coolant safety system (PCSS) which is similar to the safety systems proposed for advanced boiling water reactors. The PCSS is intended to have the capability to remove decay heat through natural circulation and floods the reactor core with emergency coolant in case of a LOCA.

The second, diverse and independent, heat removal system is the moderator safety system (MSS), which provides an additional passive layer of safety for a defense-in-depth approach. The MSS uses flash-driven natural circulation flow to remove heat from the reactor core and transfers it to a large water pool heat sink [7]. In all accident scenarios, the PCSS activates first and prevents the fuel from overheating. In extremely unlikely scenarios where total loss of coolant is accompanied by loss of PCSS function, the MSS is activated. Both the coolant-based and moderator-based cooling systems are designed to remove 100% of the decay heat independently and passively.

3.3 No-Core-Melt

Prevention of overheating and melting of the fuel and fuel cladding minimizes damage to the core and potential release of radioactive material. Prevention of core-melt, even assuming complete loss of primary coolant flow, PCSS, and station power, is achieved by passive decay heat removal through the moderator. The distribution of fuel and fuel channels in the moderator, in the case of accident conditions, is equivalent to a distribution of physically separated individual decay heat sources (the fuel in the fuel assemblies) surrounded by a large water reservoir (the moderator). The decay heat can be removed by passive radiative cooling from the fuel bundle to the insulator and conduction to the pressure tube and then the moderator, which is still at a relatively low temperature. Prior analyses have shown that levels of decay heat up to 2% of the total power can be removed without exceeding cladding and fuel melting temperatures in the hottest of the fuel pins [8].

3.4 Very Long-Term Decay Heat Removal

Maintaining the capability to remove decay heat for indefinite periods of time prevents core damage in circumstances where mitigation of accident conditions is not possible even in the long-term. The strategy for long-term decay heat removal in the SSR is to use a combination of a large water reservoir and air coolers as ultimate heat sinks in the event of a complete station blackout and loss of all emergency power supplies. This is achieved by maintaining a sufficient supply of water in the containment to serve as a heat sink until the decay power falls to 0.5% of full power (typically 24 hours after blackout) after which air cooling is sufficient to remove decay heat indefinitely.

3.5 Leak-Before-Break and Severe Accident Management

Detection of early onset of pressure tube failure can prevent severe accidents. To prevent the worst-case scenario of a complete pressure-tube rupture, two independent and diverse leak / crack detection systems are engaged to ensure that pressure tube leaks are detected before a large break. Upon detection of a leak, the reactor will be shut down and depressurized to prevent pressure tube rupture. In the unlikely event of pressure tube rupture, its effects will be mitigated by the calandria vessel, which is designed to remain intact even when multiple pressure tubes are ruptured.

In conventional heavy water pressure tube reactors, leak detection is accomplished by measuring the moisture content of the gas annulus between the pressure tube and the enveloping calandria tube. The Canadian SCWR uses an insulated pressure tube that eliminates the calandria tube and the gas annulus. Without a gas annulus moisture detection system, new leak detection techniques are required for leak monitoring. Several leak detection methods are currently being investigated for use in the Canadian SCWR, including acoustic monitoring, online measurement of light water contamination of the heavy water moderator, and monitoring of local variations in changes in channel power or reactivity.

4. Fuel Cycle Options

The reference fuel cycle for the Canadian SCWR is a once-through Pu-Th-based fuel cycle [4]. This cycle was chosen because it does not require enrichment with U-235, thus preserving natural uranium resources and thereby aiding in meeting the GIF's sustainability goals. The once through cycle will be employed in the SCWR until a sufficient reserve of U-233 in spent fuel is accumulated that can be used in a U-233 recyling-based fuel cycle [9]. The Pu-Th-based fuel cycle depends on availability and recycling of Pu from spent PWR fuel for example. Depending on the schedule to deployment, economic constraints or operational constraints, it may be desirable to employ a different fuel cycle in the SSR in the short term. Two simple alternative fuel cycles are examined here for comparison to the reference fuel cycle, a low-enriched-uranium (LEU) based cycle and a LEU-Th-based cycle. Both of these cycles are driven by initial enrichment in U-235, are once-through cycles and use fuel compositions that are uniformly distributed both within the fuel assemblies and within the SSR core.

Table 1
Comparison of SSR Fuel Cycle Parameters for the Reference (Pu-Th), LEU and LEU-Th Fuel Cycle Options

Parameter	Reference (13% PuO_2/ ThO_2)	LEU	LEU-Th (50 wt% LEU in Th)
Initial Fissile wt% HE	8.6% (Pu-239 + Pu-241)	6% (U-235)	6.5% (U-235)
Average Exit Burnup (MWd/kg)	44.3	44.2	42.9
Fissile Utilization (MWd/kg initial fissile)	515.1	738.3	681.5
Cycle Length (EFPD)	660	720	670

A core model of the SSR was constructed using the code set WIMS-AECL / RFSP in order to obtain the core-related fuel cycle parameters: cycle length, core average exit burnup, and fissile material utilization. A detailed description of the core modeling method can be found in Pencer et al. [5]. In order to compare alternative fuel options with the reference Pu-Th cycle, the fissile enrichments of the LEU and LEU-Th were adjusted so that all three fuel cycles reach the same target exit burnup in the SSR core. A comparison of fuel cycle parameters for the three fuel cycle options is provided in Table 1.

For both the LEU and LEU-Th options, the fissile requirement to reach the target exit burnup (approximately 44 MWd/kg) is reduced relative to the reference fuel option (13 wt% PuO_2 in ThO_2). Two factors contribute to the difference between the Pu-based and LEU-based fuels: the density differences among the three fuels and enhanced thermal neutron absorption in isotopes of Pu. The LEU-based fuels both have higher densities than the reference Pu-Th fuel, approximately 10.6 g/cc and 10.2 g/cc for LEU and LEU-Th, respectively, compared to 9.9 g/cc for Pu-Th fuel. The higher density of the LEU and LEU-Th fuels leads to a higher ratio of the mass of fuel to fuel cladding. The result of the higher proportion of fuel mass to cladding material is that a greater proportion of neutrons are available for the fuel to produce fissions or convert fertile material to fissile material. All isotopes of Pu in reactor grade (RG) Pu have large thermal absorption peaks, which are absent in isotopes of uranium. In addition, the thermal capture to fission ratios for the fissile isotopes of Pu in RGPu, Pu-239 and Pu-241, are almost twice that of U-235. Consequently, captures of thermal neutrons on Pu in the Pu-Th cycle reduces the number of neutrons available for fissions (or conversion of Th to U-233), thus further reducing the fissile utilization of Pu-Th fuel relative to the two LEU-based fuels.

Use of a once through LEU cycle in the short term is advantageous compared to the reference Pu-Th and the LEU-Th fuel cycle options since its longer cycle length results in a 10% higher capacity factor compared to the other fuel cycles, leading to enhanced economic benefit. As discussed above, the higher fuel density of LEU results in a higher fissile utilization compared to LEU-Th and Pu-Th, thus also contributing to the enhanced economic benefit. Use of LEU fuel could also be implemented much sooner than LEU-Th or Pu-Th based fuels because of the greater knowledge of LEU fuel behaviour based on its common use in PWR and BWR. Use of LEU in the long term is not desirable, since its use will deplete natural uranium reserves, thus having a negative impact on the sustainability of the fuel cycle.

The once through LEU-Th cycle is also appealing for use in the short term. Although the LEU-Th cycle is expected (based on cycle length) to have the same capacity factor as the Pu-Th cycle, it does show superior fissile utilization over Pu-Th, contributing to the economic benefit of LEU-Th over Pu-Th. As with LEU, use of LEU-Th in the long term is not desirable since this cycle will also deplete natural uranium reserves, negatively impacting fuel cycle sustainability.

Although the LEU and LEU-Th cycles applied in SSR in isolation appear to negatively impact fuel cycle sustainability, this negative impact may be mitigated or even reversed through the recycling and use of the U-233 created in the spent SSR LEU or SSR LEU-Th fuel in other SSRs or other reactor systems. Likewise, the use of Pu from LWR spent nuclear fuel in the Pu-Th based cycle extracts further energy from fuel that has already passed once through a reactor thus enhancing the sustainability of this cycle.

In this preliminary examination of fuel cycle options for the SSR, the impacts of fuel composition on key safety parameters such as CVR and PCR, have not been examined. The impacts of various fuel types on these safety parameters may override potential benefits gained in improved fissile utilization or capacity factor and must be assessed.

Detailed studies of potential fuel cycle scenarios [10] could be used to optimize the economics and sustainability of the SSR fuel cycle. Comparison of fuel cycle scenarios will enable the identification of synergies between the SSR and other reactor systems. Such studies could also provide recommendations for the optimal timing of transitions between different fuel cycles in the SSR and optimal balance of SSR with other nuclear energy systems or alternative energy sources.

5. Proliferation Resistance

A key stage in the assessment of the intrinsic proliferation resistance of a fuel cycle is a determination of the potential for use (i.e., "material attractiveness") in nuclear weapons of the fissile materials, i.e., uranium and plutonium, used and produced in the fuel cycle. In order to provide a quantitative metric for the material attractiveness, a figure-of-merit (FOM) formula for rating nuclear material attractiveness has been devised [11]. This metric takes into account the main factors that enhance or detract from the attractiveness of a material for use in nuclear explosives namely: critical mass, decay heat, and radiation dose rate. In addition to these, a second formula was created to take into account spontaneous neutron production from the material. It is generally thought that high spontaneous neutron production from a material may impede only the relatively unadvanced nations or sub-national groups from producing a nuclear explosive. The two formulas are:

$$FOM1 = 1 - \log_{10}\left(\frac{M}{800} + \frac{Mh}{4500} + \frac{M}{50}\left[\frac{D}{50}\right]^{1/\log_{10}2}\right) \quad (1)$$

$$FOM2 = 1 - \log_{10}\left(\frac{M}{800} + \frac{Mh}{4500} + \frac{MS}{6.8\times10^{6}} + \frac{M}{50}\left[\frac{D}{50}\right]^{1/\log_{10}2}\right) \quad (2)$$

where M is the bare critical mass in kilograms, h is the decay heat in watts per kilogram, D is the dose rate in rem/hr at a distance of 1 m, and S is the spontaneous neutron generation rate in neutrons/second/kilogram. It is important to note that these quantities are calculated for metallic uranium or plutonium, that is, the material after it has been removed from the spent fuel and processed into weapons usable form. Thus, it is implied that reprocessing capability exists, whether it is included in the nuclear energy system or is clandestine in origin.

The numerical results from the equations can then be used to describe the material attractiveness for weapons use according to Table 2.

In the analysis presented in this paper, the attractiveness of the materials have been computed for 'freshly' separated uranium, that is, directly after the material has been processed for potential weapons use. Thus, in the case of uranium derived from thorium-based fuel, the dose coming from the U-232 itself is included in the calculation; however, the dose from the decay products of U-232 is not included. Figure 4 shows the result of the FOM calculation for plutonium and uranium for the three different SSR fuel options. Plutonium remains highly attractive throughout the entire fuel burnup for all three fuel types. Only in the case of Pu-Th fuel does the FOM1 drop below 2 towards discharge burnup.

TABLE 2
Material Attractiveness: Ranking by Figure of Merit (FOM)

FOM	Weapons Utility	Material Attractiveness
>2	Preferred	High
1-2	Attractive	Medium
0-1	Unattractive	Low
< 0	Unattractive	Very Low

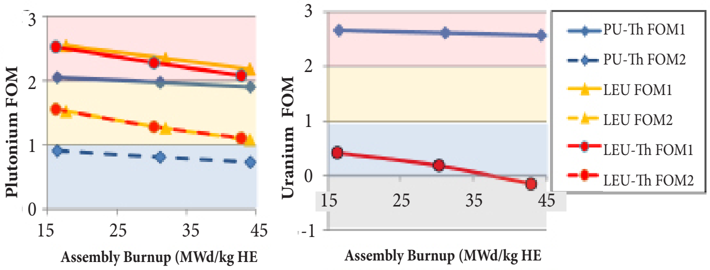

FIGURE 3
Figures of merit for plutonium (left) and uranium (right) for SSR fuel options.

The higher concentration of Pu-240 in the Pu-Th fuel leads to an unattractive rating when considering FOM2.

For uranium, FOM2 is practically identical to FOM1 due to the relatively low number of spontaneous neutrons produced and is not shown in the plots. The uranium produced in the Pu-Th fuel during irradiation is a proliferation concern as it is primarily (>90%) U-233. In the case of the LEU-Th fuel, the U-233 produced is "denatured" by the presence of U-238 and this is reflected in the unattractive (<1) FOM value. For the LEU fuel, the uranium in the fuel does not contain enough fissile material to create a critical mass and therefore unable to be used as weapons material.

In all three of the fuel cycles examined, no attempt was made to adjust the fuel compositions to reduce the material attractiveness. Strategies for reducing the material attractiveness of SSR fresh and spent fuels could include the addition of U-238 to the Pu-Th fuel to reduce the attractiveness of the U-233. Such adjustments in fuel composition would also need to be assessed for their impact on the SSR fuel cycle (e.g. exit burnup and fissile utilization) and core safety parameters such as CVR. Although material attractiveness may be reduced through the adjustment of fuel compositions, safeguards and physical protection will always be required in facilities dealing with quantities of uranium and plutonium and high levels of security will be required in reprocessing facilities. The level of security may be adjusted for storage of materials where the dose rate is deemed to be 'self-protecting'.

6. Summary and Conclusions

The concept for a small supercritical water-cooled pressure-tube reactor design was presented and includes strategies for enhanced safety, improved sustainability, improved economics and enhanced security. Development of the SSR concept takes advantage of many of the enhanced features of the SCWR, including advanced passive safety and increased thermal efficiency. In addition to the reduction in core size, changes with respect to the SCWR (e.g. the fuel cycle) may be incorporated in order to meet differences in priority with respect to economics, time to deployment, and operational constraints. Alternative fuel cycle options will also need to be assessed for their impact on core safety. Systems scenario studies could aid in enhancing the sustainability of the SSR fuel cycle and optimizing SSR deployment. Changes in fuel composition may help to reduce material attractiveness of SSR fuel options, but such changes need to be assessed both for their impact on core safety and on the fuel cycle. Refinement of the SSR conceptual design via optimization of the fuel, safety and layout is proceeding.

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FULL ARTICLE

Challenges of SMR Licensing Practices

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Abstract

This paper aims to increase the understanding of high level Nuclear Power Plant (NPP) licensing processes in Finland, France, the UK, Canada and the USA. These countries have been selected for this study because of their different licensing processes and recent actions in new NPP construction. After discussing their similarities and differences, suitable features for Small Modular Reactor licensing can be emphasized and suggested. Some of the studied licensing processes have elements that are already quite well suited for application to SMRs, but all of these different national processes can benefit from studying and implementing lessons learned from SMR specific licensing needs. The main SMR features to take into account in licensing are standardization of the design, modularity, mass production and serial construction. Modularity can be divided into two different categories: the first category is simply a single unit facility constructed of independently engineered modules (e.g., construction process for Westinghouse AP-1000 NPP) and the second is a facility structure composed of many reactor modules where modules are manufactured in factories and installed into the facility as needed (e.g., NuScale Power SMR design). Short construction schedules will not be fully benefitted from if the long licensing process prolongs the commissioning and approach to full-power operation.

The focus area of this study is to better understand the possibility of SMR deployment in small nuclear countries, such as Finland, which currently has four operating NPPs. The licensing process needs to be simple and clear to make SMR deployment feasible from an economical point of view.

This paper uses public information and interviews with experts to establish the overview of the different licensing processes and their main steps. A high-level comparison of the licensing steps has been carried out. Certain aspects of the aviation industry licensing process have also been studied and certain practices have been investigated as possibly suitable for use in nuclear licensing.

All of the current licensing processes were found to be quite heavy and time-consuming and further streamlining could be possible without compromising safety or the need for public participation in the licensing process. Some examples of the modification possibilities for SMR applications are discussed.

A profound discussion on SMR-specific licensing models, and on ways to simplify and harmonize them, will be needed in the near future in Europe too. This would be a natural continuation to the harmonization efforts underway for existing and new large reactors.

Introduction

The global nuclear industry is currently undergoing many changes. A large number of new NPP projects around the world have started a nuclear renaissance, which, however, seems to have most recently slowed down due to the increasing costs and additional challenges that emerged as a result of the Fukushima accident. This development could make smaller and simpler plant designs more appealing in the future. Both large power plant and small modular reactor designs are evolving away from traditional complex designs towards a more simple and robust direction. Small reactors, with their smaller power-to-volume and power-to-surface ratios may allow more efficient implementation of passive safety features, for example, and thereby simplification of the design.

The safety design of the plant, one of the most important areas in the nuclear industry, has been under discussion in recent years due to the increasing interest in developing new NPP designs and projects. The opinions of the different safety design approaches vary greatly from very complex active safety design features to more simple passive solutions. The main idea of the simplification of a design is to enhance the safety level of the whole power plant. Although the opinions of different stakeholders vary, the concept of a passive safety design is seen as an improvement in overall safety. This improvement is based on a decrease in the possibility of active safety systems failures and slower transient and accident sequences. A simplified design also enables the operator to better understand the features of the plant during possible operating transients and accidents.

Recently, the trend of NPP technology development is changing such that in addition to very large plants (> 1000 MWe), Small Modular Reactors (SMR) are being increasingly discussed [1]. SMR in this paper is interpreted to mean a reactor facility of less than 300 MWe using modern modular techniques [2]. Modularity is one of the design bases, in addition to the passive safety features. Modularity can be seen in many different ways, the most obvious being the modularity of several reactors on the same site with either shared systems or shared structures. Modularity can also be seen in modules manufactured in factories and assembled onsite to reduce delays and construction costs [3]. Modularity is one of the features that drives the need to rethink some aspects of the licensing process in many countries. Scaling effects have been investigated widely in different SMR publications to understand the differences between large and small NPPs. Scaling effects are divided into soft scaling effects and hard scaling effects [4]. Soft scaling effects describe cost reduction by changing the management of construction (Figure 1). Hard scaling effects include changes of applicable technologies in the design when power decreases (Figure 2).

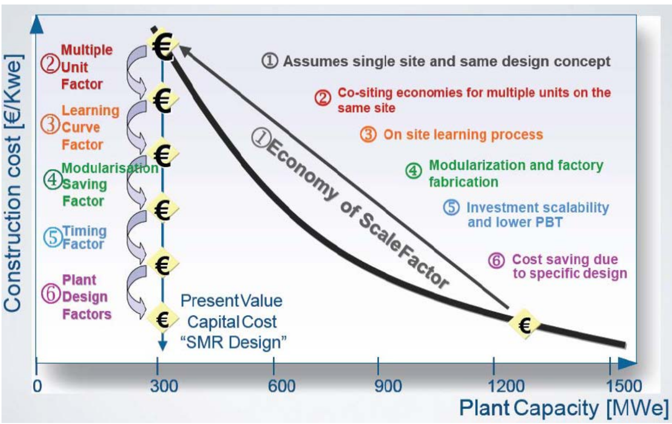

FIGURE 1
Soft Scaling Effect [4]

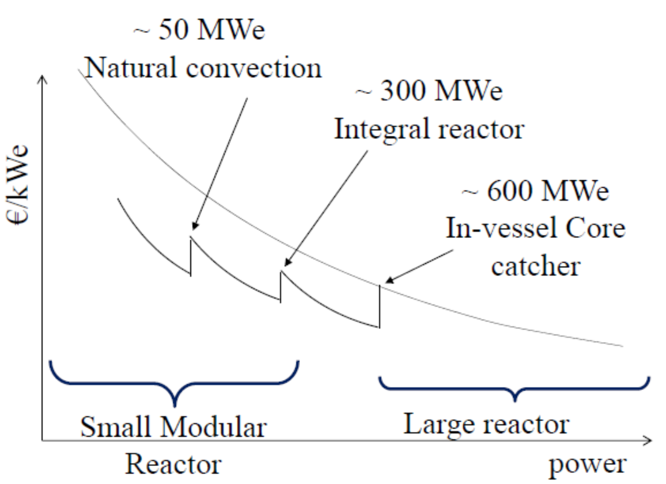

FIGURE 2
Hard Scaling Effect [4]

Other advantages are derived from the use of standardization (mass production), short construction time, serial construction (enabling self-financing) and sustainability issues. However, these issues are not discussed in depth in this paper.

Because of the public interest and unique hazards, the nuclear industry has very specific licensing needs. This being said, there are some other industry fields that have similar or comparable safety issues to deal with. This study investigates features of the nuclear licensing processes in various countries as well as aviation industry licensing. It is the aim to identify and characterize favorable features to be considered for use in SMR licensing. The focus of this study is on practices that could be feasible for a small country like Finland.

Our aim is to seek answers to two questions:

1. What are the main features of SMR licensing that are seen as different from current large NPP licensing?

2. What are the features in different licensing processes that are most suitable for SMRs?

The research methodology is presented in the Appendix.

International harmonization of licensing and technical requirements for new nuclear power plants is being pursued by several organizations. The IAEA safety standards obviously form an international benchmark. In Europe, WENRA (the Western European Nuclear Regulators' Association) is pursuing harmonization of European licensing requirements through the reference levels for existing nuclear power plants and through the safety objectives for new plants. On the industry side in Europe, the European Utility Requirements (EUR) is an attempt to harmonize the utilities' design requirements to plant vendors in a similar manner to the Utility Requirements Document (URD) in the USA. The scope of the EUR and the URD encompasses more than just safety requirements. These organizations, among others, have, however, been focusing on the requirements of the product rather than addressing the licensing process itself. In Europe, initial studies of licensing process development in the future have been, and are still being, carried out by the European Reactor Design Acceptance (ERDA) Core Group [5] under the European Nuclear Energy Forum (ENEF). This group is currently identifying ways towards licensing process harmonization in Europe and possible common European Reactor Design Acceptance in licensing. Currently, the above-mentioned efforts are not particularly focused on the specific features of SMRs. At this point in time, because this group is still in early planning stages, an opportunity may exist to introduce SMR scope to their work.

It has been a very time-consuming and challenging task to develop harmonized licensing requirements for NPP licensing in Europe. The aviation industry has been used as a good example of successful requirements harmonization in Europe and also between the USA and Europe. Also the licensing process has been harmonized well in aviation industry. The harmonization has been carried out through EU direction and this also might be the case in the future for the nuclear field, although this is only speculation and it remains to be seen in the future.

SMR special licensing features

The competitiveness of SMRs will be based on simplification of the design (without affecting safety), including extensive use of passive safety systems, standardization, mass production, short construction time and serial construction (enabling self-financing) and sustainability issues. Additionally, the effectiveness of the licensing process will be one of the key competitiveness factors. For SMR licensing, the following areas may influence the need for differences from traditional large NPP licensing:

• Smaller power output - lower decay heat, (increased use of graded approach to application of safety important mitigation measures)
• Fully passive safety features (different from most of the current NPPs),
• Modular design:
  o several reactor modules in one plant
  o modular construction with modules manufactured in factories
• Mass production (standardized design)
• Serial construction (many plants in series)

The licensing processes in most nuclear countries have been mainly developed for large NPPs that are built one at a time. This kind of licensing is not necessarily the most optimal for SMRs including several reactor modules in one plant. Licensing of SMRs should be planned keeping in mind the modularity of the design. This discussion has been opened in the USA, where the different types of licence configurations are presented as an option.

Comparison of licensing processes

In this study, the high-level licensing schemes in the USA, Canada, the UK, France and Finland have been compared. Licensing processes have been subject to development during recent years. In the USA, an important step in the development of the licensing process began in 1989, when the NRC established new alternatives for nuclear plant licensing under 10 CFR Part 52 (so called one-step-licensing). In the USA, a standardized process is easy for the licencee to understand and follow, but is still time consuming and heavy. The UK is moving in the same direction with the GDA as the USA, with separate licence for standard design and specific site followed by COL.

Canadian licensing can be seen in some aspects as being similar to the UK, with the fact that the Regulator sets high-level requirements and licensing objectives but does not set highly prescriptive rules for licencees. The licencee proposes how they will meet these requirements and the licencee's process becomes part of the licensing basis. Similarity with the UK can also be seen in Vendor Design Review process (VDR), which can be compared with the UK GDA process. It shall be noted that the VDR in Canada is not certification, but a pre-licensing activity to improve the readiness to enter the licensing process should the plant design be referenced in a specific site licence application. The licensing process has similarities with the current Finnish process (based on the USNRC 10 CFR Part 50) with Construction licence and Operating licence steps.

In Finland and France, the licensing process is issued one by one. There is not a licence for standard design, but every specific project with the selected NPP design is licenced separately. This is probably a suitable way to handle licensing in a country with less NPPs or many different designs.

The development of the licensing process in every country emphasizes the need for an early conversation with the regulatory body. Also the public is getting more and more interested in the process driving the need for robust and transparent hearings.

It appears that out of selected countries only the USA and Canada are considering SMR licensing issues. Basically, SMRs in the USA go through the standard licensing process. The main question in the USA, in addition to certain technical requirements modifications, is the number and scope of the licences [7]. The Canadian Nuclear Safety Commission (CNSC) has determined that the application of requirements for a reactor design can consider a risk-informed approach in a number of areas. That is to say, in some areas, safety mitigation can be applied, by a licence or applicant in a graded manner (grading) based on risk. For very small designs below approximately 200 MWt, additional grading may be possible due to the significantly smaller core inventory [23].

The basic licensing concepts in the studied countries are presented in Figure 3.

FIN	Decision in Principle		Construction License	Operating License
USA	Early Site Permit		Combined Construction and Operating License (COL)	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)
	Standard Design Certification			
Canada	Environmental Assessment	License to Prepare Site	License to Construct	License to Operate
France	Plan Pluriannual d'Investissement (PPI) - multiyear investment plan		The authorization decree for NPP creation	Operating License
UK	Generic Design Assessment (GDA) - separate from licensing process		Nuclear Site License (Environmental, Safety and Security review processes)	

FIGURE 3
Licensing Processes

Finland

In Finland the Olkiluoto 3 case has been used as the basis of the schedule and duration of different licensing steps

(Figure 4). Also, approximations of the durations of the Olkiluoto 4 and Hanhikivi 1 (Fennovoima) projects have been used to estimate the durations of different licensing phases.

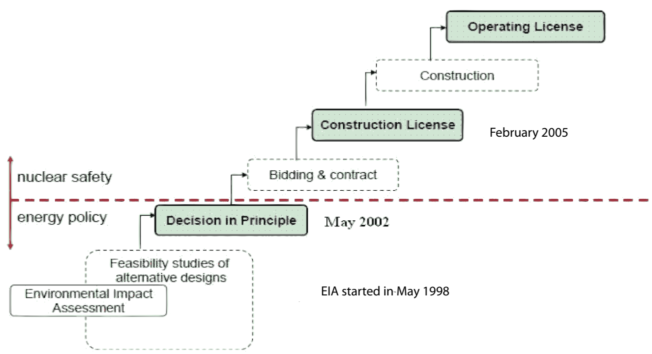

FIGURE 4
Licensing Steps of Olkiluoto 3 Project [8]

The site approval process in Finland begins with an Environmental Impact Assessment (EIA). Preliminary assessment of the suitability of the site is also carried out by the regulator as part of its preliminary safety assessment of the Decision-in-Principle application. Site approval issues are also included in the construction licence phase with activity release calculations within accidental scenarios. The EIA and preliminary safety assessment are the basis for Decision-in-Principle in Finnish nuclear licensing. This part of the decision-making process in Finland has been quite well standardized, because this process has been gone through four times in the past ten years (Olkiluoto 3 in 2002 and Olkiluoto 4, Hanhikivi 1 by Fennovoima and Loviisa 3 by Fortum in 2010). The Decision-in-Principle process formally takes more than one year; for example, Fennovoima has presented their project schedule [12]. The preparation of EIA has been estimated to take two years. For example, TVO's Olkiluoto 4 project EIA process started in the beginning of 2007 and the Decision-in-Principle was applied in 2009 [9].

The Construction Licence (CL) phase needs to be estimated not only on the basis of the Olkiluoto 3 experience, but also using the Olkiluoto 4 and Hanhikivi-1 estimated licensing schedules. For Olkiluoto 3, the construction phase was very short and it took only one year (01/2004-02/2005) [10]. This CL process has been observed to take place too early in terms of the design stage and, in upcoming projects, the design stage is expected to proceed much further when applying for a Construction Licence. The CL process has been developed by STUK and new YVL-guides have been developed taking into account the lessons learned since the Olkiluoto 3 experience. It can be estimated that, for the future NPP projects, the process will be more standardized and the CL process duration is estimated to be at least 18 months.

The pre-inspection phase is the phase in Finnish regulatory oversight between the Construction Licence and Operating Licence (OL). For example, in the Olkiluoto 3 project, this phase has taken since February 2005, and the application for the Operating Licence still has not been applied for (as of 09/2012). The duration of this phase depends strongly on the design stage that is achieved for the Construction Licence phase. If the design stage is high in the CL phase, this pre-inspection phase is particularly light, requiring only the preparation of systems design documentation for FSAR (in the Operating Licence phase). Because the only project that the experience has been gained is Olkiluoto 3, it is used as the basis for this study. The duration of this phase cannot be precisely defined, but an approximation based on the Olkiluoto 3 experience is around six to seven years. It should be observed that this phase will most probably be much shorter in the future projects, as explained above.

The duration of the Operating Licence phase cannot be specified because this process has not been issued for new NPP projects in Finland yet. The process was expected to begin for Olkiluoto 3 in 2012 [8], but the duration has not been presented in public. It can be estimated that the OL process would take around the same amount of time to review by STUK, as is expected for Construction Licence review, being about 18 months. This estimate will be used in this study, although it should be noted that this is only an estimate.

USA

The projects that are going through or are scheduled for the licensing process are used as a reference for this study. These schedules are presented in reference [12].

Figure 5 presents the NRC licensing schedules of AP1000 (Westinghouse design), EPR (Areva design) and ESBWR (GE design) [12].

The approximations of the licensing steps duration in the USA are based on these and also NRC licensing schedules for other designs. This approximation has partly been calculated by the Nuclear Energy Institute (NEI) [13].

An Early Site Permit (ESP) takes between 12 and 24 months to develop an application, depending on whether it is a "greenfield" site or a site adjacent to an existing facility. Once the applicant submits the application, the process of NRC review and approval takes approximately 33 months (including the public hearing) [13].

The Design Certification (DC) process by the NRC takes from 36 to more than 60 months to complete the review

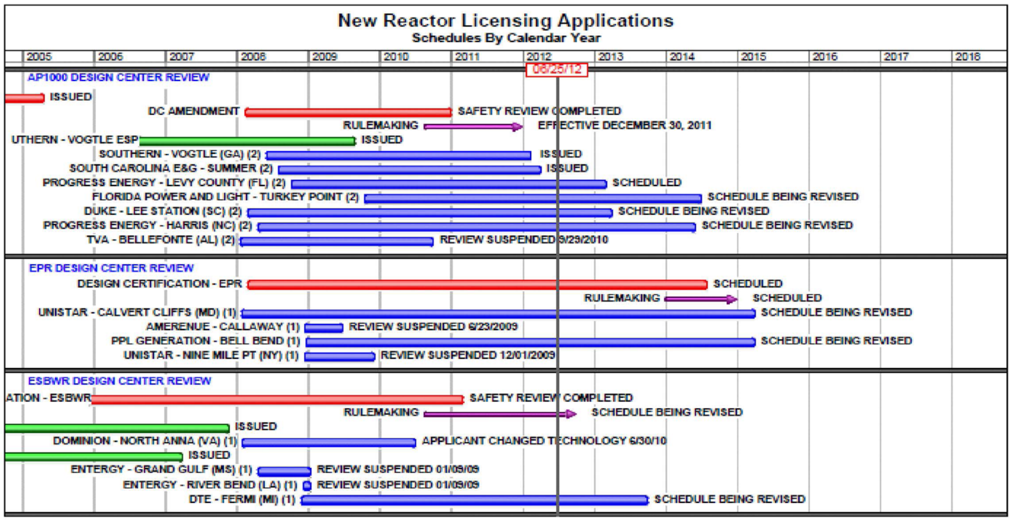

FIGURE 5
The Licensing Schedules of AP1000, EPR and ESBWR by NRC [12]

and rulemaking, depending on whether the agency has previously reviewed and approved the technology [13]. The rulemaking process takes approximately one year, including public hearings. It should be noted, however, that a certified design is generally early in the detailed design and therefore is not a complete design.

An application for a Combined Construction and Operating Licence (COL) under CFR-Part 52 is expected to reference a certified design and may also reference an Early Site Permit (ESP) or both. All issues resolved in connection with earlier proceedings associated with a standard design or site will be considered resolved for the purposes of the COL process. This makes the process more effective, allowing the NRC to focus on the remaining issues related to plant ownership, site specific design issues and organization and operational programs. The volume of open issues affects the duration of the COL process. From the schedules presented in the reference [12], it can be estimated that the COL process will take between four and five years.

In addition to these licensing processes, there is an Inspection, Tests, Analyses and Acceptance Criteria (ITAAC) process that takes four to five years. The USNRC is developing technology specific review criteria and ITAAC for some key SMR designs expected to be licenced in the near term.

Canada

In the Canadian licensing framework, the applicant's submissions are expected to address all regulatory requirements as well as applicable codes and standards. The VDR process is a pre-review process for licensing where the purpose is to increase CNSC staff knowledge of aspects of a design and give early indications to the vendor whether they understand Canadian requirements and have more development, design or analysis work to perform before entering the licensing process. The VDR process also identifies any potential fundamental barriers to licensing for the areas examined by the CNSC in the review.

The vendor design review is divided into three phases.

1. Phase 1 review – Compliance with regulatory requirements in general level, for new nuclear power plants as specified in RD-337, and for small reactors facilities in RD-367 and related regulatory requirements.

2. Phase 2 review – Pre-licensing assessment going into further detail, with a focus on identifying potential fundamental barriers to the licensing of the vendor's design.

3. Phase 3 review – Pre-construction follow-up, where the vendor can choose to follow up on one or more focus areas covered in Phase 1 and 2. The vendor's anticipated goal is to avoid a detailed revisit during the construction licence application review.

Phase 1 and 2 reviews have 19 review focus areas, representing key areas of importance for a future construction licence. The Phase 3 review is tailored on a case-by-case basis [23].

Currently, the pre-licensing Phases 1 and 2 have been completed for CANDU 6 plant, and the Phase 3 is in progress. Generation mPower (B&W) and NuScale Power are both pending start of Phase 1 in 2013.

The 19 VDR Review Areas are:
1. General Plant Description Defense in Depth, Safety Goals and Objectives, Dose Acceptance Criteria
2. Classification of Structures, Systems and Components
3. Reactor Core Nuclear Design
4. Fuel Design and Qualification
5. Control Systems and Facilities

6. Means of Reactor Shutdown
7. Emergency Core Cooling and Emergency Heat Removal
8. Containment and Safety Important Civil Structures
9. Mitigation of Design Extension Conditions
10. Safety Analysis
11. Pressure Boundary Design
12. Fire Protection
13. Radiation Protection
14. Out of Core Criticality
15. Robustness, Security and Safeguards
16. Vendor Research and Development Program
17. Management Systems of Design Process, Design QA in Design and Safety Analysis
18. Human Factors
19. Decommissioning in Design

In licensing, an applicant or licensee (typically a utility) is expected to address, in their licensing basis documents, the following Safety and Control Areas:

- Physical Design
- Safety Analysis
- Fitness for Service
- Siting & EA
- Informing the Public
- Packaging and Transport
- Security & Safeguards
- Waste Management
- Emergency Mgmt & Fire Protection
- Environmental Protection
- Conventional Occupational Health and Safety
- Radiation Protection
- Management System Framework
- Human Performance Management
- Operating Performance

For each of the Safety and Control Areas, the licensee shall address the corresponding regulatory requirements. It is important to note that it is the responsibility of licence applicants to choose the nuclear power plant technology that best meets the safety goals. Every licensee has its own Management System structure, although they can share common characteristics. Applicable codes and standards as well as applicable regulatory framework documents are to be considered, addressed and referenced in Management System documents.

The licensing steps License to Prepare Site, License to Construct and License to Operate can be conducted in series or in a staggered parallel manner. This is the decision of the licensee and depends on their business plans, licensing schedule and state of readiness. The Environmental Assessment process is an integral part of licensing as of late 2012 (in the past it was a separate process) and is, in part, used to inform the site suitability argument made by the applicant in licensing.

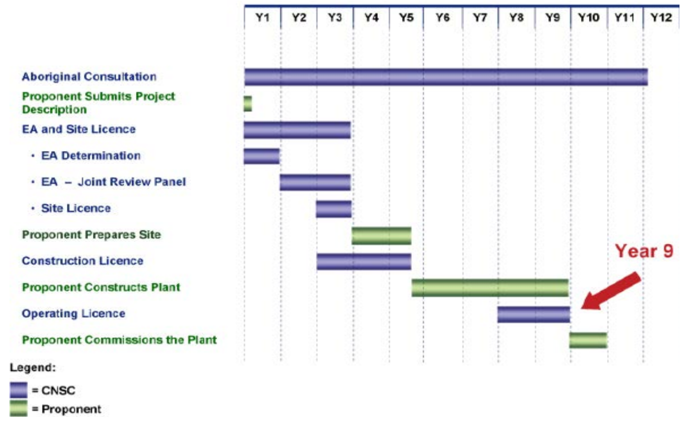

FIGURE 6
EA and Licensing Process for New Nuclear Power Plant in Canada [24]

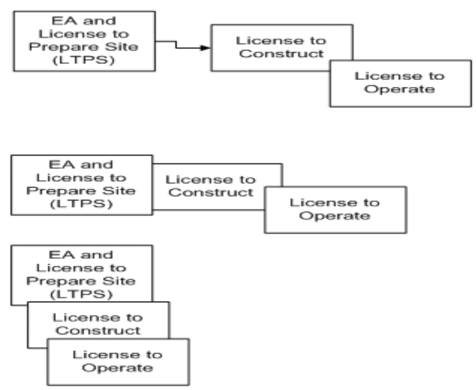

FIGURE 7
Possibilities for Canadian Licensing Process Handling [23]

An EA for a new nuclear power plant, conducted either at the comprehensive study or panel review level, provides significant opportunities for public participation. In Canada, the main focus of very small reactors is in northern Canada's isolated off-grid locations. Because of this, the very small reactors, from 2 to 25 MWe output with combined electricity production and steam plant, are of special interest. Larger SMRs are being considered for more traditional power plant roles in more southern parts of Canada. The licensing process for SMRs is currently anticipated to be the same as for NPPs with efficiencies gained from simpler reviews and staggered parallel licensing. For example, for very small SMRs, the CNSC anticipates that licensees will apply for a site license followed shortly thereafter by a combined construction and operation licence application. It is possible that for the n^{th} build, an applicant will apply for all three licences in parallel and this may reduce licensing time by a number of years. Much of the licensing time is composed of the public hearing process.

The different licensing processes can be selected according to the status of licencee readiness. In the first alternative, the

EA and LTPS are handled in parallel, and can be compared with the US practice of a Site Licence. This site licensing can still be independent from the technology selection of execution project. The third alternative is the presumption of the general licensing process and used for the FOAK-type NPPs. This type of licensing is in progress for the Darlington New Build project and this is used to estimate the duration of the licensing process (in Figure 6 above). The fourth option is typically used if expanding the site with identical technology to what is already there or for identical copies of the reactor design to be placed on new sites where the applicant has all of the information necessary to develop a safety case for operation immediately following construction (same concept as US COL process).

France

In France the Flamanville 3 case has been used as the basis of the schedule and duration of different licensing steps. These licensing steps and durations are presented in Figures 8 and 9. This project is the only new-build project in France that can be used as a basis, but in future projects the durations might be quite different from the Flamanville 3 project. Flamanville 3 can be treated as a first-of-a-kind project for the EDF Group. The project was authorized in the previous legal context. The new French law was enacted in 2007 and this may result in a different time-frame for licensing.

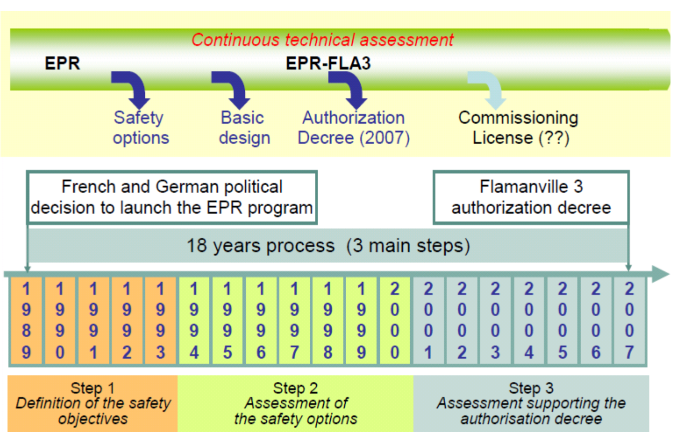

FIGURE 8
Licensing Steps Schedule of Flamanville 3 Project [14]

Future projects are to be implemented according to the new law (since 2007). Because this new process has not proceeded yet in any project, it cannot be used in this study. The next planned French NPP project is the Penly 3 project.

The Penly 3 project was "approved" by the previous parliament. However, the new elected parliament (elected in June 2012) will debate this and will probably organize "a national debate" on energy policy. The confirmation of the

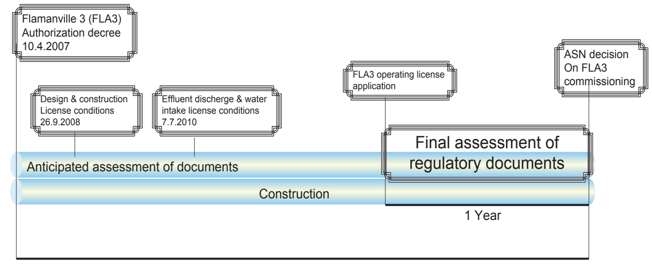

FIGURE 9
Licensing Milestones for Flamanville 3 Commissioning [14]

Penly 3 project is not assured in this context [15].

An additional step, the so-called "public debate", needs to be organized for each important investment project. This public debate is only a consultation with no blocking rights. This may take from several months up to one year to organize and conclude.

After these steps, the investor can take a decision on investment and start the actual licensing process. The instruction phase was short for Flamanville 3 (less than one year). In a more standard context and under the new legal regime it can be assumed that the instruction phase duration could be closer to two years. The approximations of project durations in France are based on interviews with licensing specialists in France. This is the same kind of direction that is seen in Finnish NPP projects.

The end of the process is presented in the Figures 8 and 9; the "Dossier pour la mise en service" (Operating licence application) is submitted one year in advance of fuel loading. If the process is handled in a "normal" way, the open issues have been addressed earlier and the duration of this stage should be reasonable.

UK

In the UK, HSE has developed the licensing process as a two-phase process; the first phase, called the Generic Design Assessment (GDA), is a review of the safety features and ultimate acceptability of a nuclear reactor design as the basis for granting a nuclear site license. The GDA is not formally mandatory, but in practice, it simplifies the overall process. If successful, the GDA leads to the issue of a Design Acceptance Certificate (DAC) by the Health and Safety Executive (HSE). The second phase involves an applicant seeking a nuclear site license to construct such a reactor at a specific site (or sites).

The framework is presented in Figure 10.

The duration of licensing in the UK is based on the on-going licensing processes of EPR and AP1000. The GDA process is predicted to take between two years nine months and

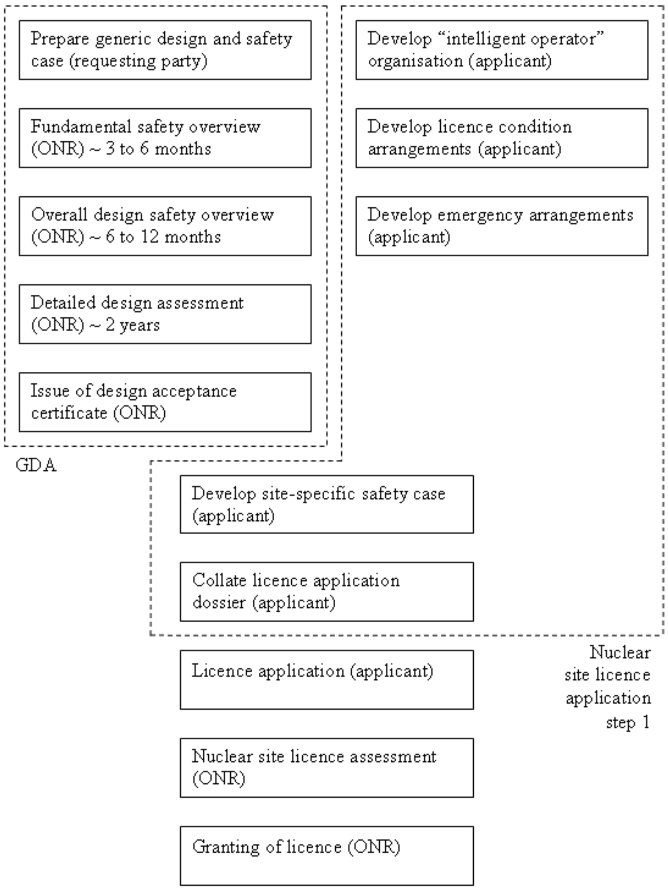

FIGURE 10
Duration of the Different Licensing Steps in the UK [16]

three years and six months, as presented in the Figure 10. Up to now, this has not been correct. For EPR and AP1000, the fundamental safety overview commenced in September 2007. The interim DAC (Design Acceptance Certificate) and a list of GDA issues were issued for both designs in December 2011. The GDA issues are to be closed and after that the (full) DAC will be issued. The first DAC was awarded for UK EPR by ONR on 14 of December 2012 [26]. The licensing process has taken longer than expected [16].

In more detail:

• The fundamental safety overview lasted from September 2007 - June 2008, taking nine months (about 50% longer than predicted).
• The overall safety design overview lasted from June 2008 - November 2009, taking 17 months (about 50% longer than predicted).
• The detailed design assessment began in November 2009 and is ongoing (the interim DAC was issued in December 2011, after two years and one month).

The GDA process has taken longer than was expected. It needs to be noted that this has been the first application of the GDA process in the UK, and so the lessons learned can affect the duration of the next GDA process. Detailed information about UK licensing process durations has been obtained through interviews with specialists.

The document describing the process for GDA also suggests that the nuclear site license application will take between six and 12 months. However, it has been suggested that in the case of Hinkley Point C licensing, 18 months would be more realistic [17].

Aviation industry licensing

The aviation industry licensing processes can be used as a basis for comparison because the safety criticality of the aviation industry is comparable to that of the nuclear field. There are, however, differences between these industries that should be acknowledged. One of the main differences is the transportation of objects, which is the case in the aviation industry and not in the nuclear field. Another major difference is that of public perception which for the nuclear industry is an order of magnitude greater than that of the aviation sector.

Even if not all aspects of aviation industry licensing are eligible to be adapted to the nuclear field, some of them are.

In civil aviation, the International Civil Aviation Organization (ICAO) is one of the main stakeholders of licensing. This is based on the Chicago Convention on International Civil Aviation [18].

The licensing of an aircraft is based on a Type Certification and registration. Type Certification is awarded to the designer or manufacturer by the national aviation authority. Type Certification is issued primarily in the country of origin, and after that in all the countries where an aircraft of that design is to be registered. In addition, every single aircraft needs an Airworthiness Certificate. These certificates can roughly be compared with nuclear licenses, Type Certification compared with Design Certification and Airworthiness Certification compares with a specific NPP operating license.

The general international framework for regulatory cooperation is provided by the Chicago Convention on International Civil Aviation, which also includes minimum safety standards. In addition to the minimum safety standards, countries have complementary national standards for safety [5].

Type Certification is not automatically valid internationally, but the authorities collaborate through bilateral agreements.

When reviewing a design, the corresponding authority finds a group of experts from aviation authorities of other major countries for the design review. This provides a Type Certificate in all the involved countries. When regulators from other countries review the design, they concentrate on their own national specific requirements to validate the Certification. After the Type Certificate is issued the further design work concentrates on design improvements, which are introduced in Airworthiness Directives [5].

With regards to the EU, in 2002 the European Aviation Safety Agency (EASA) was created to promote "the highest common standards of safety and environmental protection in civil aviation in Europe and worldwide. It is the centerpiece of a new regulatory system which provides for a single European market in the aviation industry"[27]. The agency's responsibilities include:

• Expert advice to the EU for drafting new legislation;
• Implementing and monitoring safety rules, including inspections in the Member States;
• Type-certification of aircraft and components across all EU member states, as well as the approval of organizations involved in the design, manufacture and maintenance of aeronautical products;
• Authorization of third-country (non-EU) operators;
• Safety analysis and research.

The history of EASA lies in voluntary cooperation between national aviation authorities, who founded the Joint Aviation Authorities (JAA) in 1970 [5].

The requirements of EASA are formulated such that they consist of two parts: a general requirement and the declaration for the requirement. This kind of approach keeps the actual requirement at a general level and does not direct the design in a certain direction. The declaration part gives more specific information about how the requirements can be fulfilled [19].

A large amount of work was done to harmonize the European national standards with each other and also harmonize the European standards with US standards. Later the focus has shifted towards more integrated collaboration of regulators and a common approach to certification, which was included in the Cyprus arrangements 1990 [5].

In the aviation industry, a high degree of confidence has been achieved between the USA, Europe and a few other countries. This could be the direction that the nuclear industry needs to develop with more harmonization between licensing processes such that nations can recognize and even accept licensing conclusions from other licensing jurisdictions. It is recognized that this needs to be done in such a manner so as to continue to recognize a nation's sovereign right to perform an independent licensing review that will withstand their citizens' (public and legal) scrutiny. Some initial steps towards this have been taken, e.g., in the MDEP framework, where the UK, Finland and France have shared their assessments of the same design.

Findings

SMR prospective licensees are considering scenarios that may utilize many identical modules (reactors) in one unit and probably many units constructed (and licensed) staggered in the same site. In this kind of approach, licensing needs to be repeatable via a streamlined but robust process (if the licenses of every module are separated). The part of licensing that is project specific should be minimized; the repetition of the same issues in every project should be avoided in the case of identical reactor designs. In principle, the licensing process in the USA has many features that would be suitable for SMRs. NRC has presented different alternatives for SMR licensing. The Alternative 2, the Master Facility License and Individual Reactor Module Licenses [7] would probably also be practical with slight modifications in a small nuclear country like Finland. The challenge in Finland and in many other European countries is that the approach is in general very different from the existing one and its introduction would be a major effort.

In Figure 11, the estimated durations of licensing processes in the countries studied are presented assuming that licensing would start at the beginning of 2013. This chart shows that all the licensing processes are quite long compared to the general idea of SMR deployment schemes. For these approximations, the lengths of recently issued licensing processes have been used as the basis, but it should be noticed that the designs have been first of a kind. This chart should not be used to compare the licensing process durations with each other, because of the differences of the licensing step contents and the uncertainties of the analysis.

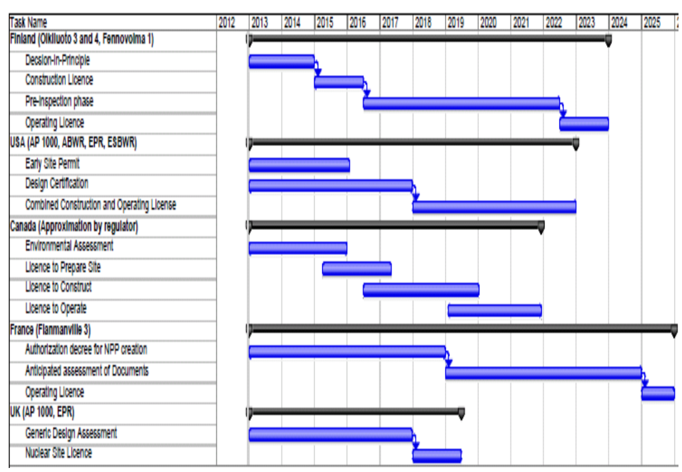

FIGURE 11
Overview of Licensing Processes Duration in Different Countries

There are differences in the contents of licensing steps in each country and therefore the comparison cannot be straightforward. Also the schedules of the handling of certain licensing issues are different; certain specific issues can be handled in one licensing process at a very early stage, while in another licensing process it can only be handled much later. There is no one right way for licensing, but each approach has its benefits and challenges. Also, there are differences in the depth of the review that is done by the regulator and the use of Inspection Organizations for the regulators. The licensing process in the USA is quite well known and it has been used as the basis for licensing process development in many countries. One of the features of the licensing in the USA is that the 1954 Atomic Energy Act (AEA) does not contain requirements concerning nuclear safety, but since the NRC was founded due to the Energy Reorganization Act of 1974, the NRC is empowered to establish specific regulatory standards. The NRC issues rules defining binding requirements and regulatory guides to provide guidance concerning the application of the rule.

The Finnish licensing process includes a political step: the Decision-in-Principle (DiP) is basically a political decision that is granted by the Council of State and has to be ratified by the parliament. The DiP process currently indicates the number of NPPs to be built or the number of reactors, because currently only plants with a single reactor in a plant have been the case. This may not be suitable for SMRs with multiple reactor modules in one plant and serial construction of many plants at a site. The DiP process is, to some extent, comparable with the French political decision PPI (Plan Pluriannual d'Investissement - multiyear investment plan) included in the new French implemented in 2006. The big difference is that the Finnish process can be initiated only by the industry, whereas the French process is initiated and conducted by the State.

In the Finnish process, the main review of technical issues is carried out during the Construction License phase and also as part of regulatory supervision during the construction (the so-called Pre-Inspection phase). The design certification issues, as well as site-specific issues, are all addressed in the CL and Pre-Inspection phases. The final configuration is reviewed for the Operating License.

The UK is, in practice, moving forward to a two-stage nuclear licensing process. In the UK the licensing is based on the Site License, but the Generic Design Assessment (GDA) has taken a comparable place in nuclear licensing to the Design Certification in the USA. In the UK, no specific set of general regulations for the safety of NPP exists. The basis of a safety case lies on a risk assessment and application of the ALARP (As Low As Reasonably Practical) principle, while in many countries the principle used is ALARA (As Low As Reasonably Achievable). The main idea of UK licensing is to have the licensee highly responsible for safety, while the regulator issues only large complexes and guiding principles. In other countries, the regulator commonly examines detailed design issues; the regulator's responsibility in the UK is to review the level of safety due for the presented safety case.

Canada has many similar licensing features to the UK, having the licensee highly responsible for safety, and also having the Vendor Design Review, which can be roughly compared to the GDA.

In addition to the licensing process differences, the level of details varies from one country to another. The depth of details has been studied in earlier studies and the estimations can be presented [22]. One of these studies has indicated differences in the level of supervision and inspection in different countries. The studied countries' (France, Finland, Canada, the USA and the UK) different types of approach to licensing have been discussed, as in some countries the licensee writes his own rules preparing the safety case for the regulator, while other countries' regulators have their adjusted set of regulations for licensing.

This analysis and study of different licensing features gives a perspective of the benefits and challenges of different licensing steps for SMR taking into account SMR-specific features, especially focusing on a country like Finland.

An early political decision is an effective way to reduce the political licensing risk. The Finnish licensing process includes a Pre-Application Review (Decision-in-Principle), including early site issues, which could be considered as also being practical for SMR licensing, ensuring political acceptance in advance. Slight modifications to the current practice regarding the format and extent of the decision might be needed in order to take into account the differences in project implementation between multiple SMRs versus a single large reactor.

A separate site approval process (e.g., the Early Site Permit in the USA) could have benefits for SMR licensing because it can be applied separately from other licensing steps and it is quite flexible. In Finland, this process is currently included in the DiP process and the CL process. This process has been used quite successfully and there is no reason to expect that it would not also suit SMR licensing in the Finnish licensing environment.

A Design Certification-type process would have benefits for SMR licensing in a country that is anticipating large numbers of identical modules to be constructed and operated. In some SMR cases, such as for the NuScale design, Design Certification could even be applied on a module-by-module basis recognizing that module designs evolve over time.

The Combined Construction and Operating Licence (COL), or Master Facility Licence in this study, would be a suitable process in principle for SMRs with some possible modification of the contents. If the Master Facility License was preceded by a separate site license and design approval, the MFL could contain only the project specific issues, external hazards, common cause failures and other possible effects that are common to all the modules. The repetition of reviewing module specific issues in every project would be minimized.

The final step of the licensing process in the USA is the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) process. The inspections, tests, analyses and acceptance criteria could be handled in SMR licensing within the licensing steps in the Requirements Management (RM) process V-curve [20]. Requirements Management is part of Systems Engineering, which has been developed in the software modeling field and used widely, e.g. in the aerospace and aviation industries [21]. Requirements Management or Requirements Engineering is a process that continues throughout the lifetime of a system or a power plant. The requirements are defined, elicited and documented at the beginning of a new NPP project. During the design, construction and operation, the requirements can change and new requirements can be elicited. The requirements exist and need to be managed over the lifetime of the system of a power plant. NPP licensing can be contrasted with Requirements Management of the licensing requirements. The use of Requirements Management as a licensing process implementation needs also an adaptation of regulations to form well-defined requirements with planned verification and validation processes and methods.

The current practices of validation and verification vary largely between countries. Some of the regulators execute very detailed technical inspections, while other regulators focus more on the processes and procedures. Also, the use of Technical Support Organizations (TSO) as support for regulators varies largely from one country to another. Some of the regulators base their decisions more on the TSO reviews, but other regulators see the need to have their own competence at a higher level in many areas. European TSOs have established a cooperation network the European Technical Safety Organization Network (ETSON), to provide more organized cooperation [5]. This network is also focusing on solving the challenging situation of many competing safety standards used in the nuclear field.

Possible high-level elements of a licensing process for SMRs are presented in Figure 12. In this kind of approach, certain licensing steps can be multiplied as required in modular design NPPs with many reactor modules and SMRs when many units are to be built staggered or in series. A Design Certification for every module could be a practical approach for SMRs with more than one reactor module in one plant (for example 12 modules in the NuScale design). While building many SMRs at the same site, only the necessary licensing steps could be selected and/or revised (as site approval), the other parts could simply be multiplied when necessary.

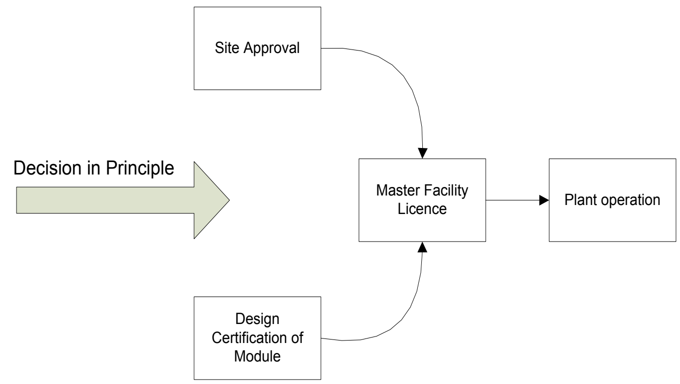

FIGURE 12
Possible Elements of a Licensing Process for SMR

The SMR-specific features were presented earlier in the paper. The focus should be on minimizing overlaps in the licensing process. The modularity and serial construction indicate certain licensing features to be more feasible to SMR licensing than others.

Possible licensing steps that could be practical for SMR licensing include:

• Decision-in-Principle
An upfront "political license" like the current Decision-in-Principle in Finland has turned out to be a good practice in reducing the political risk during the later stages of a project. This approach could be expected to also work equally well for small reactors. Slight modifications to the current practice, e.g. to conditions on the number of units, thermal power and the validity of the permission, might be needed in the case of SMRs.

• Site Approval
A site approval process similar to, for example, the Early Site Permit practices in the USA could be quite well suited to SMR licensing. It could be applied separately from other licensing steps. It should be noted that, in Finland, this process is currently included in other licensing steps and it has also been a well-suited practice in the Finnish case.

• The Standard Design Certification of a module
The Standard Design Certification type of a licence has many features that suit SMRs well. Some modifications to the contents of the Standard Design Certification could be applied for SMRs, for example issuing a design certificate for a single module. The Design Certificate could be a certification of the detailed design (almost 100% design of the module

ready) of the SMR module. The modules are assumed to have independent safety systems and, from a safety point of view, they are not dependent on the other parts of the plant. The module safety issues or design would not be reviewed again as a single module during any specific NPP licensing.

• Master Facility Licence

The Master Facility Licence (with similarities to the COL in the USA) also has many suitable features for SMR licensing. Some modification could be indicated if the Design Certificate contained only module certification, and then the Master Facility License would concentrate on safety issues that are common to the whole plant (e.g., external hazards and common cause failures). This approach would make this licensing step quite light and straightforward. The unit or project-specific part (Master Facility License) would be minimized to reduce the repetition in the licensing process.

Discussion and conclusions

While reviewing the feasibility of different licensing to SMRs, it has been noted that a technical design approval process that would already increase the licensing certainty before the start of any specific implementation project would seem to fit well to SMR-specific features. The licensing features presented in this study have been discussed keeping in mind the possibility of reducing the licensing risks while maintaining safety at the same high caliber expected of the industry in general. The focus of the technical licensing has been put into the front end of the SMR construction project. In this way, the licensing risk in later phases and delays to the construction project could be avoided.

The SMR-specific issues, like modularity, mass production and standardized design, should be taken into account while planning licensing to optimize the process. This is the way to make the SMR implementation and licensing feasible and to improve competitiveness of the SMRs against other energy production means in Europe in the future.

Licensing processes in several European countries have been under development in recent years, following the prospects for revival of nuclear new builds since the early 2000s. There are licensing processes for new builds ongoing in many projects, but none of them have been fully completed. The learning process of nuclear licensing will be seen in the future due to new NPP projects and the effectiveness of licensing will increase through experience.

In future studies, the depth of each licensing process and licensing step needs to be reviewed in more detail. Some of the specific questions are related to the use of Inspection Organizations (IO). The definition of an IO in different countries and the approach that is used with IOs should be focused on in future studies. Also, the requirements management and systems engineering process has already been introduced in the nuclear field to some extent, but this approach needs more focus in future studies, especially if, in Europe, this approach is to be applied at a large scale. Lessons learned from other industry fields should be appreciated and applied.

Appendix - Research methodology

Our aim is to seek answers to two questions:

1. What are the main features of SMR licensing that differ from current large NPP licensing?
2. What are the features in different licensing processes that are most suitable for SMRs?

The structure of this paper is the following. After a literature study, the special licensing features of SMRs are analyzed. Then, the following chapter describes the comparison of licensing processes. The Findings section reports the main findings of the study and gives some suggestions for SMR licensing process development in a country like Finland. Finally, some discussion of the findings and the study is issued and possible further research in this area is presented.

Technical issues and related requirement functions are not included in this study; they have been studied more extensively elsewhere, e.g. the need to adjust or change some of the technical requirements, such as the requirement for the use of station blackout emergency diesel generators (EDGs) [6].

Figure 13 shows the two directions of licensing and the right line (Requirements fulfillment) is the scope of this study.

Based on publicly available information, we have studied licensing practices and regulations in selected countries. Regulators' and international organizations' presentations and publications have been used as the basis of the study. The availability of information varies from country to country, some of them having a very open information policy. The difference in the level of available information has been noted and the countries with less public information available have been studied by carrying out interviews with some licensing specialists.

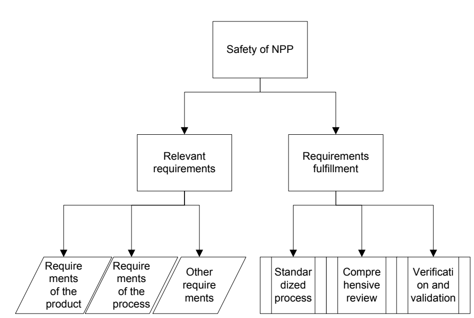

FIGURE 13
Licensing Functions Categorization in this Study

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FULL ARTICLE

A MONTE CARLO METHOD FOR ANALYZING MIXED-LATTICE SUBSTITUTION EXPERIMENTS USING MCNP

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Abstract

Critical experiments involving a small region of test fuel substituted into a reference lattice have traditionally been analyzed using diffusion codes to extract lattice physics parameters of the test fuel such as the critical buckling and the associated bias in the calculation of k_{eff}. A method that was first developed in 2006 uses a version of MCNP that was modified to allow the analyst to selectively change fission neutron production in various parts of the model. This paper describes the modification made to MCNP, demonstrates how the substitution experiment analysis is done through several examples using data from the ZED-2 critical facility, and finally, quantifies the expected uncertainties in the method.

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1. Introduction

Small critical facilities are used to investigate the behavior of assemblies of fissionable material. Typically, a single parameter, (e.g., mass, concentration, or some physical dimension) is varied until the assembly is exactly critical (k_{eff}=1.000). The experiment can be used directly to validate computer codes and data by modeling the experiment and comparing the calculated k_{eff} to unity. Many such experiments have been done to support the criticality safety discipline.

The design and licensing of nuclear reactors requires experimental benchmark data or validation data from larger and more comprehensive facilities such as the ZED-2 critical facility. A mock up of the reactor lattice being studied is assembled in the critical facility with fuel and other lattice parameters that are as similar as possible to those of the actual reactor. In a simplified or “clean” experiment, all of the fuel is the same, i.e., all “test” fuel. Measured results such as critical dimensions, critical buckling, and neutron flux profiles can be compared directly to calculated values.

However, for various reasons, it may be neither possible nor necessary to assemble a critical lattice using test fuel alone. In these cases a lattice of “reference” or “driver” fuel is used to produce a critical assembly. The test fuel replaces the reference fuel within some limited region, usually in the centre of the reference lattice where neutron flux and importance are the highest. These are called “substitution” experiments. Unlike clean experiments, substitution experiments require further analysis to extract the lattice parameters of interest; that is, to isolate the properties of the test fuel from those of the mixed lattice of test fuel and reference fuel. The process of analyzing substitution experiments to isolate and extract the properties of the test fuel is commonly known as *substitution analysis*.

Various substitution analysis methods have been developed and used within the international community and at the Chalk River Laboratories, going back to the 1960’s [1–3]. One new method that was developed and tested at CRL beginning in 2006 [4] involves the use of a modified version of MCNP [5]. The remainder of this paper discusses the MCNP-based substitution analysis method, along with sample results for a variety of substitution experiments performed in the ZED-2 critical facility.

Section 2 provides some examples of substitution experiments that were conducted in ZED-2; Section 3 presents the theory and its application to analyzing these experiments; Section 4 shows some examples of applying these analysis methods; and a discussion and the conclusions are presented in Sections 5 and 6, respectively.

2. Substitution Experiments in ZED-2

The ZED-2 critical facility contains a 3-meter by 3-meter vertical cylindrical vessel in which fuel rods or fuel channels are vertically suspended. Heavy water is introduced into the vessel to act as a moderator, and make the assembly critical; reactivity is controlled by fine adjustments of the moderator level. Fuel channels can be filled with a variety of “coolant” materials (e.g., D_2O, H_2O, air, CO_2, He, organic fluids, etc.). Neutron-flux and reaction-rate distributions are measured using various neutron activation foils. Lattice configurations can be either triangular/hexagonal or square, the lattice pitch is continuously variable, and the number of fuel rods or channels is incrementally variable.

Figure 1a shows a top view schematic of a typical ZED–2 substitution experimental setup that has a hexagonal lattice with seven test fuel sites in the centre surrounded by the reference fuel sites. The attributes that can differ between test fuel sites and reference fuel sites include fuel geometry or fissionable material, channel type or coolant, and fuel temperature, but not the moderator, or the lattice configuration or pitch. Examples of other substitution experiment layouts are shown in Figures 1b and 1c, and several of the test and reference fuels typically used in these experiments are shown in Figure 2.

Substitution experiments are used in the ZED-2 critical facility when the available number of test channels or the amount of test fuel is limited, or a lattice of pure test fuel cannot be made critical within the facility. A Monte Carlo-based technique used to extract the desired reactor physics parameters for the test fuel from these mixed-lattice experiments is the subject of the next section.

3. Analyzing Substitution Experiments

3.1 Theory

Steady-state neutron behaviour in a system containing fissionable material is governed by the time-independent Boltzmann transport equation, which can be expressed in operator notation as

$$\mathrm{M}\Phi(\vec{r},E,\hat{\Omega})=\mathrm{F}\Phi(\vec{r},E,\hat{\Omega})+S(\vec{r},E,\hat{\Omega}) \qquad (1)$$

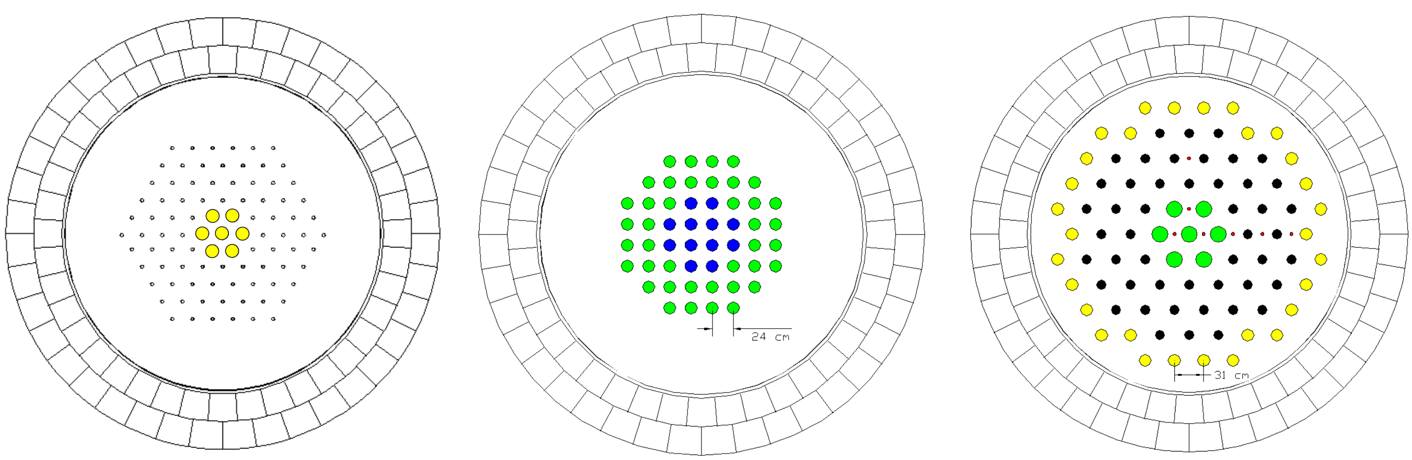
where $\vec{r}$ is the position vector, E is energy, $\hat{\Omega}$ is the unit direction vector, M is the migration and loss operator, F is the fission neutron source operator, and S is an external neutron source that is independent of neutron flux. Integration over volume, energy, and direction is implied. These operators are defined as

$$\begin{aligned}\mathrm{M}\Phi=&\,\hat{\Omega}\cdot\nabla\Phi(\vec{r},E,\hat{\Omega})+\Sigma_t(\vec{r},E)\Phi(\vec{r},E,\hat{\Omega})\\&-\int_0^\infty dE'\int_{4\pi}d\Omega'\Sigma_s(\vec{r},E'\rightarrow E,\hat{\Omega}'\cdot\hat{\Omega})\Phi(\vec{r},E',\hat{\Omega}')\end{aligned} \qquad (2)$$

and

$$\mathrm{F}\Phi=\chi(E)\int_0^\infty dE'\nu(\vec{r},E')\Sigma_f(\vec{r},E')\int_{4\pi}d\Omega'\Phi(\vec{r},E',\hat{\Omega}') \qquad (3)$$

where $\sum_t$ is the total neutron cross section, $\sum_s$ is the neutron scattering cross section, v is the average number of prompt plus delayed fission neutrons produced per fission, χ is the fission neutron energy distribution, and $\sum_f$ is the fission cross section.

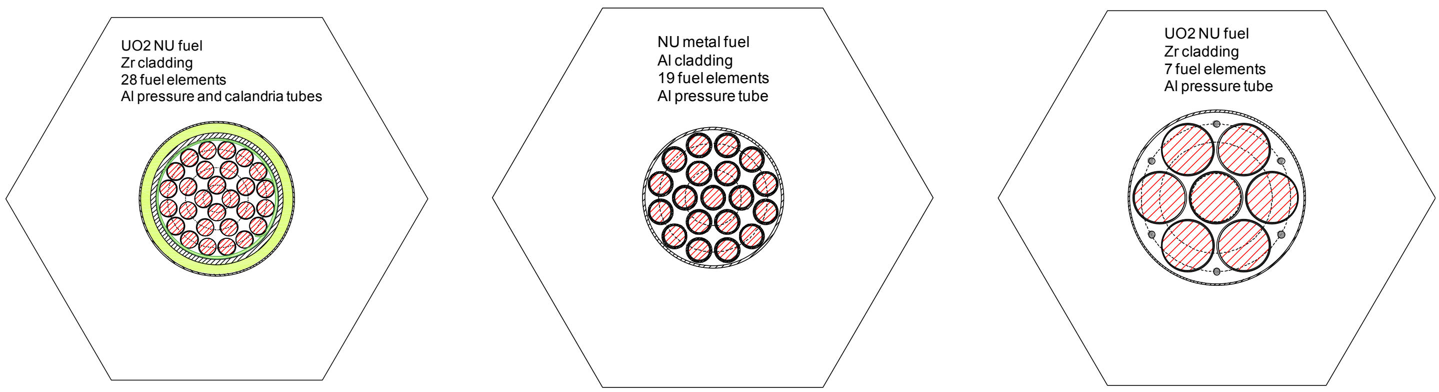


(a) Tight hexagonal pitch, 28-NU test fuel, ZEEP reference fuel (Section 4.1).

(b) 24 cm square pitch, 42-LEU test fuel, 43-SEU/RU reference fuel (Section 4.2).

(c) 31 cm hexagonal pitch, 28-NU, 19-UM, 7-NU test fuel, ZEEP and 28-NU reference fuel (Section 4.3).

FIGURE 1.
Mixed-Lattice Arrangements for the Sample Substitution Experiments



(a) 28-NU Test Fuel (b) 19-UM Test Fuel (c) 7-NU Test Fuel

FIGURE 2
Sample Test-Fuel Bundle and Channel Types in a Hexagonal Lattice Cell

If *S*=0 in Equation (1) and the left and right sides are not equal, then the flux is not constant in time and the equation has no steady state solution except the trivial solution of Φ=0. In order to allow a steady state, nontrivial flux solution an eigenvalue is introduced and Equation (1) is rewritten as

$$\mathrm{M}\Phi = \lambda \mathrm{F}\Phi \tag{4}$$

where λ is the eigenvalue used to balance the equation and Φ is the eigenfunction, called the λ-mode flux, frequently designated by Φ_λ. If Φ_λ is positive everywhere in space then λ is the fundamental eigenvalue. There is always a solution if the system being modelled contains any fissionable material.

Since the neutron multiplication constant[1] *k* is defined as the ratio of fission neutron production to total neutron loss, then

$$k = \frac{\mathrm{F}\Phi}{\mathrm{M}\Phi} = \frac{1}{\lambda} \tag{5}$$

and Equation (4) is rewritten in the more familiar form

$$\mathrm{M}\Phi = \frac{1}{k}\mathrm{F}\Phi \tag{6}$$

Although Equation (6) does not appear anywhere in MCNP nor is the neutron flux explicitly calculated, this is the form of the Boltzmann transport equation that MCNP is solving in a criticality (eigenvalue or KCODE) problem. Since λ (or *k*) is a constant, it could be taken inside the integrations that form the F operator in Equation (3) and interpreted as the factor that must be applied to one of the components of F throughout the system to make it critical. Traditionally the factor is viewed as being applied to *v*, thus, changing the effective average number of neutrons produced per fission.

If we introduce a similar adjustment factor called the *NPCF* (Section 3.2), which is applied to all fissionable materials in the model, Equation (6) then becomes

$$\mathrm{M}\Phi = \frac{NPCF}{k_1}\mathrm{F}\Phi \tag{7}$$

where $k_1 \neq k$ unless *NPCF*=1. It is easy to see that if

$$NPCF = \frac{1}{k} \tag{8}$$

then k_1=1, and the adjusted model is critical. That is, the value of *NPCF* that must be applied to all fissionable materials to make the model critical is equal to the inverse of the multiplication constant *k* from a calculation without the *NPCF*. Thus, the *NPCF* for the reference fuel to be used in a substitution experiment can be determined in a single calculation using Equation (8) with the value of *k* determined via simulation of an experiment with a whole core of that fuel type.

Now if different *NPCF* values are applied to different fissionable materials in the model, then Equation (7) becomes

$$\mathrm{M}\Phi = \frac{1}{k_2}\left(NPCF_1 \cdot \mathrm{F}_1 + NPCF_2 \cdot \mathrm{F}_2 + \ldots\right)\Phi \tag{9}$$

where the volume integrals in F_1, F_2, ... are over different regions of space, and Equation (8) is no longer valid. It is the capability of specifying a different *NPCF* for different fuel types that allows the modified version of MCNP to be used for substitution experiment analysis. However, note that the *NPCF* for the test fuel in a substitution experiment must be determined by iteration after the *NPCF* for the reference fuel has been applied. If a single *NPCF* is applied to the whole core, *k* changes, but everything else remains unchanged including the flux shape. The usefulness of the

[1] *k* is used here to represent k_∞ if the model is infinite or k_{eff} if the model is finite.

method is the ability to use different *NPCF* values to remove the calculation biases in k for different fuel types, determined experimentally. In this case, altering one of several *NPCF* values will change the flux shape.

3.2 Application of MCNP to Substitution Analysis

The use of MCNP for performing substitution analysis is somewhat similar to earlier methods of substitution analysis using approximate deterministic neutron diffusion codes such as MICRETE (1-D/2-D, source-sink, 2-group diffusion [2] or CONIFERS (3-D, 4-group diffusion) [3]. However, the use of MCNP avoids the numerous approximations inherent in neutron diffusion methods.

In an MCNP eigenvalue (KCODE) calculation, the total weight of the fission source is preserved during each cycle. The absolute number of neutrons may fluctuate statistically, but the weight of each source particle is adjusted to preserve the total source weight. The details of this process are given in the MCNP manual [5].

The *NPCF* patch in MCNP simply multiplies starting particle weights in each selected region by a user-specified Neutron Production Correction Factor[2], or *NPCF*. The *NPCF* values are specified by material (i.e., fuel composition), and effectively change the average value of ν for the affected material: decreasing ν when *NPCF*<1, and increasing ν when *NPCF*>1. *NPCF*≤0 is not allowed, and *NPCF*=1 leaves the starting weights unchanged. The capability to specify a different *NPCF* for each fissionable material allows this patch to be used to analyse substitution experiments as described below.

The following steps, illustrated in Figure 3, are used to determine geometric buckling (B^2) for test fuel from a substitution experiment in ZED-2.

1. Simulate a full core of reference fuel with MCNP, for which the critical moderator height and other data are taken from a ZED-2 experiment. The value of the $NPCF_{ref}$ needed to make the core critical is equal to $1/k$ as calculated with no *NPCF*. A comparison of the calculated flux (or buckling) with that from the experiment can be used to confirm an accurate model.

2. Simulate the substitution experiment with MCNP where the critical moderator height is again taken from the experiment, using the $NPCF_{ref}$ calculated in Step 1 for the reference fuel and using a second $NPCF_{test}$ for the substituted region.

3. Adjust $NPCF_{test}$ in the MCNP simulation of the substitution experiment (holding $NPCF_{ref}$ fixed) and iterate until k=1.000 ± δk, where δk is the desired statistical uncertainty, typically less than 0.0001. A model check can again be done by comparing calculated and measured experimental fluxes or foil activation rates.

4. Set up an MCNP simulation of an un-reflected, bare cylindrical core of test fuel using $NPCF_{test}$ applied to all of the test fuel. Adjust the radial and axial dimensions to make the core critical (k=1.000 ± δk). It is known that the critical buckling of fuel will depend somewhat on the aspect ratio (height/diameter) of the core, due to the anisotropy in the neutron leakage caused by the fuel channels. Thus, to make a consistent comparison, the aspect ratio (H/D, or B^2_z/B^2_r), or one of the buckling components (B^2_z, or B^2_r) in the substitution analysis should be as close as possible to that found in the full-core experiments.

5. Use MCNP to compute the radial and axial distributions of neutron flux (or fission energy deposition rate) in the bare critical lattice of test fuel.

6. Fit cosine and Bessel functions to the axial and radial neutron flux distributions: $\phi(z)=A_0 \cos(\alpha(z-z_{max}))$ and $\phi(r)=C_0 J_0(\lambda r)$, respectively.

7. Use the best-fit parameters for the functions to obtain the axial and radial components of buckling, and hence, the total buckling for the test fuel: $B^2=\alpha^2+\lambda^2$.

8. In the situation where the test fuel is too low in reactivity (i.e., k_{inf} < 1) and has a negative geometric buckling, use booster fuel surrounding the test fuel with an appropriate $NPCF_{booster}$ also derived as in Step 1. The booster fuel may be regarded as a second reference fuel type which may or may not be the same fuel used in Step 1. The size of the test fuel region must be sufficiently large to minimize edge effects, and to ensure that there is a large asymptotic region where the neutron energy spectrum is as independent of spatial position as possible, i.e., the ratio of fast to thermal neutron flux is essentially constant. Fit a modified Bessel function to the radial distribution within the asymptotic region: $\phi(r)=C_0 I_0(\beta r)$. Data near the edge or outside of the asymptotic region must be excluded from the curve fits. The total buckling for the test fuel is then: $B^2=\alpha^2-\beta^2$.

9. This total buckling for the test fuel can then be used for the direct validation of a lattice physics code, such as WIMS-AECL [6, 7]. The critical dimensions of the bare core can be used for the direct validation of a whole-core physics code, such as RFSP [6]. In both the lattice physics and core physics codes, the value of k_{eff} is computed, using either input critical buckling or input critical dimensions.

Since the ultimate goal is to isolate the bias in a physics code prediction of k_{eff} for the test fuel, it is generally not

[2] The term "neutron production correction factor" was first used in the documentation describing the CONIFERS-based substitution analysis method [3]. The term was retained in the MCNP-based method for continuity since *NPCF* serves the same purpose in both codes. However, the term is somewhat misleading because it does not "correct" the neutron production – it is simply a constant multiplier, applied to the F operator in Equation (7) or (9), used to remove the calculation bias in k, regardless of the cause of that bias.

necessary to do a critical core size or critical buckling search for MCNP itself, since $k_{eff\text{-}test} = 1/NPCF_{test}$. Once $NPCF_{test}$ is found, then the bias in k_{eff} for MCNP for the test fuel is simply the difference in calculated k_{eff} with and without $NPCF_{test}$ applied. The bare core dimensions and associated radial and axial bucklings are mainly of interest for validation of other codes (such as WIMS-AECL, RFSP, etc.).

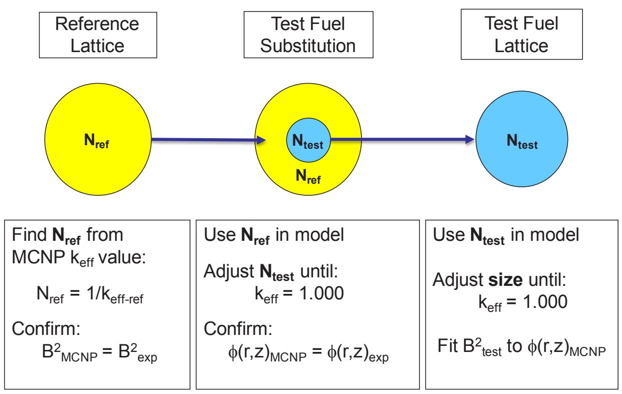


FIGURE 3
Schematic Representation of MCNP-based substitution analysis method (N is the neutron production correction factor, *NPCF*).

3.3 Uncertainty in *NPCF*

Although the derivation in Section 3.1 is rigorous, the resulting value of *NPCF* is not exact due to the statistical and propagated experimental uncertainties in calculating k for a known critical system. If the total uncertainty in k is denoted by δk, then from Equation (8) the uncertainty in the *NPCF* required to make the model critical is given by

$$\delta NPCF = \frac{\delta k}{k^2} \tag{10}$$

A substitution experiment is more complicated. Assuming there are two different fuel types in Equation (9) and that $NPCF_1$ is known from a previous full core experiment, then the uncertainty in $NPCF_2$ is given by

$$\delta NPCF_2 = \sqrt{\delta k_2^2 + \left(\frac{\partial k_2}{\partial NPCF_1}\delta NPCF_1\right)^2} \Bigg/ \left(\frac{\partial k_2}{\partial NPCF_2}\right) \tag{11}$$

where δk_2 is the estimated statistical and experimental uncertainty in k_2 and the partial derivatives are determined via sensitivity analysis using MCNP, i.e., $NPCF_1$ and $NPCF_2$ are changed independently and the impact on k_2 is determined. Ignoring the correlation between the full core and substitution experiments is conservative when estimating the uncertainty in $NPCF_2$.

3.4 Validation/Benchmarking of Substitution Analysis Method

The validation (or benchmarking) of the MCNP-based substitution analysis method can be performed three ways:

1. Buckling values derived from substitution experiments can be compared against buckling values determined from full-core flux-map experiments. Bucklings should be adjusted to common values of lattice temperature and moderator purity. This approach to validation has been used in the past [3].

2. The $NPCF_{test}$ (or $k_{eff\text{-}test}=1/NPCF_{test}$) determined from the analysis of substitution experiments can be compared against the $NPCF_{test}$ or $k_{eff\text{-}test}$ determined from the analysis of full-core experiments of test fuel.

3. An indirect, or reverse method (Figure 4) is used to determine $NPCF_{test}$ and $NPCF_{ref}$ from the analysis of full-core experiments of test fuel and reference fuel(s) *a priori*, and then to apply these values of *NPCF* in the subsequent MCNP analysis of a substitution experiment. If the values of *NPCF* are correct, then the MCNP calculation of k_{eff} for the substitution experiment should be unity, within expected uncertainties ($k_{eff}=1.000 \pm \delta k_{eff}$). This method is very convenient and has been used in recent studies [8].

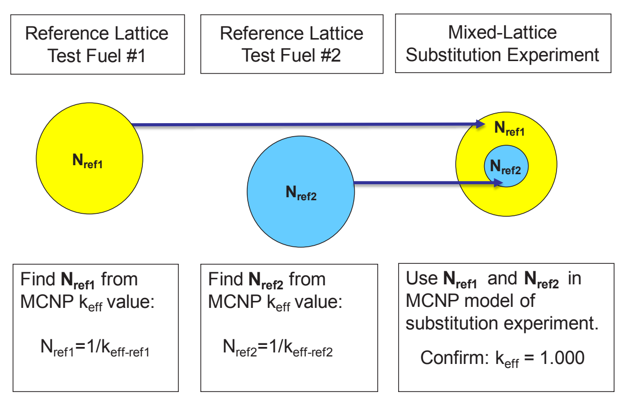


FIGURE 4
Indirect validation approach for MCNP-based substitution analysis method (N is the neutron production correction factor, *NPCF*).

4. Analysis Results for Sample ZED-2 Substitution Experiments

The following subsections describe sample results of using the MCNP-based substitution analysis method in the analysis of various substitution experiments performed in the

ZED-2 critical facility using a variety of lattice and fuel designs. The results described involve the use of one or more of the approaches to validation/benchmarking described in Section 3.4.

4.1 Tight Pitch Experiments with 28-NU Fuel

Tight pitch critical experiments were performed in ZED-2 using 91 ZEEP uranium metal rods at room temperature conditions (Figure 1a). The hexagonal lattice pitch was varied from 20 to 22.86 cm. The test fuel consisted of fuel channels (comprising aluminum pressure and calandria tubes) filled with 28-element natural uranium dioxide (28 NU) fuel bundles (Figure 2a) with five bundles per channel. The 28-NU fuel "coolant" material was either air or H_2O. More details on the ZEEP and 28-NU fuels can be found in [2] and [9].

Values of *NPCF* were determined first for the ZEEP reference lattices, and subsequently for the test fuel in the substitution experiments. Following the procedure described in Section 3.2, MCNP models of large regions of test fuel (boosted by ZEEP rods) were created, with the *NPCF* applied to the test fuel and with model dimensions adjusted to be critical. The flux distribution was calculated, and the data were curve-fitted to obtain axial and radial components of buckling. The results are shown in Table 1, along with critical buckling values derived from the flux distribution in ZED-2 experiments with large regions of test fuel, (boosted by ZEEP rods). Also shown are earlier results obtained using the CONIFERS-based substitution analysis method [3]. The buckling data is plotted for H_2O-cooled and air-cooled 28-NU fuel in Figure 5. While the CONIFERS method provides satisfactory agreement at larger lattice pitches (e.g., 22.86 cm), it is clearly demonstrated that the MCNP-based method gives better agreement in general. The MCNP-based substitution analysis results agree with the full-core results within the experimental uncertainties.

TABLE 1
Substitution Analysis Results for 28-NU Test Fuel in ZEEP/D_2O Hexagonal Lattices

Experiment Type	Lattice Pitch (cm)	Test Fuel/ Coolant	*NPCF* Ref	*NPCF* Test	B^2 Expt (m^{-2})	B^2 MCNP (m^{-2})	B^2 CONIFERS (m^{-2})
Full Core	20.00	–	1.00265	–	6.308±0.020	6.326	6.923
Full Core	21.59	–	1.00200	–	5.590±0.012	5.592	6.235
Full Core	22.86	–	1.00284	–	5.367±0.009	5.360	5.819
Substitution	20.00	28-NU/H_2O	1.00265	1.01050	-1.380±0.127	-1.411	-1.733
Substitution	21.59	28-NU/H_2O	1.00200	1.01057	-0.502±0.163	-0.524	-0.741
Substitution	22.86	28-NU/H_2O	1.00284	1.00752	-0.047±0.047	-0.052	0.081
Substitution	20.00	28-NU/Air	1.00265	1.00708	-0.318±0.114	-0.402	-0.581
Substitution	21.59	28-NU/Air	1.00200	1.00843	1.393±0.091	1.324	1.323
Substitution	22.86	28-NU/Air	1.00284	1.00876	2.322±0.068	2.259	2.294

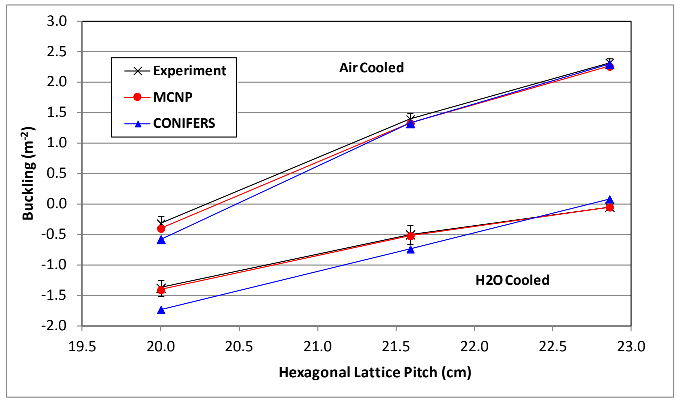


FIGURE 5
Comparison of substitution analysis methods with experiment for buckling of 28-NU test fuel in tight pitch lattices of ZEEP reference fuel (Section 4.1).

4.2 Square Pitch Experiments with 42-LEU Fuel

Substitution experiments were performed in ZED-2 using 52 channels containing 43-SEU or 43-RU fuel bundles (5 bundles per channel) at room temperature conditions (Figure 1b) as the reference fuel. The 43-SEU and 43 RU bundles were made with 43 fuel pins containing slightly enriched (0.95 wt% $^{235}U/U$) or recovered uranium (0.96 wt% $^{235}U/U$), respectively. The square lattice pitch was 24 cm. The fuel channels (pressure tube and calandria tube) were made of aluminum. Critical reference lattice experiments containing just 43-SEU/43-RU fuel were performed first, and these were either air-cooled or H_2O-cooled. Later, the central 12 lattice sites were replaced with fuel channels (comprising aluminum pressure and calandria tubes) filled with 42-element low enriched uranium (~1.7 wt% $^{235}U/U$) fuel bundles (42-LEU), with 3 to 4 bundles per channel, complemented by one or two 43-SEU bundles at the top. The 42-LEU fuel also contained a central neutron-absorbing pin made of zirconia/dysprosia/gadolinia/yttria. The 12 substituted channels with 42-LEU test fuel were cooled with either air or H_2O. The substitution experiments were set up such that the reference fuel and the test fuel had the same coolant (either air, or H_2O); however, it would have been perfectly acceptable to use an identical reference lattice for both air-cooled and H_2O-cooled test fuel substitution experiments. More details on the 43-SEU, 43-RU and 42-LEU fuel types can be found in [8, 10–12].

Values of *NPCF* were determined for the air-cooled and H_2O-cooled 43-SEU and 43-RU reference fuels from the MCNP modeling of the reference lattice experiments. Subsequently, substitution analysis was used to isolate the *NPCF* values for the 42-LEU test fuel from the MCNP simulations of the substitution experiments. Full-core experiments with 42-LEU test fuel were performed later and analyzed with MCNP to determine *NPCF* values. In addition, substitution analysis was used to obtain bucklings for the 42-LEU fuel (using 4-bundle, 5-bundle, and 6-bundle-high bare cores with radii adjusted to achieve criticality) which could also be compared with the values obtained from the full-core flux-map results.

Results are shown in Table 2. Buckling results are shown in Figure 6a. The uncertainties in the values of *NPCF* for the reference fuel are due to the statistical uncertainties in the MCNP calculations. The uncertainties in the values of *NPCF* for the test fuel are due to both the statistical uncertainties in MCNP and the propagated experimental uncertainties, using the formula shown in Equation (11). The uncertainties in the buckling derived from the substitution experiments are propagated from the uncertainties in the *NPCF* of the test fuel. The uncertainties in the buckling derived from the full-core experiments are due to the uncertainties associated with performing curve fits of the foil activation measurements.

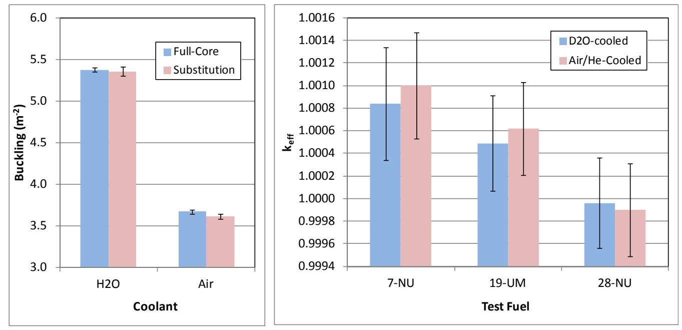
The values of buckling determined from substitution analysis differ from full-core results by 0.06 m^{-2} or less, and fall within the overlap of two standard deviations in both the substitution and full-core results. The buckling results from the substitution analysis shown in Table 2 were interpolated against axial buckling, which was set to be the same as that in the full-core experiments, and differed for the H_2O-cooled and air-cooled lattices.

TABLE 2
Substitution Analysis Results for 42-LEU Test Fuel at a 24-cm Square Pitch

Experiment Type	Ref Fuel/ Coolant	*NPCF* Ref	Test Fuel/ Coolant	*NPCF* Test	B^2 (m^{-2})
Substitution	43-SEU/RU/H_2O	1.01897	42-LEU/H_2O	1.01089±0.00128	5.358±0.053
Substitution	43-SEU/RU/Air	1.01640	42-LEU/Air	1.00883±0.00067	3.614±0.028
Full Core	–	–	42-LEU/H_2O	1.01038±0.00007	5.374±0.025
Full Core	–	–	42-LEU/Air	1.00901±0.00007	3.669±0.026

4.3 Hexagonal Lattice Experiments with Natural Uranium Fuel

Room-temperature full-core experiments were performed in ZED-2 using three types of test fuel in hexagonal lattices. The lattice for the associated substitution experiment is illustrated in Figure 1c. The lattices for the full-core experiments are not shown. The test fuels included 28-NU (described in Section 4.1), 19-element natural uranium metal (19-UM), and 7-element natural uranium oxide (7-NU), which are illustrated in Figures 2b and 2c, respectively. The substitution experiment (Figure 1c) included 55 ZEEP reference fuel rods, surrounded by 30



(a) 24-cm square pitch, 42-LEU test fuel, 43-SEU/RU reference fuel (Section 4.2)

(b) 31-cm hexagonal pitch, 7-NU, 19-UM, 28-NU test fuel, ZEEP and 28-NU reference fuel (Section 4.3)

FIGURE 6
Substitution analysis results for various test and reference fuels.

air-cooled 28-NU channels. Thus, the reference lattice comprised two different fuel types each with its own *NPCF* value in the substitution analysis. In the substitution experiments, the central seven ZEEP rods were replaced with 28-NU, 19-UM, or 7-NU fuel bundles, cooled by either air or D_2O.

Values of *NPCF* were determined for each of the test fuels, D_2O-cooled and air-cooled or He-cooled, with MCNP analysis of the full-core experiments. The experiments were conducted at various pitches and the results interpolated at a pitch of 31 cm (Table 3). The value of the *NPCF* for the ZEEP reference fuel was obtained from the MCNP model of the reference lattice with the 55 ZEEP rods and 30 air-cooled 28-NU channels. The *NPCF* for the air-cooled 28-NU was obtained from the previous full-core experiments of air-cooled 28-NU, and then applied in the MCNP model of the ZEEP reference lattice; then, the *NPCF* of the ZEEP rods was adjusted until the computed k_{eff} = 1.000.

The values of $NPCF_{test}$ determined from the full-core experiments and $NPCF_{ZEEP}$, $NPCF_{28\text{-NU-air}}$ were applied to the test fuel and reference fuel in the various substitution experiments (Figure 1c). Ideally, if the values of *NPCF* are applied to the various test fuels and reference fuels in the MCNP analysis of the substitution experiments, then the value of k_{eff} calculated by MCNP should be unity, within uncertainties. This is an indirect validation of the substitution analysis method, as discussed previously in Section 3.4 and illustrated in Figure 4. The results of the analysis of the substitution experiments with the $NPCF_{test}$ applied are also shown in Table 3 and Figure 6b. The uncertainties shown in Table 3 are due to the combined effect of statistical uncertainties in MCNP and the propagated experimental uncertainties.

It is found that k_{eff} differs from unity by no more than twice the ±0.5 mk estimated uncertainty[3]. The agreement for the 28-NU fuel is particularly good, differing from unity by less than 0.1 mk.

TABLE 3
Substitution Analysis Results for Three Test Fuels at a 31-cm Hexagonal Pitch

Experiment Type	Ref/Booster Fuel	Test Fuel/ Coolant	*NPCF* Test	k_{eff}
Full Core*	–	7-NU/D_2O	1.00727±0.00067	–
Full Core*	–	19-UM/D_2O	1.00772±0.00014	–
Full Core	–	28-NU/D_2O	1.00775±0.00008	–
Full Core*	–	7-NU/He	1.00682±0.00058	–
Full Core*	–	19-UM/He	1.00752±0.00029	–
Full Core	–	28-NU/Air	1.00791±0.00008	–
Substitution**	ZEEP/28-NU	7-NU/D_2O	1.00727±0.00067	1.00084±0.00050
Substitution**	ZEEP/28-NU	19-UM/D_2O	1.00772±0.00014	1.00049±0.00042
Substitution**	ZEEP/28-NU	28-NU/D_2O	1.00775±0.00008	0.99996±0.00040
Substitution**	ZEEP/28-NU	7-NU/Air	1.00682±0.00058	1.00100±0.00047
Substitution**	ZEEP/28-NU	19-UM/Air	1.00752±0.00029	1.00062±0.00041
Substitution**	ZEEP/28-NU	28-NU/Air	1.00791±0.00008	0.99990±0.00041

* Full-core experiments performed for 7-NU and 19-UM at various lattice pitches, showing no trend with pitch. Thus, *NPCF* at 31 cm based on an average of experiments at other pitches. The reactivity effect of He should essentially be the same as air, since both have little impact on neutron absorption or scattering.

** *NPCF* for ZEEP booster determined from MCNP model experiment where 7 central test fuel sites are replaced with ZEEP rods; $NPCF_{28\text{-NU-Air}}$ is known a *priori*; $NPCF_{ZEEP}$ is adjusted until k_{eff}=1.000.

5. Discussion

Computer capabilities have advanced to the point that whole-core Monte Carlo transport modeling for a broad range of static problems is now practical. This means that the more approximate two-step practice of using two-dimensional lattice transport calculations to produce data for three-dimensional whole-core diffusion calculations is no longer necessary for analyzing critical experiments. Historically, diffusion-based methods of analysing substitution experiments [2, 3] typically require several adjustable parameters to correct deficiencies in these methods at the interfaces between regions of different materials. Such corrections are unnecessary when using Monte Carlo transport; thus, the *NPCF* is the only test fuel parameter that must be determined from experiment.

The testing of the MCNP-based method using a single region of substituted test fuel (e.g., 7 channels or 12 channels) has shown good agreement with full-core results. Results have not been shown for MCNP analyses of progressive substitution experiments, with smaller and smaller regions of substituted test fuel. The use of progressive substitution experiments and their associated analyses have been used in the past [1–3] to help correct for deficiencies in the more approximate deterministic methods of substitution analysis. With the use of MCNP, this approach is no longer necessary. In principle, one might be able to use a single bundle of test fuel in a substitution experiment. However, the uncertainties in the derived value of *NPCF* for the test fuel will be larger. Thus, it is preferable to use a substitution region that is as large as possible, short of a full core of test fuel. Experience from these studies has shown that at least 36 fuel bundles should be adequate.

6. Conclusions

A new method for analyzing substitution experiments based on the use of a modified version of MCNP has been developed and tested. The MCNP-based method is conceptually simple, only requiring a minor change to the MCNP source code in a single subroutine to allow the application of an adjustment factor (referred as a *neutron production correction factor*, *NPCF*) to the starting weight of neutrons born in fission in a given fuel material.

The MCNP-based substitution analysis method can be used to isolate the k_{eff} (and hence the bias in k_{eff}) of a given test fuel from the bias in k_{eff} for a substitution experiment involving one or more reference fuels, provided that the *NPCF* values for the reference fuel lattice can be determined from other critical experiments. The MCNP-based substitution analysis method can also be used to determine bare core critical dimensions and buckling for a given test fuel, which can be used for the subsequent validation of other reactor physics codes.

Testing has shown that the MCNP-based method works very well, showing good agreement (within uncertainties) between substitution analysis results and full-core experimental results, with a noticeable improvement over older, more approximate deterministic methods using few-group diffusion theory codes [3].

The testing of the MCNP-based method has shown that experiments with 7 channels (in hexagonal lattices) or 12 channels (in square lattices) of substituted test fuel should be adequate. This implies that on the order of 36 bundles of test fuel are needed in experiments to determine the properties of the fuel with acceptable uncertainties (e.g., ±1 mk). The use of the MCNP-based method for the analysis of progressive substitution experiments with smaller regions of substituted test fuel could be considered for additional future studies.

The use of substitution experiments and the MCNP-based substitution analysis method will provide important validation data for various types of existing and postulated fuel materials and fuel bundle designs. Such data will be

[3] mk is the most common unit of reactivity used in Canada. 1 mk = 0.001 $\Delta k/k$ = 100 pcm.

relevant and important in validation of codes for future reactor designs, including the use of thorium-based [13] and alternative LEU-based [14] fuels in heavy-water reactors.

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TECHNICAL NOTE

MONTE CARLO CALCULATIONS APPLIED TO SLOWPOKE FULL-REACTOR ANALYSIS

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Abstract

Monte Carlo simulations are applied to the full-reactor analysis of the SLOWPOKE design. The temperature reactivity feedback calculated by using the MCNP code for either the high enriched uranium (HEU) or low enriched uranium (LEU) core is in good agreement with the experimental data, with a k-eff bias of +3.3 mk for a HEU core and +6 mk for a LEU core. Two methods that are based on existing third-party codes have been developed for use in core following: 1) MCNP (for the transport calculation) in conjunction with WIMS-AECL (for fuel burnup advancement), and 2) SERPENT (that combines both transport and burnup capabilities). Both methods show very good agreement with the experimental data for core excess reactivity and detailed power distributions versus burnup and reactivity shim.

1. Introduction

SLOWPOKE (Safe Low Power Kritical Experiment [1]) reactors are AECL-designed research reactors of pool type, loaded with either high enriched uranium (HEU, 93 wt% ^{235}U/U) in UAl metal alloy and Al clad fuel elements, or low enriched uranium (LEU, 20 wt% ^{235}U/U) in UO_2 ceramic oxide and Zr clad fuel elements. The approximate total ^{235}U core loading is 0.82 kg for the HEU fuel and 1.2 kg for the LEU fuel. The core is cooled and moderated by water and has beryllium reflectors (solid metal radially and below the core, and thin metal plates above the core). The SLOWPOKE design has a relatively small excess reactivity (subject to restoration by adding beryllium shim plates to the top reflector) but large negative temperature reactivity feedback – a very special feature assuring its safe operation.

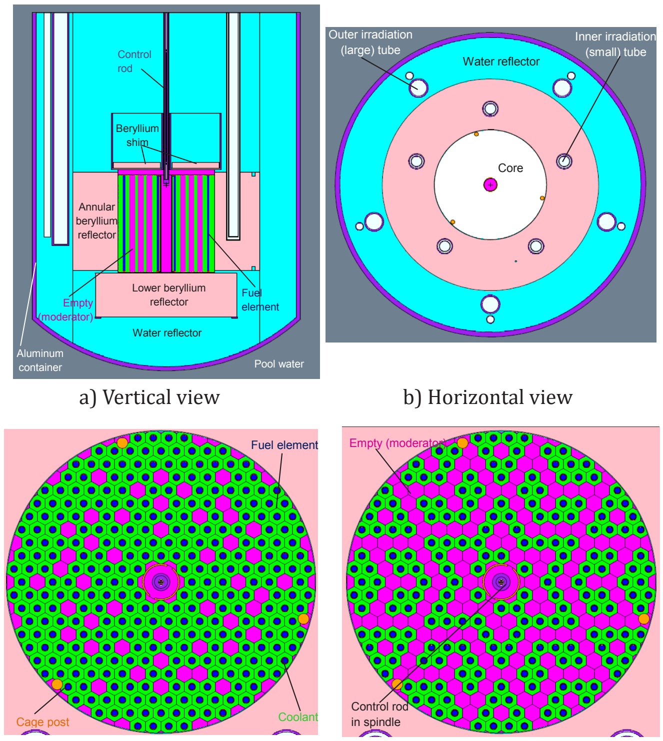
Original reactor physics analysis based on solving the few-group neutron diffusion equation, e.g., HAMMER/EXTERMINATOR [2] (used to analyze the first HEU core) or WIMS-CRNL/CITATION [3] (used to analyze the first LEU core), tended to give rise to large k-eff uncertainty and lack of power and burnup spatial distributions in fuel elements due to core homogenization. Stochastic neutron transport codes (such as MCNP [4] and SERPENT [5]), are able to eliminate these inaccuracies resulting from the diffusion approximation.[1]

The MCNP full-reactor models of SLOWPOKE (Figure 1) with a HEU or LEU core, have been created to study: 1) temperature reactivity feedback, and 2) burnup (or core following) for the SLOWPOKE design. These MCNP models include the main reactor components inside of the reactor container in detail, each of which may be changed with respect to geometry (shim thickness or control rod position), material, or temperature. Due to the simple design of the SLOWPOKE reactor and its small size, very few geometric or material approximations were required; the models are essentially exact. The models used MCNP5 Version 1.40 and an in-house multi-temperature library based on ENDF/B-VII.0 generated with NJOY [6]. When a desired material temperature was between two data sets, a combination of the two sets (interpolation of the square root of temperature) was used. Fuel elements in a hexagonal lattice are modeled to have the burnup-dependent compositions, which may vary not only from element to element but also axially within each element. Where fuel elements are expected to be in similar neutron fluxes due to location they have been grouped together to improve tally uncertainties.

2. Temperature Reactivity Feedback

Using the multi-temperature ENDF/B-VII.0 cross-section library and mixing concentrations at two library temperatures as discussed above, the material temperature in each of the main reactor components (i.e., fuel, coolant/moderator,

[1] The original deterministic codes, and the typical phase space discretization and the nuclear data they used, are all obsolete. Much better results are now possible using modern deterministic codes and data. Comparison of modern stochastic methods to modern deterministic methods is beyond the scope of this paper.



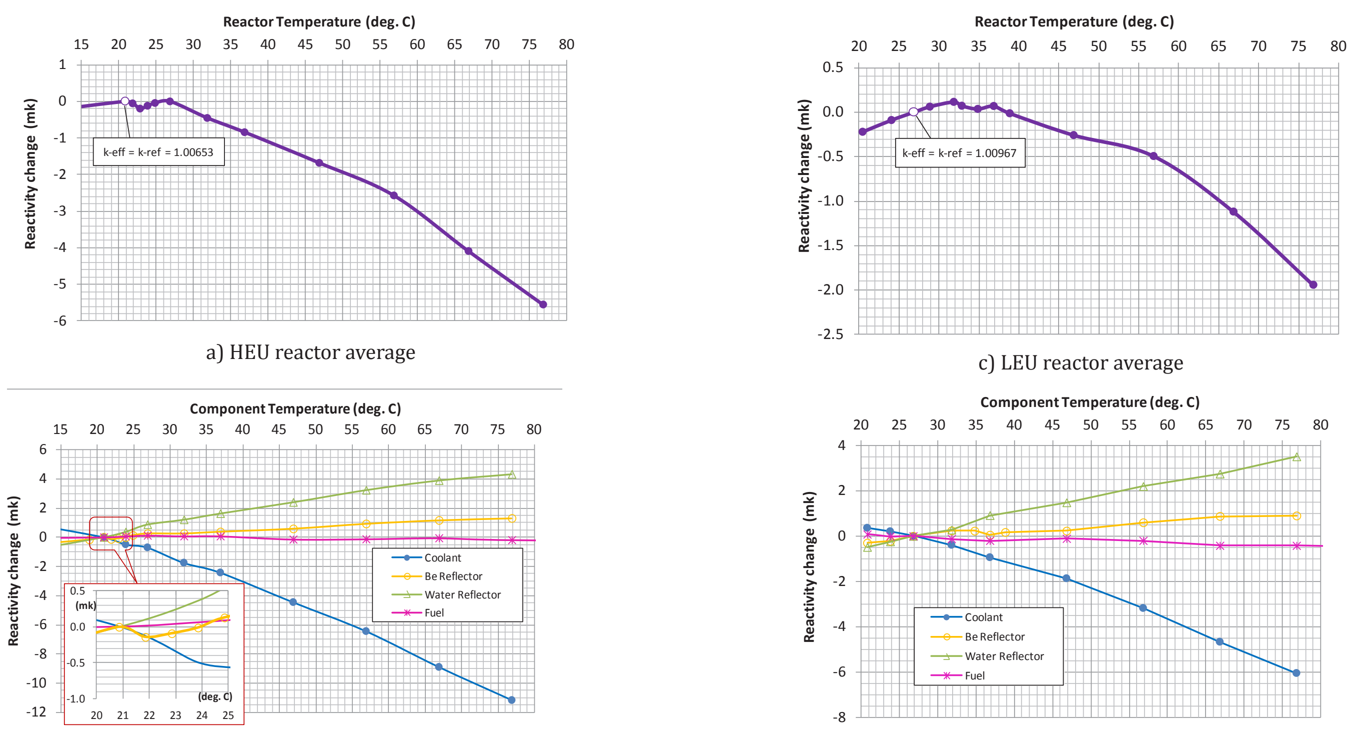
a) Vertical view   b) Horizontal view

c) HEU core loading   d) LEU core loading

FIGURE 1
Monte Carlo model of SLOWPOKE for both MCNP and SERPENT Codes.

beryllium reflector and water reflector) was set independently to a series of values for the k-eff calculations. The individual reactivity components were combined to obtain the whole core results. The core excess reactivity (i.e., the reactivity with the control rod fully withdrawn) calculated by MCNP is biased relative to the measured values, by +3.3 (±0.2) mk[2] for a HEU core and +6 (±0.2) mk for a LEU core where the uncertainty is one standard deviation. The temperature reactivity feedback for the MCNP models is consistent with the experimental data. In general, the reactivity feedback is slightly positive at low temperatures and turns negative above room temperature. Figure 2 shows the change in core reactivity from a reference state for fresh fuel as the temperature of individual components is changed while holding the others fixed, as well as the combined reactivity of changing all component temperatures together. MCNP predicts the reactivity peak to be in the range of 21–27°C for the HEU core and 32–37°C for the LEU core. The experimental data shows maximum values of excess reactivity at 20°C and 33°C for HEU and LEU, respectively, due to the combination of the individual reactor component temperature reactivity feedbacks which have different signs and values.

The reflector temperature reactivity feedback is mostly small and positive, while the fuel and coolant reactivity feedback is always negative, small for fuel and low-temperature coolant but relatively large for coolant above room temperature (-10 mk and -6 mk over a 50°C change

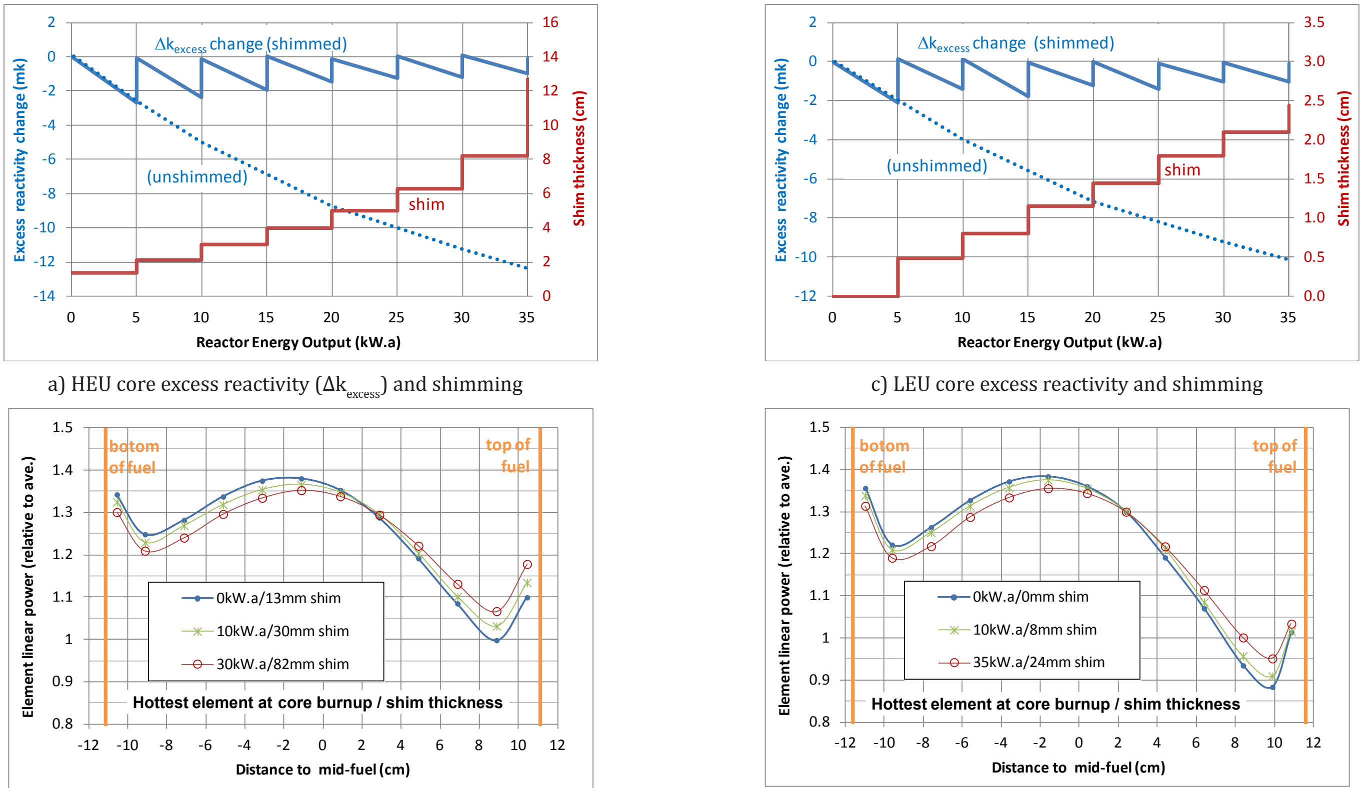


a) HEU reactor average   c) LEU reactor average

b) HEU reactor components   d) LEU reactor components

FIGURE 2
SLOWPOKE temperature reactivity feedback.

[2] The unit “mk” is the most common unit of reactivity used in Canada. 1 mk = 100 pcm = 0.001 dk/k.



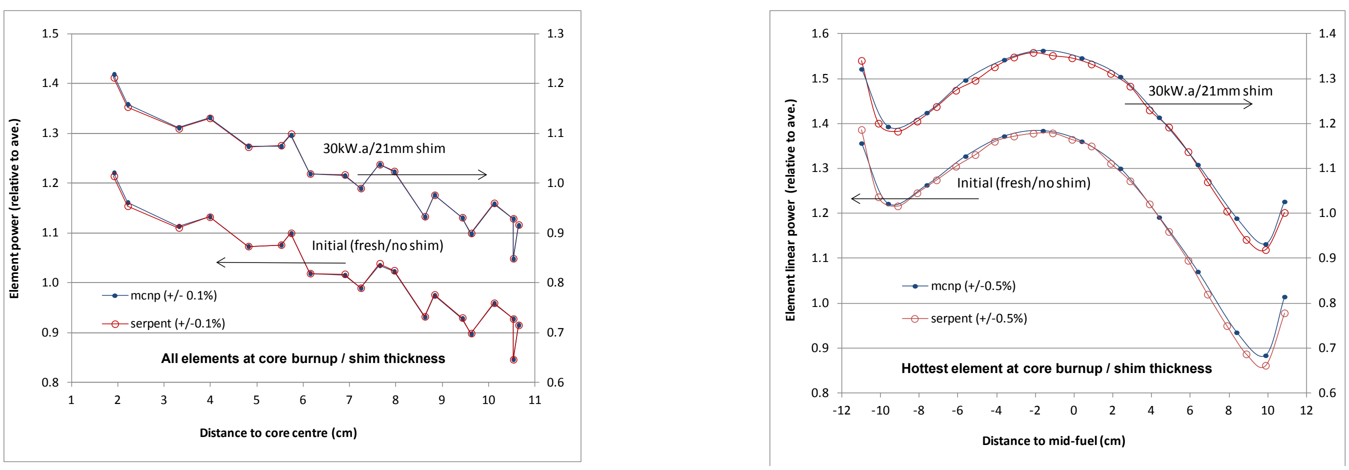
a) HEU core excess reactivity (Δk_{excess}) and shimming

c) LEU core excess reactivity and shimming

b) HEU element power

d) LEU element power

FIGURE 3
SLOWPOKE reactivity and element power distribution versus burnup.



a) Radial distribution (element powers at points)

b) Axial distribution (element linear power)

FIGURE 4
Comparison of MCNP and SERPENT power distribution results.

in the HEU and LEU core, respectively), thus, the coolant dominates the SLOWPOKE reactivity feedback at higher temperatures.

3. Burnup Analysis

For core following, the MCNP full-reactor calculation is performed to provide the three-dimensional power distribution, while a burnup code, such as WIMS-AECL Version 3.1 [7] in this study, is required for fuel burnup advancement. The burnable materials in the MCNP model are updated using the WIMS-AECL pre-computed isotopic composition as a function of burnup. WIMS-AECL is run first from fresh fuel to exit burnup using a two-dimensional model of the whole core with all fuel pins included and using the WIMS-AECL 89-group library (ENDF/B-VII.0, NJOY processed) that is the equivalent of the library that was used with MCNP; then the resulting composition tables are interpolated manually using power distributions from MCNP to advance the composition for each axial segment of fuel in MCNP for the next irradiation step. Since SLOWPOKE fuel depletes only slightly in practice, the element radial power distribution does not change significantly with burnup and top reflector shimming, but the axial power distribution does change with the shim thickness since the shims are added to the top reflector. The power calculations were done holding the control rod position fixed.

For hypothetical operation of SLOWPOKE that restores the excess reactivity by reflector shimming after every 5 kW·a[3], the reactivity loss rate due to burnup decreases with the core burnup, from ~0.4–0.5 mk/kW·a at the beginning of the core life to ~0.2 mk/kW·a at 35 kW·a. Top reflector shim effectiveness, i.e., mk gain per cm beryllium added, also decreases with the core burnup (or more correctly, with the total shim thickness), greater in the HEU core, from 3.4 mk/cm at the beginning to 0.1 mk/cm at 35 kW·a (where the shim-plate tray is full), and less in the LEU core, from ~5 mk/cm at the beginning to ~3 mk/cm at 35 kW·a (where the shim-plate tray is only ~20% full). This indicates that the LEU core can operate much longer than the period simulated (Figure 3).

To verify the MCNP/WIMS-AECL core following method, SERPENT [5] that combines both Monte Carlo transport calculation and burnup capability is used independently. Version 1.1.17 of SERPENT was run using models and cross-section libraries that are identical to those used with MCNP. For any core of a given burnup and top reflector shim thickness, the SERPENT and MCNP power distribution results agree very well, to within the statistical uncertainty of the calculations (<0.5%, see Figure 4). This provides confidence that the transport algorithms in MCNP and SERPENT, and that the WIMS-AECL and SERPENT burnup calculations, are consistent.

4. Conclusions

Monte Carlo methods can be very time-consuming compared to lower-fidelity deterministic methods and so are generally not well suited for real time core following in large reactors like NRU or CANDU that require fuel shuffling and replacement in time frames of the order of days. However, as this paper has demonstrated, Monte Carlo methods *are* practical for tracking burnup and reactivity shimming in small low-power reactors like SLOWPOKE where reactivity adjustments occur in time frames on the order of months or years. The method is also useful for analyzing reactivity coefficients and characterizing experiments. Future work could include an investigation of the calculated beryllium reflector temperature reactivity feedback and the higher burnup sustainable in the LEU core, as well as comparison of these results to recent studies using modern deterministic codes. An investigation of other burnup modelling codes and MCNP coupling techniques will also be considered.

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[3] kW.a is the time-integrated fission energy in kilowatt-years. It is the conventional unit used to express fuel burnup in SLOWPOKE reactors.

TECHNICAL NOTE

MONTE CARLO CALCULATIONS APPLIED TO NRU REACTOR AND RADIATION PHYSICS ANALYSES

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Abstract

The statistical MCNP (Monte Carlo N-Particle) code has been satisfactorily used for reactor and radiation physics calculations to support NRU operation and analysis. MCNP enables 3D modeling of the reactor and its components in great detail, the transport calculation of photons (in addition to neutrons), and the capability to model all locations in space, which are beyond the capabilities of the deterministic neutronics methods used for NRU. While the simple single-cell model is efficient for local analysis in any site of NRU, the complex full-reactor model is required for calculations of the core physics and beyond-the-core radiation. By supplementing, adjusting or benchmarking the results from the existing NRU codes, the MCNP calculations provide greater confidence that NRU remains within the licence envelope.

1. Introduction

The statistical MCNP code [1] has been satisfactorily used for reactor and radiation physics calculations to support NRU operation and analysis [2]. Advantages of MCNP over the traditional deterministic methods are its enabling 3D modeling of the reactor and components in greater detail, capability of photon calculations (in addition to neutrons), and the capability to model all locations in space (e.g., remote areas in the NRU outer structure).

Being time-consuming, MCNP is primarily used to improve (or correct) the results from the fast deterministic methods for NRU, i.e., TRIAD [3] - a 3D reactor diffusion code for physics support of NRU daily operation, and BURFEL [4] - Burnup of Fuel Elements - a code and database system for NRU loop fuel calculations, but it also provides those radiation data pertinent to photons or to the outer NRU structure, which are beyond the capabilities of the current neutronic codes.

2. MCNP models

Depending on the problems to be solved, a single cell or full reactor model may be used, and with or without the photon capability. For calculations involving energy release and deposition, two in-house patches [2, 5] are normally turned on: i) QFISS (Fission Q-values) to change the MCNP hard-wired q-fission values to the recoverable energy values customarily used, and, ii) DPERT (Direct Cross-Section Perturbation) to add the delayed beta and photon energies from decays of fission and activation products.

The single-cell model, as illustrated in Figure 1, typically represents an axial section of a core site within an approximate NRU environment (e.g., a loop test section[1] or an irradiation facility), which is rather simple but efficient in providing various power-related parameters of safety significance. On the other hand, the full-reactor model (Figure 2), including a detailed core of NRU and its outer structure, is computationally intensive and time-consuming but must be used when dealing with one or more fuel sites or any part of the outer structure. The outer structure components of the model, however, are usually simplified to save computation time and only parts particularly important in the problem are to be modeled in more detail.

Most rod types of NRU (i.e., driver fuel rods, molybdenum-99 production rods, loop fuel strings, control rods, and irradiation facilities) are modeled in detail in terms

[1] An NRU irradiation channel, cooled by high-pressure high-temperature light water, that can be loaded with a string of six modified-CANDU fuel bundles. Each bundle has the centre element removed for a tie-rod, and the remaining 36 (or 42) elements positioned in three rings – inner, intermediate and outer of 6, 12 and 18 elements, respectively (or 7, 14 and 21). A 30-element materials test bundle (MTB) has a sample holder in place of the inner ring. Element locations are numbered counter-clockwise from the outer ring (e.g., 1 or Outer#01) to the inner ring (e.g., 36 or Inner#36).

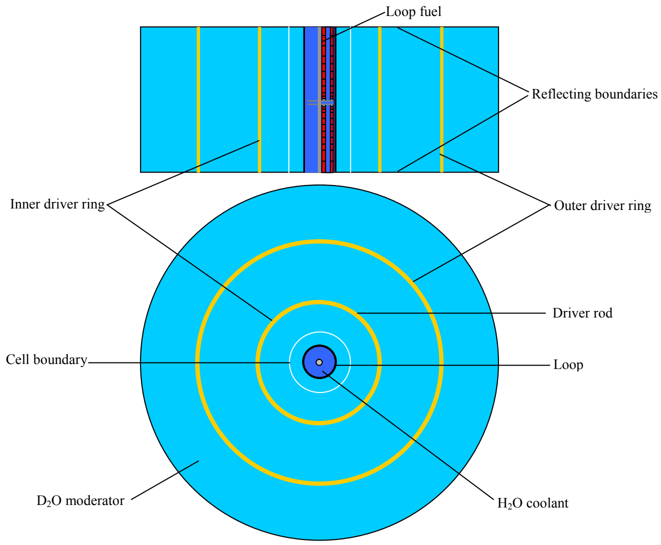


FIGURE 1
MCNP single-cell model of a loop test section in NRU.

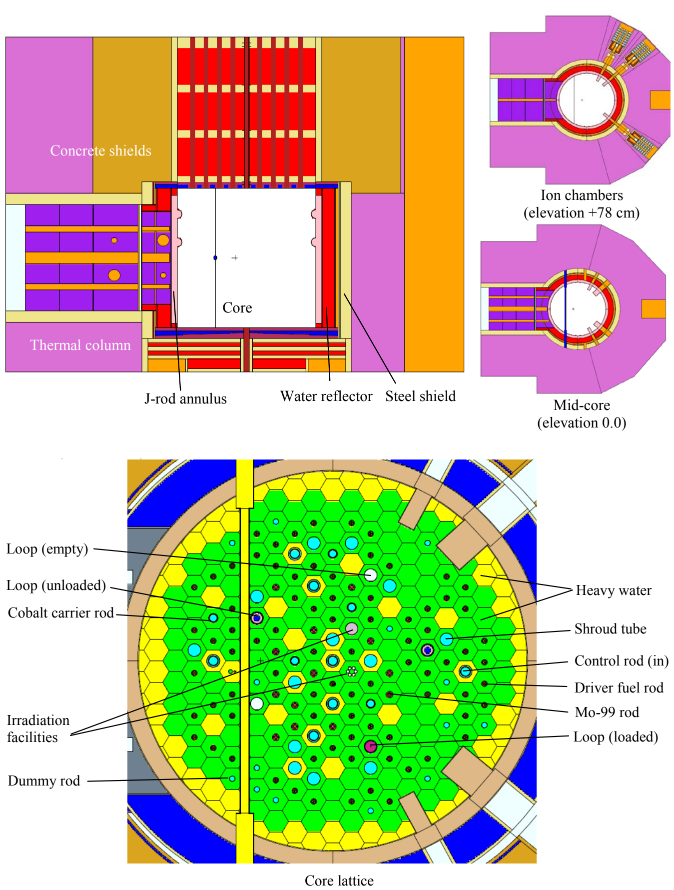


FIGURE 2
MCNP full reactor model of NRU.

of geometry and material, with only a few insignificant simplifications (such as smearing fins of a driver rod into its sheath, or aligning cobalt tiny rods instead of randomly-oriented 1-mm-sized pellets in a control rod but preserving the total mass). Individual fuel elements are axially sectioned (e.g., 14 sections in driver rods; 3 or more segments plus the end-pellet regions in loop bundles) to accommodate varying fuel compositions corresponding to the section burnups as in the TRIAD or BURFEL counterparts. The fuel isotopic compositions versus burnup are pre-computed using WIMS-AECL similar to the WIMS models used in preparing the TRIAD group constants or the BURFEL neutronics data. Also, as in TRIAD or BURFEL, the 'hot' fuel compositions from WIMS (i.e., with equilibrium poisons at the core average flux) are used in MCNP (although for specific sensitivity analyses, the 'cold' compositions may be made by adding a few extra cooling steps in the WIMS burnup calculation).

MCNP5 version 1.40, with the ENDF/B-VII multi-temperature cross-section library data generated at AECL, is used for either single-cell or full-core calculation in criticality (KCODE) mode. Material temperatures may be set to any desirable values, varying from cell to cell. Typically, at least 50 million neutron histories (on top of 0.5 to 2.5 million initial histories for source convergence) are needed for the k-eff statistical error to be within ~0.1 mk, and, generally, that is sufficient for acceptable in-core parameters (<1% of statistical errors). However, 500 million histories or more are required for obtaining *meaningful* radiation data (within a few percent of statistical errors) beyond the reactor core and, very often, additional calculations on further simplified models (i.e., using single sources and/or one of the variance reduction techniques) are needed for remote areas. In terms of computation time, on a single node of the AECL computer cluster, a single-cell calculation needs several hours, but a full-core calculation may take 3 days to one week to complete, depending on the total number of entries (cells and types) to be tallied.

For the time being, the MCNP calculated results have not been directly validated against the actual operating data (due to too many burnups and associated uncertainties of an actual NRU core for modeling), instead they are compared with the existing NRU methods (i.e., TRIAD or BURFEL) on the same, often simplified, models. TRIAD and BURFEL have known biases to the actual operating data. Generally, in addition to being mostly in good agreement, the MCNP results tend to have similar biases, but in the opposite direction, to the TRIAD or BURFEL results, suggesting credibility of the MCNP models.

3. Single-cell calculations

The single-cell model is sufficient for calculating power-related parameters in loop fuel elements and samples irradiated in an irradiation facility, such as:

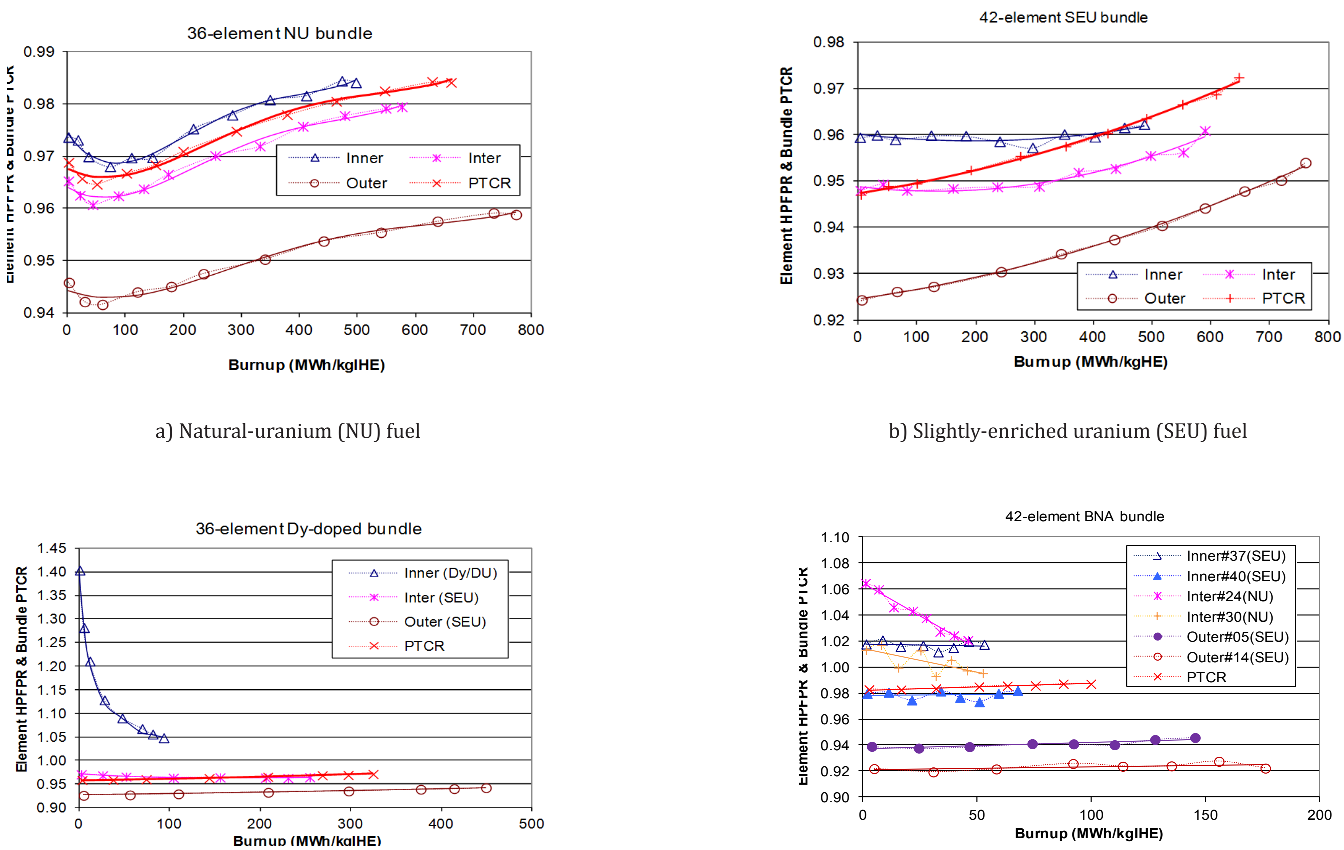
a) Heat deposition in loop fuels and coolant: The bundle power to coolant ratios (PTCR) and element heating power to fission power ratios (HPFPR) are found to vary with the fuel type (composition, location) and burnup, as in the examples illustrated in Figure 3, rather than being an invariant constant (i.e., 0.96, typical of fresh natural uranium fuel in CANDU) previously accepted. These MCNP-based factors help reduce the BURFEL burnup bias (known to be ~-9%) by a few percent, notably for enriched, poison-doped or irradiated fuels.

b) End-peaking powers in loop fuel elements: Effects of axial flux changes, including end-flux peaking in particular, on the element linear power can be properly accounted for in the 3D MCNP calculations, enabling correction of the BURFEL method (which is based on a 2D WIMS neutronics model) and, also, to avoid allowing the element power to exceed a license limit for a proposed experiment. Figure 4 illustrates an example of using MCNP to correct the BURFEL element powers in a string of typical loop fuel bundles.

c) Heating of non-fuel materials: MCNP directly provides heating powers, mostly due to photons, in any non-fuel materials irradiated in an NRU irradiation facility (e.g., capsules containing iridium or tellurium targets) to predict overheating. Depending on material and location, the radiation heat load in a sample can be realistically predicted rather than using a typical estimate of 2 W/g as was done in the past.

4. Full-reactor calculations

The MCNP full reactor model not only is capable of providing a wider diversity of data and in greater detail than TRIAD or BURFEL but also allows for extension beyond the reactor core. The constraint, however, is that the method is very computationally time-consuming. As such, it is only used when the single-cell model is inadequate, for example,



a) Natural-uranium (NU) fuel

b) Slightly-enriched uranium (SEU) fuel

c) Dy-doped depleted-uranium (Dy/DU) fuel

d) Fuel near burnable neutron absorbers (BNA)

FIGURE 3
Examples of heat deposition data in loop fuel bundles.

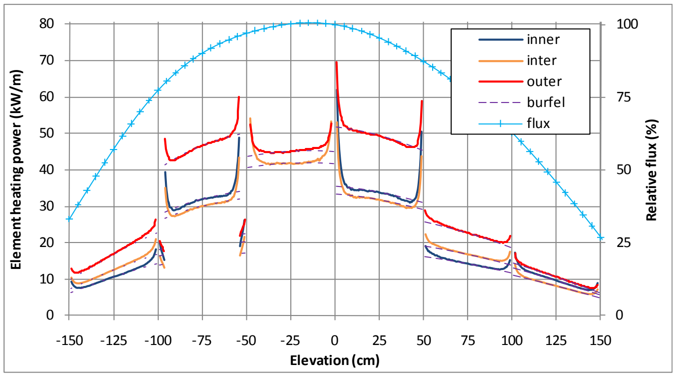


FIGURE 4
BURFEL and corrected loop element powers using MCNP.

a) Benchmarking TRIAD k-eff, reactivity worth, neutron flux and power distribution. MCNP appears to provide k-eff without the known bias from TRIAD. The key parameters obtained from the two methods are in reasonably good agreement, within 1 mk of reactivity or <10% of site flux or power (comparable to that between TRIAD simulations and the measured data but in the opposite direction). It reveals those TRIAD deficiencies resulting from diffusion theory that could be improved.

b) Providing reliable estimates of heating power ratios in an operating core, leading to the more accurate calculation of the driver fuel burnup, which is subject to a safety limit. TRIAD currently uses PTCR estimates of 0.94 for driver rods and 0.88 for Mo-99 production rods as compared to more accurate MCNP values ~0.90 and 0.92, respectively (Table 1). Fortunately, as TRIAD renormalizes rod fission powers to the total thermal power (that MCNP confirms to be a good approximation, Table 1) and due to a small power contribution from Mo-99 rods, the driver rod burnup is relatively unaffected by the values of the rod PTCR that are used (although the burnup in other rods, which are of little safety significance, would be more inaccurate).

c) Providing radiation data (flux and dose rate of both neutrons and photons) in the outer structure (e.g., the graphite thermal column and ion-chamber holes) for safety analysis and work planning. It is worth mentioning that: i) the photon dose rate in beam holes away from the reactor vessel is only 2-3% of the neutron dose rate, and, primarily due to the neutron-induced photons alone (thus, shielding neutrons effectively diminishes photons); and ii) the maximum fast (>1 MeV) neutron flux reaching the thermal column is < $3x10^9$ n.cm^{-2}s^{-1}, insufficient to result in problematic amounts of stored energy in the NRU graphite.

d) Other physics and scoping analysis of NRU:

- total photoneutron production worth ~2.5 mk;
- draining water from the light water reflector loses ~8 mk;
- installing a beryllium reflector in the empty J-rod annulus adds ~30 mk.

TABLE 1
Summary of NRU Core Heating at Reactor Fission Power of 100 MW

Category	Number of sites	Fission (MW)	Site heating (MW)	Power to Coolant (MW)	Average rod PTCR
Core (excluding loop)	-	100.0	99.1 (reactor thermal power)	-	-
Driver fuel rods	90	96.45	91.42	86.59	0.895
Mo-99 rods	15	3.55	3.84	3.28	0.925
Loaded loop (fresh NU bundles)	1	2.74	2.71	2.64	0.963
Non-fuel sites	121	0	3.80	-	-

5. Concluding remarks

The MCNP single-cell and full-reactor models of NRU provide valuable reactor physics information, not easily determined with other methods, to support its operating efficiency and safety. By supplementing, adjusting or benchmarking the results from the existing NRU codes, the MCNP calculations provide greater confidence that NRU remains within the licence envelope.

Further improvements to the MCNP method are being considered to address the inefficiency of radiation calculations at the outer surfaces of the reactor structures, a limitation on the number burnup compositions, and the lack of a burnup capability.

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TECHNICAL NOTE

Abstract

The SLOWPOKE reactor in Jamaica has been operated by the International Centre for Environmental and Nuclear Sciences, University of the West Indies since 1984, mainly for the purpose of Neutron Activation Analysis. The HEU core with current utilization has another 14 years of operation, before the addition of a large beryllium annulus would be required to further extend the life-time by 15 years.

However, in keeping with the spirit of the Reduced Enrichment for Research and Test Reactors (RERTR) program, the decision was taken in 2003 to convert the core from HEU to LEU, in line with those at the École Polytechnic and RMC SLOWPOKE facilities. This paper reports on the current status of the conversion activities, including key fuel manufacture and regulatory issues, which have seen substantial progress during the last year. A timetable for the complete process is given, and provided that the fuel fabrication can be completed in the estimated 18 months, the core conversion should be accomplished by the end of 2014.

The Status of HEU to LEU Core Conversion Activities at the Jamaica SLOWPOKE

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1. Introduction

The SLOWPOKE reactor in Jamaica has been operated by the International Centre for Environmental and Nuclear Sciences (ICENS), University of the West Indies (UWI) since 1984. The main purpose of the reactor is Neutron Activation Analysis applied to environmental / health-related studies, and mineral exploration in Jamaica. In addition ICENS has cooperated with the International Atomic Energy Agency (IAEA) in the establishment of the Caribbean Research Reactor Coalition (CRRC) [1] with reactors in Colombia and Mexico, to increase regional access to research reactor services and nuclear-related education and training.

The high enrichment uranium (HEU) core with current utilization has another 14 years of operation, before the addition of a large beryllium annulus would be required to further extend the life-time by 15 years. However, in keeping with the spirit of the Reduced Enrichment for Research and Test Reactors (RERTR) program, and the move by the international community to eliminate the civilian use of HEU, the decision was taken in 2003 to look into the possibility of converting the core to low enrichment uranium (LEU). In 2009, a formal request was made via the IAEA to the US Department of Energy (US-DOE) Global Threat Reduction Initiative (GTRI) program, which had subsumed the earlier RERTR program, to help fund the conversion of the reactor, and take back the spent fuel.

Although progress has been slow, there have been several key developments during the last year, particularly in the area of regulatory oversight, including the expected licensing requirements for defueling, commissioning and operation of the reactor. This technical note reports on the current status of SLOWPOKE-2, the selected fuel and the proposed action plan for the next two years. Provided that the fuel fabrication process can be completed in the estimated 18 months, the core conversion process should be accomplished by the end of 2014.

2. HEU Core

The ICENS SLOWPOKE core is fueled with ~ 1 kilogram of U.S. origin HEU, and has operated on average for 1300 hours per year at 10kW. It consists of an assembly of 296 fuel pins containing a total of 817 g of 93.5% enriched ^{235}U as co-extruded alloy containing 28% by weight of U in Al. A 100 mm thick pure beryllium annulus encases the fuel cage, which is a cylinder with an internal diameter of 22.1 cm and a height of 22.8 cm (wall thickness 10.8 cm). The annulus acts as a side reflector for neutrons, and a 50 mm thick beryllium disc forms the bottom reflector. The top reflectors consist of semicircular plates of beryllium each only a few millimeters thick.

There are five (5) small inner irradiation sites within the beryllium annulus; however, only four are available for use at our facility as site #2 presently houses a flux detector installed as a replacement for the originally installed flux detector which malfunctioned in 1988. In addition there is one large site outside of the beryllium and an in-pool irradiation system. The SLOWPOKE neutron flux is azimuthally symmetric about the axis of the core and extends a short distance outside the reactor container as shown in Figure 1. The maximum operational in-core flux is 1 x 10^{12} $n.cm^{-2}$ s^{-1}, the site-specific flux variation is less than ± 3%, and the flux as measured in the large outer site is approximately 51% of the nominal flux. The epithermal and fast components of the reactor neutron spectrum account for approximately 5% and 23% respectively of the total inner site flux. The fast component of neutron spectrum of SLOWPOKE is composed of both fission neutrons and those generated by (γ, n) reaction from the Be reflector.

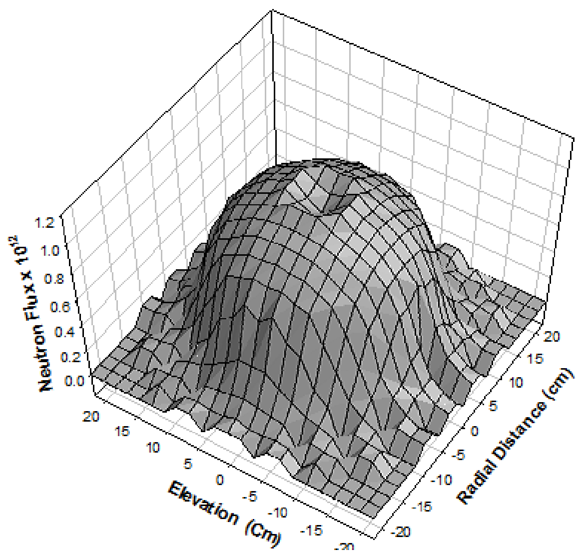

FIGURE 1
Thermal flux distribution of SLOWPOKE II.

The reactivity loss with time due to ^{235}U fuel burnup and poison buildup (primarily ^{149}Sm) is compensated for by the addition of Be shims. Current core burnup is calculated to be approximately 51% of its total lifetime (Figure 2) at October 2012. Under current utilization patterns, another 14 years of operation is possible, before the addition of a large beryllium annulus would be required to further extend the life-time. The next shim adjustment is scheduled to be required in 2015.

2.1 Estimated Fission Product Activity

The fission product activity of the used fuel, is a function of the reactor flux hours, and has in the past been estimated

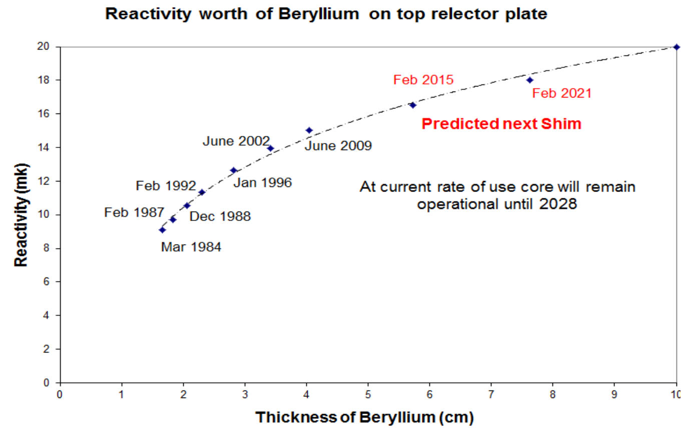

FIGURE 2
Reactivity worth of Beryllium on top reflector plate.

using an AECL calculation based on a SLOWPOKE reactor which was operated for 5 years at a neutron flux of 1 x 10^{11} $n.cm^{-2}$ s^{-1} (2 kW), then 10 hours at a neutron flux of 1 x 10^{12} $n.cm^{-2}s^{-1}$ (20 kW) (private communications). The calculated activity 30 days after shutdown was 23 TBq. Based on a maximum usage for the Jamaica SLOWPOKE of 4 hours per day 5 days per week at an average flux of 4.95 x 10^{11} $n.cm^{-2}$ s^{-1} (average over the last 29 years), the 5 years average continuous flux will be 0.57 x 10^{11} $n.cm^{-2}s^{-1}$ (1.1 kW). Based on a maximum usage for the next 5 years, the expected activity of the core 30 days after shutdown should not exceed 13 TBq. Previous experience has shown this calculation to be conservative, as in the case of École Polytechnic, a calculated maximum of ~23 TBq was measured at ~18 TBq (private communications). In their case a one month cooling period was sufficient before the conversion process could take place. These estimates are to be verified with new models being developed in conjunction with Argonne National Laboratories (ANL).

The integrity of the HEU fuel over the past 27 years has been monitored via reactor container water activity analyses. When compared to École Polytechnique with the HEU fuel, the activity of the reactor water at ICENS is almost three orders of magnitude lower, suggesting little or no blistering of our fuel (Figure 3). The residual activity of the École Polytechnique post conversion was believed to be about 1 mg of ^{235}U, which was released from the original HEU core and plated out on the beryllium reflector and other reactor container surfaces [2].

Given the relatively low activity of the ICENS HEU core water it is expected that the levels should be similar to that of RMC, post conversion.

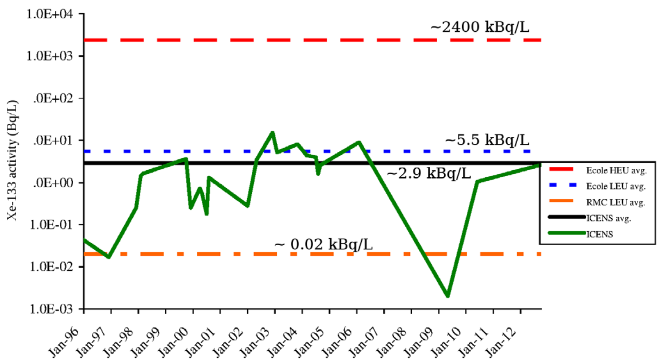

FIGURE 3
ICENS SLOWPOKE reactor core water Xe-133 activity.

2.2 LEU Fuel Composition

Based upon the operational success of the AECL-manufactured LEU cores, the ICENS SLOWPOKE will utilize fuel of identical composition and dimensions. The AECL-developed LEU fuel was fabricated from zircaloy-4 clad uranium oxide pellets and contained 1100 g of ^{235}U (total mass of U ~5600 g) with an enrichment of 19.9%, as the LEU fuel requires ~ 36% more ^{235}U to achieve the same reactivity as the HEU core. The core itself is 220 mm in diameter and 227 mm in height. Existing LEU cores at Royal Military College (operational since 1988) and École Polytechnique de Montréal (converted in 1997) achieved criticality with a total of 198 fuel pins in the fuel cage, each pin being 5.26 mm in diameter and 234 mm in length [2]. The inherent safety of the HEU core is provided by a large negative temperature coefficient [3] which is also true for the LEU core and ensures that power excursions are self-limiting [4].

The Y-12 National Security Complex and AECL will supply the UO_2 material and manufacture the fuel and fuel cage components. The characteristics of the LEU core have been well documented and modeled [2, 5, 6]; however, neutronic and thermal-hydraulic models of the reactor will be developed to support the conversion safety analyses in cooperation with the ANL.

2.3 Facility Upgrades and Aging Management

In April of 2012, an independent facility review was carried out with the objective of evaluating the adequacy of the reactor site and facilities to support the LEU conversion planning and preparation activities. The resulting report [7] identified no major issues to prevent or delay proceeding with the conversion project. It recommended among other things that the planned replacement of the analog control system with a digital one, be carried out prior to the core conversion. The system being reviewed is a digital control and instrumentation system developed specifically for the SLOWPOKE-II reactor (SIRCIS) [8], which was commissioned at the Royal Military College (RMC) reactor in June 2001. Discussions have begun with the manufacturer to resolve long term maintenance and sole source issues.

2.4 Core Replacement and Approach to Critical

In all likelihood, the core will be removed in accordance with procedures developed at Montreal. It is planned to transport the spent fuel using the AECL F-257 transportation flask, and the local regulatory body will be approached to license it.

For all SLOWPOKEs to date, the approach to first criticality has involved adding fuel pins to the core according to a prearranged schedule using a peening process [5]. For the HEU core at ICENS the fuel loading took 15 cycles. The clear advantage of this methodology is that it has been successfully utilized with seven HEU and two LEU cores. An alternative to this process is to have the fuel cage shipped partially loaded, thereby reducing the number of cycles for the approach to critical, and the risk of damage during peening. However, the feasibility of this approach will depend greatly on how closely the modeling predictions for the core loading match past data for the LEU cores.

3. Regulatory Oversight

In January of 2011, the Government of Jamaica (GOJ), via Cabinet Decision # 01/11, designated the Ministry of Industry and Commerce the parent ministry for the Radiation Safety Authority (RSA) under the auspices of the Bureau of Standards Jamaica (BSJ) (Figure 4), and enacted the Jamaica action plan (2012 – 2014) with the following:

• A draft law compliant with the International Basic Safety Standards and related IAEA publications such as GSR Part 1 Requirements [9], the Code of Conduct on the Safety and Security of Radioactive Sources [10], and supplementary Guidance on the Import and Export of Radioactive Sources [11], is scheduled for completion in March 2013. The IAEA is being requested to provide assistance in this effort.

• Stakeholder forums are to be convened for consultation on and review of a draft law (September 2012)

• Timelines are to be established for actions coming out of the stakeholder consultations, for enactment of legislation

The newly appointed RSA has established a reactor National Oversight Committee (NOC) to act as the de facto nuclear regulator, and review all operational aspects of the

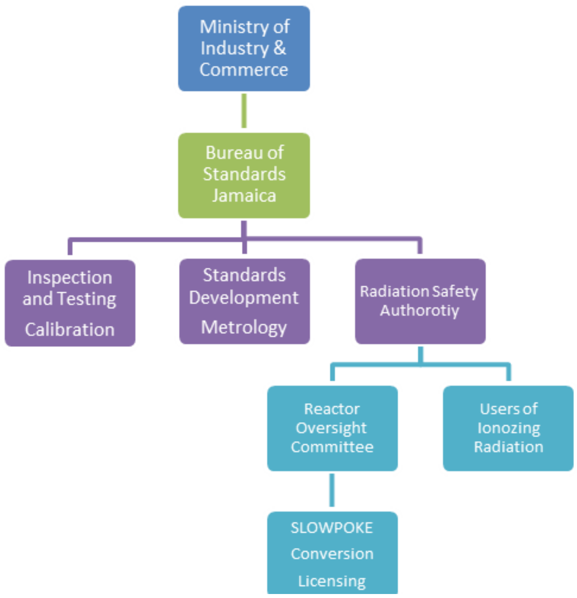

FIGURE 4
Position of Radiation Safety Authority (RSA) within the Ministry of Industry and Commerce

research reactor, to ensure that the upcoming conversion is conducted in accordance with all relevant international guidelines. This committee has thus far adopted the IAEA safety requirements document NS-R-4 "Safety of Research Reactors" as its reference document and are currently finalizing the list of documents that will be required for the conversion process.

4. Timeline of Conversion Project Activities

The core conversion is projected to take 4 years to complete (Table 1). Work has started on the documentation requirements and national regulatory framework (lightly shaded cells of Table 1), key issues which have seen significant progress in the last year. Current efforts (darker shaded cells of Table 1) are focused on completing the licensing documentation, identifying and agreeing terms for the use of various specialized equipment for the core conversion with AECL, and finalizing contracts for core conversion work. Another key issue will be the renewal of competent authority certification from the US Department of Transport for the F-257 flask, which expired on September 30, 2012.

TABLE 1
Core Conversion Activities Timeline

	2011		2012		2013		2014	
Legislation & Regulatory Oversight	Cabinet Decision # 01/11 of 10 January 2011 establishing the Radiation Safety Authority within the Bureau of Standards Jamaica (BSJ).	❑ Reactor Oversight Committee established within RSA ❑ Draft Law prepared	Identification of appropriate regulations to give effect to the proposed law	Stakeholder forum to be convened for consultation on and review of draft law	IAEA agreement on regulations	Draft Guidance Documents and Codes of Conduct	BSJ to establish a national nuclear radiation security and safety committee.	❑ Cabinet to issue drafting instructions for the regulations ❑ Operating license issued.
Documentation	Statement of Work defined		❑ Report on adequacy of reactor site and facilities ❑ Tripartite NDA (AECL-ANL-ICENS) ❑ PSA signed by Jamaican Government	Report on safety and licensing documentation requirements	Produce agreed documents (i.e. Updated SAR for HEU, Operation manual, etc)	Submit documents (updated LEU SAR, conversion procedures, and QA etc) for regulatory approval to proceed with conversion to LEU.		Request for operating license.
				PSA signed by Jamaican government, currently with US government. If no problems expected to be presented to IAEA Board of Governors Meeting December 2012.	❑ Submit documents for regulatory approval of operations and maintenance procedures. ❑ Submit documents for console upgrade			
Facilities upgrade						❑ Upgrade console ❑ Upgrade facility radiation and emission monitoring.		
Core modeling				MCNP model for HEU core being developed at ANL.	❑ Verification of HEU core model ❑ Modeling of LEU core			
Fuel fabrication					Fuel fabrication to commence and last for approx 18 months.			
Transportation					Initiate arrangements with Savannah River Acceptance Program team and AECL for use of F257 flask (re-license), scheduling for F257 flask, LEU fuel and fuel cage shipment, and HEU spent fuel take back (requires 18-24 month lead).			❑ LEU fuel delivered ❑ HEU fuel take back in F257 flask
Conversion / Installation					Finalization of AECL involvement and use of staff and equipment (contracts drawn).	Dry run of the defueling fuelling procedures as well spent fuel flask removal procedures		❑ De-fuel reactor ❑ Load new fuel and commission ❑ Commissioning tests.

5. Conclusion

Activities for the conversion of the Jamaica SLOWPOKE core have started with funding provided by the US-DOE. The work will be undertaken by AECL staff, external contractors and ICENS staff. The issues of independent regulatory oversight in the Jamaican jurisdiction are being addressed and we are awaiting final drafting instructions for the regulations to be issued by the Government. The interim oversight committee will be assisted by the IAEA in the review of safety and analysis reports and other relevant documents. It is envisaged that the conversion operation from shutdown to commissioning of the new core can be completed within six weeks, and provided that the fuel fabrication process can be completed in the estimated 18 months, the project should be accomplished by the end of 2014.

In March 2014, the ICENS Jamaica SLOWPOKE will be celebrating its 30th year of operation, and a number of events are being planned to celebrate this milestone. While the focus will be on recognizing and highlighting the efforts and outputs of the institution and its staff over the years, in carrying out research and developing competences in the field of nuclear science, it is hoped that the new core will renew efforts to promote the peaceful uses of nuclear technology, and position ICENS as a catalyst for interdisciplinary programmes within the University and the Caribbean region for the next 30 years.

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