RSP-0299

Canadian Nuclear Safety Commission

R550.1 Survey of Design and Regulatory Requirements for New Small Reactors, Contract No. 87055-13-0356

Final Report

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Executive Summary

The objectives of this report are to perform a design survey of small modular reactors (SMRs) with near-term deployment potential, with a particular emphasis on identifying their innovative safety features, and to review the Canadian nuclear regulatory framework to assess whether the current and proposed regulatory documents adequately address SMR licensing challenges.

SMRs are being designed to lower the initial financing cost of a nuclear power plant or to supply electricity in small grids (often in remote areas) which cannot accommodate large nuclear power plants (NPPs). The majority of the advanced SMR designs is based on pressurized water reactor (PWR) technology, while some non-PWR Generation IV technologies (e.g., gas-cooled reactor, lead-cooled reactor, sodium-cooled fast reactor, etc.) are also being pursued. Several international nuclear technology providers are advancing their small nuclear reactor development programs; for example, standard design approval was given for the SMART reactor in South Korea, and several others are currently under construction (e.g., CAREM in Argentina, KLT-40S in Russia) or in various licensing stages (NuScale and mPower in the USA, VBER-300 in Russia). These six advanced water-cooled reactor designs were studied in this report along with a gas-cooled reactor (StarCore) which is proposed by a Canadian company.

The SMR designs, and the ways in which they incorporate inherent and passive safety characteristics, are summarized in this report. The report also identifies, to the extent that information is publicly available, the vendors' research and development efforts in validating their designs. In the light of the Canadian Nuclear Safety Commission's interest in how SMRs may cope with a Fukushima-type event, the author attempted to identify the vendors' post-Fukushima responses. *Caveat emptor* that not all claims that were made by the vendors could be verified, since detailed SMR design information in the public domain is scarce.

The challenges, issues and considerations in licensing small nuclear reactors in Canada (from the authors' point of view) are discussed. Then, REGDOC-2.5.2, the proposed regulatory document for new nuclear power plants, along with other related regulatory documents, was reviewed against these licensing issues.

This report asserts that the SMR designs examined here could be grouped into 2 categories regarding licensing: one category being a smaller version of conventional reactor technology with some safety design enhancements, the other category being SMRs heavily relying on inherent and passive safety characteristics. While the regulatory framework in Canada appears adequate for licensing the first category of SMRs, the second category of SMRs may pose some regulatory challenges; i.e., there are no clear instructions or protocols on how inherent and passive safety characteristics are credited.

In order to address this potential gap, the following recommendations are proposed to the CNSC:

1. The existing regulatory framework is generally adequate for the large mostly-conventional SMRs such as KLT-40S and VBER-300, with a couple of major exceptions. These exceptions are a ship-based design, and the exclusion of LBLOCA from the design basis. The authors believe that these exceptions can be treated on a case-by-case basis.
2. For advanced SMRs, much of REGDOC-2.5.2 applies. However, it lacks a framework for applying risk-informed regulatory judgements on innovative features. A supplement to REGDOC-2.5.2 applicable to advanced SMRs would address this gap and deal with the high-level issues identified in Section 8. The supplement should suffice to put a pre-project design review on a good foundation.
3. REGDOC-2.5.2 should include requirements and guidance on applying a graded approach to advanced SMR designs.
4. For advanced SMRs, some of the increased safety is achieved not through inherent means (e.g., low core power) but by engineered passive systems. These may be given much more credit in the safety case than active systems, in terms of reliability – i.e., they may not (need to) be redundant. While this might be justified on a case-by-case basis, formal guidance on credit for passive systems in SMRs would be a useful regulatory tool.
5. If an operator proceeds to a construction licence, more detailed review criteria will be needed. The US is developing specific Standard Review Plans for each SMR design it is asked to review. The same model could also be used in Canada – i.e., it might be similar to the "Guidance" sections in REGDOC-2.5.2 except parts would be design-specific.
6. Since all but one of the designs that are examined in this report have been developed outside of Canada, the CNSC could liaise with the regulator of the country where the design is to be deployed first, or where it has had pre-project licensing review (e.g. Korea, Argentina, and the U.S.). The purpose would be to fast-track an understanding of the design and the regulatory issues addressed in the vendor's or first-adopter's country.

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List of Acronyms

AC	Alternating Current
ACS	Adjust and Control System
AECL	Atomic Energy of Canada Limited
AIC	Ag-In-Cd
ATWS	Anticipated Transient Without Scram
B&W	Babcock and Wilcox
BDBA	Beyond-Design-Basis Accident
BOP	Balance of Plant
BPR	Burnable Poison Rod
CANDU	Canada Deuterium Uranium
CAPCN	High Pressure Natural Circulation Rig
CAREM	Central Argentina de Elementos Modulares
CNEA	Comision National de Energia Atomica
CNSC	Canadian Nuclear Safety Commission
CRA	Control Rod Assembly
CRDM	Control Rod Drive Mechanism
CSB	Core Support Barrel
CVCS	Chemical and Volume Control System
DBA	Design-Basis Accident
DC	Direct Current
DEC	Design Extension Condition
DHRS	Decay Heat Removal System
DOE	United States Department of Energy
EA	Environmental Assessment
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EIS	Emergency Injection System
EPZ	Emergency Planning Zone
ETS	Energy Transfer System
FA	Fuel Assembly
FE	Fuel Element
FES	Fast Extinction System
FNPP	Floating Nuclear Power Plant
FOAK	First Of A Kind
FPU	Floating Power Unit

FSS	Fast Shutdown System
GCR	Gas Cooled Reactor
HF	Human Factors
HTGR	High Temperature Gas-Cooled Reactor
HTR-10	10MW High Temperature Gas-cooled reactor-test Module
HTTR	High Temperature Testing Reactor
HWR	Heavy Water Reactor
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IHX	Intermediate Heat Exchanger
INIS	International Nuclear Information System
INL	Idaho National Laboratory
IPWR	Integral Pressurized Water Reactor
IRV	Integral Reactor Vessel
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Company
LBLOCA	Large Break Loss of Coolant Accident
LMR	Liquid Metal cooled Reactor
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCP	Main Circulation Pump
MMIR	MDS Nordion Medical Isotope Reactor
MSK-64	Medvedev-Sponheuer-Karnik Scale
MSLB	Main Steam Line Break
MTC	Moderator Temperature Coefficient
MWD	Mega Watt Days
NEA	Nuclear Energy Agency
NOC	Normal Operating Condition
NPP	Nuclear Power Plant
NRU	National Research Universal (reactor)
NRX	National Research Experimental (reactor)
NSSS	Nuclear Steam Supply System
OECD	Organization for Economic Cooperation and Development
PIRT	Phenomena Identification and Ranking Table
PRHRS	Passive Residual Heat Removal System
PSV	Pressure Safety Valves
PWR	Pressurized Water Reactor
PZR	Pressurizer

R&D	Research and Development
RCCS	Reactor Core Cooling System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDT	Reactor Drain Tank
ROPS	Reactor Overpressure
RP	Reactor Plant
RPV	Reactor Pressure Vessel
RPVI	Reactor Pressure Vessel Internals
RSS	Reactor Shutdown System
SBLOCA	Small Break Loss of Coolant Accident
SDR	SLOWPOKE Demonstration Reactor
SG	Steam Generator
SLOWPOKE	Safe Low Power Critical Experiment (reactor)
SMART	System-integrated Modular Advanced ReacTor
SMR[1]	Small Modular Reactor
SSC	Structures, Systems and Components
TRISO	TRisructural ISOtropic (fuel)
USNRC	United States Nuclear Regulatory Commission
ZED-2	Zero Energy Deuterium (reactor)

[1] The abbreviation “SMR” does not have a universally-agreed definition. In this report, SMR is defined as “Small Modular Reactor”.

1. Introduction

1.1 Background

There has been a strong growth of international interest and activities in the development of small nuclear reactor technology in recent years. Currently, research is being carried out on more than 45 advanced small modular reactor (SMR) concepts in 10 International Atomic Energy Agency (IAEA) Member States[2]. These include Argentina, Canada, China, France, India, Italy, Japan, Republic of Korea, Russian Federation, South Africa, and the United States of America. SMRs encompass various reactor technology types including: light water reactors (LWRs), heavy water reactors (HWRs), gas cooled reactors (GCRs) and liquid metal cooled reactors (LMRs).

Several international vendors are developing small nuclear reactors for commercial power generation. These reactors are mostly integral pressurized water reactors (IPWRs) based on existing pressurized water reactor (PWR) technology. However, in IPWR, the main nuclear steam supply system (NSSS) including the pressurizer and steam generators are integrated with the reactor core in the pressure vessel, thus eliminating large pipe interconnections between the major components and reducing the likelihood of LBLOCA. Currently, four IPWR designs are being reviewed by the U.S. Nuclear Regulatory Commission (USNRC)[3]. They are mPower[4] from Babcock and Wilcox (180 MWe per unit, 2-unit plant), NuScale[5] backed by Fluor (45 MWe per unit, 12-unit plant), Westinghouse SMR[6] by Westinghouse Electric Company (225 MWe, single unit plant), and HI-SMUR[7] by Holtec International (160 MWe, single unit plant).

Some vendors are developing small reactors to supply both electricity and heat to remote communities. These remote communities are located far away from major electricity grids, and they have to generate their own power using diesel generators. The Government of Canada estimates that there are approximately 175 aboriginal and northern off-grid communities across Canada [1] that heavily rely on diesel-powered electricity generation. The power generation costs at these locations are very high due to the cost of the fuel and its transportation. The small reactors are meant to replace diesel generators at remote sites and their output power is comparable to those of industrial diesel generator-sets (e.g. 5~30 MWe). Examples of such

[2] Small and Medium Sized Reactors (SMRs) Development, Assessment and Deployment, http://www.iaea.org/NuclearPower/SMR/
[3] U.S.NRC, Advanced reactors and small modular reactors. http://www.nrc.gov/reactors/advanced.html
[4] Generation mPower, http://www.generationmpower.com/
[5] NuScale Power, http://www.nuscalepower.com/
[6] NexStart SMR Alliance, http://www.nexstartalliance.com/WestinghouseSMR.aspx
[7] SMR, http://www.smrllc.com/

reactors are Gen4Module[8] (25 MWe, lead-bismuth cooled fast reactor), StarCore[9] (10 MWe, gas cooled reactor) and Toshiba 4S[10] (10 MWe, sodium fast reactor).

The idea of licensing small nuclear reactors in Canada has been discussed at several nuclear technology conferences such as the International Technical Meeting on Small Reactors[11]. As part of the scope of work of this report, CNSC has requested that the current regulatory approach be assessed so it can be adjusted to address future SMR licensing in Canada. The potential SMR licensing challenges come from the following considerations:

- The Canadian regulatory requirements for nuclear power generation traditionally focused on the safety aspects of large CANDU power reactors. While the regulations have become more technology-neutral, the new reactor technologies such as IPWRs may have various safety issues that were not previously considered in the conventional power reactors.
- Modern SMRs incorporate innovative safety features to eliminate certain accident conditions such as pump seal failure, that are considered within the design basis of a conventional NPP. For instance, the IPWR aims to reduce the large break loss-of-coolant accident (LBLOCA) risk by including the pressurizer and steam generators inside the pressure vessel, thereby eliminating large interconnecting RCS piping. Thus, the design basis accidents traditionally considered in reactor licensing require re-examination with respect to their applicability in SMRs.
- It is expected that SMR vendors will request that a different regulatory process be applied to SMR licensing. For instance, the size of an SMR is expected to be a fraction of a traditional nuclear reactor, and the traditional licensing fee schedule could be disproportionately high per installed mega watt [2]. Also, some SMRs are planned to be manufactured and fuelled in a central licensed facility and transported to the operating sites [3] which may require that the reactor and the manufacturing / fuelling plant both be separately licensed.
- The difficulties in accessing small remote nuclear reactors for inspection could raise additional maintenance and security challenges.

This report identifies these potential licensing challenges for developing SMR technologies. The study includes a review of several IPWR and PWR designs that are either licensed or in the licensing process in Argentina, Russia, Republic of Korea, and the USA. The Contractor also examined a high temperature gas cooled reactor (HTGR) technology proposed by a Canadian start-up company designed to provide power and heat to Canadian remote communities, and to operate without a licensed operator on-site. These SMR technologies were examined to identify

[8] Gen4 Energy, http://www.gen4energy.com/
[9] SarCore Nuclear, http://starcorenuclear.ca/
[10] http://www.toshiba.com/tane/products_4s.jsp
[11] Canadian Nuclear Society Conferences, http://www.cns-snc.ca/conferences

the unique safety characteristics that are not commonly found in conventional power reactors. Then the Contractor assessed the applicability of the CNSC draft regulatory document for nuclear power plant design [4][12] to licensing of innovative SMR designs in Canada.

1.2 The Contractor

Hatch[13] (the Contractor) carried out the research, and prepared this final report for the Canadian Nuclear Safety Commission (CNSC) as per Contract No. 87055-13-0356 (hereafter, the Contract). Hatch is the Canadian engineering consulting company located in Mississauga, Ontario.

1.3 Scope

The scope of this study is outlined in Amendment #1 of Contract 87055-13-0356 as follows:

> *"1. Provide a concise description of the vendors' small reactor designs, with particular emphasis on their innovative safety features;*
>
> *2. Examine the CNSC regulatory framework applicable to small reactors and determine its sufficiency to address safety aspects of new small reactor designs and their claimed safety features;*
>
> *3. Discuss the identified research and development (R&D) activities that support the vendors' small reactor designs and their safety feature;*
>
> *4. Discuss the vendors' approaches in enabling their reactor designs to fully resist to the disastrous effects similar to those encountered at the Fukushima Daiichi site; and*
>
> *5. Discuss any identified potential licensing challenges that could emerge when licensing the vendors' small reactor designs in Canada."*

An addition to the scope stated:

> *"In case that collected information for Items 3 and 4 is not sufficient to address associated issues, the Contractor should give their own critical view on what else has to be considered in the R&D activities and what measures to be taken in protecting the reactor design from the exceptional geophysical events so that the vendor's reactor design can effectively respond to the anticipated transients and postulated accidents, and in mitigating the consequences of severe accidents"*

The Contract specified following seven small reactor technologies for consideration:

- KLT-40S (OKBM Afrikantov, Russia)
- VBER-300 (OKBM Afrikantov, Russia)

[12] Note: the document was issued after this document was prepared.
[13] http://hatch.ca/

- SMART (KAERI, Republic of Korea)
- CAREM (INVAP and CNEA, Argentina)
- StarCore (StarCore Nuclear, Canada)
- NuScale (NuScale Power, USA)
- mPower (Babcock and Wilcox, USA)

1.4 Definition of Small Reactor

The definition of a "small" reactor is not universally agreed. RD-367[5] defines a small reactor as:

> *"A small reactor facility is defined as a reactor facility containing a reactor with a power level of less than approximately 200 megawatts thermal (MWt) that is used for research, isotope production, steam generation, electricity production or other applications."*

The US Department of Energy (USDOE) defines[14] an SMR as:

> *"Small Modular Reactors (SMRs) are nuclear power plants that [are] smaller in size (300 MWe or less) than current generation base load plants (1,000 MWe or higher). These smaller, compact designs are factory-fabricated reactors that can be transported by truck or rail to a nuclear power site."*

The IAEA defines small and medium sized reactors as [6]:

> *"According to the classification adopted by the IAEA, small reactors are reactors with an equivalent electric power of less than 300 MW(e) and medium sized reactors are reactors with an equivalent electric power of between 300 and 700 MW(e)".*

These varying definitions can be a source of confusion: for example, among the SMRs described in this report, SMART (330 MWth, 90 MWe) and mPower (530 MWth, 180 MWe) would be considered small reactors by the US and by the IAEA, but not in Canada using RD-367.

In the authors' view, size cannot be the only determinant of what constitutes an SMR; equally important are the safety characteristics. Small reactors generally fall into two categories:

- Lower power versions of conventional reactors with some safety enhancements, and
- Advanced reactors with much more passive and inherent safety characteristics.

In most cases the existing Canadian regulatory framework will apply well to the first category: a small *conventional* CANDU or PWR should generally meet the same requirements, and be licensed the same way, as a larger version. Of course the variety of small reactor designs does not fall unambiguously into these two categories, and even reactors in the first category may

[14] http://www.energy.gov/ne/nuclear-reactor-technologies/small-modular-nuclear-reactors

have some special safety design characteristics that need special regulatory treatment. For example the VBER-300 is a relatively high-powered small PWR (917 MWth, 325 MWe), which, from the limited information publicly available, has mostly conventional PWR safety characteristics. There are however two major differences: the reactor is designed to be mounted on a ship as well as on land, and the short lengths of large-bore piping might be used to make a case to remove Large Break LOCA (LBLOCA) from the design basis. At first cut, therefore, the existing CNSC regulatory framework for power reactors would apply, with special reviews of these two characteristics.

Small reactors in the second category are not specifically accounted for in the Canadian regulatory framework. The later discussion in this report shall show that some power reactor requirements do not apply, and conversely that the compensating safety benefits of such small reactors are not reflected in the requirements. At the extreme, a reactor can have sufficient inherent/passive characteristics that: power is limited indefinitely and inherently to safe levels when control rods are withdrawn; heat can be removed indefinitely and passively to the surroundings; and extreme external events cannot cause significant doses outside the facility. SMRs in the second category have at least some of these characteristics and will be the focus of the regulatory assessment in this study.

2. Methodology[15]

2.1 Design Data Collection

The information on reactor designs and their innovative safety features was gathered from the following sources.

- International and domestic government organization databases: design information was collected from the International Atomic Energy Agency (IAEA)[16], the Nuclear Energy Agency (OECD-NEA)[17], the U.S Department of Energy (U.S. DOE)[18], and the U.S. Nuclear Regulatory Commission (U.S.NRC)[19] online databases.
- Nuclear consulting services databases were accessed: The World Nuclear Association[20] and UxC Consulting Company[21] maintain a collection of small nuclear reactor design information.
- Compendex[22] search: Compendex, the computerized version of the Engineering Index, is a comprehensive engineering bibliographic database. Compendex currently contains over 15 million records and references over 5,000 international sources including journals, conferences and trade publications. The individual reactor design and related safety features were searched in Compendex.
- Vendor brochures: the latest marketing brochures and design catalogues were directly obtained from the vendors.
- Direct vendor interviews: In cases where design data was not available from public sources, the vendors were directly contacted for design information.

2.2 Key Safety Characteristics and Technical Innovations

The new small reactor vendors claim much of their enhanced safety based on incorporation of various inherent and passive safety characteristics in addition to active safety components. The active safety devices include safety shutdown systems such as neutron-absorbing rods and liquid poison injection system, typically found in a light water cooled reactors. This report focuses on the inherent and passive safety design characteristics that are not typically found in the conventional light water reactors. The discussion of conventional active safety components is

[15] Note that formal reports referenced in this report are listed in Section 11; where a web-site was the *only* source of a piece of information, it is listed as a page footnote because of its possibly transitory nature.
[16] International Atomic Energy Agency, https://aris.iaea.org/
[17] Nuclear Energy Agency, https://www.oecd-nea.org/
[18] Idaho National Laboratory site, https://smr.inl.gov/
[19] Nuclear Regulatory Commission, http://www.nrc.gov/reactors/advanced.html
[20] WNA Reactor Database, http://world-nuclear.org/NuclearDatabase/Default.aspx?id=27232
[21] Ux Consulting Company, http://www.uxc.com/smr/
[22] Compendex Engineering Village, http://www.engineeringvillage.com/search/quick.url

brief. The definitions of inherent and passive safety used in this report are provided in APPENDIX A.

2.3 Post-Fukushima Considerations in Design

The post-accident task force reports from U.S.NRC [7], CNSC [8] and European Commission [9] were examined to understand the key areas of technical lessons-learned from the Fukushima-Daiichi accident that need to be considered by the SMR designers. These technical lessons-learned are compiled in Table 1. Then, the small reactor designs and their safety characteristics were examined to see to what extent these lessons were incorporated or addressed by the design.

The recommendations from the task forces include enhancement of the regulatory framework, security upgrades, and enhancement of the organizational accident management capabilities. Since this report is mainly concerned with the technical safety characteristics of the new small reactors, the administrative safety aspects, such as improving emergency preparedness and streamlining of the regulatory framework, are excluded from the discussion; only the technical safety aspects are discussed with respect to the lessons learned from the Fukushima event. Whenever the vendor did not specifically discuss how the design incorporated the Fukushima lessons learned, an independent assessment was made for the areas summarized in Table 1, based on the available design and safety characteristics.

Table 1 Lessons learned from the Fukushima-Daiichi accident

Key areas	Lessons Learned
Multiple concurrent external events	Major external events such as earthquake, fire and flood, normally considered separate design-basis events, can occur near- simultaneously. The design shall examine the possibility of multiple events occurring simultaneously (i.e. earthquake, flooding, internal fire, etc).
Extended loss of offsite power	The offsite power can be lost for an extended period of time due to the consequences of a natural disaster that destroys the external grid or power sources. On-site stored energy (such as batteries) may not be available for the duration before AC power is restored.
Emergency core cooling and ultimate heat sink	Diverse means to cool the core in the absence of AC power along with a loss of the ultimate heat sink shall be available.
Protection of essential equipment	Safety-related systems shall be protected from the external hazards and internal events by separation of redundant equipment, by physical protection, or by hardening.
Spent fuel pool integrity and cooling	Spent fuel pool make up capability and instrumentation to monitor the condition of the pool are required.

Containment integrity	Containment integrity shall be protected (i.e. the plant shall have means to mitigate hydrogen explosion and over-pressurization)
Multi-unit event	In a multi-unit plant, an accident occurring in a unit shall not impact the safety of other units.

2.4 Regulatory Aspect

The regulatory assessment of small nuclear reactors was performed by completing the following tasks:

- Review of the design characteristics of the seven reactor examples listed in the contract scope.
- Review of key high-level CNSC documents on reactor design:
  - The existing CNSC regulatory document, RD-367 [5], on small reactor facilities.
  - A draft of the proposed replacement for the current design requirements document for nuclear power plants (NPPs), RD-337 [10]. The scope of RD-337 covers new water-cooled nuclear power plants. Its draft replacement, REGDOC-2.5.2 [4] is an update of RD-337 which likewise applies to water-cooled power reactors, within which set it is intended to be technology-neutral. Neither RD-337 nor REGDOC-2.5.2 excludes SMRs (other than non-water-cooled designs) nor do they specifically address novel features of some SMRs.
- Brief review of selected requirements which *might* not apply well to SMRs:
  - CNSC cost recovery fees [11]
  - CNSC requirements for financial guarantees for decommissioning [12]
  - CNSC guide to decommissioning planning [13]
  - Proposed Bill C-22 on Nuclear Liability and Compensation [14]
- Compilation of typical safety-related innovations of the sampled SMRs.
- Development of potential challenges posed by SMRs to current power reactor licensing requirements and practices.
- Detailed review of draft REGDOC-2.5.2 to see how well it could be applied to SMRs. The authors also looked briefly at issues faced by other regulators in reviewing SMRs, notably the USNRC. Some of these will also be common to Canada.
- Recommendations are made to the CNSC based on the above reviews as to how they may wish to proceed in adapting their regulatory framework to SMRs.

3. Caveats

The reader should be aware of the limitations in this report:

- Only public information on SMRs was used and the technical design information is summarized in Section 4 . Unfortunately, most public information gives only general plant and reactor characteristics, and in any case is highly-coloured by marketing. There is very limited design detail available in most cases, and it was not possible to critically review or confirm the safety claims. Some designs (e.g. StarCore) have almost no public information even at the most basic level. In addition, Hatch found that the design data from different sources are often inconsistent. This is probably due to the fact that the designs are still being refined, and continuous modification is expected to occur until the final design is licensed in the vendor's country. The data condition forced the authors to use many qualified statements in this report such as "seems", "appears to" etc. However, since the purpose of this study was to identify major design innovations and the resultant licensing challenges, the authors believe that a high-level knowledge of the design was in most cases sufficient to carry out the study.
- The reports listed in Section 2.4 were reviewed as it is believed that such effort would help identify the more obvious issues in relation to SMR licensing in Canada. The authors did not review *all* CNSC reports, particularly at the detailed requirements and/or guidance levels.
- The required level of security is likely to be an issue for SMRs but there is not enough detail in public information on security (for obvious reasons) to assess the issue.

4. Reactor Technologies

This section of this report addresses the first project scope listed in Section 1.3, which says:

1. Provide a concise description of the vendors' small reactor designs, with particular emphasis on their innovative safety features.

4.1 KLT-40S

The World Nuclear Association[23] provides a brief summary of the KLT-40S reactor as follow:

"Russia's KLT-40S from OKBM Afrikantov is a reactor well proven in icebreakers and now – with low-enriched fuel – proposed for wider use in desalination and, on barges, for remote area power supply. The 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating (or 38.5 MWe gross if power only). The unit is designed to run 3-4 years between refuelling with on-board refuelling capability and used fuel storage. At the end of a 12-year operating cycle the whole plant is taken to a central facility for overhaul and storage of used fuel. Two units will be mounted on a 20,000 tonne barge to allow for outages (70% capacity factor).

Although the reactor core is normally cooled by forced circulation (4-loop), the design relies on convection for emergency cooling. Fuel is uranium aluminum silicide with enrichment levels of up to 20%, giving up to four-year refuelling intervals. A variant of this is the KLT-20, specifically designed for FNPP. It is a 2-loop version with same enrichment but 10-year refuelling interval.

The first floating nuclear power plant, the Akademik Lomonosov, commenced construction in 2007. Due to insolvency of the shipyard the plant is now expected to be completed in 2016."

The cutaway view of the FNPP vessel is shown in Figure 1 and the computer rendered picture of the reactor is shown in Figure 2. The loading of the reactor plant onto the barge is shown in Figure 3.

[23] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/

Figure 1 The floating nuclear power plant that houses two (2) KLT-40S reactor plants [15]

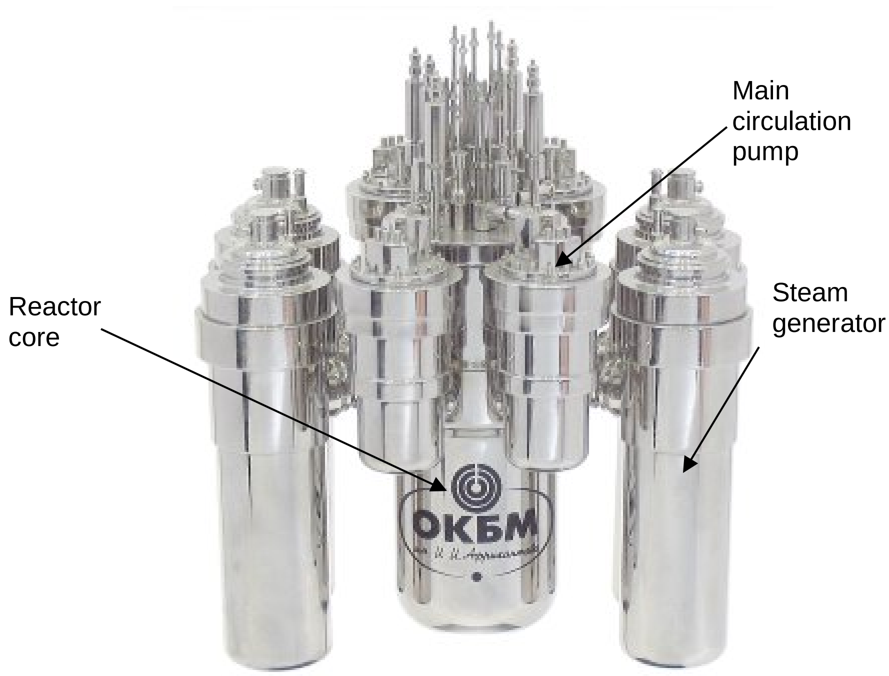

Figure 2 KLT-40S reactor plant (RP) including the reactor core, four (4) steam generator (SG) and four (4) main circulation pump (MCP) hydraulic chambers[15]

Figure 3 KLT-40S reactor plant unit are being loaded to the floating nuclear power plant (FNPP)[24,25]

The technical data of KLT-40S reactor from the IAEA Status report [16] is reproduced in Table 2 below.

Table 2 Summary of KLT-40S Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	2 × 150	MW(th)
Power output, gross	2 × 35	MW(e)
Power output, net	2 × 30	MW(e)
Plant efficiency	23.3	%
Mode of operation	Load follow	
Plant design life	40	Years

[24] http://sdelanounas.ru/blogs/41388/
[25] http://theconversation.com/russias-floating-nuclear-plants-to-power-remote-arctic-regions-19994

Plant availability	85	%
Primary coolant	Light water	
Moderator	Light water	
Thermo dynamic cycle	Rankine steam cycle	
Non-electric applications		
- **Potable water**	20,000 – 100,000	m^3/hour
- **District heat**	2 × 305	GJ/hr
Reactor core		
Fuel column height	1.2	m
Average linear heat rate	14.0	kW/m
Average fuel power density	117.8	kW/kgU
Average core power density	119.3	MW/m^3
Fuel material	UO_2 in inert metal matrix (silumin)	
FE type	Cylindrical rod	
Cladding material	Zirconium alloy	
FE outer diameter	6.8	mm
Lattice geometry	Triangular	
Number of fuel assemblies	121	
Fuel enrichment	<19.75	Weight %
Fuel cycle length	28	Months
Average discharge burnup	45.4	MWd/kg
Burnable absorber	Gadolinium	
Mode of reactivity control	Control rods	
Mode of reactor shutdown	Control rods	
Control rod absorber material	Dysprosium titanate, boron carbide	
Soluble neutron absorber	Cadmium nitrate	

Primary coolant system		
Core coolant inlet temperature	280	ºC
Core coolant outlet temperature	316	ºC
Coolant mass flow rate	761	kg/s
Operating pressure	12.7	MPa
Reactor vessel		
Inner diameter of cylindrical shell	1920	mm
Wall thickness of cylindrical shell	128	mm
Steam generator		
Type	Vertical, coiled, once-through	
Number	4	
Total tube surface area	284	m^2
Number of HX tubes	100	
Tube outside diameter	22	mm
Tube material	Titanium alloy	
Primary circulation system		
Circulation type	Forced	
Pump type	Canned, centrifugal, single-stage, vertical, double-speed	
Number of pumps	4	
Pressurizer		
Type	External, gas-operated with gas cylinders	
Total volume	8.16	m^3
Containments		
Primary	Steel	
Secondary	System of a ship compartments	

4.1.1 General Description

Construction of a small-size floating nuclear cogeneration plant with KLT-40S reactor plants (RPs) is currently under way in Russia. The rector plant refers to the reactor and other primary components that are assembled as a single cluster unit. The floating nuclear power plant houses two KLT-40S reactor plants and two turbine-generator sets with cogeneration turbines; these two sets of equipment are installed alongside the two boards of the FNPP[15].

The FNPP is designed to be a standalone power plant facility. It handles the refueling operation and reactor maintenance with the internally installed equipment. The unit includes shielded fresh and spent fuel storage, and waste storage compartments to handle liquid and solid radioactive wastes generated during the normal operation and refuelling.

The description of the KLS-40S design is taken from IAEA Status report [16], and it is summarized below.

Reactor Plant

The KLT-40S RP is an advanced variant of PWR based on the commercial KLT-40 marine propulsion reactor. The reactors used in ice breakers underwent modernization to meet the latest Russian regulatory requirements, to increase the core lifetime and to improve the reliability of the components and the safety systems. The KLT-40S is a PWR housed in a pressure-vessel, which also contains the reactivity compensating rods and emergency shutdown rods. The actuators for the reactivity control devices, actuators for the emergency protection system, and the resistance and thermoelectric temperature transducers for the reactor temperature measurement are mounted on the top cover of the vessel.

The reactor plant consists of the following design features:

- Modular RP design: reactor, SGs and main coolant pumps (MCPs) are interconnected with short nozzles rather than long pipes
- Four-loop system with combined forced and natural circulation of primary coolant
- Pressurized primary circuit with canned pumps and leak-tight bellow-type valves
- Once-through coil SG
- External gas pressurizer system
- Eight (8) actuators for the reactivity control devices
- Three (3) actuators for the emergency shutdown system

Steam Generators

The SG tubing and the casing consists a modular coil-type vertical cylindrical surface heat exchanger (HX) where the primary coolant circulates on the tube side, and the secondary coolant on the shell side. It consists of:

- Leak-tight case with a short main coaxial nozzle that connects to the reactor unit, where the annuli function as the coolant inlet and outlet. The short nozzle reduces the risk of pipe failure;
- SG internals, including the SG cover with secondary coolant supply/removal nozzles and the coiled tubing system with flow restrictors.

Main Coolant Pump (MCP)

The single-stage, centrifugal, vertical main circulation pump (MCP) is attached to the reactor unit for primary coolant circulation. The MCP is operated with a canned and shielded double-speed (double-winding) asynchronous electric motor. Similar to the SG unit connection, the MCP is attached to the reactor plant with short connection to reduce the risk of connection pipe failures.

Fuel

The KLT-40S RP core is designed based on the nuclear ice breaker reactor technologies and it uses ceramic metal (cermet) type fuel where UO_2 is embedded in an inert metal matrix (silumin). The average U^{235} fuel enrichment is 14.1%. The core has a closely packed cassette structure to maximize the number of fuel elements (FEs) and fuel volume in the core for increased fuel cycles.

The fuel element is clad with smooth cylindrical corrosion-resistant zirconium alloy with a diameter of 6.8 mm. The core consists of 121 hexahedral shrouded fuel assemblies (FAs) in a triangular lattice with 100 mm pitch. The overall length of the assembly is 1670 mm and the heated height is 1200 mm. The diameter across of the assembly measures to be 98.5 mm. The fuel elements are placed in FAs at a regular triangular lattice pitch of 9.95 mm. Gadolinium-based burnable poison rods (BPRs) are used in FAs to compensate the reactivity change due to fuel burnup.

Spent fuel storage

The core is refueled after 3-4 years of operation with the fresh fuel supply available onboard. After the core refuelling, the spent nuclear fuel is also stored in the floating power unit (FPU). The spent fuel assembly (FA) and solid radioactive waste storage includes two components: the wet storage for decay heat removal immediately after FAs are defueled, and the dry storage for subsequent air cooling.

Three independent wet storage tanks are provided onboard for spent fuel and solid radioactive waste storage. Each tank has the storage capacity to hold the spent FAs from one reactor core. During normal operation, the wet storage is actively cooled by pumping the decay heat to the seawater through two intermediary loops. The wet storage is also passively cooled by evaporating water from the storage tank cofferdam into the ventilation system.

The dry storage consists of four independent storage containers in which leak-tight canisters are placed. Each storage container is capable of holding the inventory of spent FAs from one reactor core. The heat from the canisters in the storage containers is removed by open-loop ventilation.

Fuel handling complex

The fuel handling complex is used to perform a range of refueling operations including unloading of spent FAs from the reactor, their transportation and placement into wet storage tanks and subsequent transportation to dry storage containers, unloading of neutron sources, lower ends of emergency protection, resistance thermometer and thermoelectric transducer sleeves, and loading of fresh FAs into the reactor. The fuel handling complex is designed to handle high heat FAs because refuelling is performed only thirteen (13) days after reactor shutdown to maintain a high capacity factor, when the residual heat releases from spent FAs are still high.

The fuel handling complex includes:

- Shielded refuelling containers for spent FAs
- Alignment mechanisms installed on the reactor vessel, wet storage tanks and dry storage containers
- Hydraulic jacks, pump station, observation device, etc.

Pressurizer system

The KLT-40S uses an external gas pressurizer system to develop and keep the primary circuit pressure within the prescribed limits in all operation modes. The system includes:

- Four gas pressurizers
- Two main gas cylinder groups (6 cylinders in each group)
- Stand-by cylinder group (6 cylinders)
- Gas compressor
- Piping, valves, instrumentation

In the nozzle connecting the system to the reactor vessel, there is a flow restriction device intended to reduce primary circuit outflow rate in case of a pipe rupture.

4.1.2 Key Safety Characteristics and Technical Innovations in Design

The KLT-40S reactor plant incorporates various inherent and passive safety characteristics to mitigate accidents [16], in addition to the active safety systems.

Inherent safety features

The KLT-40S aims to eliminate or reduce hazards by incorporation of the following inherent safety features in the design.

- Small source term: the reactor size is much smaller than a conventional reactor and the core power density is lower, which make the decay heat removal better manageable in case of reactor shutdown.
- Negative reactivity coefficient for fuel temperature, coolant temperatures, and steam density, and negative power coefficients of reactivity.
- High thermal conductivity of the cermet fuel keeps the fuel at relatively low temperature and the stored energy in the matrix is correspondingly low.
- Use of gas pressurizer system excludes the pressure perturbation due to electric heaters failures.
- Compact modular design which connects the main equipment (reactor, SG and MCP) with short co-axial nozzles reduces the possibility of large and medium LOCAs due to absence of long large-diameter primary pipes.
- Installation of flow restrictors in the nozzles connecting the primary circuit systems with the reactor limits the coolant outflow rate in case of a break.
- Pressurized primary system with welded joints, glandless canned pumps and leak-tight bellow-type valves prevent the small break loss of coolant accident (SBLOCA) associated with pump of valve seal failure.
- Use of once-through SGs limiting the cooldown of the secondary circuit (and hence the increase in reactivity) in case of steam pipe break.
- SG with lower tube-side pressure (than sheet-side pressure) during normal operation to reduce the probability of inter-circuit leaks.

Passive safety systems

The passive safety systems in KLT-40S use natural convection and gravity, and stored energy such as compressed gas, to activate the safety equipment without a supply of external power. The following passive safety features are incorporated in the design:

- Normal reactor shutdown system with insertion of control rods into the core by gravity.
- Insertion of emergency shutdown rods using accelerating springs when the locking electromagnets are de-energized.
- Gas pressurizer system means that failure of electric heaters need not be considered.
- Passive system of emergency cool down through the SG unit.
- System of emergency water supply from hydraulic accumulators.

- Passive emergency heat removal system with natural coolant circulation and evaporation of the water stored in the tanks.
- Natural circulation flow in the primary circuit to cool the reactor core in case of MCP switches off.
- Passive reactor vessel bottom cooling system for a severe accident that ensures retention of the molten corium inside the reactor vessel.

Additional discussions

The floating nuclear plant comes with its own safety benefits and challenges. For instance, the availability of coolant in case of an emergency is much greater than a land-based reactor since the reactor is on water. However, there are additional safety considerations that are not necessary in a land based reactors. In addition to the nuclear safety systems, the structures, systems and components (SSCs) of the KLT-40S floating power unit have to incorporate the marine vessel safety features against the natural and man-caused impacts that can be encountered at the deployment location and the towage routes. The plant is designed to meet the requirements of OPB-88/97[17], the Sea Shipping Register of Russia [18] and other safety regulations that apply to a marine vessel. Equipment, machinery, safety-related systems and their attachment units withstand shock loads with acceleration not lower than 3 g in any direction, and maintain operability during heaving which is typical during FPU operation.

4.2 VBER-300

The World Nuclear Association describes the VBER-300 PWR[26], as shown in Figure 4, as follows:

OKBM Afrikantov's VBER-300 PWR is a 917 MWt, 295-325 MWe unit, the first of which is planned to be built in Kazakhstan. It was originally envisaged in pairs as a floating nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr. The reactor is designed for 60-year life and 90% capacity factor. It has four external steam generators and a cassette core with 85 standard VVER fuel assemblies enriched to 5% and 48 GWd/tU burn-up. Versions with three and two steam generators are also envisaged, of 230 and 150 MWe respectively. Also, with more sophisticated and higher-enriched (18%) fuel in the core, the refuelling interval can be pushed from two years out to five years (6 to 15 years fuel cycle) with burn-up to 125 GWd/tU. A 2006 joint venture between Atomstroyexport and Kazatomprom set this up for development as a basic power source in Kazakhstan, then for export. It is also envisaged for use in Russia, mainly as cogeneration unit. It is considered likely for near-term deployment.

Figure 4 VBER-300 Reactor Plant including steam generator and main coolant pumps [19]

[26] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/

The technical data summary of VBER-300 reactor is taken from the IAEA status report [20] and shown in Table 3.

Table 3 Summary of VBER-300 Reactor Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	917	MW(th)
Power output, gross	325	MW(e)
Power output, net	305.6	MW(e)
Plant efficiency	33	%
Mode of operation	Load follow	
Plant design life	60	Years
Plant availability	Over 90	%
Primary coolant	Light water	
Moderator	Light water	
Thermo dynamic cycle	Indirect Rankine steam cycle	
Reactor core		
Active core height	3.53	m
Equivalent core diameter	2.285	m
Average linear heat rate	95	kW/m
Average fuel power density	21.3	kW/kgU
Average core power density	63.4	MW/m^3
Fuel material	UO_2	
FE type	Smooth-rod, Cylindrical	
Cladding material	Zirconium alloy	
Lattice geometry	triangular	
Number of fuel assemblies	312	
Fuel enrichment	4.95	Weight %

Fuel cycle length	72	Months
Average discharge burnup	47	MWd/kg
Burnable absorber	Gd_2O_3	
Mode of reactivity control	Control rods	
Mode of reactor shutdown	Control rods	
Control rod absorber material	Dysprosium titanate, boron carbide	
Soluble neutron absorber	H_3BO_3	
Primary coolant system		
Core coolant inlet temperature	292	ºC
Core coolant outlet temperature	327.5	ºC
Operating pressure	16.3	MPa
Reactor vessel		
Inner diameter of cylindrical shell	3400	mm
Wall thickness of cylindrical shell	205	mm
Total height	8265	mm
Steam generator		
Type	Vertical, coiled, once-through	
Number	4	
Total tube surface area	2624	m^2
Number of HX tubes	3850	
Tube outside diameter	10	mm
Tube material	Titanium alloy	
Primary circulation system		
Circulation type	Forced	
Pump type	Canned, centrifugal, single-stage, vertical, double-speed	

Number of pumps	4	
Pressurizer		
Type	External, steam	
Total volume	42	m^3
Containments		
Primary	Cylindrical Steel	
Secondary	Cylindrical concrete	

4.2.1 Description

The following description of the VBER-300 is from the IAEA Status report [20].

The VBER-300 reactor plant (RP) is a medium-size thermal neutron power reactor for ground-based nuclear power plants and nuclear cogeneration plants, as well as for floating nuclear power plants (FNPPs) and desalination complexes.

The VBER-300 RP design is an evolution based on modular marine propulsion reactors such as those used in the nuclear ice breakers. The vessel size and fuel mass are increased for greater thermal power output while the RP appearance and main design solutions are kept as close as possible to those of marine propulsion reactors. The design is being developed using the experience of VVER-type reactors operation and achievements in the field of NPP safety.

Characteristic features of the VBER-300 RP are:

- Possible application for both ground-based and floating NPPs.
- Use of nuclear ship building technologies, engineering solutions and operating experience of VVER reactors.
- Compliance with the safety requirements for new-generation NPPs.
- Possibility to enlarge or reduce the source power using only unified VBER-300 RP equipment (RPs consisting of two to six loops) as shown in Figure 5, where 4 loop RP is the reference design [19].

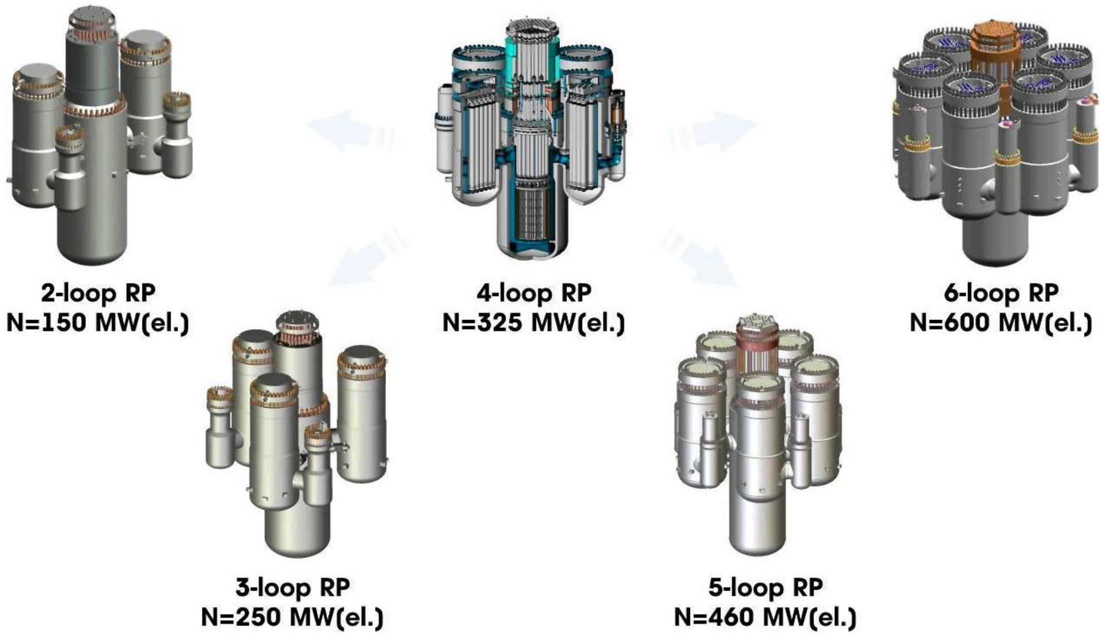

Figure 5 Various power level configurations of VBER reactor plants

Reactor Plant

VBER-300 is a PWR using light water as the primary coolant and moderator. The primary coolant is cooled in the once-through steam generators (SGs) that generate slightly superheated steam and direct it to the turbine. Part of the steam is taken off from the turbine to heat up the district heating circuit fluid.

Main RP design features are as follows:

- Modular RP design: reactor, SGs and main coolant pumps (MCPs) are interconnected with short nozzles without long pipes.
- Four-loop system with forced and natural circulation of primary coolant.
- Pressurized primary circuit with canned pumps and leak-tight bellow-type valves.
- Once-through coil SG.
- External steam pressurizer system.

The general view of the reference VBER-300 reactor unit is shown in Figure 6.

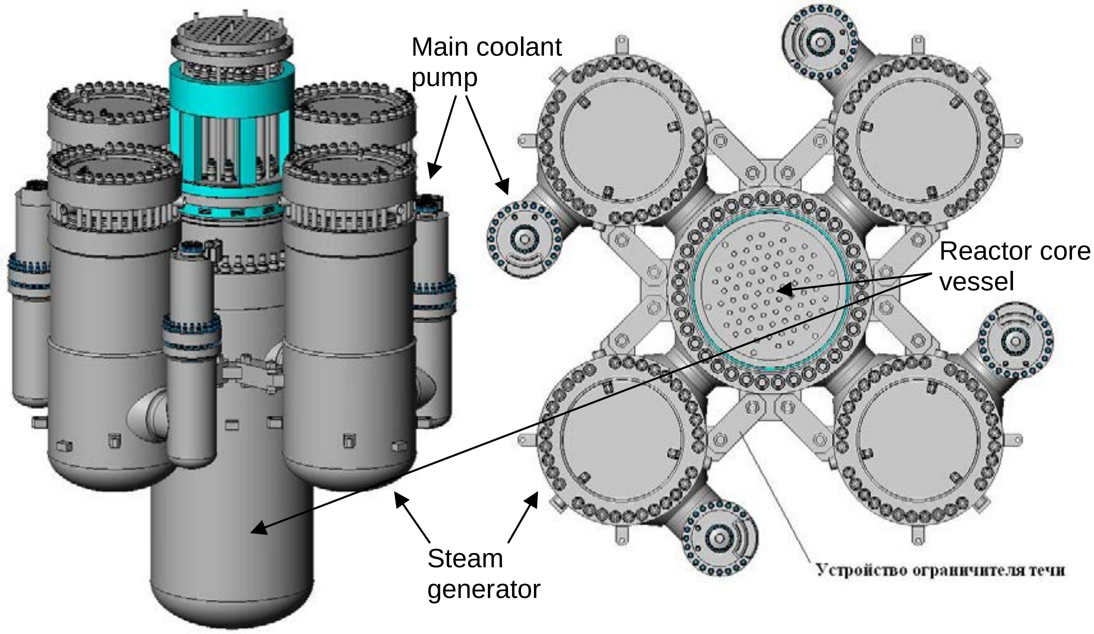

Figure 6 The reference 4-loop design of VBER-300 reactor plant

The 60 year life vessel system consists of a reactor vessel and four ‘SG and MCP’ unit vessels connected to the reactor vessel. The reactor vessel is a welded cylindrical shell with an elliptical bottom, four main nozzles and a flanged upper portion. The reactor vessel accommodates the reactor core, reactor cavity and reactivity control devices. Each ‘SG and MCP’ unit consists of a SG with its vessel connected with the MCP hydraulic chamber by a short coaxial nozzle. The modular design helps to minimize reactor unit mass, overall dimensions and reactor structural volume. The design excludes long main circulation pipes and greatly reduces the risk of large and medium break LOCAs.

Reactor core

The VBER-300 employs the cassette design of the reactor core. The fuel is below 5% enriched uranium dioxide pellets of 7.6 mm diameter. The fuel assemblies (FAs) of the VBER-300 core are shroud-less skeleton type TVSA[27] FA which was originally developed by OKBM for the VVER-1000 reactor.

Burnable gadolinium-spiked uranium dioxide pellets are used to compensate the reactivity change due to fuel burnup. The reactivity compensating elements have the same geometry as the regular fuel element geometry. The gadolinium added fuel elements and boric acid solution injection into the primary coolant are the main means of compensating for initial excess reactivity and burnup.

[27] http://www.tvel.ru/wps/wcm/connect/tvel/tvelsite.eng/products/nuclearproducts/vverreactorfuel/

The cluster reactivity compensation system is used to compensate for temperature and power reactivity effects, reactivity margins for core poisoning by xenon-135 and samarium-149, operating margins to change reactivity during reactor power variation, and to provide core sub-criticality at reactor shutdown. Clusters are bundles of 18 absorber rods on a common cross bar moving inside guide tubes of E-635 zirconium-alloy with the outer diameter of 12.6 mm and wall thickness of 0.6 mm. Each cluster is moved by its own drive.

All control rods fulfill both reactivity compensation and emergency protection functions (e.g. emergency shutdown). Emergency protection functions are performed when the control rods are passively inserted into the core by gravity from any position when the drives are de-energized upon the emergency alarm signals from the control system.

Steam generator

The SG tubing together with the casing is a modular coil-type vertical cylindrical surface heat exchanger (HX) where the primary coolant circulates through the tube side, and the secondary coolant through the shell side. The tubing heat exchange surface consists of 55 identical coiled steam generating modules; three of these modules are inspection-ready modules that can be removed from the SG without touching the rest of the tubing system.

Main circulation pump

The MCP consists of an axial flow pump and a canned electric motor as a single module. The pump flow channel includes a guide flange, an axial-type console impeller and a guide vane. The guide flange and the guide vane shape the flow at the impeller inlet and move the coolant from the impeller to the pressure chamber.

The cross sectional view of the steam generating module and main circulation pump unit is shown in Figure 7 below.

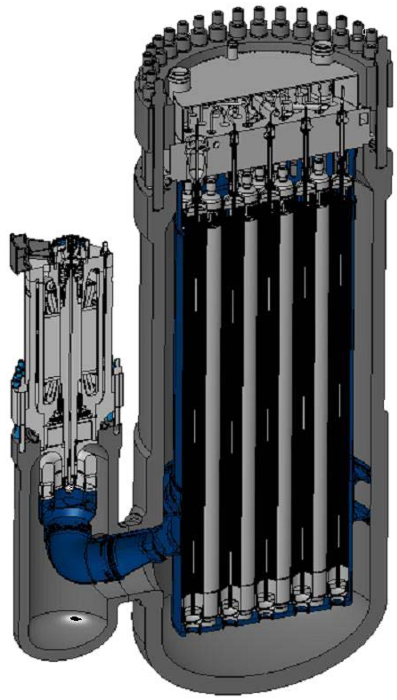

Figure 7 The cross sectional view of the steam generating unit and main circulation pump of VBER-300 reactor

Fuel cycle

The VBER-300 RP design concept allows adopting a flexible fuel cycle for the reactor core with standard VVER FAs. The interval between partial refuelling is one or two years. The number of FAs replaced in the refuelling batch is either 15 or 30 depending on the selected fuel cycle. In case of 30 fresh FAs are reloaded at 5% enrichment, the maximum fuel burnup does not exceed 60.0 MWd/kgU.

Maximum power peaking factors in both fuel cycles do not exceed the maximum values established for VVER-1000 reactors. Natural uranium consumption is within 198-240 g U/MWd. The two possible fuel cycles of VBER-300 are shown in Table 4.

Table 4 Main characteristics of the VBER-300 fuel cycles

Characteristics	Value	
Cycle length	1 years	2 years
Maximum uranium enrichment (%)	4.25	4.95
Number of FAs in the reloading batch	15	30
Refuelling fraction	1 / 5.67	1 / 2.83
U mass in FA (kg)	446.6	446.6
Average final burnup (MWd/kgU)	52.0	59.2

Fuel Handling System

Transportation of spent FAs from the reactor into the decay storage pool and then from the storage pool into the transportation container installed in the pit is performed by the fuel handling machine, which serves both the reactor and the decay storage pool.

The fuel reloading employs the dry method where the reloaded FA is in water inside the fuel handling machine protective tube. The fuel handling machine design includes a refuelling tube which accommodates one FA with the coolant, thus providing the biological shielding and heat removal from the FA during refuelling. Each spent FA is unloaded from the reactor by the fuel handling machine, transported into the decay storage pool and installed onto an assigned storage rack shelf. Then, the fuel handling machine takes one fresh FA from the fresh fuel storage rack, transports it into the reactor and installs into the assigned core cell.

The dry transportation of in-vessel equipment is also performed in a shielded transportation container that provides biological protection for the servicing personnel.

Fuel storage

Acceptance and storage of fresh fuel, as well as its preparation for loading into the core, are provided by the fresh fuel storage. Fresh fuel is delivered to the site in containers in a special railway carriage whereas each container includes four cassettes. The fresh fuel storage accommodates the number of FAs needed to reload the core with a 20% margin. The fresh fuel storage and the reactor compartment are connected with an internal railway with a flat car.

Pressurizer system

The VBER-300 uses an external steam pressurizer system. The system includes:

- One (1) steam pressurizer.
- 14 electric heaters consuming 90 kW of power each.

- Piping, control and safety valves, primary measurement transducers.

The steam pressurizer is a vessel filled with saturated water and steam with a phase interface at rated pressure of 16.2 MPa. The water space of the pressurizer where electric heaters are located is connected to the hot leg of the primary circuit. The steam space is connected to the cold leg of the primary circuit in the area of MCP pressure chamber; water is supplied to the pressurizer from this chamber when the valves in the injection line are open.

In order to prevent the formation and accumulation of explosive hydrogen compounds in the pressurizer steam space, the steam-gas mixture is continuously removed from under the pressurizer cover along the steam removal line into the bubbler. Primary coolant chemistry in the VBER-300 RP is actively controlled.

Containment

Double containments are used for a ground-based VBER-300 nuclear plant, consisting of an internal steel shell and an outer non-preloaded concrete shell. The steel shell is cylindrical, 28.0 m in diameter and 34.0 m high. The concrete shell is made of monolithic non-preloaded reinforced concrete with the external diameter of 34.0 m and height of 42.2 m.

4.2.2 Key Safety Characteristics and Technical Innovations in Design

The IAEA Status Report [20] describes that the VBER-300 incorporates various inherent and passive safety features.

Inherent safety features

The VBER-300 incorporated the following inherent safety features in the design:

- Negative fuel and coolant temperature reactivity coefficients, negative reactivity coefficient on coolant density, negative steam and integral power reactivity coefficients.
- Lower core power density compared to the marine propulsion reactors and VVER-1000 reactors.
- Connecting the majority of primary circuit pipes to “hot” sections of the circuit and arranging the nozzles on the reactor vessel above the core level, which ensures that steam outflow take place and decreases requirements for the emergency core cooling system (ECCS) flow rate.
- The reactor unit has short load-bearing nozzles between the main equipment units, without lengthy large-diameter primary pipes.
- Small-diameter flow restrictors in the nozzles of primary circuit auxiliary systems; these restrictors, in combination with the modular layout of the main equipment, reduce the probability of large and medium break loss of coolant accident.

- Canned motor reactor coolant pump which has no pump seals and thus loss of coolant associated with pump seal failure is prevented.
- Absence of penetrations in the reactor vessel bottom to enhance the fuel coverage with coolant.
- Once-through SGs that limit the increase of secondary circuit heat removal power (overcooling of the primary circuit coolant) in case of a steam line rupture.

Passive Safety Features

The passive safety systems in VBER-300 uses the natural forces such as natural convection and gravity, and stored energy such as compressed gas to activate the safety equipment without a supply of external power. The following passive safety features are incorporated in the design:

- Gravity driven emergency shutdown cooling system.
- The control rods are passively inserted into the core by gravity from any position when the drives are de-energized upon the actuation of emergency alarm signals.
- Natural convection flow in the primary heat transfer circuits providing decay heat removal from the shutdown reactor.
- Double containment:
    - Passive containment heat removal system limiting containment pressure in LOCAs
    - System of fuel retention in the reactor vessel in accidents with severe core damage
    - Iodine and aerosol air purification system for the space between the containment and the protective enclosure purifying air from radioactive leaks from the containment in accidents with containment overpressure

4.3 SMART

The World Nuclear Association[28] describes SMART reactor (shown in Figure 8) as follows:

South Korea's SMART (System-integrated Modular Advanced Reactor) is a 330 MWt pressurised water reactor with integral steam generators and advanced safety features. It is designed by the Korea Atomic Energy Research Institute (KAERI) for generating electricity (up to 100 MWe) and/or thermal applications such as seawater desalination. Design life is 60 years, fuel enrichment 4.8%, with a three-year refuelling cycle. Residual heat removal is passive. While the basic design is complete, the absence of any orders for an initial reference unit has stalled development. It received standard design approval from the Korean regulator in mid 2012 and KAERI plans to build a 90 MWe demonstration plant to operate from 2017. A single unit can produce 90 MWe plus 40,000 m^3/day of desalinated water.

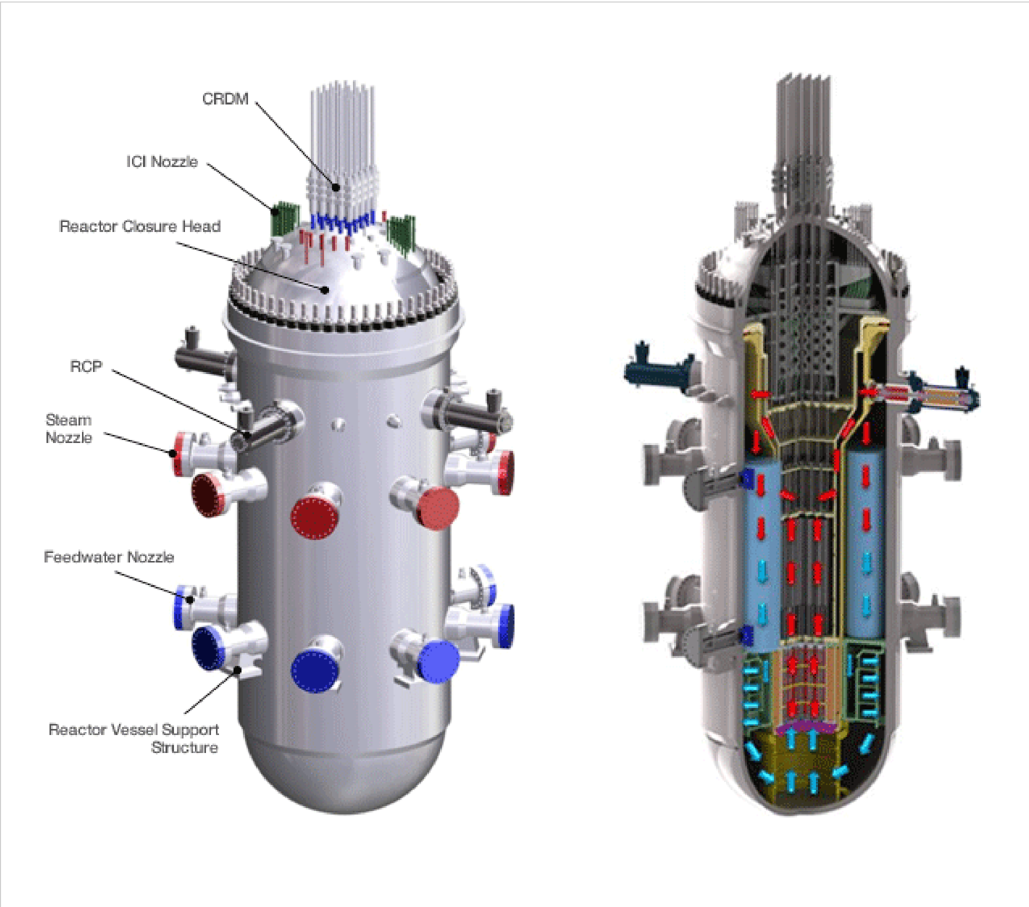

Figure 8 Cutaway view of the SMART Integral reactor

[28] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/

The summary of SMART technical data taken from the IAEA report [21] is provided in Table 5.

Table 5 Summary of SMART Reactor Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	330	MW(th)
Power output, gross	100	MW(e)
Power output, net	90	MW(e)
Plant efficiency	30.3	%
Mode of operation	Load follow	
Plant design life	60	Years
Plant availability	Over 95	%
Primary coolant	Light water	
Moderator	Light water	
Thermo dynamic cycle	Indirect Rankine cycle	
Reactor core		
Active core height	2.00	m
Equivalent core diameter	1.83	m
Average linear heat rate	10.97	kW/m
Average fuel power density	23.08	kW/kgU
Average core power density	62.62	MW/m^3
Fuel material	UO_2	
FE type	17 X 17	
Cladding material	Zircaloy-4	
Lattice geometry	Square	
Number of fuel assemblies	57	
Number of fuel elements in fuel assemblies	264	

Fuel enrichment	4.80	Weight %
Fuel cycle length	36	Months
Average discharge burnup	36.1	MWd/kgU
Burnable absorber	Gd_2O_3-UO_2	
Mode of reactivity control	Control rods and integrated burnable absorber	
Mode of reactor shutdown	Control rods, soluble boron	
Control rod absorber material	Ag-In-Cd	
Soluble neutron absorber	H_3BO_3	
	Primary coolant system	
Core coolant inlet temperature	295.7	ºC
Core coolant outlet temperature	323	ºC
Operating pressure	15	MPa
	Power conversion system	
Working medium	Steam	
Working medium temperature	200	ºC
	Reactor vessel	
Inner diameter of cylindrical shell	5332	mm
Wall thickness of cylindrical shell	331	mm
Total height	15500	mm
	Steam generator	
Type	Helically coiled, once-through, integrated cassette	
Number	8	
Total tube surface area	500	m^2
Number of HX tubes	375	
Tube outside diameter	17	mm

Tube material	Inconel 690	
Primary circulation system		
Circulation type	Forced	
Pump type	Integrated	
Number of pumps	4	
Pressurizer		
Type	Integrated, electrical	
Total volume	61	m^3
Containment		
Primary	Cylindrical, concrete	
Design leakage rate	0.2 volume % /day	

4.3.1 Description

SMART (System-integrated Modular Advanced ReacTor) is an integral type reactor with a rated thermal power of 330 MW. The description of SMART reactor in this section is summarized from the latest IAEA report [21] on the design.

Reactor Vessel Assembly

The reactor assembly of SMART contains its major primary systems such as fuel and core, eight (8) steam generators (SG), a pressurizer (PZR), four (4) reactor coolant pumps (RCP), and twenty five (25) control rod drive mechanisms (CRDM) in a single pressurized reactor vessel. The integrated arrangement of these components enables the removal of the large size pipe connections between major reactor coolant systems, and thus fundamentally eliminates the possibility of large break loss of coolant accidents (LBLOCA) in a pipe. The reactor coolant pumped by RCPs installed horizontally at the upper shell of the RPV flows upward through the core, and enters the shell side of the SG from the top of the SG. The secondary side feedwater enters the helically coiled tube side from the bottom of the SG and flows upward to remove the heat from the shell side eventually exiting the SG in a superheated steam condition. The large free volume in the top part of the RPV located above the reactor water level is used as a PZR region. As the steam volume of a PZR is designed to be sufficiently large, a spray is not required for a load maneuvering operation. The primary system pressure is maintained constant due to the large PZR steam volume and a heater control. The core exit temperature is programmed to maintain the primary system pressure constant during a load change. In this way, the reactor always operates at its own operating pressure range matched with the system condition. Eight (8)

SGs are located at the circumferential periphery with an equal spacing inside of the RPV and relatively high above the core to provide a driving force for a natural circulation of the coolant.

Core Design and Fuel Management

The SMART core is designed to produce a thermal energy of 330MW with 57 fuel assemblies of a 17x17 array. Core reactivity control during normal operation is achieved by control rods and soluble boron. The SMART core design is characterized by:

- Longer than conventional PWR cycle operation with a two-batch reload scheme
- Low core power density
- Thermal safety margin of more than 15 %
- Inherently free from xenon oscillation instability
- Minimum rod motion for load following with coolant temperature control

SMART fuel management is designed to achieve a long cycle length between refuelling. A simple two-batch refuelling scheme without reprocessing provides a cycle of 990 effective full power days (EFPD) for a 36 month operation.

Primary Circuit Components

SMART has eight (8) identical SG cassettes which are located in the annulus formed by the RPV and the core support barrel (CSB). Each SG cassette is of a once-through design with a number of helically coiled tubes. The primary reactor coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flows upward through the tubes. The secondary feedwater is evaporated in the tube and exits the SG cassette nozzle header as 30°C superheated steam at 5.2 MPa. In case of an abnormal reactor shutdown, the SG is used as the heat exchanger for the passive residual heat removal system (PRHRS). Each SG cassette contains about 375 tubes with orifices to prevent flow instability. In the case of a tube leak, each tube can be plugged, up to 10% of the total heat transfer area. The design temperature and design pressure of the SG cassette are 360°C and 15MPa, respectively.

Reactor Coolant Pump

The SMART RCP is a canned motor pump which does not require pump seals. This characteristic basically eliminates a SBLOCA associated with a pump seal failure which is one of the design basis events for conventional reactors.

SMART has four RCPs horizontally installed on the upper shell of the RPV. Each RCP is an integral unit consisting of a canned asynchronous three phase motor and an axial-flow, single-stage pump. The motor rotor and the impeller shaft are connected by a common shaft rotating on two radial bearings and one axial thrust bearing, which uses a specialized graphite-based material. The cooling of the motor is accomplished with component cooling water which flows through the tubes wound helically along the outer surface of the motor stator. The rotational speed of the pump rotor, 3600 rpm, is measured by a sensor installed in the upper part of the motor.

Secondary System

The secondary system receives superheated steam from the NSSS. It uses most of the steam for electricity generation and seawater desalination. The main steam pressure is kept constant during a power operation. A load change is accommodated by changing the feedwater flow rate. A seawater desalination system can be connected to the secondary system.

Fuel Handling and Storage

Since the SMART fuel is almost identical to the 17x17 standard PWR fuel [22] except for its shorter height, the fuel-handling system of SMART is similar to that of the conventional PWRs. Fuel-handling equipment is used for safe handling of fuel assemblies and control rod assemblies (CRAs) under all specified conditions and for the assembly, disassembly, and storage of the reactor vessel head and internals during refuelling.

The major components include a refuelling machine, a fuel transfer system, a spent fuel-handling machine, a new fuel elevator, CRA change fixture, and fuel storage facility. This equipment transfer new and spent fuel between the fuel storage facility, the containment building, and the fuel shipping and receiving areas during core loading and refuelling operations. Fuel is inserted and removed from the core using the refuelling machine. During normal operations, irradiated fuel and CRAs are always maintained in a water environment.

The storage of used fuel for the reactor life time is provided by the large spent fuel storage pool located in the secured fuel building.

4.3.2 Key Safety Characteristics and Technical Innovations in Design

Inherent Safety Characteristics

The IAEA status report [21] explains that the following inherent safety features are incorporated into the SMART design:

- Integration of primary components in a vessel: SMART contains major primary components such as a core, eight (8) steam generators, a pressurizer, four (4) reactor coolant pumps, and twenty-five (25) control rod drive mechanisms (CRDMs) in a single reactor pressure vessel (RPV). The integral arrangement of the primary system removes large size pipe connections between major components and thus, fundamentally eliminates the possibility of LBLOCA due to a pipe break.
- Smaller source term: the reactor size is smaller than a conventional reactor and the core power density is lower, which make the decay heat removal more manageable in case of reactor shutdown.

- Burnable poison rods create flat radial and axial power profiles, which result in increased thermal margin for the core.
- A nearly constant reactor coolant average temperature, controlled by the reactor regulating system, gives a stable pressure and water level within the pressurizer and so improves load following performance.
- The canned motor RCP eliminates the need for an RCP seal, and basically eliminates a potential for a SBLOCA associated with the seal failure.
- The modular type once-through steam generators are located relatively high above the core to provide a driving force for natural circulation flow. This design feature along with low flow resistance enables the system to have residual heat removal with natural circulation, when the normal means to transfer residual heat from the core are not available.
- A large volume semi-passive PZR can accommodate a wide range of pressure transients during system transients and accidents.
- Low core power density lowers the fuel element temperature rise under accident conditions and increases the thermal margin.
- A negative moderator temperature coefficient (MTC) assists core self-stabilization, and limits the reactor power during accidents.
- The low power density design with about a 5 wt% UO_2 fuelled core is proven to provide a thermal margin of more than 15 % to critical heat flux, to accommodate any design basis transients. This feature ensures the core thermal reliability under normal operation and any design basis events.

Passive Safety Systems

The IAEA report [21] also indicates that SMART reactor is also equipped with the following passive safety design features that use either stored energy or natural forces to operate safety related devices without active operator intervention:

- Gravity driven Reactor Shutdown System (RSS): The shutdown signal de-energizes the control rod drive mechanism and then the control rods drop into the reactor core by the force of gravity.
- Automatic Safety Injection System (SIS): The SIS is provided to prevent core damage during a SBLOCA. The core is protected during a SBLOCA and covered by a large primary coolant inventory. When the pressure drops below 10 MPa in the PZR, the SIS is actuated automatically and cold water is injected immediately into the reactor coolant system. The SIS consists of four independent trains with a 100% capacity for each train.

The system refills the reactor vessel so that the decay heat removal system can function properly in a long-term recovery mode following an accident.

- Passive Residual Heat Removal System (PRHRS): The PRHRS passively removes the core decay heat and sensible heat by natural circulation in case of an emergency such as a station blackout. The PRHRS may also be used for long-term core cooling during a repair or refuelling. The PRHRS consists of four independent trains, each with a 50% capacity to remove the decay heat from the whole core. Each train is composed of an emergency cool-down tank, a heat exchanger and a makeup tank. The system is designed to maintain the core un-damaged for 36 hours without any corrective actions by operators for the postulated design basis accidents. In the case of a normal shutdown, the residual heat is removed through the steam generators to the condenser with a turbine bypass system.
- Reactor Overpressure Protection System (ROPS): The system consists of two (2) pressurizer safety valves (PSVs), which are installed on top of the reactor head assembly. The steam discharge lines of the PSVs are combined to a single pipe and connected to the containment atmosphere through the reactor drain tank (RDT). When the primary system pressure increases over the predefined set-point, PSVs are opened to discharge the steam into the RDT.

4.4 CAREM

The World Nuclear Association provides the introduction of the CAREM reactor[29], shown in Figure 9, as follows:

The CAREM Reactor is being built by the Argentine National Atomic Energy Commission (CNEA). It is designed to be used for electricity generation or as a research reactor or for water desalination (with 8 MWe in cogeneration configuration). The 25 MWe prototype unit is being built next to Atucha, on the Parana River in Lima, 110 km northwest of Buenos Aires, and the first larger version (probably 150 MWe) is planned in the northern Formosa province, 500 km north of Buenos Aries, once the design is proven. Some 70% of CAREM-25 components will be local manufacture. The IAEA lists it as a research reactor under construction since April 2013, though first concrete was poured in February 2014, marking official start of construction.

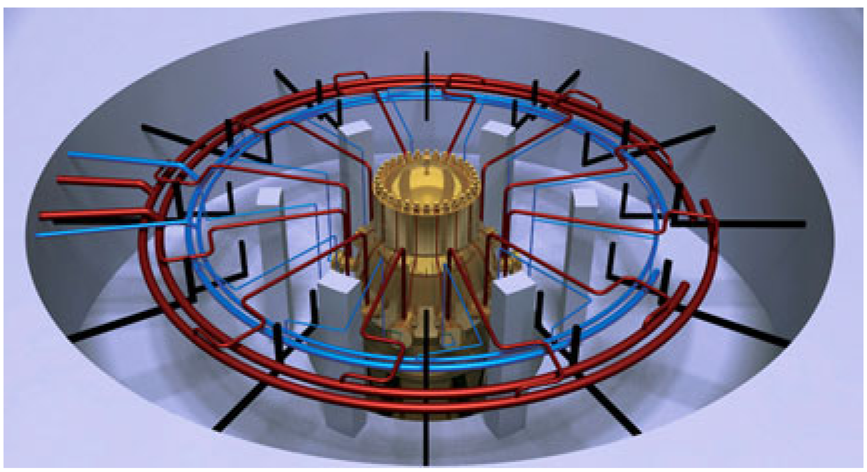

Figure 9 CAREM-27 reactor vessel with the steam generator pipes[30]

The technical details of the CAREM reactor are summarized in Table 1 as taken from [23] and [24].

Table 6 Summary of CAREM Reactor Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	100	MW(th)
Power output, net	27	MW(e)
Plant efficiency	25	%

[29] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/
[30] Comision Nacional de Energia Atomica – Proyectos, http://www.cnea.gov.ar/proyectos/carem.php

Mode of operation	Load follow	
Plant design life	60	Years
Primary coolant	Light water	
Moderator	Light water	
Thermo dynamic cycle	Indirect steam Rankine	
Reactor core		
Active core height	1.4	m
Average linear heat rate	~10.8	kW/m
Average fuel power density	N/A	kW/kgU
Average core power density	N/A	MW/m^3
Fuel material	UO_2	
FA type	Hexagonal	
Cladding material	N/A	
Lattice geometry	Triangular	
Number of fuel assemblies	61	
Number of fuel elements in fuel assemblies	108	
Fuel enrichment	1.8 to 3.1[31]	Weight %
Fuel cycle length	14	Months
Average discharge burnup	N/A	MWd/kg
Burnable absorber	Gd_2O_3	
Mode of reactivity control	Control rod	
Mode of reactor shutdown	Control rod	
Control rod absorber material	Ag-In-Cd	
Soluble neutron absorber	None	
Primary coolant system		

[31] http://www.cnea.gov.ar/proyectos/carem/caracteristicas_tecnicas/componentes.php

Core coolant inlet temperature	N/A	ºC
Core coolant outlet temperature	326	ºC
Operating pressure	12.25	MPa
Power conversion system		
Working medium	Steam	
Working medium temperature	291	ºC
Reactor vessel		
Diameter	3200	mm
Height	11000	mm
Steam generator		
Type	Helically coiled, once-through	
Number	12	
Total tube surface area	N/A	m^2
Number of HX tubes	N/A	
Tube outside diameter	N/A	mm
Tube material	N/A	
Primary circulation system		
Circulation type	Natural convention	
Pump type	None	
Number of pumps	0	
Pressurizer		
Type	Self-pressurized	
Total volume	N/A	m^3
Containment		
Primary	Cylindrical, concrete	

4.4.1 Description

The description of the reactor in this section is a summary based on design descriptions reported by CNEA [23], [24], [25]. CAREM is a jointly developed project between Comision National de Energia Atomica (CNEA) and INVAP, and it is currently under construction in Argentina. CAREM is a project for an advanced, simple and small nuclear power plant which relies on passive safety features. The CAREM is an indirect cycle, a 27 MWe power reactor with some distinctive features that greatly simplify the reactor and also contribute to a higher level of safety:

- Integrated primary cooling system.
- Primary cooled by natural circulation.
- Self-pressurized.
- Safety systems relying on passive features.

The CAREM NPP is a light water cooled and moderated integrated reactor in which the whole high-energy primary system including core, steam generators, primary coolant and steam dome, are contained inside a single pressure vessel. The emphasis has been put on minimizing the dependence on active components and operators' actions. The flow rate in the reactor primary systems is achieved by natural circulation. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form pressure-losses, producing a flow rate in the core that provides a high thermal safety margin. The driving force for the coolant's natural convection is produced by the location of the steam generators above the core. Self-pressurization of the primary system in the steam dome is the result of the liquid-vapour equilibrium. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Due to self-pressurization, the core outlet bulk temperature corresponds to saturation temperature at primary pressure. Heaters and sprays typical of conventional PWR's are thus eliminated.

Reactor core and fuel assembly

The core has 61 fuel assemblies (FA) of hexagonal cross section. The fuel is UO_2 enriched at 1.8% and 3.1% (note: fuel enrichment is not finalized and several different values have been reported). An 8% weight of Gd_2O_3 is used as burnable poison to flatten the power distribution along the fuel cycle. The fuel cycle can be tailored to customer requirements, with a reference design of 330 full-power days and 50% of core replacement.

Each absorbing element (AE) consists of a cluster of rods linked by a structural element (e.g. spider), so the whole cluster moves as a single unit. Absorber rods made of Ag-In-Cd alloy fit into the guide tubes, at 18 positions in the FA not occupied by fuel rods.

Absorbing elements (AE) are used for reactivity control during normal operation (Adjust and Control System), and to produce a sudden interruption of the nuclear chain reaction when required (Fast Extinction System). Chemical shim is not used for reactivity control during normal operation.

Steam generator

Twelve identical 'Mini-helical' vertical steam generators, of the "once-through" type are placed equally distant from each other along the inner surface of the Reactor Pressure Vessel (RPV) area. They are used to transfer heat from the primary to the secondary circuit, producing 30 ºC super-heated dry steam at 4.7 MPa.

The location of the steam generators above the core produces natural circulation in the primary circuit. The secondary system circulates upwards within the tubes, while the primary does so in countercurrent flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system which guarantees that the entire stream of the primary system flows through the steam generators. In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized by changing the number of tubes per coil layer. For safety reasons, steam generators are designed to withstand the pressure from the primary up to the steam outlet /water inlet valves even without pressure in the secondary.

The natural circulation of the coolant produces different values of flow rate in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained.

Pressurizer

Due to the self-pressurizing of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all operating conditions, sufficient margin exists to maintain the stable RPV pressure. The control system is designed to keep the reactor pressure at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit combined with the self-pressurization features make this behaviour possible with minimum control rod motion.

Residual heat removal system

The passive residual heat removal system (PRHRS) has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple system that operates condensing steam from the primary system in emergency condensers. The diagram of primary system and isolation condenser is shown in Figure 10.

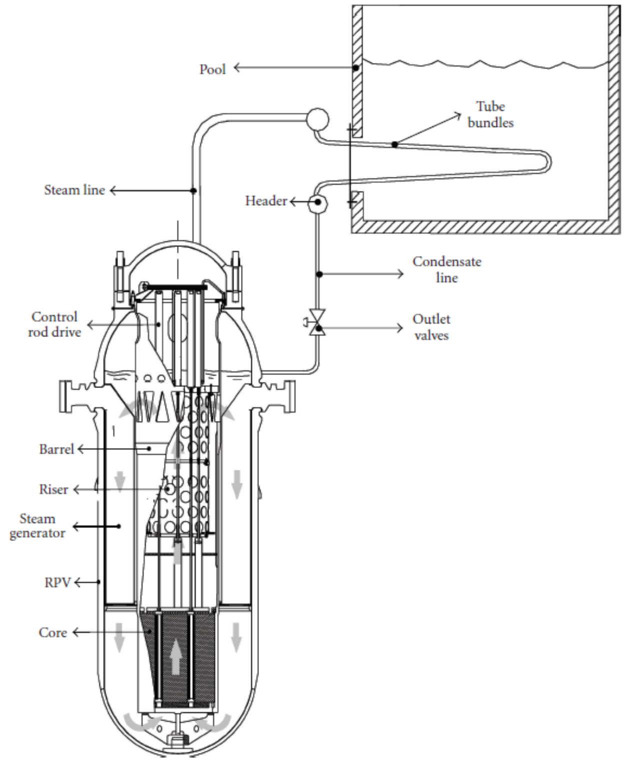

Figure 10 CAREM primary system and isolation condenser

The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected via piping to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside of the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water in the pool which boils to remove heat. This evaporated water is then condensed in the pressure suppression pool of the containment.

Emergency Injection System

The Emergency Injection System prevents core exposure in case of LOCA. In the event of such accident, the primary system is depressurized with the help of the emergency condensers to less than 1.5 MPa, with the water level over the top of the core. At 1.5 MPa, a low pressure water injection system comes into operation. The system consists of two tanks with borated water connected to the RPV. The tanks are pressurized to 2.1 MPa, thus when during a LOCA the pressure in the reactor vessel reaches 1.5 MPa, the rupture disks break and the flooding of the RPV starts.

Containment

The CAREM nuclear island is placed inside a containment system, which includes a pressure suppression feature to contain the energy and prevent a significant fission product release in the event of a postulated design basis accident.

The primary system, the reactor coolant pressure boundary, and important auxiliary systems are enclosed in the primary containment, a cylindrical concrete structure with an embedded steel liner. The primary containment is of pressure-suppression type with two major compartments: a drywell and wetwell. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition floor and cylindrical wall separate the drywell from the wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber.

The building surrounding the containment has several structural layers and it is placed on a single reinforced concrete foundation mat. It supports all the structures with the same seismic classification, allowing the integration of the RPV, the safety and reactor auxiliary systems, the spent fuels pool and other related systems in one block.

4.4.2 Key Safety Characteristics and Technical Innovations in Design

Inherent safety features

The safety functions in CAREM rely on the following inherently safe features that are commonly found in other IPWR-type SMRs:

- Integrated primary coolant system, eliminating large break LOCA.
- Long characteristic times in the event of a transient or severe accident, due to large coolant inventory and the use of passive systems.
- Small source term: the reactor size is much smaller than a conventional reactor and the core power density is lower, which make the decay heat removal much more manageable in case of reactor shutdown.
- Low core power density provide additional thermal safety margin.
- Internal CRDMs reduce the number and size of reactor vessel head penetrations to reduce small break LOCA probability.
- Elimination of liquid boron reactivity control to reduce the possibility of boron dilution accident.

- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The large water volume between the core and the wall leads to a very low fast neutron dose over the RPV wall.
- Negative reactivity effects and coefficients.
- Hydraulic control rod drive mechanisms located completely inside the RPV eliminates control rod ejection accidents.
- Natural convection core cooling eliminates loss of flow accidents.
- Self-pressurization of the primary system in the steam dome contributes to damping of eventual pressure perturbation and elimination of heaters and sprays which reduces equipment failure DBA.

Passive safety features

CAREM reactor heavily relies on the passive safety features that are described below.

- First Shutdown System (FSS): hydraulic and gravity driven safety shutdown system is designed to shut down the core by dropping a total of 25 neutron-absorbing elements into the core by the force of gravity when the water flow in the vessel, which suspends the elements above the core, is interrupted.
- Second shutdown system (liquid poison injection system): an automatic gravity-driven injection system of borated water at high pressure.
- Hydraulically moved absorbing elements (AEs) avoid the use of mechanical shafts passing through, or the extension of the primary pressure boundary, and thus eliminates possibilities of small Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV.
- Reduction of hydrogen concentration in the containment by catalytic recombiners.
- The self-pressurization of the RPV eliminates the sprays.
- Passive residual heat removal system
- Safety relief valves: Triple redundant (3) safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the core power and the power removed by the systems. The blow-down pipes from the safety valves are routed to the pressure suppression pool.
- Containment with pressure suppression features.

4.5 StarCore

StarCore[32] is a high temperature gas cooled reactor (HTGR) being proposed by StarCore Nuclear in Canada. The 20 MWe plant is proposed to provide electricity and heat to frontier mines and villages through unmanned plant operation. The architect's rendering of the twin unit plant is shown in Figure 11.

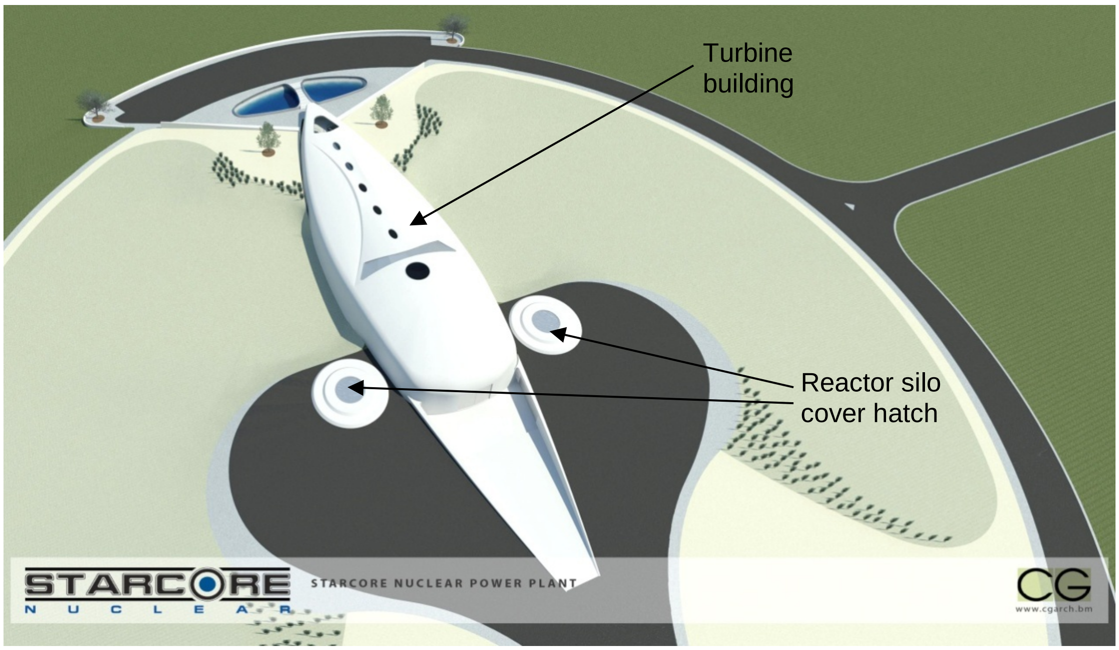

Figure 11 StarCore twin reactor unit nuclear power plant [26]

The StarCore technical data is obtained directly from the vendor and it is summarized in Table 7 below.

Table 7 Summary of StarCore Reactor Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	2 X 25	MW(th)
Power output, net	2 X 10	MW(e)
Rejected heat available for heat supply	2 X 15	MW(th)

[32] StarCore Nuclear, http://starcorenuclear.ca/

Plant electrical efficiency	40	%
Mode of operation	Load follow	
Plant design life	60	Years
Plant availability	Over 95	%
Primary coolant	Helium	
Moderator	Carbon graphite	
Thermo dynamic cycle	Indirect Rankine cycle	
	Reactor core	
Active core height	2.5	m
Core diameter	1.5	m
Average linear heat rate	N/A	kW/m
Average fuel power density	N/A	kW/kgU
Average core power density	N/A	MW/m^3
Fuel material	UO_2	
Fuel element type	TRISO particles in graphite	
Cladding material	Silicon carbide	
Fuel element geometry	Truncated cuboctahedron	
Fuel element diameter	60	mm
Number of fuel elements	26800	
Number of TRISO particles in a fuel element	~2000	
Fuel enrichment	19	Weight %
Fuel cycle length	5 – 10	Years
Average discharge burnup	N/A	MWd/kgU
Burnable absorber	None	
Mode of reactivity control	Control rods (quick), stowable reflectors (long term)	

Mode of reactor shutdown	Control rods	
Control rod absorber material	N/A	
Soluble neutron absorber	none	
Primary coolant system		
Primary coolant medium	Helium	
Coolant outlet temperature	850	ºC
Operating pressure	7.5 MPa	MPa
Intermediary heat transfer fluid		
Working medium	Nitrogen	
Working medium pressure	6.8	MPa
Power conversion fluid		
Working medium	steam	
Reactor vessel		
Inner diameter of cylindrical shell	2500	mm
Wall thickness of cylindrical shell	N/A	mm
Total height	6500	mm
Heat exchanger		
Type	Once-through	
Number	3	
Total HX surface area	N/A	
HX dimension	N/A	
Tube material	N/A	
Primary circulation system		
Circulation type	Forced	
Pump type	Integrated	
Number of pumps	N/A	

Pressurizer	
Type	No pressurizer
Containments	
Primary	Steel lined concrete silo

4.5.1 Description

Due to lack of public information on the reactor design, the design description was directly obtained from the vendor [26] and the vendor's website[33].

StarCore is a HTGR being proposed by StarCore Nuclear in Canada. The StarCore nuclear power plant consist of two (2) co-generation 25 MWth gas cooled reactors that is based on the pebble bed modular reactor technology. In terms of size and basic technical characteristics, the reactor is similar to HTR-10[34], a prototype pebble bed modular reactor built and operated in China, and HTTR[35], a 30 MWth experimental gas cooled reactor in Japan.

The company plans to deploy the NPP in remote villages and mines for power generation and heat supply. The reactors are to be constructed on an assembly line and transported and final assembled on site for rapid deployment. The reactor operation is performed remotely via triple-redundant satellite network; local operators will not be stationed at the plant.

Plant configuration

The plant is designed for unmanned operation. The reactors are placed at the bottom of 55 meters deep double-walled high-strength concrete silos. The nuclear heat from the core is carried away by helium coolant to the first heat exchanger that is located at the top of the silos. The intermediary loop heat transport fluid is nitrogen gas at slightly lower pressure than helium in the first loop and it carries the heat to the second heat exchanger. Finally, the second heat exchanger superheats steam and it is sent to a gas turbine generator. The side view of the reactor silo, heat exchanger and gas turbine generator layout is shown in Figure 12.

The turbine room is 10 meters high internally and the building in 100 meters long and 30 meters in width. There is an access to the two silos by means of personnel access tunnels from the turbine room, and there is an interface room at the top of the silo for reactor replacement operations.

[33] StarCore Nuclear, http://starcorenuclear.ca/wp-content/uploads/2014/01/StarCore-Design-Website.pdf
[34] Institute of Nuclear and New Energy Technology, http://www.inet.tsinghua.edu.cn/publish/ineten/5696/index.html
[35] Japan Atomic Energy Agency, http://httr.jaea.go.jp/eng/index.html

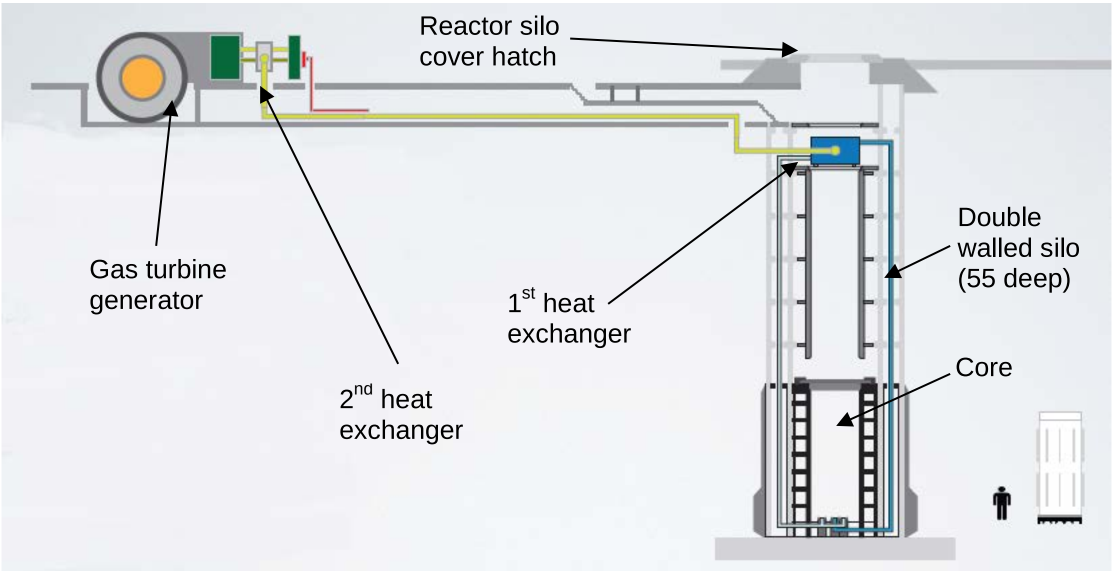

Figure 12 Side view of the power plant (not to scale)

Reactor core and fuel elements

The reactor uses TRISO (TriStructural Isotropic) [27] fuel in the form of micro-spheres of less than 1 mm diameter, which are then formed into fuel compacts (FCs). There are approximately 2,000 microspheres in each fuel compact, and around 26,800 FCs in the core. The reactor uses a Rectilinear Core that uses a fuel compacts in the form of Truncated Cuboctahedrons, which look like a cube with the corners cut off, as shown in Figure 13. This design has the advantage of allowing the fuel packing density and gas channels to be optimized, as well as providing a stable self-supporting matrix in the core.

Figure 13 Graphite fuel elements in Truncated Cuboctahedron shape

The core is nominally 1.5 meters in diameter and 2.5 meters high, with 18 control rods and 12 stowable reflectors with neutron-absorbent cores. These are pneumatically operated (at 6 MPa) by the gas pressure of the first stage Energy Transfer System (ETS-1), as are the automatic gas shut-off valves at the base of the Reactor Pressure Vessel (RPV), which is 2.5 meters in diameter and 6.5 meters high. They can also be closed pneumatically by command from the control center. The core also has reflectors and burnable poisons embedded in it to maintain the reactivity within design limits for the five year life span. The layout of the core is shown in Figure 14 and Figure 15.

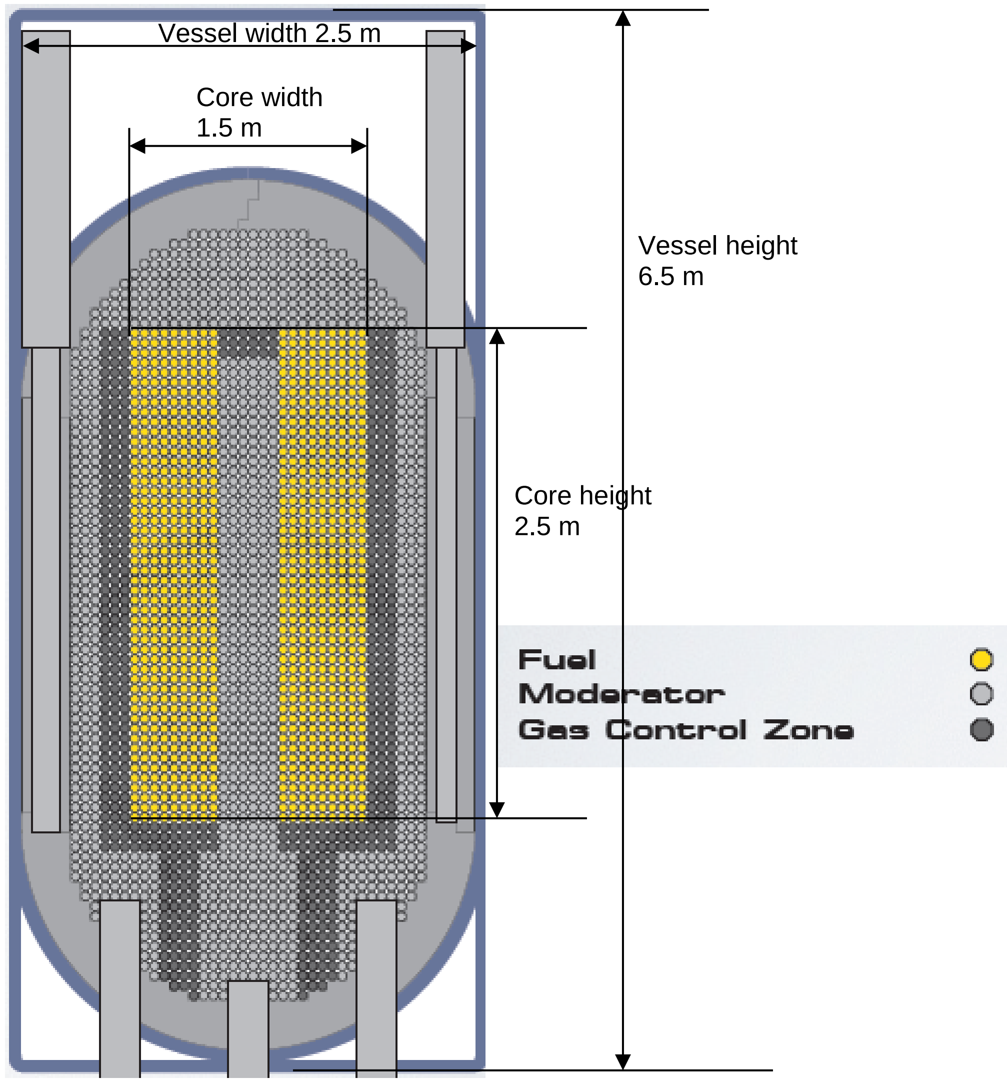

Figure 14 StarCore reactor core filled with fuel and moderator blocks where yellow blocks represent fuel elements

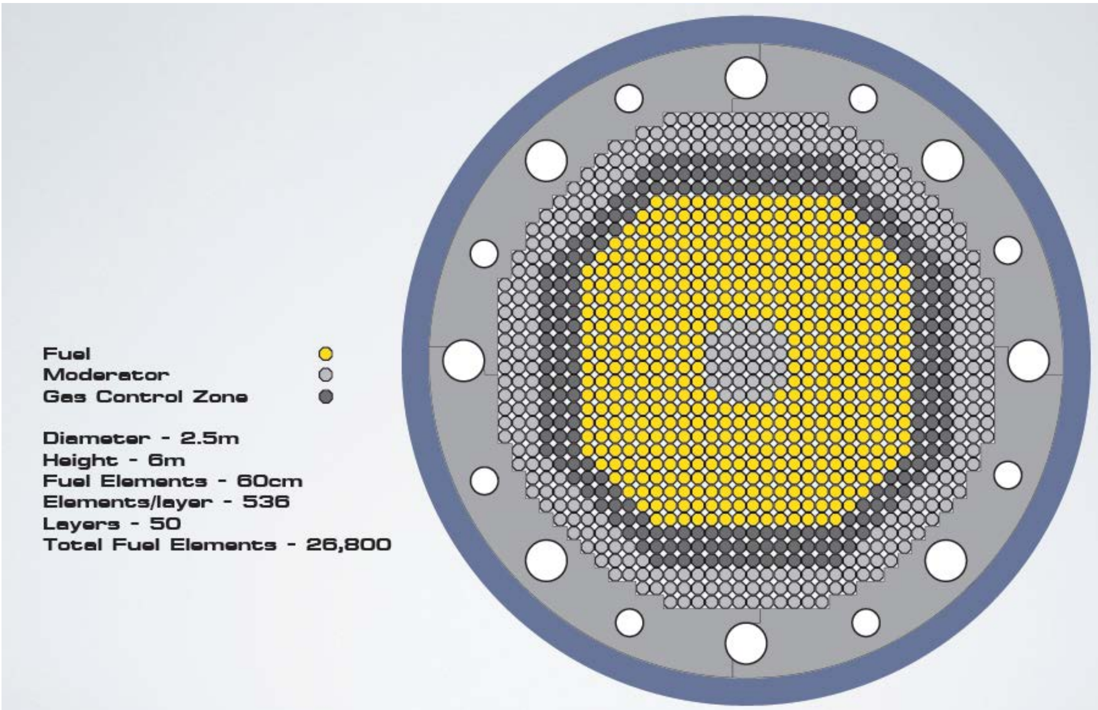

Figure 15 Cross sectional view of the StarCore reactor

Heat exchangers

The plant uses a three-stage heat transfer process from helium to nitrogen, and then from nitrogen to steam which is sent to a steam turbine to generate electricity. This three stage process allows StarCore to prevent helium migration at the first stage heat exchanger by balancing pressure across this interface; in addition a passive fractional scrubber and condenser in the first stage system remove any radioactive trace elements in the form of dust or heavier gaseous isotopes that may occur. The first stage ETS-1 uses helium gas at 7.5 MPa; this ends at the first Intermediate Heat Exchanger (IHX-1) where the second stage (ETS-2) takes over the transfer process. This system uses Nitrogen pressurized at 6.8 MPa to minimize pressure gradient across IHX-1, and then takes the hot gas to the co-generation heat exchanger (IHX-2) which produces super-heated steam, and then to the third heat exchanger (IHX-3) in the steam turbine.

Fuel

The StarCore fuel is TRISO fuel with 19% enriched UO_2. The TRISO particle forms the first layer of containment.

Fuel cycle and handling

The StarCore reactor is expected to operate 5 years without on-site refueling. At the end of a fuel cycle, the entire reactor core vessel will be replaced with a new vessel containing fresh fuel. The removed core will be transported back to the centralized facility with all reactivity control devices placed in the vessel to maintain the reactor sub-critical during the transport. The spent fuel will not be stored at the plant site, and there is no fuel handling at the site.

Autonomous operation

The StarCore power plant is an autonomous operation with a load-following and a shutdown mode. The plant condition is remotely monitored and controlled from the central control facility via triple redundant satellite communication. The large negative temperature coefficient of the fuel allows flexible load-following operation without large control rod movements (i.e. as more thermal energy is drawn from the coolant, the reactor temperature drops and produces more power, and vice versa).

4.5.2 Key Safety Characteristics and Technical Innovations in Design

Inherent Safety Features

The StarCore reactor design and operation are strongly based on the inherent safety features.

- Strong negative temperature reactivity coefficient of TRISO particles in graphite blocks.
- Achievement of increased reactor self-control in anticipated transients without scram, without exceeding safe operation limits for fuel. In a recent HTR-10 reactor test [28], the control rods were withdrawn and the all coolant systems shut down while the reactor was at 30% rated power. The reactor reached effectively zero power output in about 6 minutes, and then thermal stasis after about two hours. At this time the power level settled at about 200 kW, which was mainly driven by the energy transfer rate out of the RPV. StarCore initial estimates show that the power levels will stabilize at around 500 kW in the StarCore reactor under similar conditions. This low power allows the reactor core to be cooled by natural convection air flow almost indefinitely without human intervention or external power supply.
- De-rating of accident scenarios rated as potentially severe in reactors of other types, including LOCA, loss of flow, and reactivity initiated accidents; for example, helium release from the core in the HTGR can be a safety action and not the initiating event for a potentially severe accident.
- Achievement of a large temperature margin between the operation limit and the safety limit, owing to inherent fission product confinement properties of TRISO fuel at high temperatures and fuel burnups [29].
- Truncated Cuboctahedron fuel elements: The stackable fuel elements allow much improved core modeling and the reduced chance of coolant gas flow path blockage ensures that the fuel elements do not experience a localized heating outside of the calculation. The temperature limit for TRISO particles to safely retain fission products in the matrix is 1,600 ºC whereas the core average temperature is expected to be below 1,000 ºC.

- Gas coolant: The reactor coolant and all heat transfer medium are in a gaseous state, and there is no phase changes in case of a coolant pipe break or rupture. Since the coolant does not go through significant volume expansion, the steel and concrete containment structures can maintain the design pressure margin. The primary coolant and the intermediary heat transport medium are both chemically inert, helium and nitrogen, respectively. The helium coolant can be scrubbed to be free of radiation just in case of coolant leakage.
- Relatively high heat capacity of the reactor core and reactor internals and low core power density, resulting in slow progression of the transients.
- Underground placement: At the site, the nuclear reactors are installed at the depth of 57 meters below grade in a double-walled stainless steel containment structure with passive thermal management of heat output, and with double-walled high performance concrete silos extending to the surface.

Passive Safety Features

The design of StarCore reactor is still in conceptual stage and there was no information on passive safety features.

4.6 NuScale

The World Nuclear Association[36] provides the following description of NuScale which is shown in Figure 16:

NuScale is a multi-application small reactor, a 165 MWt, 45 MWe integral PWR with natural circulation. The NuScale Power Company was spun out of Oregon State University in 2007, though the original development was funded by the US Department of Energy. After NuScale experienced problems in funding its development, Fluor Corporation paid over $30 million for 55% of NuScale in October 2011. In August 2013 Rolls Royce joined the venture to support an application for DOE funding. In December 2013 the US Department of Energy (DOE) announced that it would support accelerated development of the design for early deployment, with up to $226 million on a 50-50 cost share basis.

NuScale expects to lodge an application for US design certification in 2015, and is already engaged with NRC, having spent some $130 million on licensing to November 2013. It expects the NRC review to take 39 months, so the first unit could be under construction late in 2019.

In March 2012 the US DOE signed an agreement with NuScale regarding constructing a demonstration unit at its Savannah River site in South Carolina. In mid 2013 NuScale joined Energy Northwest and Utah Associated Municipal Power Systems for a demonstration project deploying the reactor by 2014, potentially in Idaho. It would likely be owned by a consortium of utilities and operated by one of them.

[36] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/

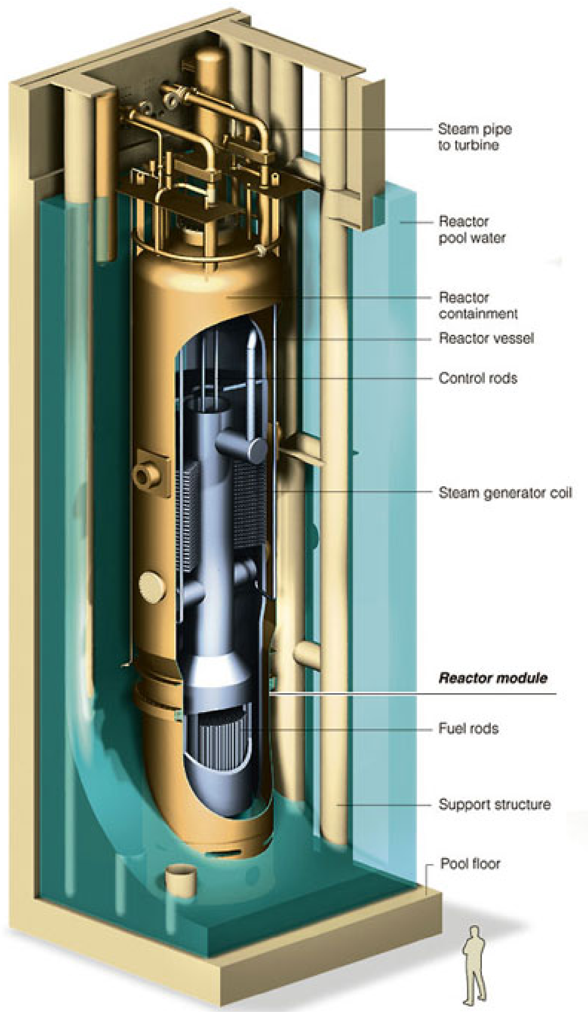

Figure 16 Cross-cut view of the NuScale reactor plant unit

The technical parameters of the reactor are obtained from the IAEA report [30] and they are reproduced in Table 8.

Table 8 Summary of the NuScale Reactor Technical Data

Parameter	Values	Units
General plant data (single module)		
Thermal output power	160	MW(th)
Power output, net	45	MW(e)
Plant efficiency	> 30	%

Mode of operation	Base load	
Plant design life	60	Years
Plant availability	92 ~ 95	%
Primary coolant	Light water	
Moderator	Light water	
Thermo dynamic cycle	Indirect Rankine steam cycle	
Reactor core		
Active core height	2.0	m
Average linear heat rate	N/A	kW/m
Average fuel power density	N/A	kW/kgU
Average core power density	N/A	MW/m^3
Fuel material	UO_2	
FE type	Smooth-rod, Cylindrical	
Cladding material	Zircaloy-4	
Lattice geometry	Square array	
Number of fuel assemblies	37	
Fuel enrichment	4.95	Weight %
Fuel cycle length	24	Months
Average discharge burnup	>30	MWd/kg
Burnable absorber	Gd_2O_3	
Mode of reactivity control	Control rods	
Mode of reactor shutdown	Control rods	
Control rod absorber material	N/A	
Soluble neutron absorber	H_3BO_3	
Primary coolant system		
Core coolant inlet temperature	218.4	ºC

Core coolant outlet temperature	327.5	ºC
Operating pressure	8.72	MPa
Reactor vessel		
Inner diameter of cylindrical shell	~2830	mm
Total height	~20,000	mm
Steam generator		
Type	Helically coiled, vertical	
Number	2	
Feedwater temperature	149	ºC
Primary circulation system		
Circulation type	Natural convection	
Number of pumps	0	
Pressurizer		
Type	Internal electric heaters and sprays	
Containments		
Primary	Cylindrical, steel, vacuum	
Secondary	Concrete building	

4.6.1 Description

The following technical description of the NuScale Reactor is a summary of the IAEA report [30]. In 2003, Oregon State University, in collaboration with the Idaho National Engineering Laboratory, and Nexant-Bechtel, completed a project to develop a preliminary design for an innovative reactor called the “Multi-Application Small Light Water Reactor”, or “MASLWR.” In 2007, NuScale Power Inc. was formed to commercialize the concept, and MASLWR was renamed to the NuScale Plant to reflect the significant improvements made to the original design. In early 2008, NuScale Power notified the U.S. Nuclear Regulatory Commission of its intent to begin Pre-Application discussions aimed at submitting an application for Design Certification of a twelve-module NuScale Power Plant. Fluor Corporation became the majority investor of NuScale Power in 2011 and will provide engineering, procurement and construction services for plant deployments.

A NuScale plant consists of 1 to 12 independent modules, each capable of producing a net electric power of 45 MWe. Each module includes an integral Pressurized Light Water Reactor operated under natural circulation primary coolant conditions. Each reactor is housed within its own high pressure containment vessel which is submerged underwater in a stainless steel lined concrete pool. The details of the system parts are discussed further below.

Description of the Nuclear Systems

The NSSS can be entirely prefabricated off site and shipped by rail, truck or barge. The NuScale design also utilizes a compact movable modular containment, in contrast to a traditional cast in-place concrete design. Similar to the reactor vessel, the use of a compact containment that is prefabricated off site and shipped by rail, truck or barge, enhances the economics and deployment flexibility

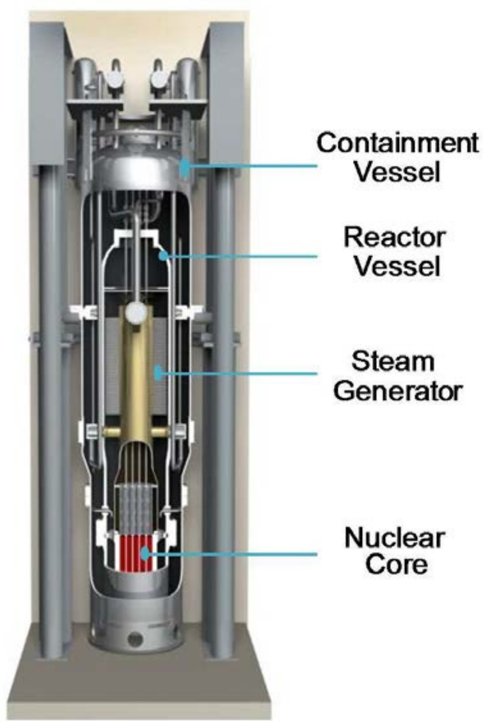

Figure 17 The basic configuration of a single NuScale reactor module

The basic configuration of a single NuScale reactor module is shown schematically in Figure 17. The integrated nuclear reactor pressure vessel contains the nuclear core, a helical coil steam generator, and a pressurizer. It is approximately 20.0 m (65 ft) long by 2.8 m (9.3 ft) in diameter. The nuclear core consists of an array of reduced height LWR fuel assemblies and control rod clusters at 4.95 wt% enrichments. The helical coil steam generator consists of two independent sets of tube bundles with separate feedwater inlet and steam outlet lines.

Feedwater is pumped into the tubes where it boils to generate superheated steam. A set of pressurizer heaters is located in the upper head of the vessel to provide pressure control. The entire Nuclear Steam Supply System (NSSS) is enclosed in a steel containment that is 24.6 m (80 ft) long by 4.6 m (15 ft) in diameter.

The nuclear core is cooled entirely by natural circulation. The coolant is heated in the core and the hotter and lower density water rises upwards through the riser placed above the core. The helical coils wrapped around the outside of the riser provide a heat sink that cools the water, causing its density to increase. The density difference acting over an elevation difference results in a buoyancy force that drives the fluid flow around the loop. Natural circulation operation provides a significant advantage in that it eliminates pumps, pipes, and valves and hence the maintenance and potential failures associated with those components. It also reduces in-house plant loads.

Plant layout

A NuScale plant consists of up to 12 NSSS modules and corresponding steam turbines. A cutaway view of a single module is shown in Figure 18. For a 540 MWe plant with 12 modules, 6 of these modules are lined up in 2 rows. The modules share a refueling building where an NSSS module is transported and refuelled. The refueling building also houses the spent fuel storage pool. Each module resides under a biological shield in a three-sided bay that is open to a common, stainless-steel lined, pool.

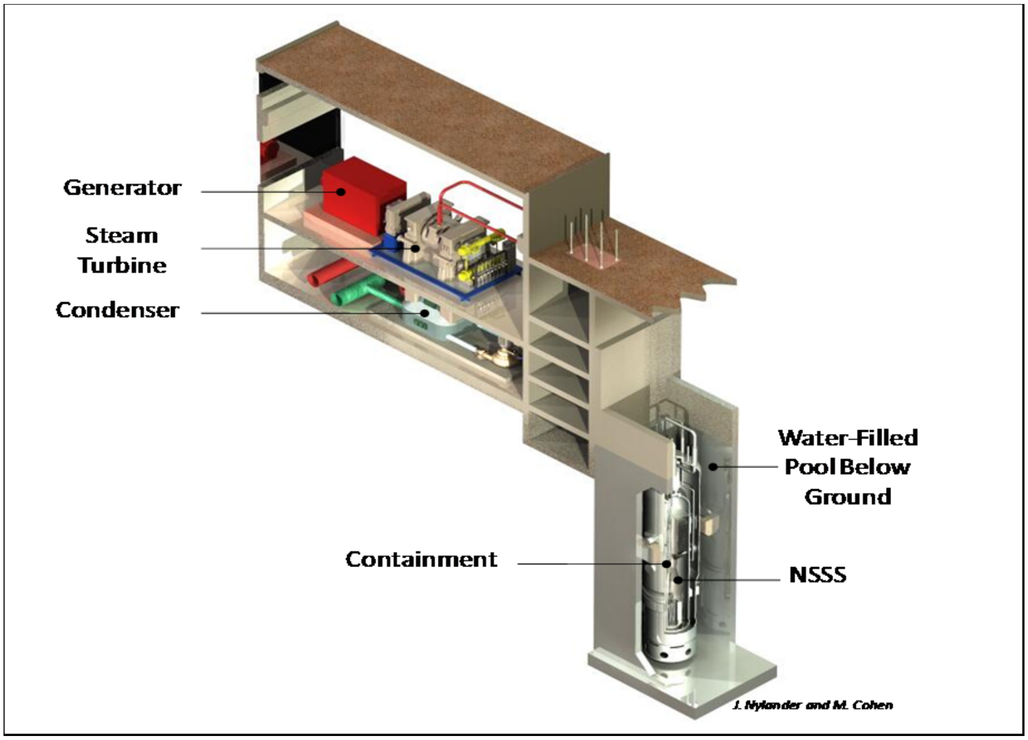

Figure 18 Cutaway view of a NuScale module

Module placement

The NuScale module is located below grade in a pool of water which serves multiple purposes. First, it provides passive containment cooling and decay heat removal; the pool provides a heat sink with a capacity to absorb all the decay heat produced by a fully mature core for greater than 30 days, as depicted in Figure 19. Second, it provides a means of dampening seismic events. Third, it provides an additional fission product barrier. Fourth, it provides radiation shielding outside containment. Last, the below grade pool provides physical security. Each power unit has its own dedicated turbine-generator set.

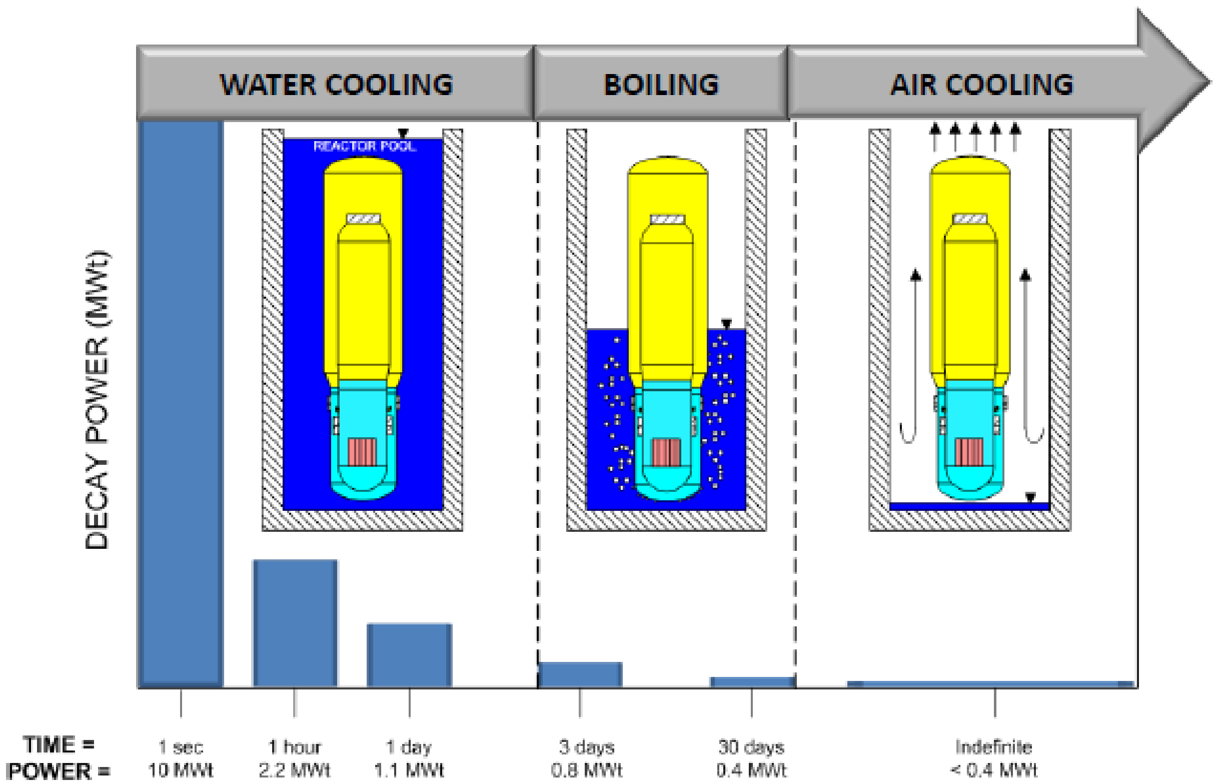

Figure 19 NuScale placement in a pool for passive decay heat removal

Reactor containment vessel

The NSSS module is composed of a reactor core, a pressurizer, and two steam generators integrated within the reactor pressure vessel and housed in a compact steel containment vessel. The reactor pressure vessel consists of a steel cylinder with an inside diameter of approximately 2.8 m and an overall height of approximately 20.0 m and is designed for an operating pressure of approximately 12.8 MPa.

Flanges are provided on the containment vessel to allow for disassembly and to allow access to the reactor pressure vessel during refueling operations and maintenance. Manways are located in the upper head and in the vessel circumference to allow access to the steam generator headers during refueling outages. Penetrations located on the vessel upper head provide access for process piping to the reactor pressure vessel and to the containment vessel interior. Additional penetrations are provided for electrical and instrumentation connections.

The vessel is vertically and laterally supported by connection to the reactor pool walls. A support skirt attached to the containment vessel lower head allows the vessel to be supported laterally. Internal to the containment, the reactor vessel is laterally and vertically supported by connections to the containment vessel wall.

The containment vessel is submerged in the reactor pool, which provides a passive heat sink for the containment heat removal under LOCA conditions. Although not credited, the reactor pool provides an additional means of fission product retention beyond that of the fuel, fuel cladding, reactor pressure vessel, and the containment vessel for certain events. The containment vessel is designed to withstand the environment of the reactor pool as well as the high pressure and temperature of any design basis accident.

The containment vessel pressure is maintained at a deep vacuum under normal operating conditions. Maintaining a deep vacuum provides for reduced moisture that could contribute to component corrosion and impact the reliability of instrumentation and other systems within the containment vessel. The deep vacuum essentially eliminates convection heat transfer removing the need for "direct-contact" reactor pressure vessel insulation. Due to a lack of appreciable amounts of air, the deep vacuum also enhances steam condensation rates that would occur during an accident with ECCS actuation and would limit the formation of a combustible mixture of hydrogen and oxygen during a severe accident.

Following an actuation of the ECCS, heat removal through the containment vessel rapidly reduces the containment pressure and temperature and maintains them at less than design conditions. Steam is condensed on the inside surface of the containment vessel, which is passively cooled by conduction and convection of heat to the reactor pool water. Because the containment vessel is evacuated to a low absolute pressure during normal operation, few non-condensable gasses are present inside the containment vessel. This is beneficial, because the presence of non-condensable gasses has a tendency to reduce condensation heat transfer rates.

Core and fuel assembly design

The NuScale reactor design features a 37-fuel assembly core composed of conventional 17x17 square lattice array fuel assemblies. The core includes sixteen control rod assemblies (CRAs). Each fuel assembly includes 264 fuel pins, 24 control rods, and one instrument tube.

The fuel assembly design will utilize axial zoning. The purpose of the axial zoning is to maximize neutron economy and control the axial power shape and peaking. This is achieved by using reduced enrichment axial blankets that are on the top and bottom of the fuel stack. This lower enrichment creates an effective reflector for the neutrons within the core, reducing the axial leakage.

Reactivity Control

Reactivity control is achieved using soluble boron and control rod assemblies (CRAs). The NuScale reactor core is cooled and moderated by light water primary coolant containing natural boron. The soluble boron is used throughout the cycle to compensate for the excess reactivity of the fresh fuel loaded into the core at the beginning of the cycle and is adjusted throughout the cycle to control slow reactivity changes resulting from burn-up effects. The soluble boron is also adjusted to compensate for large reactivity changes during reactor heatup and cooldown.

The NuScale reactor design incorporates a total of 16 identical CRAs for providing rapid shutdown and power distribution control. Each CRA spans only one fuel assembly and consists of 24 individual absorber rods integrated in a single spider assembly. The absorber rods are composed of silver-indium-cadmium absorber material, extending the full length of the core. Stainless steel tubes encapsulate the absorber material, isolating it from the reactor coolant.

The CRAs are used for reactor startup and shutdown, following load changes, and controlling small transient changes in reactivity. The CRAs can shut the reactor down at all times, even with the most reactive rod stuck out of the core.

The CRAs are grouped to form two control rod banks. Control group 1 consists of the inner four CRAs and is used for core reactivity regulation. Control group 2 includes the outer 12 CRAs and is used for reactor shutdown. All of the CRAs in control groups 1 and 2 are composed of the same materials and axial composition.

Steam generator

Each NSSS module uses two once-through helical-coil steam generators for steam production. The steam generators are located in the annular space between the hot leg riser and the reactor pressure vessel wall. The steam generator consists of tubes connected to upper and lower plenums with tubesheets. Preheated feedwater enters the lower steam generator plenum through nozzles on the reactor pressure vessel (see Figure 20). As feedwater rises through the interior of the steam generator tubes, heat is added from the reactor coolant and the feedwater boils and exits the steam generator as superheated steam.

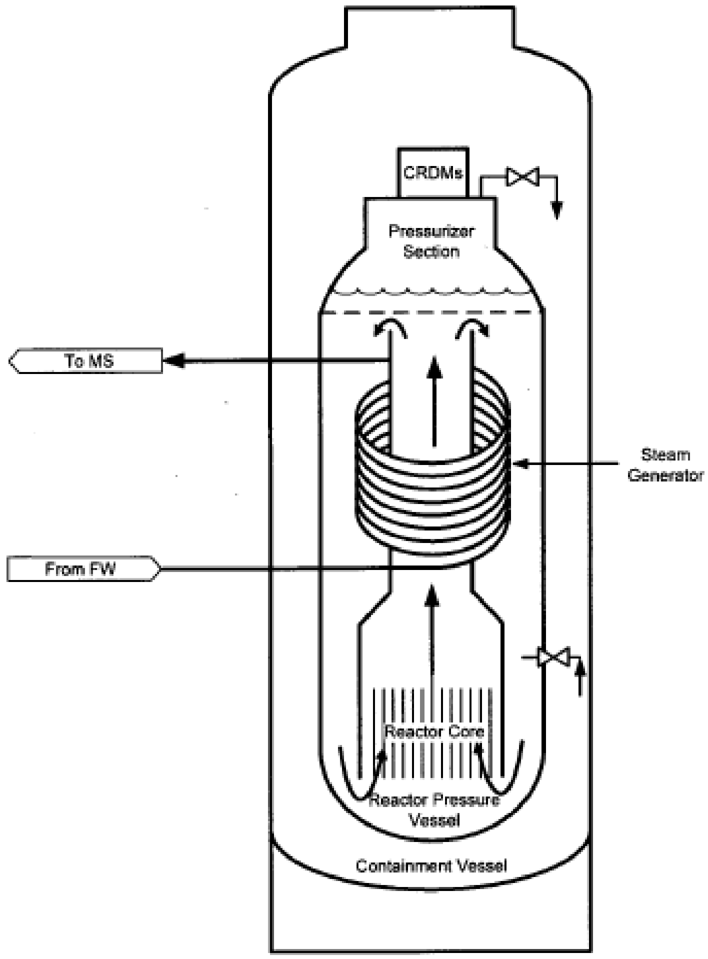

Figure 20 Steam generator and reactor flow

Pressurizer

The pressurizer is integrated into the reactor pressure vessel (RPV) and provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant pressure during operation. Reactor coolant pressure is increased by applying power to a system of pressurizer heaters in the reactor pressure vessel head. The heater penetrations are installed above the pressurizer separator plate. Pressure in the reactor coolant system is reduced using spray provided by the chemical and volume control system (CVCS).

Off-the shelf turbine-generator systems

Each NuScale reactor would implement a small 45MWe conventional steam turbine generator set optimized to NuScale steam generator outlet conditions. These are readily available and widely used in the fossil fuel power generation industry and can be dedicated for service in nuclear plants. The small size allows for air cooling of the generator, thus avoiding maintenance and safety issues associated with hydrogen cooling of the generator.

Multi-Module Plant

Citing the reduced operating requirements due to the design simplification and the advancement in digital controls, NuScale proposes that all reactors in the plant are controlled from a single operating room. The multi-module configuration offers refueling cycle advantage and distributed operational risks. First, it eliminates single shaft risk. That is, the temporary shutdown of a single unit does not require shutdown of the entire plant. Second, the layout permits refueling to be completed in a single module while the other modules continue to generate power. This staggered refueling of individual modules can be done by a small, well-trained, permanent team rather than employing a large temporary workforce. Refueling operations could be conducted throughout the year while keeping the remaining modules online. In addition, skid mounted turbine generator sets and ample lay down areas in the turbine building permit ease of maintenance and turbine replacement to reduce single unit down-times. A 12 module plant includes two independent turbine buildings each housing 6 steam turbine generator sets, a central reactor building which houses all of the modules, 2 sets of forced air evaporative coolers, and supporting electrical and mechanical equipment.

Spent Fuel and Waste Management

The multi-module NuScale plant spent fuel pool consists of a steel lined concrete pool located underground. It is designed with the capability of storing and cooling all of the fuel offloaded from 12 modules, as well as an additional 10 years of used nuclear fuel. After 10 years of storage in the spent fuel pool, air-cooled interim storage is possible as with conventional nuclear plants.

4.6.2 Key Safety Characteristics and Innovative Designs

The integral configuration eliminates the traditional Large Break Loss of Coolant Accident (LBLOCA) by design. For design basis small break assumptions, there is no scenario in which the core becomes exposed or uncovered – it will always be under water. Thus cooling pathways are always available to remove decay heat. Because of the assured heat removal path and the fail-safe nature of the emergency core cooling system (ECCS) valves, which passively open upon loss of power, the reactor can be safely cooled for an unlimited time with no AC or DC power, no operator action, and no additional water.

Inherent safety features

The inherent safety features of NuScale reactor include:

- The absence of large reactor coolant system (RCS) piping: the main NSSS components such as the reactor core and steam generator are placed in a pressure vessel and they are not connected with large diameter pipes.
- The core pressure is created by the inherently safe self-pressurization feature.

- The reactor has negative power and temperature coefficients.
- The natural convection cooling of the core does not rely on pumps. Thus, the core cooling can be provided in absence of external AC power. Also, it eliminates the small LOCA due to pump seal failure.
- The reactor does not have any penetration below the core such that, in case of a small loss of coolant accident, the core will not be uncovered by the loss of coolant outflow and provides much longer response time.
- Each module has a smaller fuel inventory and therefore a reduced source term compared to a conventional light water reactors.
- The containment vacuum eliminates the need for combustible gas control inside containment; i.e., there is little or no oxygen in the containment atmosphere.
- The steel containment immersed in a stainless steel lined pool eliminates the potential for molten core-concrete interactions.
- The ability to reliably equilibrate containment and reactor pressure prevents the possibility of a high-pressure “corium” melt ejection.

Passive safety features

- Additional fission product barriers: in addition to the fuel pellet, cladding, reactor vessel, and containment, NuScale design provides additional barriers which further reduce the potential for severe accident releases, including the containment cooling pool, the stainless steel lined containment pool structure, biological shield, and the reactor building.
- Reactor Pool: The reactor pool consists of a large, below-grade concrete pool with a stainless steel liner that provides stable cooling for the containment vessel for a minimum of 72 hours following any LOCA. During normal plant operations, heat is removed from the pool through a closed loop cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. In an accident where offsite power is lost, heat is removed from the reactors and containments by allowing the pool to heat up and boil. Water inventory in the reactor pool is large enough to cool the reactors for at least 72 hours without adding water. After 72 hours, reactor building pool water boil-off, and ultimately passive air cooling of the containment vessels, provide adequate cooling for long-term decay heat removal.
- Emergency cooling water injection by gravity: As shown in Figure 21, the emergency core cooling system (ECCS) consists of two independent reactor vent valves and two independent reactor recirculation valves. The ECCS provides a means of decay heat removal in the event-of a loss of coolant accident or a loss of the main feedwater flow in conjunction with the loss of both trains of the DHRS.

The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the containment vessel. It allows heat conduction through the containment vessel walls and heat conduction and convection to the water in the reactor pool. Long-term cooling is established via recirculation of reactor coolant to the reactor pressure vessel via the ECCS recirculation valves, which when opened provide a return flow of cooled water to the reactor.

The ECCS is initiated by opening the two (2) reactor vent valves in lines exiting the top of the reactor pressure vessel (the pressurizer region) and the two (2) reactor recirculation valves on lines entering the reactor pressure vessel in the downcomer region at a height above the core. Opening the valves allows a natural circulation path to be established. Water that is vaporized in the core leaves as steam through the reactor vent valves, is condensed and collected in the containment vessel, and is then returned to the downcomer region inside the reactor vessel through the reactor recirculation valves.

Following a LOCA or other condition resulting in an actuation of the ECCS, heat removal through the containment vessel rapidly reduces the containment pressure and temperature and maintains them at acceptably low levels for extended periods of time. Steam is condensed on the inside surface of the containment vessel, which is passively cooled by conduction and convection of heat to the reactor pool water. Since the containment vessel is evacuated to a low absolute pressure during normal operation, only a small amount of non-condensable gas will be present inside the containment vessel.

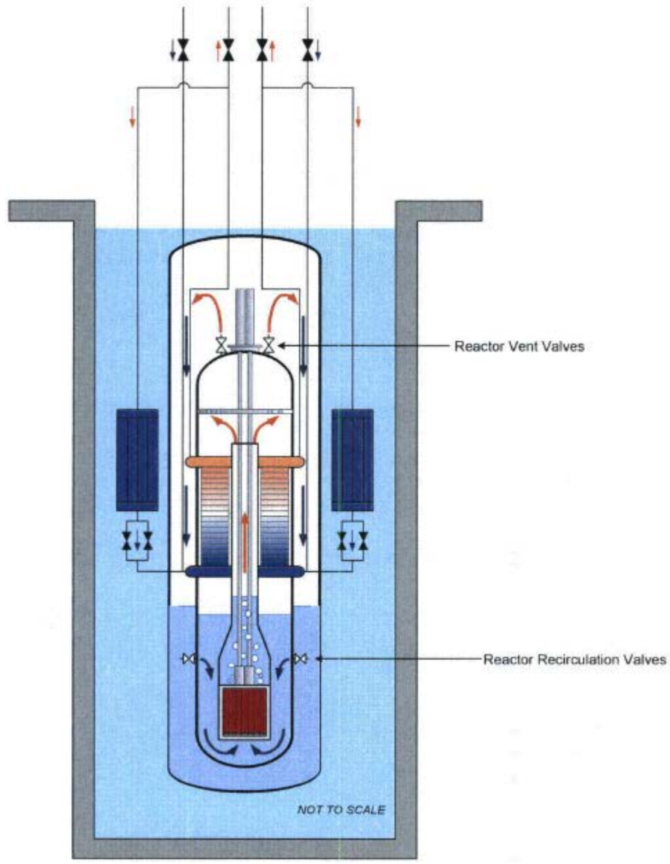

Figure 21 Emergency core cooling and containment heat removal system schematic

- Vacuum containment: during normal power operation, the containment atmosphere is evacuated to provide an insulating vacuum that significantly reduces heat loss from the reactor vessel. As a result, the reactor vessel does not require surface insulation. This eliminates the potential for sump screen blockage. Furthermore, the deep vacuum improves steam condensation rates during any sequence where safety valves vent steam into this space. Further, by eliminating containment air, it prevents the creation of a combustible hydrogen mixture in the unlikely event of a severe accident (i.e., little to no oxygen), and eliminates corrosion and humidity problems inside containment, and eliminates the need for hydrogen recombiners. Finally, because of its relatively small diameter, it has been designed for a maximum pressure of approximately 5.5 MPa. As a result, the equilibrium pressure between the reactor and the containment vessels in the event of a small break LOCA is achieved within a few minutes and will always be below the containment design pressure.
- Decay heat removal system: the decay heat removal system (DHRS) provides secondary side reactor cooling when normal feedwater is not available. The system is a closed-loop,

two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each steam generator loop. Each train is independently capable of removing a 100 % of the decay heat load and cooling the reactor coolant system. Each train has a passive condenser submerged in the reactor pool. The condensers are maintained with sufficient water inventory for stable operation.

Upon receipt of an actuation signal, the DHRS valves open. This allows water from the decay heat removal condensers to flow into the steam generators and cool the reactor coolant as it boils. The steam then travels through the steam line back to the decay heat removal condenser where it is condensed by the reactor pool water, and the cycle is repeated. Heat is removed via the steam generators, thus preserving natural circulation within the reactor coolant system.

4.7 mPower

The brief description of the mPower reactor, as shown in Figure 22, is taken from the World Nuclear Association website[37].

In mid-2009, Babcock & Wilcox (B&W) announced its B&W mPower reactor, a 180 MWe integral PWR designed to be factory-made and railed to site. B&W Nuclear Energy Inc. has set up B&W Modular Nuclear Energy LLC (B&W MNE) to market the design, in collaboration with Bechtel which joined the project as an equity partner to design, license and deploy it. B&W's 90%-owned subsidiary, Generation mPower LLC (GmP), reports into B&W MNE. B&W expects both design certification and construction permit in 2018 and commercial operation of the first two units in 2022.

When B&W announced the launch the mPower design in 2009, it said that Tennessee Valley Authority (TVA) would begin the process of evaluating Clinch River at Oak Ridge as a potential lead site for the mPower reactor, and that a memorandum of understanding had been signed by B&W, TVA and a consortium of regional municipal and cooperative utilities to explore the construction of a small fleet of mPower reactors. It was later reported that the other signatories of the agreement are First Energy and Oglethorpe Power. In February 2013 B&W signed a contract with TVA to build up to four units at Clinch River, with design certification and construction permit application to be submitted to NRC in 2014 or 2015. In July 2012 B&W's GmP signed a memorandum of understanding to study the potential deployment of B&W mPower reactors in FirstEnergy's service territory stretching from Ohio through West Virginia and Pennsylvania to New Jersey.

[37] World Nuclear Association, http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Small-Nuclear-Power-Reactors/

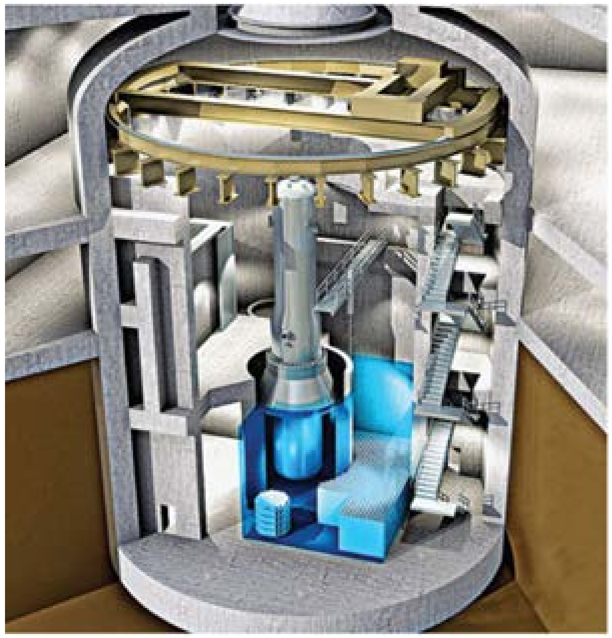

Figure 22 Cross cut view of the mPower NSSS system with containment

The technical data of mPower reactor is summarized in Table 9 which is taken from the series of Generation mPower corporate presentations available on the Ux Consulting company website[38]. Other than this marketing-oriented information, the mPower design information is not generally available in the public domain.

Table 9 Summary of the mPower Reactor Technical Data

Parameter	Values	Units
General plant data		
Thermal output power	530	MW(th)
Power output	180	MW(e)
Plant efficiency	33.9	%
Mode of operation	Base load	
Plant design life	60	Years
Plant availability	N/A	%
Primary coolant	Light water	

[38] http://www.uxc.com/smr/uxc_Library.aspx?dir=Design Specific/mPower

Moderator	Light water	
Thermo dynamic cycle	Indirect Rankine steam cycle	
Reactor core		
Active core height	2.4	m
Equivalent core diameter	2	m
Average linear heat rate	11.5	kW/m
Average fuel power density	N/A	kW/kgU
Average core power density	~70	MW/m^3
Fuel material	UO_2	
FE type	Smooth-rod, Cylindrical	
Cladding material	Zircaloy-4	
Lattice geometry	Square	
Number of fuel assemblies	69	
Fuel enrichment	< 5	Weight %
Fuel cycle length	4+	years
Average discharge burnup	N/A	MWd/kgU
Burnable absorber	Gd_2O_3	
Mode of reactivity control	Control rods	
Mode of reactor shutdown	Control rods	
Control rod absorber material	Ag-In-Cd and B_4C	
Soluble neutron absorber	None	
Primary coolant system		
Core coolant inlet temperature	297.2	ºC
Core coolant outlet temperature	320	ºC
Operating pressure	14.13	MPa
Reactor vessel		

Diameter of cylindrical shell	4500	mm
Wall thickness of cylindrical shell	N/A	mm
Total height	25000	mm
Steam generator		
Type	Straight tube, Once through	
Number	N/A	
Total tube surface area	N/A	m^2
Number of HX tubes	N/A	
Tube outside diameter	N/A	mm
Tube material	N/A	
Primary circulation system		
Circulation type	Forced	
Pump type	Vertical	
Number of pumps	8	
Pressurizer		
Type	Internal, electrically heated	
Total volume	N/A	m^3
Containments		
Primary	Cylindrical concrete	

4.7.1 Description

The following description of mPower reactor and plant is reconstructed from the vendor presentations that are available on Ux Consulting Company database[39].

The mPower project is being developed by the alliance between Babcock and Wilcox (B&W) and Bechtel. The mPower is a Generation III+ IPWR with a modular design that uses light water as coolant and moderator. The reactor and steam generators are located in a single integrated reactor vessel located in an underground containment facility that would also store all of the spent fuel. The modular unit has a diameter of 4.5 metres and it is 25 metres high. The reactor

[39] [39] http://www.uxc.com/smr/uxc_Library.aspx?dir=Design Specific/mPower

core size is about 2 m in diameter and 2 meters in height. The steam condenser is designed to be cooled by air or water depending on the availability of cooling water. The choice of condenser cooling method determines the electrical output which varies from 150 MWe to 180 MWe. In its pre-application review with the Nuclear Regulatory Commission, the reactor's rated capacity was described to be 500 MW of thermal power and 160 MW of electrical power, but the thermal power of 530 MW and electrical power of 180 MW are also cited in the design presentations. The reactor has an expected lifetime of 60 years.

Plant layout

The default plant configuration is a 'twin-pack' where 2 fully underground reactor systems are located in one NPP as shown in Figure 23. The installation shares a common spent fuel pool located between the reactors.

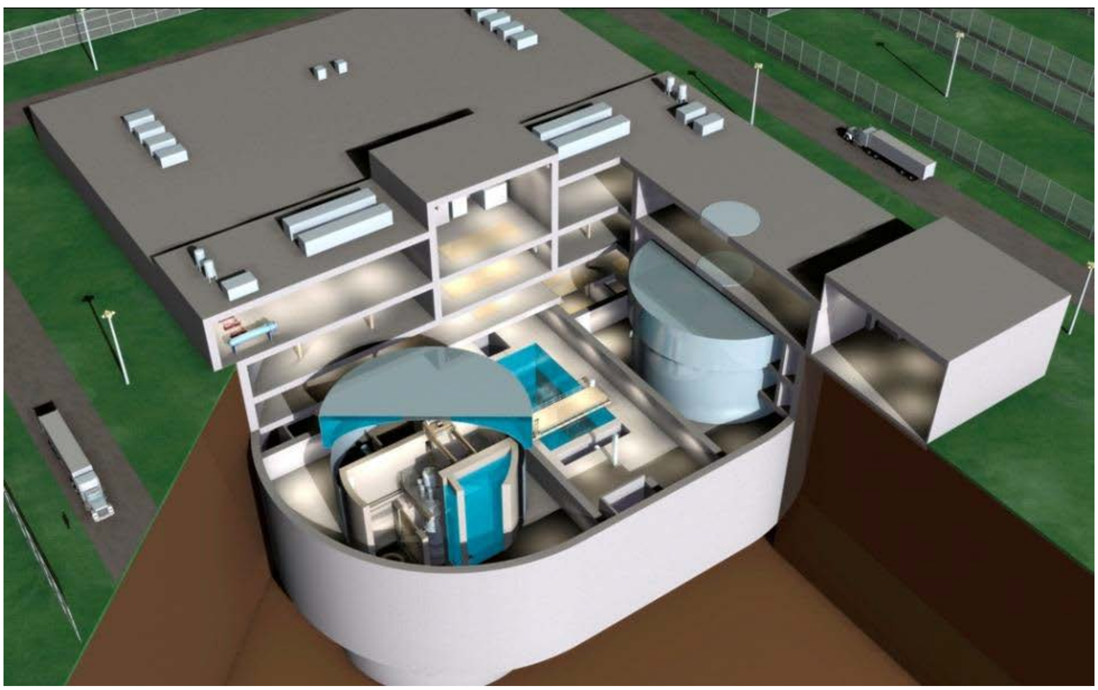

Figure 23 The underground construction of a 2 unit mPower NPP

Nuclear steam supply system

The overall structure of the mPower NSSS module is shown in Figure 24.

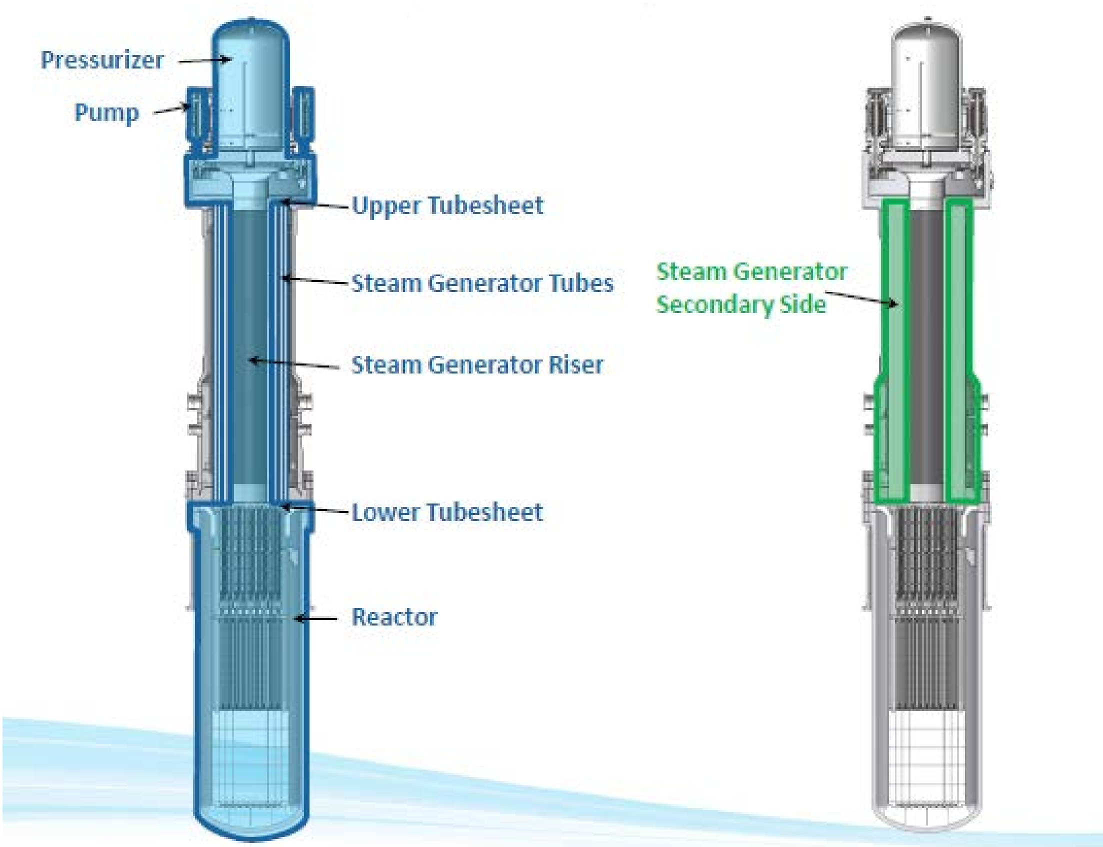

Figure 24 mPower integral reactor vessel structures

The reactor core is placed at the bottom of the vessel and the control rod mechanism and the rod guide frame are also placed in the bottom section of the pressure vessel. The bottom section does not have any penetration. The upper part of the vessel is connected to the lower section by flange connection. The upper vessel consists of the coolant riser, electrically heated pressurizer, reactor coolant pump and steam generator. The hot and buoyant coolant water rises from the core to the top of the vessel through the riser and then it comes down through the outer layer to pass through the once-through steam generator. The coolant flow is due to natural convection and 8 small pumps.

Reactor core and assembly design

The reactor core consists of 69 reduced-height, but otherwise standard, Westinghouse 17×17 PWR fuel assemblies (FAs). The FAs are of a conventional square lattice design on a 21.5 cm inter-assembly pitch with a fixed grid structural cage. The assembly has been shortened to an active length of 241.3 cm and optimized specifically for the mPower reactor.

The mPower uses fuel enriched to less than 5% with Gd_2O_3 spiked rods to control the long-term reactivity changes due to fuel burnup. Ag-In-Cd (AIC) and B_4C control rods with a 3% shutdown margin also form the part of fuel assembly known as RCCA spider (short for rod cluster control assembly spider). There is no soluble boron present in the reactor coolant for reactivity control.

Fuel cycle

The reference core loading and fuel management plan for a four-year cycle is based on its presumed attractiveness to potential customers. This option is a once-through fuel cycle in which the entire core is discharged and replaced after four years, unlike traditional reactors, which require fuel handling and movement of individual fuel rods during a refuelling outage. The entire used core, once removed, can be placed in storage in the spent fuel pool next to the integral reactor vessel (IRV) in the containment, which is designed to hold an entire 60 years' worth of used fuel, and is accessible by the containment gantry crane located above the IRV within the containment.

In addition, a conventional fuel utilization strategy, employing a periodic partial reload and shuffle, was developed as an alternative to the four-year once-through fuel cycle. The alternative fuel cycle option of mPower provides flexibility to customers who consider the once-through fuel cycle unacceptable from a fuel utilization standpoint. When compared to the once-through concept, reloads of the mPower reactor will achieve higher batch average discharge exposure, will have adequate shut-down margin, and will have a relatively flat excess reactivity profile across the core at the expense of slightly increased power peaking factor.

Reactivity control

The long term core reactivity is controlled by spiking the fuel rod with Gd_2O_3 rather than using soluble boron in the coolant. The burnable poison assures that the excess reactivity is suppressed below the shutdown margin with 3%.

The control rods are placed in the fuel assembly and their movements are guided with control rod guide tubes. The control rods are connected to the drive mechanism to form a RCCA spider.

Steam generator

The mPower module uses a once-through steam generator for steam production. The steam generators are located in the annular space between the hot leg riser and the reactor pressure vessel wall. The steam generator consists of tubes connected to upper and lower plenums with tubesheets. Heated coolant rises through the riser placed above the core to the upper plenum. Then it comes down through the annular space where steam generator is located. The coolant transfers heat to the feedwater in the steam generator tubes; the coolant becomes colder and denser, and sinks to the bottom of the reactor, creating the natural convection flow in the vessel. The feedwater boils and exits the steam generator as superheated steam.

Coolant pumps

The reactor uses eight internal coolant pumps with external motors driving 3.8 m^3/s of primary coolant through the core. The coolant flow in the core is assisted by the natural convection force.

Pressurizer

The integrated pressurizer at the top of the reactor is electrically heated and the reactor coolant pressure is nominally at 14.1 MPa.

4.7.2 Key Safety Characteristics and Innovative Designs

Inherent safety features

The safety of the mPower reactor is mainly characterized by the following innovations in the design.

- Integral vessel design: a large pipe break LOCA is not possible because the primary components are located inside the pressure vessel and the maximum diameter of the external connected piping is less than 7.6 cm.
- Internal control rod drives eliminate the possibility of rod ejection accident.
- Low core linear heat rate: it reduces the fuel and clad temperatures during accident. Also, lower power density allows lower flow velocities to minimize the flow induced vibration effects.
- Large reactor coolant system (RCS) volume compared to the core power: large RCS volume allows more time for safety system response in the event of an accident. In case of a small break LOCA, more coolant is available to provide continuous cooling to protect the core.
- No penetration in the lower part of the vessel: the high penetration locations increase the amount of coolant left in the vessel after a small break LOCA. Also, it reduces the rate of energy release to containment resulting in lower containment pressures.
- Underground placement: the reactor module is placed in a secure underground containment which reduces the consequences of aircraft impact or other natural disasters.
- Soluble boron is not used for reactivity control, which eliminates a boron dilution accident.

Passive safety features

The mPower reactor also incorporates various passive emergency safety features. The emergency core cooling system is connected with the reactor coolant inventory purification system and removes heat from the reactor core after anticipated transients in a passive manner.

The mPower reactor deploys a decay heat removal strategy with a passive heat exchanger connected with the ultimate heat sink, an auxiliary steam condenser on the secondary system, water injection or cavity flooding using the reactor water storage tank, and passive containment cooling.

5. Discussion of Small Nuclear Reactor Safety Innovations

Based on the assessment of SMR design data in the previous section, the innovative safety features of SMRs that a regulator should be able to explicitly account for in its review are summarized in Table 10. No single SMR necessarily has all of these aspects; however, the seven reactor designs discussed in this report, as well as other water-cooled SMRs that are not included in this report, are expected to possess a subset of these innovative safety features. Each safety characteristic is categorized as inherent or passive, using the very precise definitions in APPENDIX A.

The active safety systems are not discussed in this report, as they are not much different from those currently used in conventional power reactors. One area where there is no significant improvement is in reactivity control: movement of control rods is under active control and they still present a potential to be withdrawn in error. Loss of reactivity control therefore remains a conventional Design Basis Accident for SMRs. Similarly most of the shutdown systems are also conventional (release of control rods into the core).

Table 10 SMR Safety-related Innovations

Safety Characteristic	Category	Comments
		Related to the reactor core
Low total fission product inventory	Inherent	This is a function of the thermal output of the core and the burnup. It can lead to a low accident fission product source term.
Low free fission product inventory	Inherent	The free (gaseous) fission product inventory in a fuel element is a function of the fuel element rating and burnup. It can also lead to a low accident source term.
Negative reactivity feedback	Inherent	A negative reactivity coefficient such as fuel or moderator temperature may slow down or arrest a reactivity excursion at a safe stable power level. It is not equivalent to shutting the reactor down. See [31] for details on how reactivity feedback is related to safety. If power is stabilized, there still needs to be some means of removing it.
No use of soluble poison such as boron in the coolant (PWRs)	Inherent	Eliminates boron dilution accidents

Advanced Fuel	Inherent	KLT-40S design uses ceramic metal fuel that exhibits high thermal conductivity and low operating temperature. StarCore plans to use TRISO fuel which can withstand high operating temperature, and whose layers serve as micro-containment to retain fission products.
Once-through steam generators	Inherent	Limits rate of cooldown following a main steam line break (MSLB) - and hence the positive reactivity insertion rate. Relevant to reactors with a negative reactivity feedback coefficient for coolant / moderator / fuel temperature. The trade-off is much less available inventory in the steam generators (compared to "light-bulb" type steam generators as in CANDU and in many PWRs) in case of a loss of heat sink.
Control rods inside pressure boundary	Inherent	If the integral pressure-vessel design also encloses the entire control rod, then pressure-assisted control rod ejection is not possible.
Related to the Reactor Coolant System (RCS)		
Integral RCS design	Inherent	By putting the steam generators, the pressurizer, the pumps and the core all inside a single pressure vessel, large RCS pipes can be eliminated and hence LBLOCA can be claimed to be beyond design basis. None of the sample designs assume LBLOCA is credible. However this approach places more emphasis on ensuring the integrity of the integral pressure vessel. There are also subsidiary issues such as the basis for the ECCS design and the containment design.
Large thermal inertia	Inherent	A large ratio of RCS inventory to reactor power slows down the progression of accidents with a loss of heat removal.
Advanced pressurizer	Inherent/passive	Water-cooled SMR reactors are self-pressurized (e.g. SMART, CAREM, NuScale, mPower) or use 'soft' gas pressurizer (e.g. KLT-40S) which damps the pressure perturbation in a DBA. In some designs, electrical heaters and sprays are eliminated from the design, so that their failure is not part of the list of DBAs.

Natural circulation	Passive	CAREM and NuScale rely on natural convection flow to cool the core at power. Thus, reactor coolant pumps are not necessary and that eliminates accidents related to pump failure. In other water-cooled reactors, the reactor internal layout is set up such that there is some degree of natural convection flow. Natural convection mitigates the consequences of a loss of forced flow and also removes decay heat during reactor shutdown.
Passive core flooding on LOCA	Passive	Even for designs where LBLOCA is claimed to be a Beyond-Design-Basis Accident (BDBA), small LOCA may still be a Design Basis Accident (DBA). A passive Emergency Core Cooling System (ECCS) (flooding the core by gravity) could in principle result in high reliability of core cooling. A detailed review would need to confirm the degree of dependence on active components such as valves – i.e. the category of passive safety as described in APPENDIX A
Canned pump	Inherent	The lack of pump seal either reduces or eliminates small break LOCA due to seal failure.
Passive core heat removal with intact RCS boundary	Passive	For non-LOCA events, a design may have passive decay heat removal using natural circulation in one or more loops to an ultimate heat sink. Two designs even use natural RCS circulation at operating power. The comments above on degree of passivity also apply.
Passive core cooling after a LOCA	Passive	Post-LOCA, generated heat may be removed from gas-cooled reactors by natural circulation using atmospheric air. Typically this mechanism also relies on fuel that can withstand high temperatures (e.g. TRISO).
Pressure vessel penetration	Inherent	Many water cooled SMRs do not have large penetrations in the RPV. Where small size penetrations are made, they are made at much higher elevation than the top of the reactor core to prevent core uncovery.
	Containment / Severe Accidents	
Passive containment heat removal	Passive	As with the RCS, an SMR may have a passive means of cooling containment – e.g. immersing the entire containment in a water pool. As with other passive systems, a detailed review would need to confirm the degree of dependence on active components such as valves – i.e. the category of passive safety as described in APPENDIX A

Inerted containment	Inherent	An inerted containment, or one kept under high vacuum, would prevent hydrogen deflagration after a severe accident, due to the absence of oxygen.
RPV in a pool	Passive	The NuScale design places the RPV in a pool which serves as a heat sink to remove the majority of decay heat following an accident. The evaporation of the pool allows time for the reactor decay heat to fall to a level that can be indefinitely removed by natural air convection.
Underground siting	Passive	If the reactor and all (or most) of its containment are located underground, there is passive protection for severe external events such as aircraft crash, malevolent acts, extreme weather etc.
External vessel flooding / containment flooding	Passive	One means of arresting a severe core damage accident at the pressure-vessel boundary is to flood the pressure-vessel from the outside (or flood the inside of containment).

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6. Relevant R&D Activities

This section of this report addresses the third scope of the contract as listed in section 1.3:

> *3. Discuss the identified research and development (R&D) activities that support the vendors' small reactor designs and their safety feature;*
>
> *In case that collected information for Items 3 and 4 is not sufficient to address associated issues, the Contractor should give their own critical view on what else has to be considered in the R&D activities and what measures to be taken in protecting the reactor design from the exceptional geophysical events so that the vendor's reactor design can effectively respond to the anticipated transients and postulated accidents, and in mitigating the consequences of severe accidents.*

The previous section listed the SMR innovative safety features that are not generally found in conventional water-cooled power reactors. Hatch found that the vendors, through academic publications, conference presentations or marketing materials, claim that their designs and safety innovations are supported with various R&D activities. However, it was not possible to validate these claims using public information; the details of the R&D information are likely to be available not to the public but only to the regulators as part of a licensing application.

The identified R&D information that is relevant to the SMR development is discussed below. Hatch also found that there are several IAEA technical reports that discuss a few innovative safety features listed in this report.

6.1 KLT-40S

KLT-40S is being licensed in Russia and the reactor is already loaded on to the floating power unit[40]. OKBM Afrikantov reports [32] that the following R&D activities have been performed in regards to reactor safety:

- Design of emergency reactor shutdown systems
- Emergency heat removal systems
- Emergency core cooling system
- Emergency containment pressure reduction system
- External hazards resistance analysis
- Severe accident analysis
- Radiation and environmental safety

[40] http://theconversation.com/russias-floating-nuclear-plants-to-power-remote-arctic-regions-19994

- Stress analysis including total station blackout.
- Level 1 probabilistic safety analysis

The IAEA status report [16] summarized the technology development status as in Table 11.

Table 11 KLT-40S Technology Development Status

Reactor Technology	Status
Modular PWRs for Russian nuclear vessels	Widely used reactor technology, operation experience of marine multipurpose reactors exceeds 6500 reactor-years
Icebreaker type KLT-40S reactor for the FOAK FNPP	The RP and FPU designs were developed; FPU construction license was obtained from the regulatory authority; fabrication of RP and steam-turbine plant equipment is under way
Nuclear shipbuilding technology	The total of 11 nuclear vessels was constructed (icebreakers, one lighter carrier), FPU construction is in progress
Reactor of the nuclear cogeneration plant (AST-500) regarding safety ensuring approaches and solutions	Under construction; IAEA performed Operational Safety Review[33]

The reactor and FNPP are being developed by Russian organizations and enterprises that have experiences in designing, building and operating nuclear reactors for the Navy and civil fleet. The list of participants is shown in Table 12.

Table 12 Main participants of KLT-40S project

Company	Responsible Project Area
JSC “Afrikantov OKB Mechanical Engineering” (OKBM), Nizhny Novgorod	Chief designer of the RP, package supplier of the RP
RRC “Kurchatov Institute”, Moscow	Scientific supervisor of the RP design
JSC “TsKB “Iceberg”, Saint Petersburg	General designer of the FPU
Krylov Shipbuilding Research Institute	Scientific supervisor of the FPU design
JSC “Atomenergo”	Design of coastal and hydraulic engineering facilities

United Shipbuilding Corporation, "Baltiysky Zavod", Saint Petersburg	FPU builder
Rosatom, JSC "Energoatom Concern", Directorate of FNPPs under Construction	Project customer and investor
JSC "Kalouzhsky Turbine Plant"	General designer and supplier of the turbine set

6.2 VBER-300

According to the IAEA status report [20], OKBM claims that the key R&D work for the VBER-300 RP was completed (optimization of circuitry and layout, reactor and SG material studies, purging of models for reactor and MCPs, etc.). The basic VBER-300 RP production technologies mastered on the commercial scale are as follows:

- Technology of vessel system welding.
- Technology of SG tubing manufacture of titanium alloys.
- Technology of manufacturing and assembling the in-vessel coaxial elements that provide coolant circulation.
- Canned MCPs development and fabrication technology.
- Technology of VVER skeleton-type TVSA FAs[41].
- Technology of fabrication of normal operation system and safety system elements (self-actuated devices, pressurizer, tanks, HXs, pumps, filters).

The current status of technology development that forms the basis of VBER-300 RP development is given in Table 13 below.

Table 13 Development status of technologies relevant to VBER-300

Reactor Technology	Status
Modular PWRs for Russian nuclear vessels	Widely used reactor technology, operation experience of marine multipurpose reactors exceeds 6500 reactor-years
VVER-1000 power reactors (reactor cores)	Widely used reactor technology (20 operating reactors)
Icebreaker type KLT-40S reactor for the FOAK FNPP (for the floating variant)	The RP and FPU designs were developed; FPU construction license was obtained from the regulatory authority; fabrication of equipment and construction of

[41] http://www.okbm.nnov.ru/english/reactor-cores-and-fuel

	FPU are under way.
Reactor of the nuclear cogeneration plant (AST-500) regarding safety ensuring approaches and solutions	Under construction; IAEA performed Operational Safety Review[33]

The designs are being developed by Russian organizations and enterprises having experience in designing, building and operating nuclear reactors for the Navy and civil fleet. The main participants in the VBER-300 projects are shown in Table 14.

Table 14 Main participants of the VBER-300 project

Company	Responsible Project Area
JSC "Afrikantov OKB Mechanical Engineering" (OKBM), Nizhny Novgorod	Leading designer of the NPP, chief designer of the VBER-300 RP
RRC "Kurchatov Institute", Moscow	Scientific supervisor of the RP design
Scientific Research and Design Institute " Atomenergoproyekt" (NIAEP), Nizhny Novgorod	General designer of the ground-based NPP
JSC " TsKB Lazurit", Nizhny Novgorod	General designer of the floating NPP
JSC " Power Machines"	Chief designer of the turbine generator set
FSUE "NIIIS"	Chief designer of the automated process control system

6.3 SMART

The companies and institutes that participate in the SMART project are presented in Table 15 [34].

Table 15 Main participants of the SMART project

Company	Responsible Project Area
KAERI	Technology validation
KAERI	NSSS design
KEPCO NF	Fuel design
KEPCO E&C	BOP design

Doosan	Component design

The existing proven PWR technologies are basically utilized for the SMART design. However, it also adopts new and innovative design features and technologies that must be proven through tests, experiments, analyses, and/or the verification of design methods.

According to the IAEA ARIS report [21], a series of fundamental tests and experiments have been carried out throughout the SMART development phases to examine the physical phenomena related to the specific SMART design concepts. The main purpose of these experiments was two-fold: to understand the thermal-hydraulic behaviour of the specific design concepts and to obtain fundamental data to be used, in turn, for further feedback to the optimization of design. Among the experiments conducted, specific SMART design-related experiments are as follows:

- Boiling heat transfer characteristics in the helically coiled steam generator tube.
- Experiment for natural circulation in the integral arrangement of the reactor system.
- Two-phase critical flow tests with non-condensable gases to investigate the thermal-hydraulic phenomena of critical flow with the existence of non-condensable gases.
- Critical heat flux measurement for SMART-specific UO_2 fuel rod bundles.
- Water chemistry and corrosion tests at a loop facility to examine the corrosion behaviour and characteristics of fuel cladding, internal structural materials, and steam generator tube materials at reactor operating conditions.
- Experiments on wet thermal insulation to determine the insulating effects for the internal PZR design and to derive a heat transfer coefficient for the design.
- Experiments on phenomena and characteristics of heat transfer through the condensing mechanism of the heat exchanger inside PRHRS tanks.

The overview of the SMART design and technology validation [35] is presented in Figure 25.

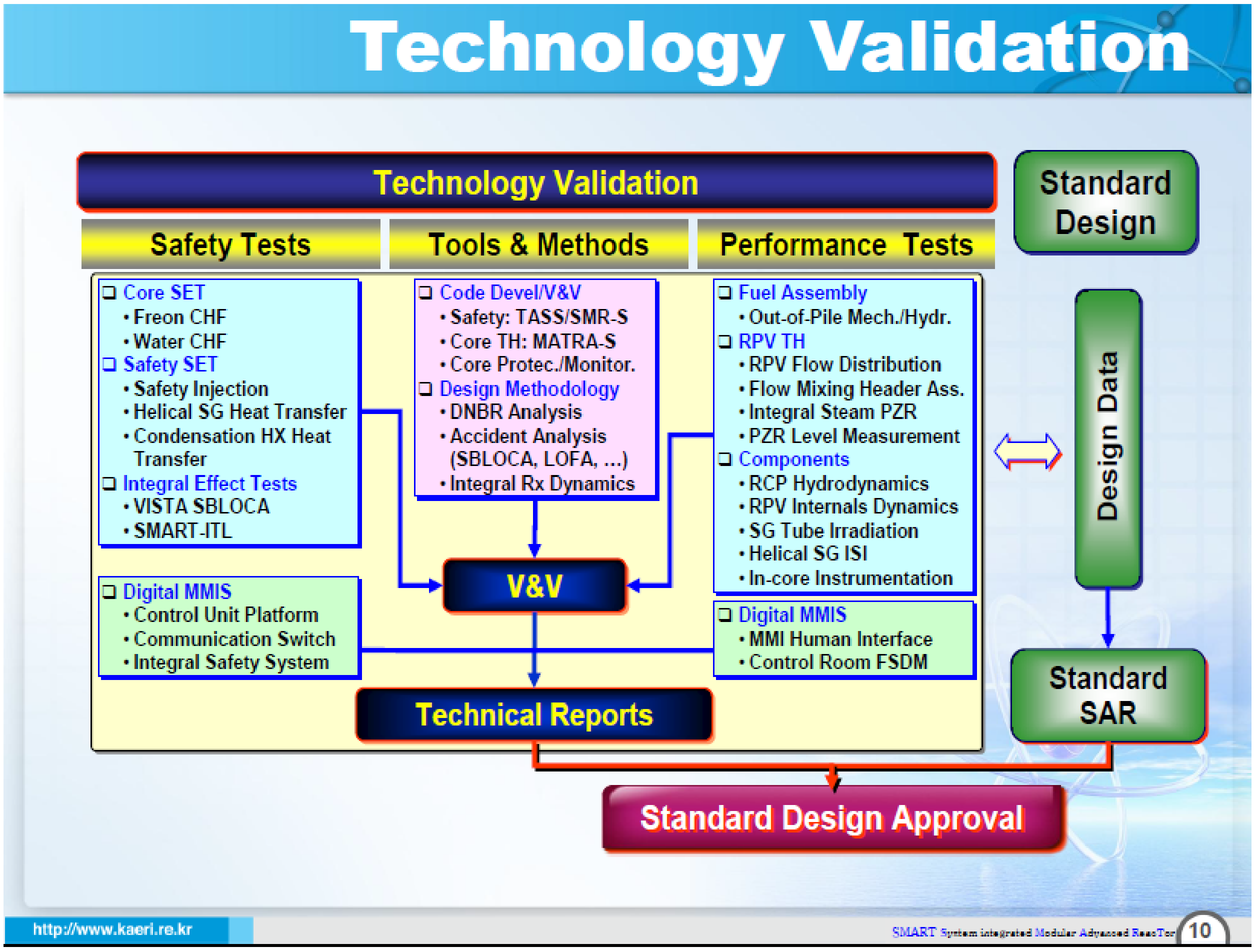

Figure 25 Overview of SMART technology validation

KAERI claims [34] that 22 validation experiments were selected based on PIRT [36] and expert opinions from regulators, industries, institutes and universities. The experimental validation envelops fuel, core, thermalhydraulics, safety, mechanics, components and digital I&C. In addition, various software validations are undertaken to validate key design tools and methods (e.g. core physics, core thermal-hydraulics, safety analysis, etc.).

Described below are brief introductions of the typical tests and experiments implemented to verify the SMART design characteristics.

6.3.1 Two-Phase Critical Flow Test with a Non-Condensable Gas

The early SMART concept adopted an in-vessel pressurizer type with an inherent self-pressure regulating capability designed to operate via the thermo-pneumatic balance between the water, steam, and nitrogen gas which are the three fluids that fill the pressurizer. In the event of a rupture of a pipe line connected to the pressurizer at a high-system pressure, a mixture of water, steam, and nitrogen is discharged through the break at critical flow conditions. The computer codes for the safety analysis of SMART need to use a verified and validated model for this critical flow. To investigate the thermal-hydraulic phenomena of the critical flow affected by the non-condensable gas entrained in the two-phase break flow, a separate effects test facility [37] was designed and installed at the KAERI site. The test facility can be operated at a temperature of 323C and pressure of 12 MPa with a maximum break size of 20 mm in diameter. A nitrogen gas flow rate of up to 0.5 kg/s can be injected and mixed with a two-phase mixture in the test section to simulate the transient behaviour expected during a LOCA. The test data from the facility was used for the development and verification of the critical break flow model for SMART.

6.3.2 Integral Effect Test

The thermal hydraulic integral test facility, VISTA (Experimental Verification by an Integral Simulation for a Transients and Accidents) [38], has been constructed to simulate the SMART-P, the prototype for SMART reactor. The VISTA facility has been used to understand the thermal-hydraulic behavior including several operational transients and design basis accidents. During the past five years, several integral effects tests have been carried out and reported, including performance tests, RCP transients, power transients and heatup or cooldown procedures, and safety related design basis accidents. VISTA will contribute to verifying the system design of the reference plant.

6.3.3 Major Components Performance Test

A performance test of the major components such as the RCP, SG, and CRDM was carried out in dedicated facilities [39].

6.4 CAREM

The majority of CAREM development activities are available from the project status report [40].

Within CAREM Project, the effort has been focused mainly on the nuclear island (inside containment and safety systems) where several innovative design solutions require concept development (to assure they comply with functional requirements). This comprises mainly: the Reactor Core Cooling System (RCCS), the Reactor Core and Fuel Assembly, internals of the Reactor Pressure Vessel (RPV), and the First Shutdown System (FSS). An extensive experimental plan has been prepared, including the design and construction of several experimental facilities to fulfill the Project's requirements.

The RCCS modelling and qualification are boosted by the tests performed in a High Pressure Natural Circulation Rig (CAPCN), covering Thermal Hydraulics (TH), reactor control and operating techniques. The CAPCN rig reproduces all the dynamics phenomena of the RCCS, except for 3-D effects.

The Core Design involves different aspects: i.e., study of thermal limits, neutronic modeling, structural mechanical and fuel assembly design. Neutronic modeling needs may be covered by benchmark data available world-wide and by experimental data from the Critical Facility RA-8. As for Fuel Assembly Design, CNEA has vast experience in the technology of nuclear fuels and their structural and hydrodynamic tests will be carried out in low and high pressure rigs.

The following is a brief description of some of the most relevant development tasks and their facilities that are carried out or foreseen as part of the CAREM project:

Dynamic RCCS Tests

The purpose of the High Pressure Natural Circulation Rig (CAPCN) is mainly to study the thermal-hydraulic dynamic response of CAREM primary loop, including all the coupled phenomena that may be described by one-dimensional models. This includes the validation of the calculation codes and models of the rig, and the extension of validated models to the safety analysis of the CAREM reactor. The main tool used in thermal-hydraulic calculations is the RETRAN-02 computer code.

CHF Tests and Thermal Limits

The TH design of CAREM reactor core was carried out using an improved version of 3-D, two fluid model THERMIT code. In order to take into account the strong coupling of the thermal-hydraulic and neutronic of the core, THERMIT was linked with the neutronic code CITVAP. This coupled model allows the “drawing” of a 3-D map of power and thermal-hydraulic parameters at any stage of the burn-up cycle.

The prediction of the thermal-limits (to phenomenon like critical heat flux) of the fuel elements during operation and transients is considered of the utmost importance. Mass flow rate in the core of the CAREM reactor is rather low compared to typical light water reactors and therefore correlations or experimental data available are not completely reliable in the range of interest. Thus analytical data must be verified by ad-hoc experiments.

The experiments were conducted at the thermal-hydraulic laboratories of the Institute of Physics and Power Engineering (IPPE, Obninsk, Russian Federation). The main goal of the experimental program [41] was to generate a substantial database to develop a prediction methodology for CHF applicable to the CAREM core, covering a wide range of T-H parameters around the point of normal operation,

Hydraulic CRD Tests

One of the most innovative systems behind the CAREM concept is the Hydraulic (in-vessel) Control Rod Drive mechanism. Two designs are under development: “Fast Extinction” and “Adjust & Control” CRD, since the latter presents major challenges related to the design.

The development plan refers to four separate stages and includes the construction of several experimental facilities which will eventually test the system performances under RPV operating conditions. The four different stages and their (built or foreseen) facilities are:

- Preliminary tests (conceptual verification): The aim of this test was to prove the feasibility of the theoretical approach, to have a first idea of some of the most sensitive controlling parameters and to determine points to be reviewed during design. Tests were undertaken on a rough device with promising experimental results, and good agreement with first modeling data was obtained.
- First prototype tests: This stage pointed to determining preliminary operating parameters on a full-scale mechanism as a first approach towards detail engineering. These parameters include a range of flow, ways to produce hydraulic pulses, etc. Manufacturing hints that simplified and reduced costs of the first design were also found. Tests were carried out in a custom built rig and as part of this experimental development it was decided to separate the regulating and fast-drop requirements in different devices.
- Test on a low pressure loop: This stage was carried out with the CRD at atmospheric pressure, and with feed-water temperature regulation up to low sub-cooling. The feed-water pipeline simulated alternative configurations of the piping layout with a second injection line (dummy) to test possible interference of flow fluctuations.

  The ad-hoc test loop was designed to allow automatic control of flow, pressure and temperature, and its instrumentation produces information of operating parameters including pulse shape and timing. The tests included the characterization of the mechanism and the driving water circuit at different operating conditions, and the study of abnormal events such as: increase in drag forces, pump failure, loss of control of water flow or temperature, saturated water injection, suspended particle influence, and pressure "noise" in feeding line.

  The tests carried out at turbulent regime, which are the closest conditions to operation obtained in this loop, showed good reliability and repetitiveness as well as sensitivity margins for the relevant variables within control capabilities of a standard system.
- Qualification Tests: A high-pressure loop (CAPEM, Circuito de Alta Presión para Ensayo de Mecanismos) is being designed in order to reach the actual operating conditions ($P = 12.25$ Mpa, $T \approx 326$ °C). The main objectives are to verify the behaviour of the mechanisms, to tune up the final controlling parameter values and to perform endurance tests. After this stage, the system under abnormal conditions, such as the behaviour during RPV depressurisation, simulated breakage of feeding pipes, etc. will be tested.

RPV Internal Tests

The mechanical structure of the core, supporting guides and all parts of the First Shutdown System are of particular interest. Complex assemblies and structures like the Steam Generator

Safety • Quality • Sustainability • Innovation

Units or ad-hoc mechanical solutions require the evaluation of manufacturing and assembly process, before finishing the design stage.

In sum, internals must be verified in order to define manufacturing, assembling allowances, and other detail engineering parameters to comply with their function during the RPV lifetime. Most tests are performed on mock-up facilities at 1:1 vertical scale. The following is a brief description of some of these devices, experimental plans and current status.

- Full Scale Core-Sector: This is a complete-vertical representation of the core up to an extension of three fuel elements (i.e. structure, upper and lower grids, dummy FE, absorbing element guides, etc.) and major devices involved (i.e. absorbing fuel rods and connecting bar). All of the structure can be perturbed by a hydraulic-driven actuator, which simulates minor vibrations and horizontal seismic loads on a wide range of frequencies and magnitudes.
- Full Scale (Vertical) Structural Barrel, Core and Cinematic Chain: An important series of experiments to verify structural and dynamic behaviour of the Barrel will be performed after finishing those at the core sector. Since this structure is very slender, the experiments deal with alignment, clearances in linear bearings, mass, momentum and dynamic analysis. The latter determines natural frequencies, mode shapes and responses of the system under various external perturbations, which resemble seismic conditions and other vibrations.

6.5 StarCore

As a start-up company, the StarCore's corporate activities seem to be concentrated on business case and design concept developments. There have not been any published R&D activities. However, considering that the reactor is based on the pebble bed modular reactor technology, the R&D data from the high temperature gas cooled reactor developments would be relevant. They include:

- High temperature structural materials: the reactor will be operating at a temperature over 850 ºC. The structural materials will be exposed to high temperature as well as irradiation.
- TRISO safety: the TRISO fuel has been studied in depth since it was first developed in the UK. There is an extensive research data on TRISO behaviour.
- Existing HTR results: HTR-10[42] in China and HTTR in Japan[43] would offer very valuable technical data since StarCore is very similar to these reactors.

[42] High Temperature Reactor, Tsinghua University, http://www.tsinghua.edu.cn/publish/ineten/5696/index.html
[43] High Temperature engineering Test Reactor, Japan Atomic Energy Agency, http://httr.jaea.go.jp/eng/index.html

6.6 NuScale

The design features of NuScale reactor mostly rely on well-established light water reactor technology. However, there are several key research and development activities that must be completed to demonstrate the safety of the technology such as listed below:

- Integral reactor design [42]: NuScale has a comprehensive test program designed to validate its safety analysis and design computer codes and to assess the performance of its safety systems. A unique feature of the NuScale design process is that an integrated system test facility was built early in the concept development phase and has been used throughout the design process to inform design choices. The NuScale Integral System Test (NIST) facility is located at Oregon State University. The NIST is a 1/3rd NuScale height facility, designed to operate at the NuScale coolant pressure and temperature. It models the reactor core (electrically heated rod bundle), the pressurizer, the helical coil steam generator, the passive safety systems and the safety logic. The facility provides integral system data for steady-state operating conditions and transient conditions such as SBLOCA. The data is used for system characterization, safety code validation, informing operational procedures, and safety methodology development. The test facility is required to meet NQA-1 quality assurance and is being used to support NuScale design certification by the NRC.
- Multi-unit cluster control from a single operating room [42]: The control of multiple units from a single operating room was not previously demonstrated and it is outside of the current regulatory practices. Comprehensive human factors engineering and human/system interface studies are underway to determine the optimum number of reactors that can be effectively and safely controlled by a single operator and a control room simulator has been constructed to provide a technical basis for the selected strategy.

  In 2012, NuScale constructed a main control room simulator laboratory. The simulator is being used to support NuScale's NRC design certification effort. The control room laboratory includes 12 individual module stations, each capable of modeling the neutronic and thermal-hydraulic behavior of each module using state-of-the-art computer codes. The entire balance of plant is modeled and coupled to all 12 modules. This unique facility is being used to perform the human factors engineering studies and the human-system interface studies needed to ensure that a multi-module NuScale plant can be operated safely from a single, all digital, state-of-the-art control room. In January 2013, NRC human factors staff conducted a 3 day visit of the NuScale control room simulator.

The R&D activities and the Oregon State University facilities used to validate NuScale design are described in Reference [43].

6.7 mPower

Hatch found that the R&D information for mPower design is very scarce and the only available sources for the information are corporate presentations. The recent presentation at IAEA SMR technical meeting [44] shows the overview of the mPower test program which is reproduced in Figure 26 below.

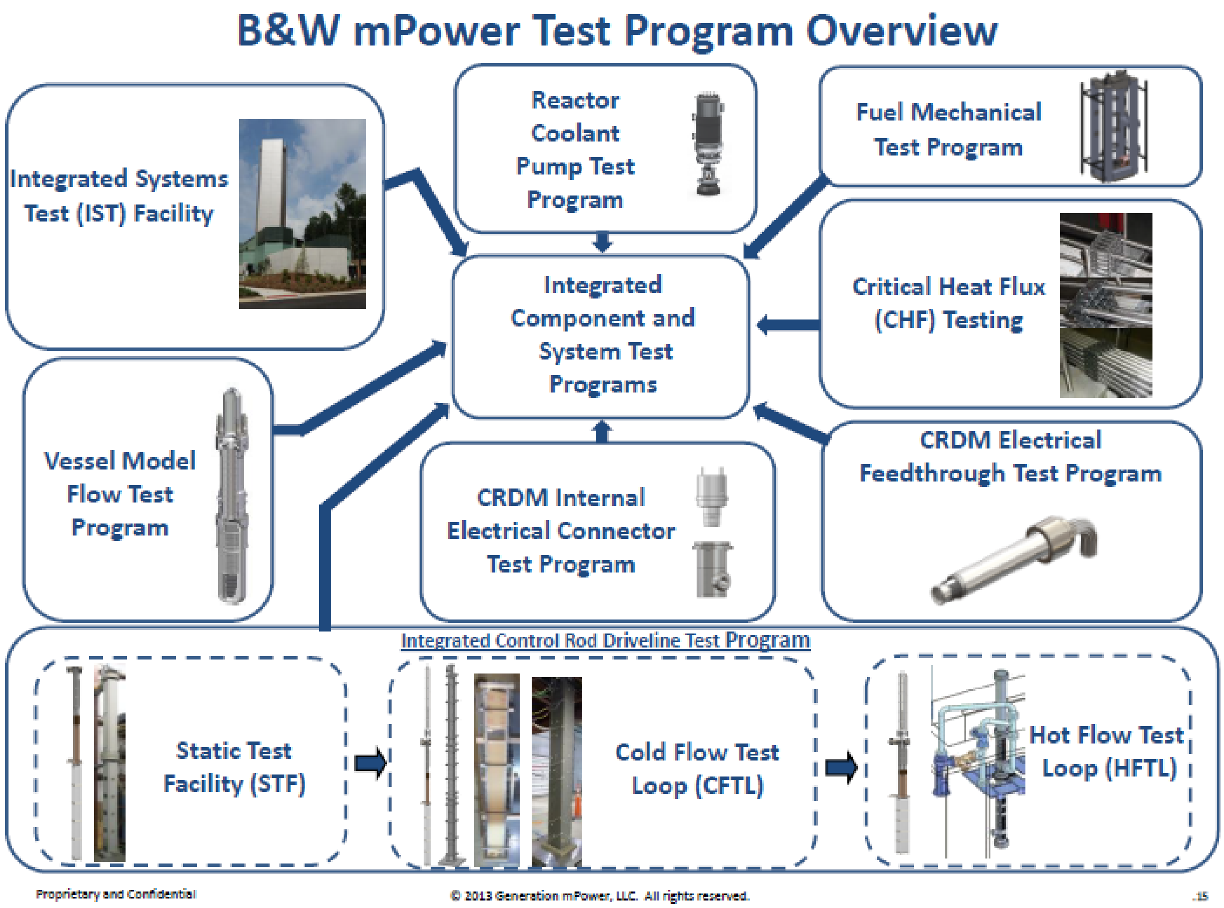

Figure 26 mPower test program overview

The integrated components and system test program consists of:

- Integrated systems test facility
- Reactor coolant pump test program
- Fuel mechanical test program
- Critical heat flux testing
- CRDM electrical feedthrough test program
- CRDM internal electrical connector test program

- Vessel model flow test program

The integrated control rod driveline test program is carried out in 3 stages using:

- Static test facility,
- Cold flow test loop, and
- Hot flow test loop

Also, the company is performing various integrated systems test, including:

- Heat transfer phenomena
- Steam generator performance
- LOCA response
- Pressurizer performance
- Reactor control
- mPower Simulator

There are very limited publications from the company other than several corporate presentation materials. However, Hatch found that mPower has filed extensive number of patents[44] (over 100) related to the reactor design and safety systems.

6.8 Relevant IAEA Documents

Many innovative features of advanced water-cooled SMR and gas-cooled reactors have been discussed in IAEA technical reports. The following list includes the technical reports that may be of interest to the readers in understanding the SMR design features discussed in this report.

- International Atomic Energy Agency, "Studies on fuels with low fission gas release," Proceedings of a Technical Committee meeting held in Moscow, Technical report IAEA-TECDOC-970, 1996.

- International Atomic Energy Agency, "Development status of metallic, dispersion and non-oxide advanced and alternative fuels for power and research reactors," Vienna, Technical report IAEA-TECDOC-1374, 2003.
- International Atomic Energy Agency, "Natural Circulation Phenomena and Modelling for Advanced Water Cooled Reactors," IAEA-TECDOC-1677, Vienna, 2012.

[44] Google Patents search with keyword ‘mPower nuclear Babcock & Wilcox’, *https://www.google.com/?tbm=pts*

- International Atomic Energy Agency, " Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends," IAEA-TECDOC-1451, Vienna, 2005.
- International Atomic Energy Agency, "Advanced Nuclear Power Plant Design Options to Cope with External Events," IAEA-TECDOC-1487, Vienna, 2006.
- IAEA, "Evaluation of Advanced Thermohydraulic System Codes for Design and Safety Aanlysis of Integral Type Reactors," IAEA-TECDOC-1733, Vienna, 2014.
- International Atomic Energy Agency, "Small Reactors without On-site Refuelling: Neutronic Characteristics, Emergency Planning and Development Scenarios," IAEA-TECDOC-1652, Vienna, 2010.
- International Atomic Energy Agency, “Status of Advanced Water Cooled Reactor Designs,” IAEA-TECDOC-1391, Vienna, 2004.
- International Atomic Energy Agency, “Passive Safety Systems in Advanced Water Cooled Reactors,” IAEA-TECDOC-1705, Vienna, 2013.

7. Assessment of Fukushima Lessons Learned

This section of this report addresses the fourth scope of the contract as listed in section 1.3:

4. Discuss the vendors' approaches in enabling their reactor designs to fully resist to the disastrous effects similar to those encountered at the Fukushima Daiichi site.

In case that collected information for Items 3 and 4 is not sufficient to address associated issues, the Contractor should give their own critical view on what else has to be considered in the R&D activities and what measures to be taken in protecting the reactor design from the exceptional geophysical events so that the vendor's reactor design can effectively respond to the anticipated transients and postulated accidents, and in mitigating the consequences of severe accidents.

7.1 KLT-40S

Hatch found that there is very limited information on the preparedness of KLT-40S reactor in Fukushima-type accident. The OKBM corporate presentation at INPRO Dialogue Forum claims the following stress analysis results [32]:

- Seismic impact of magnitude MSK-64 scale 10: no radiological consequences for public and environment
- Tsunami waves, casting the floating power unit ashore: no radiological consequences for public and environment
- Total blackout with all off-site, auxiliary and emergency power sources unavailable: no radiological consequences for public and environment
- Total blackout and core meltdown in the floating power unit reactors:
  - The reactor remains deeply subcritical
  - The leaking reactor coolant and non-condensing gases are localized within the containment
  - No melting through the reactor pressure vessel with heat being removed by the passive system of reactor vessel external heat removal

The assessment of KLT-40S reactor concept with respect to the post-Fukushima lessons learned (Table 1) is provided in Table 16.

Table 16 Fukushima lessons learned against KLT-40S

Key areas	Lessons Learned	Design Assessment
Multiple concurrent external events	Major external events such as earthquake and flood can occur simultaneously. The design shall examine the possibility of multiple events occurring simultaneously (i.e. earthquake, flooding, internal fire, etc).	Insufficient information is available on the current safety assessment for concurrent postulated events.
Extended loss of offsite power	The offsite power can be lost for an extended period due to consequences of a natural disaster that destroys the external grid or power sources. The stored energy (such as batteries) may not be available for sufficient duration while AC power is restored.	Two autonomous passive channels for emergency heat removal for 24 hours without water makeup. The reactor relies on passive methods to remove the decay heat from the core which is significantly lower than the Fukushima Daiichi plants. The placement of the vessel being on water, and high thermal conductivity of the steel vessel would enhance the cooling of the plant in an emergency.
Emergency core cooling and ultimate heat sink	Diverse means to cool the core in absence of AC power and loss of the ultimate heat sink shall be available.	The core is cooled with gravity driven passive emergency cooling water.
Protection of essential equipment	The safety-related systems should be protected from the external hazards and internal events by the separation of redundant equipment, by physical protection or by hardening.	The vessel is compartmentalized to separate the safety related equipment physically.
Spent fuel pool integrity and cooling	The spent fuel pool make up capability and instrumentation to monitor the condition of the pool are required.	There is an active and a passive means to cool the spent fuel pool. The small amount of spent fuel in the storage pool makes the cooling easier than in conventional plants.
Containment integrity	Containment integrity shall be protected (i.e. the plant shall have means to mitigate hydrogen explosion and over-pressurization)	Passive emergency containment pressure reduction system is incorporated.

Multi-unit event	In a multi-unit plant, an accident occurring to a unit shall not impact the safety of other units.	Insufficient information available.

7.2 VBER-300

Hatch was not able to find any vendor discussion on how VBER-300 addresses the Fukushima lessons learned. However, VBER-300 plant is based on the conventional VVER NPP with an exception of the RP which is adapted from marine propulsion reactors. The Russian regulator, Rostechnadzor, published a list of post-Fukushima action items[45] that are applicable to VVER-440 and VVER-1000 NPPs. These action items are applicable to VBER-300 NPP.

7.3 SMART

According to the Korean regulator presentation at INPRO [45], the regulatory authority in Korea performed detailed safety inspections on operating plants and 50 safety action items were derived to enhance the safety following the Fukushima accident. The incorporations of them were recommended for SMART. Out of 50, 34 items were applicable to the SMART licensing: 10 items for standard design, 8 items for construction, and 16 items for operation. KAERI implemented the Fukushima in the standard design as follows:

- Adding automatic seismic trip system at earthquake > 0.18g
- Strengthening the seismic design for MCR panel
- Providing watertight door & drain pump
- Securing mobile generator & battery
- Preparation of measure to cool-down spent fuel pool
- Providing external safety injection flow path
- Providing passive autocatalytic hydrogen recombiner

KAERI also performed assessment of SMART behaviour in Fukushima-type accidents, such as:

- External injection into RCS [46]
- SMART station blackout analysis [47]

7.4 CAREM

The latest project status update from CNEA [48] shows that the following post-Fukushima actions were taken for CAREM-25 design:

[45] Federal Environmental, Industrial and Nuclear Supervision Service of Russia, *http://en.gosnadzor.ru/international/Post-Fukushima/Rosenergoatom/WWER/*

- Seismic design basis review
- Loss of heat sink and station blackout
  - Provisions to use fire extinguishing system or an autonomous system to remove core decay heat and to provide containment cooling after the grace period
  - Provisions to supply electricity to safety related systems using autonomous generation systems.

In general, the details of post-Fukushima lessons implementation for CAREM reactors are seldom found in the public domain. The assessment of CAREM reactor concept with respect to the post-Fukushima lessons learned is provided in Table 17.

Table 17 Fukushima Lessons Learned against CAREM Reactor

Key areas	Lessons Learned	Design Assessment
Multiple concurrent external events	Major external events such as earthquake and flood can occur simultaneously. The design shall examine the possibility of multiple events occurring simultaneously (i.e. earthquake, flooding, internal fire, etc.).	Seismic and tsunami type events are not expected to cause serious consequences as the reactor will shut down when the external power is interrupted and it is passively cooled.
Extended loss of offsite power	The offsite power can be lost for an extended period due to consequences of a natural disaster that destroys the external grid or power sources. The stored energy (such as batteries) may not be available for sufficient duration while AC power is restored.	The core is passively cooled, thus, the reactor will remain safe in case of an extended power loss with a grace period of 36 hours.
Emergency core cooling and ultimate heat sink	Diverse means to cool the core in absence of AC power and loss of the ultimate heat sink shall be available.	Emergency core cooling is maintained passively.
Protection of essential equipment	The safety-related systems should be protected from the external hazards and internal events by the separation of redundant equipment, by physical protection or by hardening.	The safety equipment is passively operated and the control and shut off rods are placed in the pressure vessel (i.e. physically protected from an external event)

Spent fuel pool integrity and cooling	The spent fuel pool make up capability and instrumentation to monitor the condition of the pool are required.	The spent fuel pool monitoring and cooling capability in the design basis accident will have to be incorporated.
Containment integrity	Containment integrity shall be protected (i.e. the plant shall have means to mitigate hydrogen explosion and over-pressurization)	The means to protect the containment integrity will need to be incorporated.
Multi-unit event	In a multi-unit plant, an accident occurring to a unit shall not impact the safety of other units.	The CAREM reactor is designed as a standalone unit at this point.

7.5 StarCore

The Fukushima-type event does not impact a gas-cooled reactor safety in the same manner that impacts a water cooled reactor safety. Gas cooled reactors are subject to different kind of challenges. The inherent safety features of the StarCore reactor is based on the strong negative temperature and power coefficients of the fuel and high temperature resistance of TRISO particles. The basic safety concept relies on the fact that the reactor core has much lower power density in a gas-cooled reactor than a conventional power reactor. When a postulated initiating event occurs, and the reactor temperature increases, the reactor output power reduces significantly, allowing natural convection air cooling. The high thermal conductivity and high temperature resistance of the fuel enables the reactor to be maintained in hot shutdown state indefinitely.

The assessment of StarCore reactor with respect to the Fukushima lessons learned criteria are provided in Table 18 below.

Table 18 Fukushima lessons learned against StarCore Reactor

Key areas	Lessons Learned	Design Assessment
Multiple concurrent external events	Major external events such as earthquake and flood can occur simultaneously. The design shall examine the possibility of multiple events occurring simultaneously (i.e. earthquake, flooding, internal fire, etc.).	The safety analysis has to be performed for the reactor to identify potential risks and the plant response.

Extended loss of offsite power	The offsite power can be lost for extended period of time due to consequences of a natural disaster that destroys the external grid or power sources.	The reactor can be passively cooled with natural convection air flow provided that air flow path can be provided.
Emergency core cooling and ultimate heat sink	Diverse means to cool the core in absence of AC power and loss of the ultimate heat sink shall be available.	The core can be passively cooled with natural convection air flow.
Protection of essential equipment	The safety-related systems should be protected from the external hazards and internal events by the separation of redundant equipment, by physical protection or by hardening.	The reactor is physically separated from potential external hazards (i.e. 57 m below the grade)
Spent fuel pool integrity and cooling	The spent fuel pool make up capability and instrumentation to monitor the condition of the pool are required.	There is no spent fuel on site
Containment integrity	Containment integrity shall be protected (i.e. the plant shall have means to mitigate hydrogen explosion and over-pressurization)	Since there is no coolant phase transition, the containment is not subject to extreme pressure increase.
Multi-unit event	In a multi-unit plant, an accident occurring to a unit shall not impact the safety of other units.	The 2 cores are physically separated by placing them in separate silos with independent safety systems. However, the system could be vulnerable to a common cause event such as an earthquake.

7.6 NuScale

NuScale Power LLC claims that NuScale reactor is generally immune to Fukushima-like accidents due to its different design from the conventional water cooled reactors. Table 19 is a reproduction of NuScale corporate presentation [49] that shows the comparison of NuScale and Fukushima-type plants.

Table 19 Comparison of NuScale to Fukushima-Type Plant

Fukushima	NuScale
Reactor containment	

Safety Emergency Diesel Generators Required	Safety Emergency Diesel Generators Not Required
External Supply of Water Required	Containment immersed in 30 day supply of water
Coolant Supply Pumps Required	Not Required
Forced flow of water required for long term cooling	Long term (Beyond 30 days) cooling by natural convection to air
Spent Fuel Pool	
Water Cooling of Spent Fuel	Extended Cooling Capability *4 times the water of conventional spent fuel pools per MW power*
Elevated Spent Fuel Pool	Deeply Embedded Spent Fuel Pool
Limited Access to Back-up Supply of Water	Accessible Back-up Supplies of Water

The company presented that no major impact to the NuScale design is anticipated following the Fukushima accident since NuScale design fully addresses decay heat removal for prolonged station blackout [49]. However, the following activities are planned to ensure the design safety:

- Add long term air-cooling test to NuScale Integral System Test Matrix and SIET decay heat removal tests to demonstrate effectiveness of passive air-cooling with an empty reactor building pool.
- Review Spent Fuel Pool Cooling capability under air-cooled conditions.
- Examine role of “Island Mode” operation for multi-module plant.
- Confirm adequacy of existing seismic design basis for NuScale (0.5g zero period acceleration) and ensure efforts are consistent with ongoing industry efforts.
- Review NRC recommendations when they become available and determine applicability to NuScale.

7.7 mPower

The mPower Generation claims that the reactor has various protection mechanisms against Fukushima-type events via corporate presentations [50] and [51]. The design features that provide the protection are shown in Table 20.

Table 20 mPower defence features against Fukushima type accident

Events and Threats	mPower Design Features
Earthquakes and Floods	• Seismic attenuation: deeply embedded reactor building dissipates energy, limits motion • Water-tightness: separated, waterproof reactor compartments address external flooding risks.
Loss of Offsite Power	• Passive safety: AC power, offsite or onsite, not required for design basis safety functions • Defense-in-depth: 2 back-up 2.75 MWe diesel generators for grid-independent AC power
Station Blackout	• 3-day batteries: safety related DC power supports all accident mitigations for 72 hours • APU back-up: auxiliary power units inside reactor buildings recharge battery systems • Long-duration “station keeping”: 7+ day battery supply for plant monitoring/control
Emergency Core Cooling	• Gravity driven, not pumps: natural circulation decay heat removal; water source in containment • Robust margins: core power density (11.5 kW/m) and small core (530 MWth) limit energy • Slow accidents: maximum break size is small compared to reactor inventory ($4.7X10^{-5}$ m^2/m^3)
Containment Integrity and Ultimate Heat Sink	• Passive hydrogen recombiners: prevention of explosions without need for power supply • Internal cooling source: ultimate heat sink inside underground shielded reactor building • Extended performance window: up to 14 days without need for external intervention
Spent Fuel Pool Integrity and Cooling	• Protected structure: underground, inside auxiliary containment, located on basemat • Large heat sink: 30+ days before boiling and uncovering of fuel with 40 years of spent fuel

8. Small Reactor Licensing Challenges

This section of this report addresses the fifth project scope listed in Section 1.3, which says:

> *5. Discuss any identified potential licensing challenges that could emerge when licensing the vendors' small reactor designs in Canada."*

Most small reactor designs, either implicitly or explicitly, challenge some of the licensing requirements for conventional power reactors. The innovative SMRs use the inherent or passive safety characteristics described in section 5 to make the case that some conventional power reactor requirements are irrelevant, inapplicable, or could be relaxed while still meeting the safety intent.

This section describes some of the major challenges in SMR licensing under the current regulatory framework. The discussion is based on the common elements across most of the small reactor designs; the list of challenges may not apply in entirety to any one particular SMR design.

The challenges are sometimes phrased as issues or questions, reflecting the lack of regulatory experience. None of these challenges are insurmountable but they are called challenges because there is no framework to consistently address them.

8.1 Need for a Prototype

The section 6 of this report has summarized the R&D that supports each design. In most cases the core seems relatively conventional, notwithstanding the lack of detailed design information in some cases. If for some reason the *first* SMR of a specific design is licensed in Canada, the need for a prototype reactor must be addressed. The following questions should be addressed:

- Is the SMR design sufficiently removed from mainstream experience that a prototype is needed?
- If not, should there be constraints on operation, or requirements for additional instrumentation and tests for the First-Of-A-Kind (FOAK) of an SMR, to offset lack of operating experience?

With a power reactor, a requirement for a prototype can be an economic show-stopper but this may not be as serious a market barrier for the first SMR of a large number. However, to be fair to a licensing applicant, the regulatory decision-making framework needs to be defined beforehand.

8.2 Integral Vessel Designs

All the PWR-based SMRs discussed in this report, except for the KLT-40S and the VBER-300, have truly integral designs as described in Table 10 (section 5). The two exceptions have very short large-diameter pipes connecting the pumps and steam generators to the reactor pressure vessel (RPV). In both cases, the design concepts are then used to exclude LBLOCA from DBA-space. While such an argument has been made in special cases previously with the CNSC for existing plants (e.g. for pipe-whip protection, and to determine more realistic LBLOCA margins), it has not been used on a new design. It raises a number of important issues:

- The two concepts – integral reactor and short lengths of large-bore RCS piping – are qualitatively different. For the first, a large pipe break is claimed to be ruled out inherently since there are no large bore RCS pipes. For the second, the large pipe break probability is just lower than conventional PWR LBLOCA (if the case is made that pipe break probability is proportional to pipe length).
- In integral designs, there are pressure-vessel penetrations to let the (secondary) steam out. It would have to be shown that failure of the nozzle is incredible, or does not lead to a primary LOCA as well as a secondary LOCA.
- Elimination of LBLOCA from DBA-space leaves the design basis of the ECCS system and the containment system ill-defined. Certainly small LOCA is one of the design bases. However, the required level of robustness of both safety systems to mitigate larger LOCAs needs to be defined; i.e. is LBLOCA to be treated as a Design Extension Condition (DEC), BDBA beyond DEC, or considered “practically eliminated”? A related question is the size of the largest credible leak in a pressure-vessel, since (depending on the ECCS and containment design basis) pressure vessel integrity may become a single line of defence against LBLOCA.

8.3 Operating Organization

To date in Canada, the operating organization of power reactors has been a large electrical utility, and they were also the sole owners in most cases. Under the Nuclear Safety and Control Act (NSCA) [52] one of the requirements for a licence is that the licensee, in the opinion of the Commission:

“(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on;…”, etc.

The electrical utilities in Canada have the resources to be an “intelligent customer” and to a large extent have become the design authority, with some elements of the core design delegated to AECL. The intelligent customer concept (although not stated in those terms in Canada) is relevant here: it was developed by the UK Office for Nuclear Regulation [53] and is defined as:

“As an intelligent customer, in the context of nuclear safety, the management of the facility should know what is required, should fully understand the need for a contractor's services, should specify requirements, should supervise the work and should technically review the output before, during and after implementation. The concept of intelligent customer relates to the attributes of an organisation rather than the capabilities of individual post holders”.

While this was not an issue for large electrical utilities, it may become so for SMRs; the owner and the operator may not be the same and the former may be small disparate organizations whose main business is not nuclear generation. In Canada, the operator has primarily responsibility for safety but the owner must provide resources. The regulator will need to be satisfied that adequate resources for safe operation are indeed provided. The issue is certainly solvable (e.g. by having one experienced operating organization for all SMRs of a certain type across the country) but any solution will need a careful review.

8.4 Siting

Unlike new power reactors, SMRs would normally be located very close to the load centre. Most SMR vendors claim or imply that a small or non-existent exclusion area beyond the plant boundary is adequate, with either a small Emergency Planning Zone (EPZ), or none at all outside the facility.

The size of the exclusion area for nuclear power plants in Canada is *not* prescribed by regulation; it has traditionally been 914 m radius on the land side of the plant. The Emergency Planning Primary Zone for a power reactor such as Pickering is about 10 km in radius [54]. By contrast, for the 3 MWth McMaster Nuclear Reactor, the campus population can routinely walk next to the outer surface of the concrete containment, and the emergency planning zone is about 100 metres in radius[46], adjusted upwards where necessary so as include complete buildings.

For SMRs, the zones would be based on risk. For example the USNRC [55] is developing a dose/distance approach for SMRs, in which the EPZ *"could be scaled to be commensurate with the accident source term, fission product release, and associated dose characteristics for the designs."* One would also have to consider the number of units at a site, and the level of confidence that passive safety systems can rule out significant releases.

Two SMR designs examined in this report are ship-based (e.g. KLT-40S and VBER-300). They have unique characteristics and unique accidents (e.g. ship sinking, collisions with other ships, grounding, security, interactions with severe weather etc.). To the author's knowledge Canada has no framework to deal with a civilian ship-based nuclear reactor continuously operating in a port – precedents are either military vessels (with inaccessible information) or the cancelled Polar 8 nuclear ice-breaker which would probably not have run its reactor(s) in port.

8.5 Multiple Units

Some SMRs envisage multiple units on the same site; for example, NuScale may have up to 12 modules generating 45 MWe each in the same overall station. Canada has experience with multi-unit stations, but the larger number of modules on the same site raises several questions and issues as follows:

[46] "McMaster Emergency Guidebook", http://security.mcmaster.ca/campus_emergencies_guide.html

- Does a station consisting, for example, of 12 x 45 MWe modules have a significantly different risk profile from a conventional CANDU 6, since they have similar total fission powers? While each SMR module may have a low risk from internal events due to its inherent / passive safety characteristics, external events such as earthquakes, malevolent acts, and extreme weather could impact all modules.
- Should each module have its own licence, or should all 12 modules share a common licence? Should the numerical safety goals in e.g. [4] be applied per-module or for the set of all twelve modules?
- Many multi-unit designs may share safety-related Systems, Structures and Components (SSCs) – even if they do not share safety systems (e.g. the turbine hall, the reactor buildings, the containment heat sink, a common control room, common cooling towers, and for vessel-based SMRs, possibly the same ship). Such practice runs counter to the prevailing safety philosophy on power reactors.
- Some multi-unit designs have a single operator controlling more than one unit / module. One would have to review the implications of accidents simultaneously occurring in more than one unit (e.g. due to external events) – or an accident in one unit when the other unit(s) under the operator's supervision are running. One design (StarCore) is supposed to be unattended and controlled by satellite, although there is no public information on how this will be achieved safely.

8.6 Security

Conventional nuclear power plants must withstand a spectrum of severe external events, one of which is aircraft crash. For a new nuclear power plant, a deliberate crash of a large commercial aircraft onto the plant is part of the design basis. Most of the SMRs, however, do not seem to consider deliberate large aircraft crash in the design: for example the KLT-40S withstands the crash of an aircraft with the mass of 10 t from the height of 50 m – e.g. a helicopter or a Cessna. SMRs presumably discount a deliberate large aircraft crash based on small site footprint (i.e. the probability of aircraft striking the facility is lower) and the placement of much of the SMR underground. It is not clear the extent to which a *nearby* large aircraft crash is considered (i.e. the effect being shock waves transmitted through the ground), the effect of large amounts of fuel spilling and igniting, etc. One of the potential disadvantages of SMRs in this case is the close proximity of the SSCs, both of a single unit and of multiple units.

Conventional security *may* be an issue but there is not enough publicly available information to comment. One issue, if there is one, is not technical but economic; one would expect a SMR operator to argue that in view of the inherent / passive safety characteristics, a smaller security force per MW can be justified than for a conventional NPP. The Threat-Risk-Assessment could be used to justify a smaller security force.

Some of the requirements of the Nuclear Security Regulations [56] may not work well with very small sites: e.g. 5 metre gap between the two fences enclosing the protected area (although an equivalent structure can be used instead of a double fence); a 5 metre unobstructed gap on either side of the double fence or structure; 5 metres between the inner barrier of the protected area and the barrier enclosing the inner area. This comment is somewhat speculative as scale drawings of the site layouts for the various designs are not available.

One of the SMRs examined in this report, StarCore, is designed for unmanned operation – it will be difficult to justify not having any security presence on site.

8.7 Safeguards

In principle the safeguards arrangements will be defined by the IAEA, with the CNSC acting as a consultant. None of the SMR reactors described in this report are refuelled at power, so PWR-type safeguards will in general apply. However, some designs remove / replace / transport the entire core, which will no doubt require special techniques to verify what went in and what is removed from the core.

In addition, KLT-40S uses 15 ~ 19% enriched fuel and StarCore, 19%. Low enriched uranium with less than 20% enrichment is permitted for civilian purposes but the proposed fuels have much higher enrichment than those used in most operating power reactors. Some SMR vendors propose to take back the used core and replace it with a new one – if the vendor is not in Canada, then such operation will require import and export licences and a Nuclear Cooperation Agreement with the vendor's country.

There may be some technical challenges with safeguards for SMRs, as outlined in [57], e.g., remote location with limited access, long-life sealed core, high initial excess reactivity, etc. Some of these challenges are also potential benefits; i.e., a remote location makes it more difficult for the IAEA to inspect, but also more difficult for diversion; the same is true of a sealed long-life core.

8.8 Nuclear Liability

Current nuclear liability in Canada for an operator is limited to a maximum of $75 Million. The proposed Energy Safety and Security Act (Bill C-22) (Part 2 - Nuclear Liability and Compensation Act) raises liability for nuclear operators to a maximum of $1 Billion [14]. An SMR operator may argue that the risk posed by his facility is much smaller than that of a conventional nuclear power reactor, and therefore that his limit of liability should be lower. This is provided for in Bill C-22, Section 24:

"(2) The Governor in Council may, by regulation,

(a) amend subsection (1) to increase any amount of liability; or

(b) reduce the amount of liability applicable to an operator of a nuclear installation, or operators of a class of nuclear installations, having regard to the nature of the installation and the nuclear material contained in it."

Possibly this should be done as a class (SMRs), with the CNSC given discretion to set a smaller limit within a range. However, the issue of insurance for multi-unit sites with many SMRs needs resolution, for the same reasons as discussed in Section 8.5 – i.e., should the insurance be based per module or on the total thermal power of the site?

8.9 Fuelling and Transport

For some SMR designs, the vendor (not the operator) performs the defueling and refuelling, sometimes removing the entire reactor core in its vessel [3]. One of the generic Licence Conditions for operating nuclear power plants in Canada states that:

"Prior to loading any fuel bundle or fuel assembly into a reactor, the licensee shall obtain written approval of the Commission, or of a person authorized by the Commission, for the use of the design of that bundle or assembly."

CNSC requires any fuel design to be "approved" [58]. Typically, an approval includes a review of the qualification of the fuel design, including tests. For SMRs, CNSC approval may need to include inspection of the vendor's facilities in another country.

Furthermore, in addition to the safeguards issues discussed in Section 8.7, the transportation of new and used cores must be considered. While Canada has Certified Packages for used nuclear fuel, the packages for new and used reactor cores would likely be developed by the vendor, approved in his country, and then re-validated in Canada. Transport is covered by requirements in [59] which references [60]; a cursory review of these documents does not indicate any show-stoppers for SMRs, although the author is not an expert in the area of nuclear materials transportation; in any case, it is up to IAEA to comment on the applicability of their own document to SMR core shipments. Since the used and new cores both contain fissile material, prevention of both tampering / diversion during transit and criticality will be areas of detailed review.

8.10 Decommissioning

The requirements for decommissioning are covered in CNSC documents G-219 [13] (decommissioning planning) and G-206 [12] (financial guarantees). G-219 already allows scaling flexibility so that there should not be an issue for SMRs decommissioning. Similarly, G-206 is scale-independent. Thus, the author concludes that the existing decommissioning documents adequate serve SMRs.

8.11 Regulatory Review Process

Current regulatory review fees are $225 / hour, which would apply to SMRs [11]. The cost of the first review, environmental assessment and licensing of an SMR will be substantial, and will likely *not* scale with thermal power. However, the economics of SMRs requires that many units be sold, and if this cost is repeated for every site, it could pose a barrier to the technology development. The CNSC already deals with multi-unit sites in Ontario, and with repeat designs at different sites (e.g., Point Lepreau and Gentilly-2). The following further suggestions would allow an efficient licensing review for SMRs where multiple units at different sites are planned:

- A generic design review (e.g., similar to the US Standard Design Certification or the UK Generic Design Approval) which would be done only once for each design. It could be an extension of the current CNSC process for pre-project review of vendor designs, except it would be binding as a basis for the construction licence.
- A generic Environmental Assessment (EA). While an EA would still have to be adapted to a specific site, it could be designed to envelope most potential *sites* so that most technical issues would have to be dealt with only once. This differs from defining an EA envelope for a specific site, allowing a range of *designs* to be sited there. Of course the two concepts could be combined, so that a “generic” EA would cover a range of sites and a range of SMR designs, leaving only minor customization for a specific site.
- A combined construction / operating licence. This is already (i.e., effectively) permitted in Canada but has not been used to date. An extension for the *n*th of a kind would be combined site, construction and operating licenses.
- Combined public hearings for a fleet of SMRs of similar design.
- Splitting a plant licence into certification of a module and a Master Facility Licence that would cover project-specific aspects common to the whole plant, an idea proposed in [61].

8.12 Experience with the Technology

The CNSC is responsible for licensing review in Canada. While all licensed power reactors have been CANDUs with an exception of Gentilly-1, the CNSC and its predecessor, the Atomic Energy Control Board, has also reviewed, or licensed, and in some cases continues to license, a number of different technologies: e.g. the research reactors such as NRX, NRU, ZED-2, McMaster Nuclear Reactor, and SLOWPOKE-2; the SDR heating reactor at Whiteshell; pre-licensing reviews of the SES-10 heating reactor; and an operating licence for the MMIR isotope production reactor. In addition, CNSC has performed Phases 1 and 2 Pre-Project Design Review of the Westinghouse AP1000 reactor design.

Of the sample SMRs in this report, all but one design are PWRs with which the CNSC already has some familiarity. However, one of the candidate designs (StarCore) is gas-cooled reactor and uses TRISO fuel. CNSC has no experience in this type of reactor licensing; i.e., CNSC's requirements and regulatory philosophy are historically based on water-cooled reactors.

It is assumed that if an SMR has been, or is being licensed in other jurisdictions, then CNSC will set up the appropriate regulatory cooperation agreements with the vendor's domestic regulator in order to fast-track its familiarization with the technology. This assumes that the regulator of the SMR developer's country is competent and independent of the vendor. CNSC will also likely require access to independent review resources, and may have to go outside of Canada to find them. Moreover, it has been proposed [62] that IAEA revise its Safety Standards to cover SMRs, and an SMR Regulator's Forum be set up. So there are a number of venues for CNSC to access outside experience in the process of granting SMR licences.

A more pressing issue is the possibility of a vendor requesting a pre-project design review before CNSC becomes fully expert in the design, especially if the request is for a FOAK. The concern would be that existing regulatory requirements on design might not apply well (as discussed here and in section 9); thus, the review would be judgement-based and the judgement may change as CNSC becomes more expert in the technology later on. Therefore, it is important for the CNSC to set down in writing some SMR requirements early on, at least at a high level.

8.13 Codes and Standards

The CNSC has historically required compliance with applicable Canadian Codes and Standards for nuclear power plants (e.g. CSA N- and Z-series), relevant ASME and IEEE Standards. Alternatively, an applicant can show that the applicant's design uses codes and standards that are equivalent in intent. The issue with SMRs is that the vendor may claim that some of the codes and standards do not apply because of the passive / inherent design characteristics. There is little detailed public information on this topic, although it seems to be an open issue – from [62], K. Ennis, Director for ASME Nuclear Codes & Standards, noted that *"…the SMR industry has not yet engaged in any significant depth with codes and standards organizations to address specific changes necessary for deployment of SMRs…"* and *"The diverse types of SMRs currently being promoted also indicate that in some cases, new codes and standards will need to be developed and supported by appropriate R&D."*

8.14 Defence in Depth

A common theme of many of the issues identified above has been the request for a change in traditional safety design practices, to account for the safety benefits of the items listed in Section 5. This could be portrayed as a re-characterization of defence-in-depth. (Note: it is assumed the reader is familiar with levels of defence-in-depth – see APPENDIX B for a summary.) For example, *if* a large RCS pipe break is not possible because of the absence of large RCS pipes, *and* spontaneous pressure-vessel failure (e.g. at a nozzle) is practically eliminated, *and* there are no accidents such as un-terminated loss of reactivity control or loss of heat sink which would induce vessel failure, *then* the ECCS *and* the containment need not be designed for LBLOCA – i.e. Level 3 defence-in-depth is downgraded for LBLOCA. Oversimplifying a bit, advanced

SMRs typically put much more emphasis on Levels 1 and 2 of defence-in-depth in the design than do conventional power reactors, and require less on Levels 3, 4 and 5. The concept is illustrated in Figure 27. The reader shall note that the case made by SMRs is not that Level 5 is weak, but that it is made unnecessary by Levels 1 to 4.

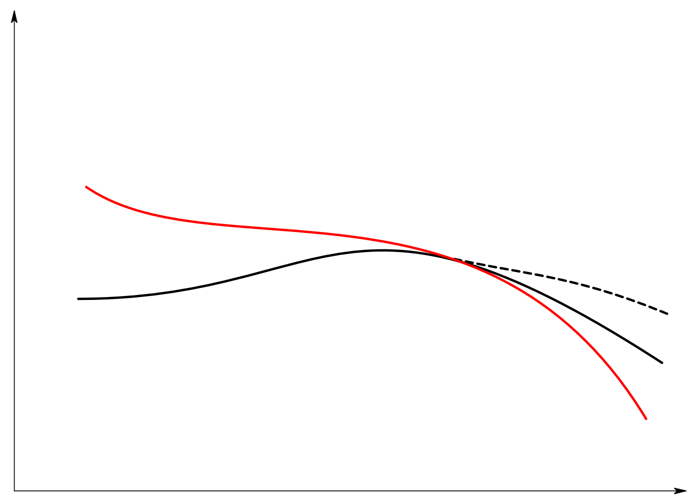

Figure 27 Defence in Depth Concept for SMRs

The CNSC's challenge is that there is no formalized way of addressing these tradeoffs in the regulations for conventional power reactors. Reference [24] is a useful summary of the various approaches for a number of specific SMR designs, but does not attempt to provide an overall decision-making framework. However, the Objective Trees presented in Appendix II of Reference [63], which gives a framework for evaluating defence-in-depth for conventional power reactors, could be adapted for SMRs with some effort.

8.15 Issues Raised by Other Regulators

For the reader's interest, the *policy* issues raised by the USNRC [55] are listed below, as it anticipates design certification or licensing of SMRs. "Policy issues" means that the Commission proper must approve the resolution. They are:

- Implementation of the Defence-In-Depth Philosophy for Advanced Reactors
- Appropriate Source Term, Dose Calculations, and Siting for SMRs
- Appropriate Requirements for Operator Staffing for Small or Multi-Module Facilities
- Security and Safeguards Requirements for SMRs

The policy issues will not necessarily be the same as those discussed above for Canada, due to the different regulatory structure, but there is an overlap.

9. Applicability of draft REGDOC 2.5.2 to SMRs

This section of this report addresses the second project scope listed in Section 1.3, which says:

> *2. Examine the CNSC regulatory framework applicable to small reactors and determine its sufficiency to address safety aspects of new small reactor designs and their claimed safety features;*

The existing CNSC regulatory document (RD-367) on small reactor facilities [5] was reviewed briefly. It is aimed at quite small reactors (e.g., 200 MWth or less). It is a high-level regulatory document, and while it does not preclude advanced designs, it does not provide much guidance to designers or to the CNSC as to how to meet the requirements. Since the author understands that REGDOC-2.5.3 will replace RD-367, the review of RD-367 was not performed in detail.

CNSC document RD-337 "Design of New Nuclear Power Plants" [10] currently sets the requirements for the design of new nuclear power plants in Canada. RD-337 is being replaced by REGDOC-2.5.2, "Physical Design: Design of Reactor Facilities: Nuclear Power Plants" [4]. At the time, a draft revision of REGDOC-2.5.2 was reviewed to see how its specific requirements might apply, or not, to SMRs with respect to the issues discussed in Section 8. The intent of the review was to capture major issues, but *not* to review every phrase.

Overall, REGDOC-2.5.2 is a very comprehensive successor to RD-337. The scope applies well to CANDUs and to mainstream PWRs. It should also apply more or less directly to the larger, more conventional "small" reactors in this report, as discussed in Section 1.4, and with the exceptions noted there.

Much of REGDOC-2.5.2 also applies to the innovative or advanced SMRs. However, it is not risk-informed when applied to innovative SMRs – i.e., it gives no criteria on how to balance conventional requirements against passive/inherent safety characteristics, small fission product inventory, etc. As noted already, the REGDOC applies only to water-cooled NPPs – but one small reactor reviewed in this report (StarCore) is a gas-cooled reactor. The document does allow for a case-by-case review of non-water-cooled reactor designs, but there is really not enough guidance provided to CNSC Staff as to how to perform such a review, or to a vendor as to what requirements should be met.

Possible issues in applying REGDOC-2.5.2 to innovative SMRs are identified in Table 21. Since many of the issues have been discussed in section 8, the sections in REGDOC-2.5.2 are cross-referenced to the subsections in section 8 where issues were discussed.

Table 21 Potential issues in applying REGDOC-2.5.2 to innovative SMR licensing

DRAFT REGDOC-2.5.2 [4] Section	Issue	XRef. to Sec. 8 of this report	Comment
2	Reactor types	8.12	The REGDOC applies only to water-cooled reactors. One reactor in the sample is gas-cooled. Liquid-metal cooled SMRs have also been proposed [6]
3	Owner / operator model	8.3	The owner / operator model may be very different for SMRs than large power reactors. The level of knowledge of the owner, and his duties to the operator, need careful consideration.
4.2.1	Dose limits	8.5	The dose limits are the same as for a large power reactor. Since the benefits of an SMR are smaller than for a large power reactor, should the dose limits be reduced?[47]
4.2.2	Numerical safety goals	8.5	The numerical safety goals are the same as for a large power reactor. Since the benefits of an SMR are smaller than for a large power reactor, should the numerical safety goals be reduced? Do they apply per reactor or per site? REGDOC-2.5.2 Section 6.6.1 deals just with common-cause failures but even without those, the risk increases arithmetically with number of units.

[47] It was not in the scope of this report to propose regulatory positions. However, one might envisage the regulatory position to be developed using the following considerations:

- Dose limits are set to protect the public and the environment against harm from radiation. Therefore, they should not be affected by the activity, and in particular the reactor size
- There is however greater opportunity for dose reduction in a small reactor (versus a large one), especially if the small reactor has the enhanced safety characteristics discussed earlier. Hence application of the ALARA principle would lead to smaller dose targets than for a power reactor and hence lower actual public doses
- Offsetting this argument is the lack of evidence of any harmful effect of radiation at low doses, and the possibility of a health benefit.

4.2.4, 4.3.1, 6.1, 6.5	Defence in depth	8.14	The defence-in-depth concept may need review for SMRs, especially for Level 5. SMRs typically emphasize Levels 1 and 2 much more than conventional power reactors, and may claim that Level 5 is intrinsically addressed by Levels 1 to 4. How much weight should be given to passive features in determining reliability and effectiveness with respect to defence-in-depth? Under what conditions, can off-site emergency response be accepted as unnecessary? Can an SMR have an exclusion area at the "normal" plant boundary? REGDOC-2.5.2 Section 6.5 is flexible but does not really address plant safety characteristics.
5, 5.1	Design authority	8.3	The SMRs may not be owned or operated by large utilities. How does one ensure a knowledgeable operator? Who has real design authority? It may not be realistic for a local operator to have a deep understanding of the plant design. The design authority may be a vendor located outside of Canada. If the core is removed and replaced in entirety, what is the responsibility of the local operator in validating the change?
5.1, 5.2, 5.3	Review of design	8.3	If a design developed elsewhere is to be licensed in Canada, how will CNSC audit the design process after the fact, especially if done under a regulator that is not as recognized as the US NRC?
5.4, 5.5, 8.1.2	Need for a prototype	8.1	What (if anything) would trigger the need for a prototype in a case where the first plant is to be built in Canada? e.g., substantially novel core design, unique dependence on inherent reactivity characteristics.
5.4	Codes and standards	8.13	Most SMRs will not have been designed to Canadian Codes and Standards and in any case the vendor may argue that some power reactor standards do not fully apply.
7.3.3	Design Basis Accidents	8.2	If an SMR rules out LBLOCA, either inherently (no large RCS pipes) and / or on the basis of low frequency, what constitutes a robust case, especially for a new reactor design with little operating experience? Should LBLOCA be eliminated in entirety or remain part of DEC? What is the design basis of the safety systems?

7.3.4	Severe Accidents / Defence in Depth	8.14	A truly passive small reactor may not need any Complementary Design Features, although they are prescribed here. Moreover, a case may be advanced that a passive system, if sufficiently reliable, can provide both preventative and mitigating functions.
7.6	Safety system reliability	8.14	The appropriate safety system reliability is highly design-dependent. For example, it could depend on how many lines of defence there are for a given safety function. SMRs may ascribe a very high reliability to a passive safety function, but it may also be hard to prove – i.e. confidence in the predicted reliability is as important as the number. How would one ascribe the degree of reliability to natural convection cooling in CAREM or NuScale?
7.6.5.2	Sharing of SSCs	8.5	Where a station consists of a number of SMR modules, safety-related SSCs (such as containment immersion, control rooms, turbine halls etc.) may be shared. A ship-based design may have SMRs that share the ship.
7.16	Commissioning multiple units	8.5	If a plant has many SMR modules, should there be a provision for reduced commissioning for units after the first one? Can passive systems such as containment heat removal be adequately tested during commissioning?
7.19	Transport and packaging	8.9	There will need to be approved packages for an entire core (new and used) in some designs.
7.21	Human factors	8.5	There will need to be HF requirements for one operator managing multiple units, in both NOC and accidents.
7.22	Malevolent acts	8.6	There will need to be generic guidelines on deliberate large aircraft crash – under what circumstances can it be ruled out on the basis of probability? If not, how is an appropriate event defined? Is there enough R&D to support the protection provided by underground siting (e.g. from military experience)?
7.22.4	Cyber security	8.6	The proposal by one vendor for satellite control of SMRs would need far more detailed requirements, verification, testing etc., and probably development of an industry standard.

8.4	Shutdown redundancy	8.14	It is not clear if all sample SMRs have two means of shutdown (some do). Most LWRs analyse Anticipated Transient Without Scram (ATWS), but not DBAs + failure of the fast shutdown system (rods), which are dismissed on the grounds of very low probability. The latter case may be challenging not just for SMRs but also for conventional LWRs. It can be solved (e.g. at the Sizewell plant in the UK) by having two fast shutdown systems. The REGDOC requires the use of stored energy in shutdown systems – this seems overly prescriptive, unless gravitational potential energy is included.
8.8	Heat sink redundancy	8.14	In some SMRs there may be just one passive decay heat removal system for the reactor and the containment and ECCS. The case would be that the system has high reliability because it is passive, and decay heat need only be removed once.
8.9	AC power redundancy	8.14	If an SMR does not depend on AC power for safety, these requirements could be significantly relaxed – e.g. there may be no need for alternate AC power.
11	Alternative approaches	N/A	In principle this section allows a vendor to propose an alternative approach to any part of the REGDOC, subject to a demonstration of equivalent safety. In practice for a different concept such as an advanced SMR, it is too open-ended to guide the vendor and for regulatory certainty.

10. Discussion and Recommendations

Based on the review of CNSC regulatory documents and the assessment of emerging SMR technologies, the following recommendations are made to the CNSC:

1. The existing regulatory framework is generally adequate for the large mostly-conventional SMRs such as KLT-40S and VBER-300, with a couple of major exceptions. These exceptions are that they are ship-based designs, and they exclude LBLOCA from design basis. We believe the exceptions can be treated on a case-by-case basis.
2. For advanced SMRs, much of REGDOC-2.5.2 applies. However, it lacks a framework for applying risk-informed regulatory judgements on innovative features. A supplement applicable to advanced SMRs would fulfill this purpose. It should deal with the high-level issues identified in Section 8. That supplement should suffice to put a pre-project design review on a good foundation.
3. Effectively what is missing from REGDOC-2.5.2 are requirements and guidance on applying a graded approach to advanced SMR design. The graded approach is defined in Reference [64], in general as:

   "a structured method by means of which the stringency of application of requirements is varied in accordance with the circumstances and the regulatory and management systems used";

   and in detail as:

   "For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control".

   "An application of safety requirements that is commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures."

   Reference [64] is focussed on research reactors but the concepts could be applied more broadly. CNSC permits the use of a graded approach to safety for small reactors [65] – the focus there is on safety analysis rather than design, but it notes that *"The graded approach is a method in which the stringency of the design measures and analyses applied are commensurate with the level of risk posed by the reactor facility."*

   It is suggested that the supplement to REGDOC-2.5.2 discussed in item 2 above contain such requirements and guidance on a graded approach.
4. For advanced SMRs, some of the increased safety is achieved not through inherent means (low core power) but by engineered passive systems. These may be given much more

credit in the safety case, in terms of reliability, than active systems – e.g. they may not (need to) be redundant. While this might be justified, guidance on credit for passive systems in SMRs would be a useful regulatory tool. Reference [25] notes that *"There is the need to demonstrate the understanding of the key thermal- hydraulic phenomena that are selected for characterizing the performance of passive systems"*; and *"… a systematic effort for evaluating the level of understanding of thermal-hydraulic phenomena for passive systems and connected code capabilities appears to be limited and in general lacking."*

5. If an operator proceeds to a construction licence, more detailed review criteria will be needed. The US is developing specific Standard Review Plans for each SMR design it is asked to review. That model could also be used in Canada – it might be similar to the "Guidance" sections in REGDOC-2.5.2 except parts would be design-specific.
6. Since all but one of the designs that are examined in this report has been developed outside of Canada, the CNSC could liaise with the regulator of the country where the design is to be deployed first, or where it has had pre-project licensing review (e.g. Korea, Argentina, and the U.S.). The purpose would be to fast-track an understanding of the design and the regulatory issues addressed in that country.

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APPENDIX A - Definition of Safety Terminology

The following definitions of *inherent, passive,* and *grace period* are used in this report, as taken from [66]:

1) Inherent safety characteristic
Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept

2) Passive Component
A component which does not need any external input to operate

3) Active component
Any component that is not passive is active

4) Passive system
Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation

5) Active system
Any system that is not passive is active

6) Fail-safe
The term describes the behaviour of a component or system, following a failure (either internal or external). If a given failure leads directly to a safe condition, the component or system is fail-safe with respect to that failure

7) Grace period
The grace period is the period of time during which a safety function is ensured without the necessity of personnel action in the event of an incident/accident

Note that [66] deprecates the use of an "inherently safe reactor". Such a reactor would somehow have to eliminate or exclude inherent hazards such as radioactive fission products and their associated decay heat, excess reactivity, and energy releases due to high temperatures, high pressures and energetic chemical reactions. This concept might largely apply to a reactor such as the 20 kWth SLOWPOKE-2 [67], but is not easily conceivable for reactors in the MW range. Even for SLOWPOKE, there are a few aspects which are not inherently safe, such as the *physical* ability to insert fissile material in an irradiation tube – this must be prohibited by interlocks or procedure. We therefore do not use the concept of an inherently safe *reactor* in this task – just inherently safe characteristics.

Even passive safety has degrees. Often systems are claimed to be passively safe in terms of using natural circulation to remove heat, but in order to operate them, valves must be opened or closed, using AC or DC power, or stored energy. Reference [66] provides a gradation of degrees of passive safety, summarized in Table 22 below:

Table 22 - Categories of Passive Safety

Characteristic	Category A	Category B	Category C	Category D
Signal Inputs of Intelligence	**No**	**No**	**No**	**Yes**
External power sources or forces	**No**	**No**	**No**	**No**
Moving mechanical parts	**No**	**No**	**Yes**	**Either**
Moving working fluid	**No**	**Yes**	**Yes**	**Either**
Example	**Barriers such as fuel clad, containment; core cooling relying only on radiation or conduction to outer structural parts**	**Heat removal by natural circulation to heat exchangers in water pools, from the core or containment**	**Rupture disk or spring loaded valve for overpressure protection; accumulator isolated by check valve**	**Shutdown System #1 and #2 in CANDU**

APPENDIX B – Defence-in-Depth Summary

The definitions of defence-in-depth levels are taken from Reference [63] and reproduced in Table 23 below.

Table 23 - Levels of Defence-in-Depth

Levels of Defence in Depth	Objective	Essential means for achieving the objective
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control and abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant release of radioactive materials	Off-site emergency response

APPENDIX C - Questions/Comments received following the presentation to CNSC in Ottawa on June 16, 2014

CNSC Question

NuScale has a CRDM – Control Rod Drive Mechanism on top of the RPV, while mPower's CDRM is just above the core. Could you comment on what factors could have influenced such designer's decisions and what would be advantage/ disadvantage of such design concepts? What could be potential concerns from the regulatory point of view?

Author's Response

Placing a CRDM inside the pressure vessel reduces the number of vessel penetrations and it removes the possibility of control rod ejection. However, the drive mechanism is exposed to the primary system environment which may contribute to additional component degradation. In addition, CRDM inspection and maintenance would be more difficult; i.e. mPower design requires the top and bottom vessel components to disconnect to allow an access to CRDM for inspection and part replacement. The reliable operability of the CRDM needs to be demonstrated in the primary system condition.

NuScale's CRDM is similar to the conventional PWR CRDM except that the control rods assemblies are attached to longer rods, which may require additional mechanical component performances such as resistance to bending, etc. The positioning of the drives allows easier inspection and maintenance.

CNSC Comment

The vendor of the NuScale stated that quantity of water in the stainless steel-lined concrete pool is large enough to provide 30 days of core and containment cooling by ECCS (emergency core cooling system), i.e., a Long-Term Cooling (in the case of a complete station blackout) without additional cooling or water addition. After 30 days when water pool is emptied, the LTC is enabled by the air through natural convection.

This claim should be supported either by adequately performed experimental study and/or by numerical simulation using advanced predicting models. Recently performed simulation of natural convection under BDBA conditions obtained at the MASLWR test facility by using best estimate T-H code TRACE have shown that buoyancy dominated flow and heat transfer cannot be predicted accurately, so that the predicted inlet and outlet core coolant temperatures considerably overpredict the experimental results, and that trend of growth is worsen with time.

Author's response

We agree with CNSC's view that experimental evidences or numerical simulation results need to be provided for this important safety claim.

<u>CNSC question</u>

Could you elaborate on the fuel qualification criteria for SMR designs with fuel cycles longer than four years (in particular for SMRs with very high fuel temperature, and burnup)?

<u>Author's response</u>

This is beyond the scope for this study. However, it is noted that most water-cooled SMR designs utilize shortened PWR fuel assemblies such as Westinghouse 17 × 17 FA or VVER hexagonal fuel assemblies with burnups comparable to those of conventional PWR fuel. Only two SMR designs aim to achieve very high burnup, and they are KLT-40S using cermet fuel, and StarCore using TRISO fuel particles. Since these types of fuel have been utilized in operating power reactors, we expect that some qualification data would exist for these fuel types.

<u>CNSC comment</u>

Any commercial benefit of SMR reduces safety margin, so that everything what is claimed as a benefit has to be analyzed from the safety point of view (e.g., small number of operational staff or remote operator, remote location, etc.).

<u>Author's Response</u>

The commercial benefit can be realized in various ways including shorter construction time due to reduced earthwork and modularization. These savings do not necessarily reduce safety margin.

Some SMRs' inherent safety characteristics can actually improve the safety margin compared to that of conventional reactors. However, designer's attempt to improve the design's commercial benefit may result in lowering safety margin. From regulatory point of view, the additional safety margin (due to inherent and passive safety characteristics) and the reduction of safety margin (due to efforts to improve commercial benefit) should result in a net safety margin for an SMR that is equivalent or better than that of conventional reactors.

<u>CNSC Question</u>

Could you provide information (found/not-found in the used documents) for considered SMRs in the matrix format for the following issues:

a) Set of transients and DBAs for each design, especially internal LOCA (on the interface between 1st & 2nd circuits) for integrated reactors (e.g., is the location of LOCA on the dividing plate or the tube deck, gasket between 1st & 2nd parts of reactor vessel, SG tube, 1st circuit relief valve, etc.). *Vendor must demonstrate that MSLB will not lead to the criticality after reactor shutdown due to negative Doppler and coolant temperature* *reactivity effects. In-core LOCA for integral reactors could have more severe consequences than LBLOCA for non-integral SMRs (KLT-40S & VBER-300) because the pressure and temperature of the 1st & 2nd circuit become equal and, as a consequence, after isolation valve of the SGs is closed, heat transfer from the 1st to 2nd circuit will stop.*

b) Reactivity accident: if there is no boron in the SMR coolant, than it should be another reactivity device for compensation of Xe and Sm, as well as Doppler and temperature effects. Extraction or ejection of such device could occur.

Author's comment → soluble boron is used for long term reactivity suppression, not necessarily for Xe reactivity compensation.

c) Set of the SMR's BDBAs and DECs for the single reactor and for the sight (i.e., more than a single reactor) with any complementary design features to prevent accident progression and to mitigate the consequences of DECs and BDBAs.

d) Fukushima lessons learned for inert (N2 gas) containment: Filtered venting; prevention of H2 deflagration/explosion inside and outside of the containment; long time of hydrogen suppression (*after a few blowouts nitrogen could be entirely or partly replaced with the generated H2 or O2*) (Vaccum, ventilation, etc)

e) Common cause failure for multi-unit NPP (e.g., 10 units)

f) Are the Human factors and human reliability for small number of operators addressed by the vendor?

Note: For all of the considered SMR reactors, please inform whether the issues (in the 1st column) are discussed/mentioned or simply not considered in the reviewed SMR's documents. For some SMR you can say that the listed issue is N/A; for some the issue is not discussed/mentioned (but it is important for pertinent SMR; and for some others, the issue is discussed/mentioned (in this case, it would be enough just to refer the reference [ ].

	KLT-40S	VBER-300	CAREM	SMART	StarCore	mPower	NuScale
a)	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope
b)	Cadmium nitrate used	Soluble boron used	Soluble neutron poison not used	Soluble boron used	N/A	Not discussed	Soluble boron used
c)	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope
d)	Not found	Not found	Not found	Discussed in [45]	N/A	Discussed in [50]	Discussed in [49]
e)	2 unit NPP. Any damage to the vessel will endanger both reactors	N/A. Single unit plant	N/A Single unit plant	N/A Single unit plant	Satellite failure	2~4 units NPP Similar to conventional multi-unit plant	12 units in a common pool. Operator room controls multiple units.
f)	Conventional operation	Conventional operation	Not found (CAREM-27 is a prototype demonstration)	Conventional operation	Excludes human participation in the operation	Conventional operation	Discussed in [49]